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North Carolina Electric Membership Corporation
Saluda River Electric Cooperative, Inc.

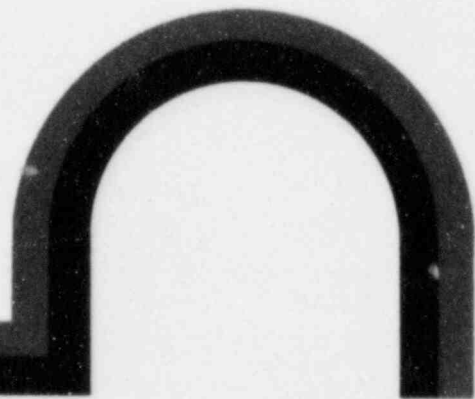
**Catawba Nuclear Station
Unit 1
STARTUP REPORT**

Docket No. 50-413, License No. NPF-35

50-413

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DUKE POWER COMPANY
NORTH CAROLINA ELECTRIC MEMBERSHIP CORPORATION
SALUDA RIVER ELECTRIC COOPERATIVE, INC.

CATAWBA NUCLEAR STATION

UNIT NO. 1

DOCKET NO. 50-413

LICENSE NO. NPF-35

STARTUP REPORT

September 27, 1985

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1.0 INTRODUCTION

The Catawba Nuclear Station Unit 1, located on Lake Wylie in South Carolina, consists of a Westinghouse 4 loop pressurized nuclear supply system rated at 3411 MWt and a General Electric turbine-generator rated at 1205 MWe with unit net of 1145 MWe. The station is located approximately 19 miles southwest of Charlotte, North Carolina and 6 miles north of Rock Hill, South Carolina off Highway 274 near Newport, South Carolina.

The design and fabrication of the initial core, supplied by Westinghouse, is of the optimized fuel design and consists of 1.6%, 2.4% and 3.1% nominal enrichments. The core consists of 193 assemblies each containing 17 x 17 fuel rod array with 264 fuel rods.

The Catawba Nuclear Station was designed and constructed by Duke Power Company. Duke Power Company is responsible for operation and maintenance of the unit on behalf of the principle owners: Duke Power Company, Saluda River Electric Cooperative, Inc., and North Carolina Electric Membership Cooperative.

Construction started at the Catawba site under Limited Work Authorization on May 16, 1974. Construction Permit CPPR-116 was issued by the Atomic Energy Commission on August 7, 1975. Facility Operating License NPF-24 was issued on July 18, 1984.

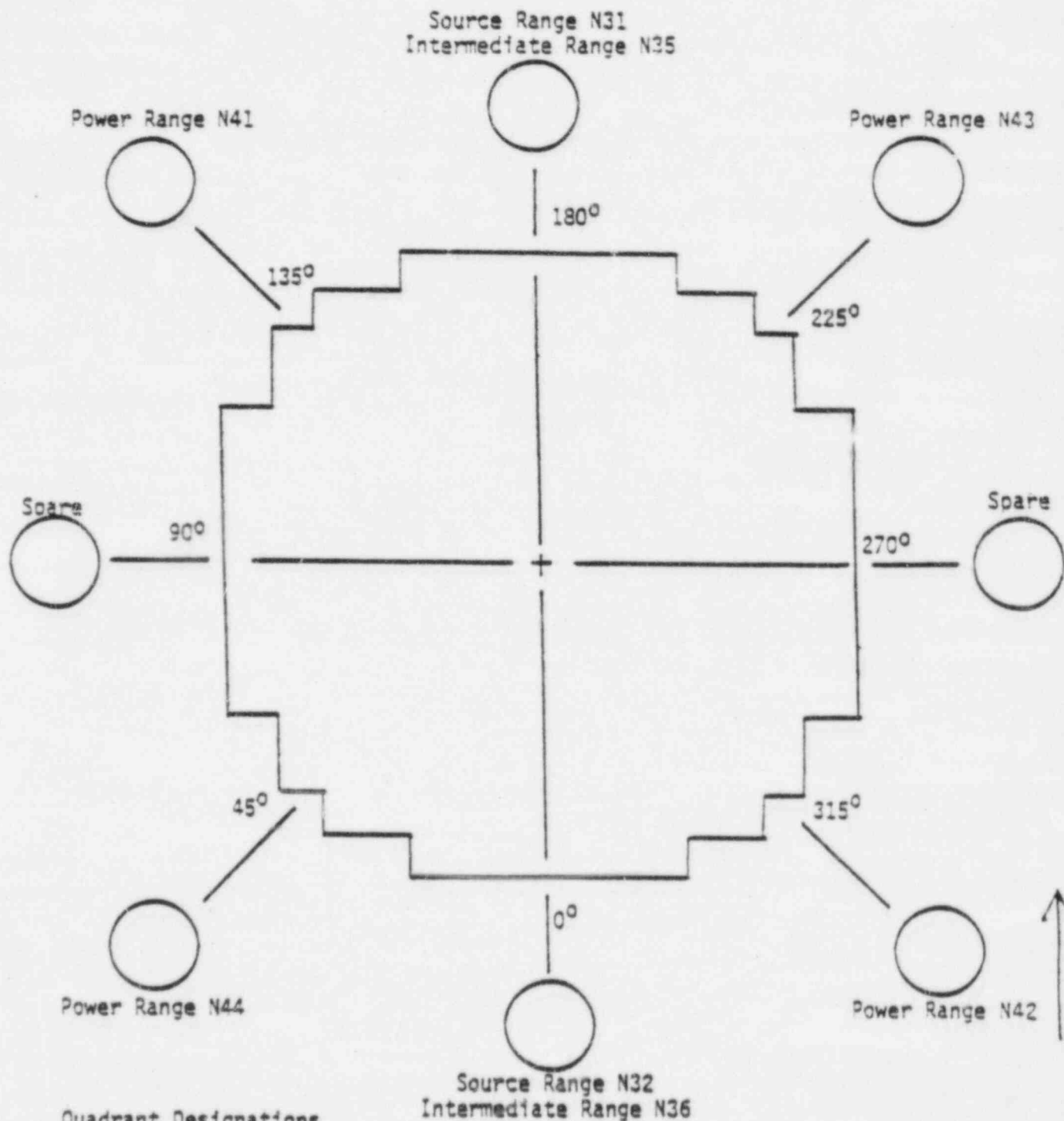
This report is prepared in accordance with the requirements of Regulatory Guide 1.16 Revision 4 and 1.68 Revision 2 and Catawba Unit 1 Technical Specifications 6.9.1. It addresses the results of preoperational and startup testing, as described in Catawba FSAR Chapter 14 through Revision 11, conducted between Initial Fuel Loading and Commercial Operation.

Table 1.0-1 provides a reference of abbreviations and acronyms used throughout this report. Figures 1.0-1 through 1.0-4 show general information for Catawba 1 reactor core referenced in this report.

CATAWBA ABBREVIATIONS AND ACRONYMS

ARO	-	All Rods Out
BB	-	Steam Generator Blowdown System
CA	-	Auxiliary Feedwater System
CF	-	Feedwater System
CRDM	-	Control Rod Drive Mechanism
D/G	-	Diesel Generator
DRPI	-	Digital Rod Position Indication
FCM	-	Full Core Map
HFT	-	Hot Functional Testing
IAE	-	Instrumentation and Electrical Personnel
I/R	-	Intermediate Range
ITC	-	Isothermal Temperature Coefficient
NB	-	Boron Recycle System
NC	-	Reactor Coolant System
NCP	-	Reactor Coolant Pump
ND	-	Residual Heat Removal System
NEO	-	Nuclear Equipment Operator
NI	-	Safety Injection System
NIS	-	Nuclear Instrumentation System: Excore Source Range, Intermediate Range and Power Range Detectors
NV	-	Chemical and Volume Control System
OAC	-	Operator Aid Computer
PZR	-	Pressurizer
QCM	-	Quarter Core Map
RC	-	Condenser Cooling Water System
RCCA	-	Rod Control Cluster Assembly
RN	-	Nuclear Service Water System
S/G	-	Steam Generator
SRO	-	Senior Reactor Operator
T/C	-	Thermocouple
T/D	-	Turbine Driven
VE	-	Annulus Ventilation System
VF	-	Spent Fuel Pool Ventilation System
VI	-	Instrument Air System
WL	-	Liquid Radioactive Waste System
WS	-	Solid Radioactive Waste System
YV	-	Containment Chilled Water System
ZPPT	-	Zero Power Physics Testing

REACTOR CORE MAP
EXCORE DETECTOR LOCATIONS



Quadrant Designations

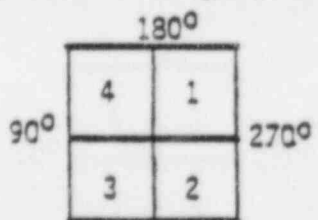


Figure 1.0-1

REACTOR CORE MAP
CONTROL ROD LOCATIONS

	R	P	N	M	L	K	J	H	G	F	E	D	C	B	A
1															
2				SA-4 GR-2		CB-4 GR-2		CC-1 GR-1		CB-1 GR-1		SA-1 GR-1			
3					SD-4 GR-1		SB-4 GR-2		SB-1 GR-1		SC-1 GR-1				
4	SA-4 GR-1			CD-2 GR-2				SE-1 GR-1				CD-1 GR-1		SA-1 GR-2	
5			SC-4 GR-1										SD-1 GR-1		
6	CB-4 GR-1					CC-4 GR-2		CA-1 GR-1		CC-1 GR-2				CB-1 GR-2	
7			SB-4 GR-1										SB-1 GR-2		
8	CC-4 GR-1		SE-4 GR-1		CA-2 GR-2		CD-3 GR-2		CA-1 GR-2		SE-2 GR-1		CC-2 GR-1		
9			SB-3 GR-2										SB-2 GR-1		
10	CB-3 GR-2					CC-3 GR-2		CA-2 GR-1		CC-2 GR-2				CB-2 GR-1	
11			SD-3 GR-1										SC-2 GR-1		
12	SA-3 GR-2			CD-2 GR-1				SE-3 GR-1				CD-1 GR-2		SA-2 GR-1	
13					SC-3 GR-1		SB-3 GR-1		SB-2 GR-2		SD-2 GR-1				
14				SA-3 GR-1		CB-3 GR-1		CC-3 GR-1		CB-2 GR-2		SA-2 GR-2			
15															

LEGEND:

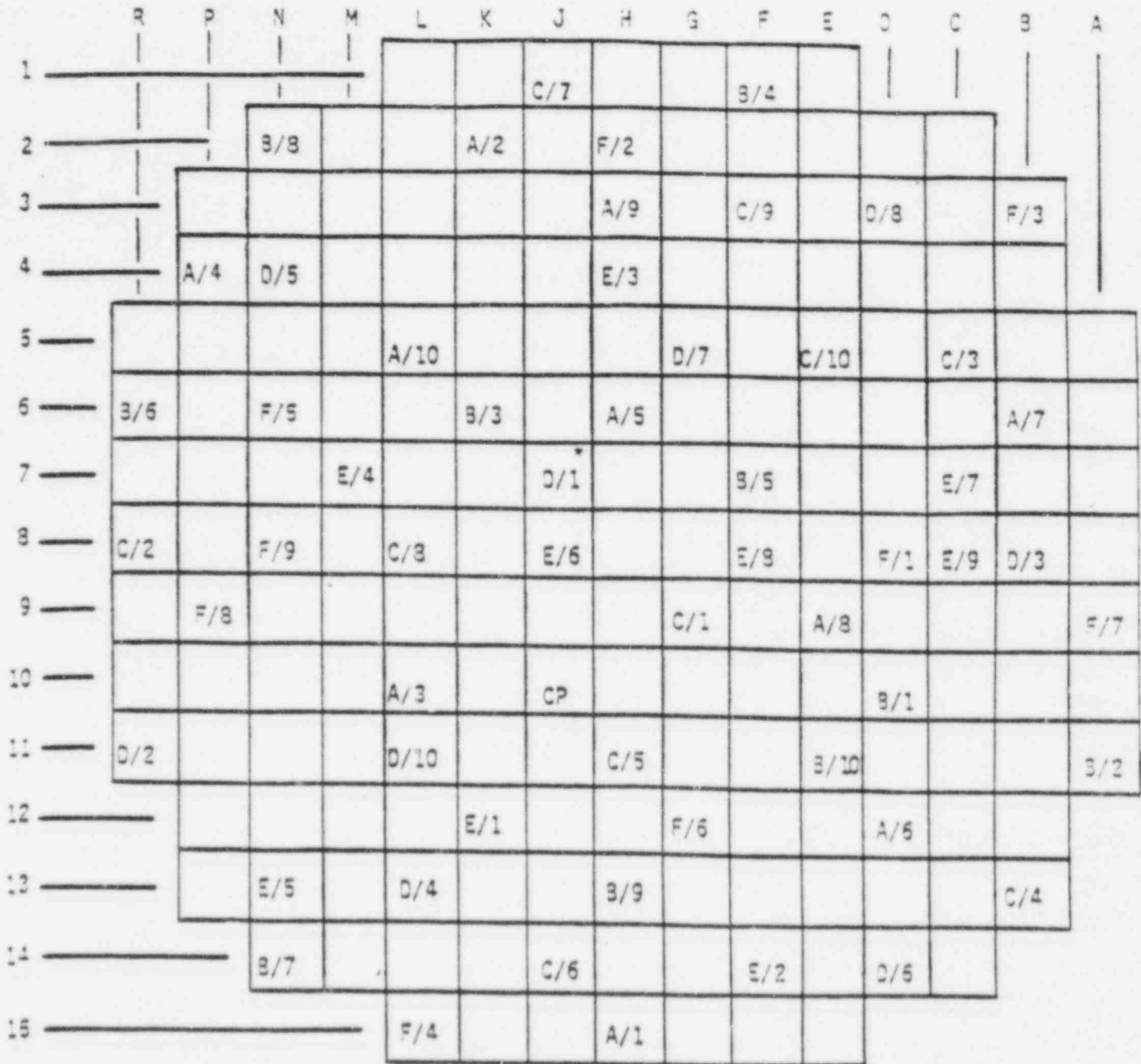
xx-y
GR A

xx - Bank Name
y - RCC No.
A - Group No.



Figure 1.0-2

REACTOR CORE MAP
MOVABLE INCORE DETECTOR
THIMBLE LOCATIONS



* This location will be used for the RVLIS system.

LEGEND:

- x - Detector No.
- x/yy - Detector X
- Normal Mode
- IC Path Position
- CP - Calibrate Path

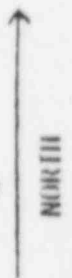
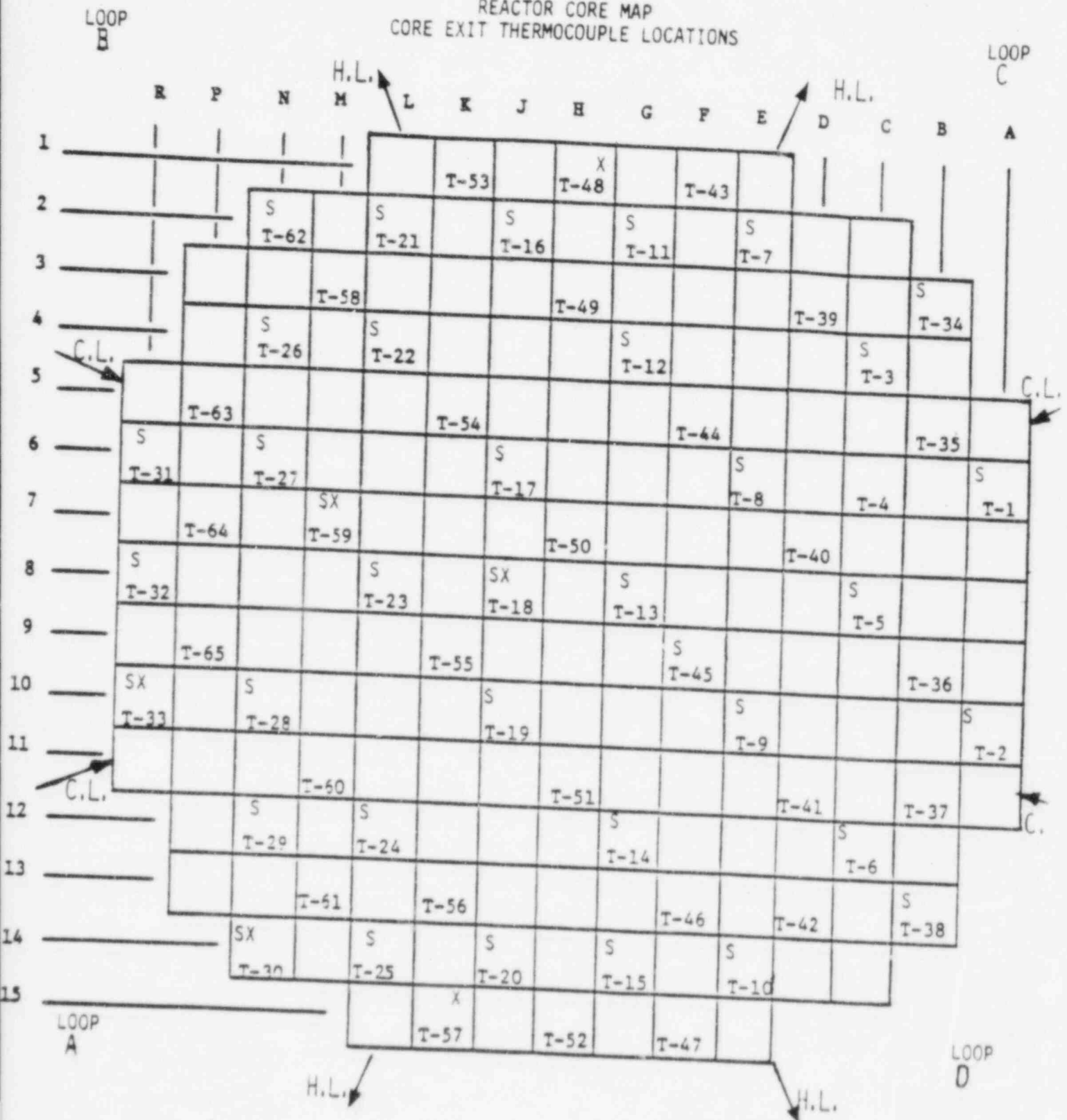


Figure 1.0-3

REACTOR CORE MAP
CORE EXIT THERMOCOUPLE LOCATIONS



LEGEND:
 S X
 T-xx xx - Thermocouple No.
 S - Safety Grade
 X - Unreliable Indication
 H.L. - Hot Leg
 C.L. - Cold Leg

Figure 1.0-4

2.0 SUMMARY

Significant startup milestones and events for Catawba Unit 1 initial startup are listed below:

Receipt of Fuel Loading and Precritical Testing License (NPF-24)	July 18, 1984
Start of Initial Fuel Loading	July 18, 1984
Completion of Initial Fuel Loading	July 23, 1984
Outage to repair thermocouple (T/C) column leak	October 23, 1984- November 17, 1984
Receipt of Low Power Testing (< 5%) License (NPF-31)	December 6, 1984
Outage to repair control rod connecting rods (IE Information Notice 85-14)	December 8, 1984 January 5, 1985
Initial Criticality	January 7, 1985
Start of Zero Power Physics Testing	January 7, 1985
Receipt of Full Power Operating License (NPF-35)	January 17, 1985
Completion of Zero Power Physics Testing	January 20, 1985
Commenced Power Escalation Testing	January 21, 1985
First turbine roll with nuclear-generated steam	January 22, 1985
Outage to repair Condenser Cooling Water Piping leak	February 9, 1985- February 28, 1985
Outage to remove turbine valve temporary strainers and to perform other maintenance	April 19, 1985- June 10, 1985
Commercial Operation	June 29, 1985

Catawba Unit 1 startup and power escalation testing as addressed in this report are summarized below.

A. Initial Fuel Loading

Initial Fuel Loading began on July 18, 1984 and was completed on July 23, 1984. The core loading was verified on July 24, 1984. A "leaning" assembly could not be placed in its designated location by the loading sequence. This assembly was "boxed" into its designated location, which is a routine practice for reload cores. No fuel handling equipment malfunctions were encountered. Refer to Section 3.0 for details.

B. Testing Prior to Initial Criticality

Following initial fuel loading, various tests were performed prior to initial criticality. Some of these tests required the core to be assembled, upper vessel internals, and vessel head installed, and instrumentation to the vessel complete. The Pressurizer Functional Test, NC Flow Test, and NC Flow Coastdown Test failed to meet the FSAR Acceptance Criteria. FSAR revisions were implemented for these tests prior to initial criticality. Various test were repeated, as they will be following each refueling, due to a vessel T/C column leak and inspections of the control rod latch assemblies. Refer to Section 4.0 for details.

Other testing prior to initial criticality included the completion of various preoperational tests. Refer to Section 12.0 for details.

C. Initial Criticality

Initial criticality was achieved on January 7, 1985 at 2008 hours with Control Bank D at \approx 141 steps, withdrawn. The critical boron concentration was 969 ppmB. Refer to Section 5.0 for details.

D. Zero Power Physics Testing

Zero Power Physics Testing began on January 7, 1985 and was completed January 17, 1985. Fundamental nuclear characteristics were measured and compared to predictions. An unexpected core power tilt was measured. The tilt was confirmed by multiple measurements and evaluated as acceptable. Refer to Section 6.0 for details. Natural Circulation was determined on January 19 and 20, 1985. A 29.2 degree delta T was observed to confirm the natural circulation characteristics. Refer to Section 7.0 for details.

E. Power Escalation Testing

The power escalation testing program was designed to provide initial startup data in areas of core physics, controls and instrumentation, plant transients, chemical control, and behavior of the plants radiological environment. Testing began on January 21, 1985.

The core power tilt was routinely measured to verify that its magnitude decreased as power increased. The resulting core power tilt during performance of the Below Bank Test was greater than expected and delayed subsequent testing by approximately 7 days. Refer to Sections 8.0 through 12.0 for details.

F. Special Reports

Special Reports are provided for the Loose Parts Monitoring and Post Accident Liquid Sampling Systems. Refer to Section 13.0.

Tables 2.0-1 through 2.0-12 provide a detailed monthly summary of operations from fuel loading through power escalation to declaration of commercial operation. Figures 2.0-1 through 2.0-6 provides an overview of the monthly power history.

MONTHLY UNIT OPERATION SUMMARY

July 1984

<u>Date</u>	<u>Time</u>	<u>Event</u>
07/18/84		Received Fuel Loading and Precritical Testing License NPF-24
07/19/84	0536	Entered Mode 6 Commenced TP/1/A/2650/01, Initial Fuel Loading
	0536	First Fuel Assembly loaded into core (C04)
07/23/84	1426	Last fuel assembly (C44) loaded into core
	1914	Commenced PT/1/A/4550/03C, Core Verification
07/24/84	1821	Completed Core Verification and Initial Fuel Loading
	1830	Commenced precritical vessel activities

TABLE 2.0-1

CATAWBA 1

MONTHLY UNIT OPERATION SUMMARY

August 1984

<u>Date</u>	<u>Time</u>	<u>Event</u>
08/03/84	1905	Entered Mode 5

TABLE 2.0-2

CATAWBA 1

MONTHLY UNIT OPERATION SUMMARY

September 1984

<u>Date</u>	<u>Time</u>	<u>Event</u>
09/01/84	1100	Commenced RCCA Drop Timing, DRPI Operability, and CRDM Timing Testing
09/11/84	2050	Completed control rod testing
09/28/84	1234	Entered Mode 4

MONTHLY UNIT OPERATION SUMMARY

October 1984

<u>Date</u>	<u>Time</u>	<u>Event</u>
10/03/84	0800	Commencing TP/1/A/1150/08A, Thermal Expansion of ASME Piping and IP/0/A/3231/01, T/C - RTD Cross Calibration
10/04/84	2000	Commencing TP/1/A/1100/06, D/G 1B Testing
10/09/84	1517	Commencing TP/1/A(B)/1400/05B CO ₂ Testing for D/G Rooms
	2120	Cooldown commencing for ND System Vibration measurement per PT/1/A/4200/15A
10/11/84	0815	Commencing NC System Cooldown to repair S/G Level instrument root valve
	1850	Entered Mode 5
10/12/84	0800	Commencing ND system piping/pump repairs
	1600	Commencing RN pump run per TP/1/A/2650/02, Essential Transformer Tap Readings
10/15/84	0850	Commencing TP/1/A/1200/21, Steady State Piping Vibration Measurements for D/G 1B Support Systems
10/17/84	2320	Commencing NC System Fill/Vent for Heatup
10/20/84	1309	Entered Mode 4
10/22/84	1504	Entered Mode 3
10/23/84	2145	Commencing NC System cooldown to repair T/C column leak and clean vessel head/components
10/24/84	0026	Entered Mode 4
	1104	Entered Mode 5
10/31/84	1042	Entered Mode 6

TABLE 2.0-4

MONTHLY UNIT OPERATION SUMMARY

November 1984

<u>Date</u>	<u>Time</u>	<u>Event</u>
11/08/84	1846	Entered Mode 5
11/10/84	1115	Began NC System heatup
11/11/84	1215	Commencing DRPI Alignment
11/14/84	1021	Received Feedwater Isolation due to instrumentation calibration on S/G levels
11/15/84	1658	Entered Mode 4
11/17/84	0430	Entered Mode 3
	1145	Commencing Thermal Expansion and T/C-RTD Testing at = 450°F
11/18/84	0900	Commencing NC System Heatup
11/20/84	1241	NC System at = 557°F, 2235 psig, Commencing Hot No-Load testing:
		TP/1/A/1150/08A Thermal Expansion Testing
		TP/1/A/1200/21 Steady State Piping Vibration
		TP/1/A/3231/01 T/C-RTD Cross Calibration
		TP/1/A/2150/08 RTD Bypass Flow Verification
		TP/1/A/2150/01 NC Flow Test
		TP/1/A/2150/13 Pressurizer Functional
		TP/1/B/2600/06 CRDM Timing
		TP/1/A/1200/26 Post Transient Piping Surveys
		TP/1/A/2600/07 CRDM Drop Timing
		TP/1/A/2650/02 Essential Transformer Voltage Measurements (NC Pump)
		TP/1/A/2150/02 NC Flow Coastdown Test
		TP/1/B/2600/05 CRDM Alignment
		TP/1/A/2600/04 CRDM Position Indication
		TP/1/B/2650/09 Secondary Systems Testing (CA T/D Pump)
		TP/1/A/2650/04 S/G Blowdown Testing

MONTHLY UNIT OPERATION SUMMARY

December 1984

<u>Date</u>	<u>Time</u>	<u>Event</u>
12/01/84		D/G 1A/1B and Hot No-Load Testing in progress
12/06/84		Receipt of Low Power Testing (<5%) License
12/08/84	1100	Commencing NC System Cooldown to investigate and repair control rod connecting rod latching pin. Vessel head removal required.
12/09/84	0350	Entered Mode 4
	1158	Entered Mode 5
12/13/84	0145	Entered Mode 6
12/20/84	0440	Entered Mode 5
	0445	Entered Mode 6
	0515	Entered Mode 5
12/25/84	2201	Opened NC32B NC Pressurizer PORV during fill/vent process
12/27/84	0200	Commencing NC System heatup
12/31/84	2226	Entered Mode 4

MONTHLY UNIT OPERATION SUMMARY

January 1985

<u>Date</u>	<u>Time</u>	<u>Event</u>
01/01/85	2140	Commenced heating up NC System to $\approx 345^{\circ}\text{F}$
01/02/85	0230	Holding at 345°F NC Temperature
	1610	Entered Mode 3
01/03/85	0240	Tripped Reactor Coolant Pump (NCP) 1B due to emergency high vibration
	0300	Started NCP 1B
	0314	NEO's dispatched to NCP 1B to monitor vibration with hand held vibration meter. Vibration still present.
	0400	NCP 1B vibration alarms cleared
	1430	NC temperature at 557°F
	1437	Commenced IP/0/A/3220/01 - Rod Drop Testing
01/04/85	1350	Rx trip signal generated when instrumentation personnel took channel II turbine impulse pressure to test - inserted P-7 signal causing a reactor trip on low pressure. Feedwater isolation and CA pumps 1A & 1B auto start.
01/06/85	0500	Commenced PT/1/A/4150/01A - Reactor Coolant System Leak Test
	0930	Commenced PT/0/A/4150/01C - Reactor Coolant System Controlled Leakage Verification
	1726	Began baseline data for initial approach to criticality
01/07/85	0237	Began NC System Dilution
	1457	Entered Mode 2
	2008	Rx Critical
	2030	4×10^{-10} amps on I/R, 972 ppm, 557°F

TABLE 2.0-7

CATAWBA 1

MONTHLY UNIT OPERATION SUMMARY

January 1985

<u>Date</u>	<u>Time</u>	<u>Event</u>
01/07/85	2224	Completed 1/M Approach to Criticality commenced Zero Power Physics Testing Controlling Procedure
01/08/85	0345	Rx taken subcritical to take data on source range/int. range overlap
	0425	Rx again critical
	0601	Rx power @ = 10^{-10} amps
	0639	Rx power @ 10^{-8} amps
	0800	Increased Rx power to 10^{-7} amps to increase sensitivity of reactivity measurement equipment
	1400	Temporary mod completed to switch reactivity signal from N44 to N42 due to excessive noise on N44
	1528	Attained Nuclear Heat 4×10^{-6} amps on I/R CH 36. Decreasing power to 1×10^{-6} per Performance
	1636	Pulled rods to achieve Nuclear Heat, Nuclear Heat at 4×10^{-6} amps on I/R Ch 36
	1655	Decreasing Rx power to 3×10^{-8} amp per Performance
	1712	Rx power @ 3×10^{-7} amps
	1745	Increased Rx power to 2×10^{-7} amps per Performance in efforts to eliminate noise on N42
01/09/85	0445	Began PT/1/A/4150/12A - ARO ITC Measurement Results invalid due to noise on N42
	1470	SSPS Train B surveillance test procedure inadequacy caused actuation of SR Hi Flux Rx Trip which caused 'B' Rx trip breaker to open. 'B' bypass breaker was closed.

MONTHLY UNIT OPERATION SUMMARY

January 1985

<u>Date</u>	<u>Time</u>	<u>Event</u>
01/09/85	1720	Resolved N42 noise problems by installing inline filter and increasing physics test band
	1840/2300	Repeated PT/1/A/4150/11A ARO ITC Measurement
01/10/85	0205	Increased Rx power to $\approx 3\%$ to allow low power flux mapping of core
	0545	Began ARO Core Power Distribution
	1230	Commence Rx shutdown due to both trains of VE inop., shutting down @ 10%/hr under T.S. 3.0.3
	1246	Secured from Rx shutdown. "A" Train VE restored
	1915	Decreased Rx power to 1×10^{-7} amps on N-35 per Test Coordinator for Rod Group testing
	2000/2345	NC System dilution for Control Bank D worth
01/11/85	0245	SRO declares "Notification of Unusual Event" due to the leakage > than Tech Spec limit
	0438	Placed 1B seal injection filter in service - secured 1A seal injection - NC leakage rate decreased substantially
	0545	Secured from Unusual Event - SRO making verbal notifications
	1030/1330	Performed PT/1/A/4150/11A - Control Bank D Inserted ITC Measurement
	1542	Rx power increased to $\approx 3.5\%$ for Core Power Distribution
01/12/85	0442/1100	Dilution of NC system per Performance Test Coordinator for Control Bank C worth
	1420/1800	Performed PT/1/A/4150/11A - Control Banks D and C inserted ITC Measurement
	1840/2115	NC Dilution to determine Control Bank B worth
01/13/85	0115/0430	NC Dilution to determine Control Bank A worth

TABLE 2.0-7 (cont.)

CATAWBA 1

MONTHLY UNIT OPERATION SUMMARY

January 1985

<u>Date</u>	<u>Time</u>	<u>Event</u>
01/13/85	1000	NC Dilution to determine shutdown E, D, C worth
	1600	Manual Rx trip per Zero Power Testing Procedure, entered Mode 3, NC system boration for shutdown margin
	2238	Commenced Rx Startup
01/14/85	0050	Entered Mode 2
	0136	Rx critical
	0240	Began NC dilution to get shutdown C inserted
	0640	Drove S/D Bank C into < 100 steps - Shutdown Margin < 1.3% $\Delta K/K$ operating under Special Test Exception 3.10.1
	1015	Began PT/1/A/4150/24 - Stuck Rod Test H-14 swap with Shutdown B and A
	1319	YV Chiller tripped due to loss of VI. VI lost to Cont. chill water due to ruptured line behind D/G Bldg.
	1321	Attempted swap of YV to RN, RN valves would not open
	1357/1404	Secured NCPs due to high stator temp.
	1425	Started 1B NCP to establish NC forced circulation. Stator temp reduced.
	1426	Secured 1B NCP due to emergency high vibration
	1427	Manual Rx trip from control room on recommendation from Rx Group Test Engineer in charge of Core Physics testing
	2130	Reactor trip investigation complete
01/15/85	0110	Commenced Rx startup for zero power testing
	0142	Entered Mode 2

MONTHLY UNIT OPERATION SUMMARY

January 1985

<u>Date</u>	<u>Time</u>	<u>Event</u>
01/15/85	0208	Rx critical
	0220	Critical @ 10^{-8} amps
	0410	NC Dilution to setup restart of Stuck Rod test
	1145/2100	Performed Stuck Rod Test
	2152	Tripped reactor per Test Procedure and started emergency boration for shutdown margin and to re-establish criticality with shutdown banks out/control banks in.
01/16/85	0335	Commenced Rx startup
	0358	Entered Mode 2
	1015/1415	NC Boration for Control Banks in Overlap worth to Zero power insertion limit
	1530	Rx power increased to = 4% to obtain Core Power Distribution
01/17/85	0100	Decreased power to 1.5%
	0300/2400	Pseudo Ejected Rod Test (Part 1).
	1125	Rx power increased to 3.5% for Full Core Flux Map
	2130	Decreased power to 2×10^{-7} amps
	0530/1100	NC System Boration for Control Rod Worth in Overlap
01/18/85	1455	Rx power increased to 2% for Turbine shell warming
		Received Full Power License
01/19/85	0400/1130	N-44 inoperable due to noise prohibiting setpoint checks
	1700	Rx power increased to 3% for Natural Circulation Test

MONTHLY UNIT OPERATION SUMMARY

January 1985

<u>Date</u>	<u>Time</u>	<u>Event</u>
01/19/85	2144	All NC Pumps tripped - Natural Circulation established
	2330	Decreased power to $i \approx 10^{-10}$ amps (Keff = 0.99)
01/20/85	0128	Entered Mode 3 due to cooldown to 541°F when NC Pump 1B was restarted (cold S/G water slugged through vessel).
	1708	Commenced Rx startup.
	1717	Reactor Critical with Control Bank D @ 35 steps wd Critical Boron Conc. = 900 ppmB Power Level = 1×10^{-8} amps
	1750	Rx power increased to 2×10^{-6} amps
	1800/1830	Nuclear Instrument Baseline at $\approx 0\%$
01/21/85	0310	Rx power increased to $\approx 3\%$ for Steam Dump Test
	0505	Rx power increased to $\approx 4\%$ for Steam Dump Test
	0540	Rx power decreased to $\approx 3\%$.
	0615/1300	NIS Baseline at $\approx 3\%$
	1445	Commenced power escalation
	1451:30	Entered Mode 1
	1455	Stopped escalation at 6% Rx power
	1500/1830	NIS Baseline at $\approx 6\%$
	1835	Increased Rx power to 9%
	1930/2100	NIS Baseline at $\approx 9\%$
01/22/85	0210	Turbine/Generator on line (Rx power = 10%)

MONTHLY UNIT OPERATION SUMMARY

January 1985

<u>Date</u>	<u>Time</u>	<u>Event</u>
01/22/85	0250	Turbine tripped due to high shaft vibration (Rx power = 12%)
	1551	Increased Rx power to 15%
	1757	Turbine/Generator on line (Rx power = 17%)
	2100	Rx power decreased to 6%
01/23/85	0230	Increased Rx power to 16%
	0330	Increased Rx power to 18%
	0620	Increased Rx power to 20%
	1447	Reactor trip from 20% due to Lo Lo S/G Level - Root causes Loss of Aux. Steam Pressure to Main Feedwater Pump Turbine. Entered Mode 3.
	0601	Commenced Reator Startup
01/25/85	0623	Reactor critical with Control Bank C @ 86 steps wd. Critical Boron Conc. = 857 ppmB Power Level = 1×10^{-8} amps
	0730	Increased Rx power to 4%
	0900	Entered Mode 1 (Rx power = 5%)
	1150	Increased Rx power to 7.5%.
	1305	Increased Rx power to 10.5%.
	1450	Increased Rx power to 14.5%.
	1605	Increased Rx power to 17.5%.
	1840	Increased Rx power to 20%.
	1856	Turbine/Generator on line (Rx power = 20%).
01/29/85	1220	Reactor Trip from 20% per Station Blackout Test. Entered Mode 3.
01/30/85	2307	Commenced Reactor Startup.

TABLE 2.0-7 (cont.)

MONTHLY UNIT OPERATION SUMMARY

January 1985

<u>Date</u>	<u>Time</u>	<u>Event</u>
01/30/85	2330	Reactor Critical with Control Bank D @ 31 steps wd. Critical Boron Conc. = 843 ppmB Power Level = 1×10^{-8} amps
	2345	Increased Rx power to 3%.
01/31/85	0014	Entered Mode 1 (Rx power = 5%).
	0040	Increased Rx power to 10%.
	0350	Turbine/Generator on line (Rx power = 10%).
	0410	Increased Rx power to 18%.
	0511	Increased Rx power to 20%.
	1424	Reactor trip from 20% per Loss of Control Room Test. Entered Mode 3.

MONTHLY UNIT OPERATION SUMMARY

January 1985

The Core Burnup Calculation for the month of January was based on 8 days of power operation which resulted in 4170 MWD of depletion. Burnup rates generated from a 20% F.P. Flux Map (FCM/1/01/005) taken on Jan. 28, 1985 were utilized by the Burnup Program. The results of the computer calculation are as follows:

Burnup for Period _____	51.355 MWD/MTU (1.222 EFPD)
Total Energy for Period _____	344.02 x 10 ⁹ BTU
Cummulative Cycle B/U _____	51.355 MWD/MTU (1.222 EFPD)
Cummulative Cycle Energy _____	344.02 x 10 ⁹ BTU

Other Data:

Generator hours on line	<u>100.27</u>
Reactor hours critical	<u>478.75</u>
Unit Electrical Gross (MWH)	<u>10060</u>
Unit Net Generation (MWH)	<u>-19986</u>

MONTHLY UNIT OPERATION SUMMARY

February 1985

<u>Date</u>	<u>Time</u>	<u>Event</u>
02/02/85	0100	Reactor Startup Commenced
	0845	Mode 2 Entered
	0849	Reactor Critical @ 38 steps wd. on Bank D Critical Boron = 886 ppm B
	1425	Mode 1 Entered
	1740	Rx Power increased to 10%
	1900	Rx Power increased to 17%
	2227	Turbine/Generator placed on line
02/03/85	0200	Rx Power increased to 20%
	0505	Feedwater swapped from Aux nozzles to Main nozzles per S/G Water Hammer Test
02/04/85	0340	Rx Power increased to 23%
	0445	Rx Power increased to 26%
	0555	Rx Power increased to 29%
	0700	Rx Power increased to 30%
02/06/85	0300	Rx Power decreased to 20% to prepare to take Turbine offline due to hydraulic fluid (LH) leak from a Turbine Control Valve
	0321	Rx Power at 10% Turbine/Generator taken off line
	0335	Rx Power decreased to 4% Entered Mode 2
	0453	Entered Mode 1
	0600	Rx Power increased to 8%
	0800	Rx Power increased to 12%
	1210	Entered Mode 2

MONTHLY UNIT OPERATION SUMMARY

February 1985

<u>Date</u>	<u>Time</u>	<u>Event</u>
02/06/85	1220	Entered Mode 3. Reactor shutdown to allow an Operator-In-Training to perform Reactor Startup
	1708	Reactor Startup Commenced
	1727	Entered Mode 2 Reactor Critical @ 85 steps wd. on Bank D Critical Boron = 751 ppm B
	1900	Entered Mode 1
	2000	Rx Power increased to 10%
02/07/85	0000	Rx Power increased to 17%
	0030	Rubbing on Turbine/Generator Shaft in Exciter Housing discovered
	0200	Rx Power decreased to 12% to reduce Turbine/Generator RPMs
	0500	Rx Power decreased to 7%
	0545	Entered Mode 2
	0610	Rx Power decreased to 1×10^{-8} amps on Intermediate Range
02/08/85	0441	Commenced increasing Rx Power from 1×10^{-8} Amps to 8%
	0449	Mode 1 Entered
	0500	Rx Power increased to 8%
	0510	Rx Power decrease initiated due to loss of both Main Feedwater Pumps because of low vacuum
	0512	Mode 2 Entered
	0600	Rx Power decreased to 1×10^{-8} Amps on Intermediate Range
	0909	Rx Power increased to 1%

TABLE 2.0-8 (cont.)

CATAWBA 1

MONTHLY UNIT OPERATION SUMMARY

February 1985

<u>Date</u>	<u>Time</u>	<u>Event</u>
02/08/85	1026	Mode 1 Entered
	1045	Rx Power increased to 8%
	1150	Rx Power decrease initiated due to problem with Condenser Dump Valves which has made steam pressure control erratic
	1157	Mode 2 Entered
	1200	Rx Power decreased to 1×10^{-7} Amps on Intermediate Range
	2230	Rx Power increase initiated
	2247	Mode 1 Entered
	2319	Rx Power increased to 17%
02/09/85	0245	Rx Trip from 17% with Safety Injection due to S/G B Low Steam Line pressure signal. Event caused by voltage spike in Process Cabinets which falsely provided SI/Rx Trip logic. Mode 3 Entered
	1320	Unit Shutdown commenced to repair ruptured RC System pipe. Condenser vacuum must be broken and RC System shutdown to repair piping.
	2229	Mode 4 Entered
02/10/85	1000	Mode 5 Entered
02/24/85	0200	NC System Heatup commenced
	0405	Mode 4 Entered
02/27/85	0754	Mode 3 Entered
	0900	Rx startup commenced
	0905	Mode 2 Entered Rx critical @ 48 steps wd. on Bank D Critical Boron = 882 ppm B
	0947	Rx Power increased to 1%

TABLE 2.0-8 (cont.)

CATAWBA 1

MONTHLY UNIT OPERATION SUMMARY

February 1985

<u>Date</u>	<u>Time</u>	<u>Event</u>
	0952	Mode 3 Entered
02/27/85	1000	First of a series of Reactor Startups for the purpose of operator training commenced
	1012	Mode 2 Entered Rx critical @ 45 steps wd. on Bank D Critical Boron = 882 ppm B
	1042	Rx Power increased to 1%
	1048	Mode 3 Entered
	1100	Rx Startup commenced
	1110	Mode 2 Entered Rx critical @ 54 steps wd. on Bank D Critical Boron = 882 ppm B
	1130	Rx Power increased to 1%
	1137	Mode 3 Entered
	1153	Rx Startup commenced
	1211	Mode 2 Entered Rx Critical @ 53 steps wd. on Bank D Critical Boron = 928 ppm B
	1225	Rx Power increased to 1%
	1228	Mode 3 Entered
	1240	Rx Startup commenced
	1245	Mode 2 Entered Rx critical @ 54 steps wd. on Bank D Critical Boron = 909 ppm B
	1309	Rx Power increased to 1%
	1313	Mode 3 Entered
	1325	Rx Startup commenced

TABLE 2.0-8 (cont.)

CATAWBA 1

MONTHLY UNIT OPERATION SUMMARY

February 1985

<u>Date</u>	<u>Time</u>	<u>Event</u>
02/27/85	1333	Mode 2 Entered @ 57 steps wd. on Bank D Critical Boron = 908 ppm B
	1350	Rx Power increased to 1%
	1357	Mode 3 Entered
	1415	Rx Startup commenced
	1422	Mode 2 Entered Rx critical @ 54 steps wd. on Bank D Critical Boron = 923 ppm B
	1445	Rx Power increased to 1%
	1452	Mode 3 Entered
	1527	Rx Startup commenced
	1536	Mode 2 Entered Rx critical @ 56 steps wd. on Bank D Critical Boron = 927 ppm B
	1605	Rx Power increased to 1%
	1610	Mode 3 Entered
	1621	Rx Startup commenced
	1627	Mode 2 Entered Rx critical @ 57 steps wd. on Bank D Critical Boron = 907 ppm B
	1659	Rx Power increased to 1%
	1703	Mode 3 Entered
	1721	Rx Startup commenced
	1729	Mode 2 Entered Rx critical @ 56 steps wd. on Bank D Critical Boron = 913 ppm B
	1800	Rx Power increased to 1%
	1806	Mode 3 Entered

TABLE 2.0-8 (cont.)

CATAWBA 1

MONTHLY UNIT OPERATION SUMMARY

February 1985

<u>Date</u>	<u>Time</u>	<u>Event</u>
02/27/85	1820	Rx Startup commenced
	1825	Mode 2 Entered Rx critical @ 57 steps wd. on Bank D Critical Boron = 930 ppm B
	1830	Rx Power increased to 1%
	1852	Mode 3 Entered
	1922	Rx startup commenced
	1958	Mode 2 Entered Rx critical @ 56 steps wd on Bank D Critical Boron = 882 ppm B
	2008	Rx Power increased to 1%
	2033	Mode 3 Entered
	2038	Rx startup commenced
	2046	Mode 2 Entered Rx critical @ 58 steps wd. on Bank D Critical Boron = 882 ppm B
	2103	Rx Power increased to 1%
02/28/85	0052	Mode 3 Entered
	0053	Rx startup commenced
	0100	Mode 2 Entered Rx critical @ 62 steps wd on Bank D Critical Boron = 882 ppm B
	0105	Rx Power increased to 1%
	0157	Mode 3 Entered
	0200	Rx startup commenced
	0202	Mode 2 Entered Rx critical @ 62 steps wd on Bank D Critical Boron = 882 ppm B

MONTHLY UNIT OPERATION SUMMARY

February 1985

<u>Date</u>	<u>Time</u>	<u>Event</u>
02/28/85	0222	Rx power increased to 1%
	0242	Mode 3 Entered
	0248	Rx startup commenced
	0250	Mode 2 Entered Rx critical @ 63 steps wd on Bank D Critical Boron = 882 ppm B
	0305	Rx power increased to 1%
	0345	Mode 3 Entered
	0349	Rx startup commenced
	0352	Mode 2 Entered Rx critical @ 63 steps wd on Bank D Critical Boron = 882 ppm B
	0409	Rx power increased to 1%
	0439	Mode 1 Entered
	0447	Rx power increased to 10% for Turbine Shell Warming
	1814	Turbine/Generator on line
	2100	Rx power increased to 20%

The Core Burnup Calculation for the month of February was based on 6.5 days of power operation which resulted in 3777.83 MWD of depletion. Burnup rates generated from a 30% F.P. Flux Map (FCM/1/01/006) taken on March 5, 1985 were utilized by the Burnup Program. The results of the computer calculation are as follows:

Burnup for Period _____	46.16 MWD/MTU (1.108 EFPD)
Total Energy for Period _____	309.4 x 10 ⁹ BTU
Cummulative Cycle B/U _____	97.542 MWD/MTU (2.33 EFPD)
Cummulative Cycle Energy _____	653.42 x 10 ⁹ BTU

TABLE 2.0-8 (cont.)

MONTHLY UNIT OPERATION SUMMARY

February 1985

Other Data:

Generator hours on line	<u>80.38</u>
Reactor hours critical	<u>220.98</u>
Unit Electrical Gross (MWH)	<u>15112</u>
Unit Net Generation (MWH)	<u>-5465</u>

MONTHLY UNIT OPERATION SUMMARY

March 1985

<u>Date</u>	<u>Time</u>	<u>Event</u>
03/01/85	0000	Reactor Power 12%
	0023	Power Change of 10% - Main Turbine in Standby
	0100	PT/1/A/4250/02D - Turbine Overspeed Test
	0148	Overspeed Test Complete
	0300	Rx Power 10%
	0331	Generator On-Line
	0400	Rx Power 19%
	0700	Rx Power 26%
	1300	Reactor Power 30%
	1707	TP/1/A/2600/11 - Pressurizer Pressure and Level Control System Test
	2400	TP/1/A/2600/11 - Complete
03/02/85	0700	Reactor Power 31.7%
03/03/85	0500	TP/1/A/2600/10 - Control Rod System at Power
	1230	TP/1/A/2600/10 - Complete
	1442	Reactor Power Drop 30.5% to 23% - no apparent cause
	1450	Begin to return reactor power to 30%
	1556	Turbine Runback to 105 MW
	1600	Begin Load Increase
	1649	Turbine Runback = 80 MW Drop
	1650	Load Increase at 3%/hour
	1930	Rx Power 24%
	2020	Rx Power 28%

TABLE 2.0-9

CATAWBA 1

MONTHLY UNIT OPERATION SUMMARY

March 1985

<u>Date</u>	<u>Time</u>	<u>Event</u>
03/03/85	2140	Rx Power 30%
	2217	Runback to 16%
	2225	Rx Power 22%
	2235	Rx Power 25%
03/04/85	0035	Rx Power 28%
	0135	Rx Power 30%
	0750	Runback to 20%
	0805	Turbine Off-Line - leak in Turbine Hydraulic Control piping Rx Power 22%
03/05/85	1924	Turbine On-Line, Rx Power 27%
	0445	Turbine Load Reduction
	0451	Manual Turbine Trip
	0452	Rx Power Reduced from 30% to 26%
	2230	Pseudo-ejected rod test - enter Special Test Exception 3.10.2 RCCA D-12 > 12 steps misaligned
03/06/85	0712	Rod D-12 realigned, out of 3.10.2
	0745	Pseudo-Ejected Rod Complete
	1026	Turbine On-Line, Rx Power 30%
	1928	"A" Condensate Booster Pump Trip on lo suction press. Turbine load reduction 250 MW - 25 MW Reactor Power maintained at 30%
	2036	Turbine loaded to 255 MW and holding
03/07/85	0124	TP/1/A/2150/04 - Doppler Only Power Coefficient Verification, turbine load reduced to 200 MW

TABLE 2.0-9 (cont.)

CATAWBA 1

MONTHLY UNIT OPERATION SUMMARY

March 1985

<u>Date</u>	<u>Time</u>	<u>Event</u>
03/08/85	0136	Turbine at 255 MW
	0400	TP/1/A/2150/04 - Complete
	2340	Reduce turbine power 10% (80 MW) per TP/1/A/2650/05 - Unit Load Transient Test
	0020	Increase turbine power to 270 MW per TP/1/A/2650/05, Test Complete
03/09/85	1730	Commence power escalation to 48% with 3%/hour limit
	1920	Rx Power 37%
	2200	Rx Power 40%
03/10/85	0030	Rx Power 49%
	0930	Feedwater Transient - CF Swings
	1140	Feedwater Transient - CF Swings
	1320	Feedwater Transient - CF Swings
03/11/85	0948	CF Flow Swings
	1401	CF Flow Transient
	2240	CF Flow Transient
	2255	CF Flow Transient
03/12/85	0005	Commence TP/1/A/2150/04 Doppler Only Coefficient Test
	0200	TP/1/A/2150/04 - Complete
	1350	Turbine Runback 460 MW - No apparent reason
	1355	Turbine Load Increase - Rx Power Maneuvering Limit 3%/hr
	1900	Rx Power 49%

TABLE 2.0-9 (cont.)

CATAWBA 1

MONTHLY UNIT OPERATION SUMMARY

March 1985

<u>Date</u>	<u>Time</u>	<u>Event</u>
03/13/85	1030	CF Reg. Valves Swinging
	1220	Commence Power Reduction at 10%/hr per 3.7.1.3 'ACTION' statement - CA Pump Turbine (CAPT) Out of Service
	1303	CAPT tested, determined operable power reduction stopped
	1406	Commenced increasing Rx Power to = 47%
	1541	Removed NI-44 from service to eliminate noise
03/14/85	0216	Rx Trip due to loose pin in Power Range 44 drawer while testing N-41 in progress. "NI Hi ϕ rate P/R Rx Trip" occurred
03/15/85	0610	Rx Startup Commenced
	0615	Entered Mode 2
	0642	Rx Critical
	0930	Entered Mode 1
	1320	Turbine/Generator on line
03/16/85	0330	Rx Power @ 48%
	1435	Commenced Unit Load Transient Test
	1930	Turbine Load Dropped, Rods at 192 steps
	2200	Rods borated out to 205 steps
03/17/85	1045	Bypassed Channel D T-Avg Loop, Higher T-Avg for D causing unwarranted rod insertion
	1500	Commenced Base Case TP/1/A/2150/05, Flux Map for Below Bank Rod Test
	2030	Completed Map
03/18/85	0110	Dilution Started for Below Bank Rod Test

TABLE 2.0-9 (cont.)

CATAWBA 1

MONTHLY UNIT OPERATION SUMMARY

March 1985

<u>Date</u>	<u>Time</u>	<u>Event</u>
03/18/85	0630	Mapping Began
	0730	OAC not accepting flux map data
	1417	Retrieved Bank D Control Rod D-12
	2020	Reducing Power due to ΔT and NIS mismatch from Below Bank Test. Could not meet daily surveillance procedure
	2350	Dilution to Control T-avg during Xenon oscillation
03/19/85	0010	Rx Power @ 37%
	0430	Rx Power @ 41%. Commenced shutdown at 10%/hr. Could not meet surveillance requirement for Mode 1.
	0700	Rx Power @ 20%
	0855	Adding 1200 gallons of Rx M/U water to drive Bank D rods to 160 steps to help stop radial Xenon oscillation
	0920	Dilution and rod drive complete
	1000	Commenced Power Reduction to < 15%
	1950	Tech Spec Surveillance req. during Core Tilt incorporated into surveillance test. Commenced Power increase to 25%.
	2100	Rx Power @ 25%
	2300	Rx Power @ 32%
03/20/85	1320	Holding Rx Power @ 37% per NRC request
	1431	Received NRC approval to increase power
03/21/85	0230	Rx Power @ 48.5%. Tilt is oscillating but converging
	0830	PT/1/A/4150/13B Calorimetric Rx Coolant Flow Measurement

TABLE 2.0-9 (cont.)

CATAWBA 1

MONTHLY UNIT OPERATION SUMMARY

March 1985

<u>Date</u>	<u>Time</u>	<u>Event</u>
03/21/85	1800	Replaced Detector "C" Moveable incore probe.
03/22/85	0400	N41, N42, N43, N44 trip setpoints set @ 70%
	0800	Cut in Secondary Thermal Power Calc. on OAC; dropped best est. Thermal Power = 2%
	1140	PT/1/A/4600/06A - Incore Instrumentation Detector Calibration
	1300	Completed PT/1/A/4600/06A
	1730	Begin Below Bank Rod Test - TP/1/A/2150/05 (Retest)
03/23/85	0055	Control Rod D-12 on bottom
	0120	Began taking a map
	0400	Map Complete
	0424	Began boration and withdrawal of Control Rod D-12
	0545	Began taking a map, D-12, 25 steps low
	0720	Map Complete
	0815	Control Rod D-12 realigned. Below Bank Rod Test Completed.
	0830	OAC Trouble
03/24/85	2045	N41 High Flux Setpoint adjusted to 95%
	2243	N42 High Flux Setpoint adjusted to 95%
	2359	N43 High Flux Setpoint adjusted to 95%
03/25/85	0210	N44 High Flux Setpoint adjusted to 95%
	0236	Commenced Rx Power increase to 49%
03/26/86	1830	Rods in Auto N41, N42, N43, & N44 High Flux Trip Setpoints at 80% FP.

TABLE 2.0-9 (cont.)

CATAWBA 1

MONTHLY UNIT OPERATION SUMMARY

March 1985

<u>Date</u>	<u>Time</u>	<u>Event</u>
03/26/86	2020	Operations preparing to add water to begin power increase to 68%
	2026	Rx Power @ 50.1%
	2125	Rx Power @ 52%; Commenced Dilution and Rod Withdrawal to obtain 68% Rx Power @ 3%/hr
03/27/85	0249	Rx Power leveled off @ 68% FP
	0600	Began ENA Detector Calibration Performance Test
	0800	ENA Calibration Test Complete
	2141	Tripped Turbine per TP/1/A/2650/07
	2145	Rods in manual-holding Rx Power @ = 15%
	2149	Completed AP/1/A/5500/02 - Turbine Trip Test
	2317	Generator Back on Line
	2355	Power Increase commenced (toward 75%) Power @ 51%
03/28/85	0205	Rx Power @ 54%
	0230	Discrepancy noted between Control Board Axial Flux Difference (AFD) indications and Nuclear 06 printout. NSSS log review showed AFD monitor inoperable since 1400 on 3/27. Points for excores not updating.
	0330	AFD monitor now operable. Determined to have been inoperable since turbine trip test. Penalty time updated on this assumption.
	0500	Excore NI Trip Setpoints reduced to 55%
	1900	AFD indicates axial Xenon oscillations in progress
03/29/85	0324	N41 Setpoint raised to 95%

TABLE 2.0-9 (cont.)

CATAWBA 1

MONTHLY UNIT OPERATION SUMMARY

March 1985

<u>Date</u>	<u>Time</u>	<u>Event</u>
03/29/85	0356	N42 Setpoint raised to 95%
	0435	N43 Setpoint raised to 95%
	0459	N44 Setpoint raised to 95%
	0500	Commenced escalation of Rx Power @ 3%/hr. toward 75%, currently @ 48%
	1100	Suspended escalation @ 66% to obtain valid calculation of NC leakage.
	1300	Escalation - 75% resumed
	1815	Starting Testing @ 74% FP
	2100	Problem with Best Est. power indication noted
	2215	Rx Power Best Est. been invalid since 1600 hours
	2130	Primary Thermal Output showing 82.4%. Commenced power de-escalation to = 74 - 75%
	2250	OAC Back in Service
	2345	Terminated Power Reduction. Primary Thermal Power @ 74.6%
03/30/85	1935	Reduced Rx Power from 79% to 75%
03/31/85	0030	Began Doppler Only Test
	0345	Completed 3 load swings
	1030	Begain Unit Load Steady State
	1815	Increased Rx Power to 77% to checkout input from Thermal Power Best Est.
	1835	Reduced Rx Power to 75%
	2030	Commenced NIS Calibration

MONTHLY UNIT OPERATION SUMMARY

March 1985

<u>Date</u>	<u>Time</u>	<u>Event</u>
03/31/85	2146	OAC Out of Service
	2223	OAC Back in service
	2300	Borated "D" out from 192 to 215 steps

The Core Burnup Calculation for the month of March was based on 31 days of power operation which resulted in 43309 MWD of depletion. Burnup rates generated from a 50% F.P. Flux Map (FCM/1/01/005) taken on March 13, 1985 were utilized by the Burnup Program. The results of the computer calculation are as follows:

Burnup for Period _____	529.492 MWD/MTU (12.698 EFPD)
Total Energy for Period _____	3546.993 x 10 ⁹ BTU
Cummulative Cycle B/U _____	627.034 MWD/MTU (15.036 EFPD)
Cummulative Cycle Energy _____	4200.416 x 10 ⁹ BTU

Other Data:

Generator hours on line	<u>663.56</u>
Reactor hours critical	<u>720.25</u>
Unit Electrical Gross (MWH)	<u>287004</u>
Unit Net Generation (MWH)	<u>250400</u>

MONTHLY UNIT OPERATION SUMMARY

April 1985

<u>Date</u>	<u>Time</u>	<u>Event</u>
04/01/85	0700	Rx Power 75%, 840 MWe
04/02/85	0356	During PT/1/A/4600/05A, Incore and NIS Correlation Check, enter Special Test Exception 3.10.2 due to AFD out of limits.
	1210	Large CF Swings on all S/G's, place Reg. Valves in manual and stabilize.
	1528	Large CF Swings on all S/G's, place Reg. Valves in manual and stabilize.
04/03/85	0010	Exit Special Test Exception 3.10.2 due to 960 minutes AFD penalty time, per Tech Spec 3.2.1.
	1715	Commence TP/1/A/2650/05, Unit Load Transient Test
	1716	10% Step Load Reduction to 665 MWe
	1733	10% Step Load Increase to 803 MWe Acceptance Criteria not met on load reduction
04/04/85	0112	10% Step Load Reduction per TP/1/A/2650/05, Large CF Swings, CF Reg. Valves into manual and stabilize. Return to 74%.
	0135	10% Step Load Reduction per TP/1/A/2650/05, Large CF Swings, CF Reg. Valves into manual and stabilize.
	0145	Reduce Power to 55%, to shutdown and repair 1B CF Pump vent piping.
	1135	Reactor High Level Trip Point 109%
	1941	Ramp to 75% in progress
	2305	Reactor Power 75%
	2320	10% Step Load Reduction per TP/1/A/2650/05 690 MWe Not Acceptable - CF Swings - return to power at 10%/min. load rate, adjust CF Reg Valve Controller.

MONTHLY UNIT OPERATION SUMMARY

April 1985

<u>Date</u>	<u>Time</u>	<u>Event</u>
04/05/85	0245	10% Step Load Reduction per TP/1/A/2650/05 to 690 MWe Acceptance Criteria met.
	0700	Reactor Power 74%
	2235	Commence Power Increase to 90% at 3%/hr.
04/06/85	0306	Stop power increase at 88% OPAT Turbine Runback Block Rod Withdrawal Alert Control.
	1633	Turbine Runback from 920 MWe to 850 MWe on 2/4 OPAT (cause unknown until 4/8/85) initiate load increase at 3%/hr.
	1808	Turbine Runback 920 MWe to 865 MWe on 2/4 OPAT (cause unknown until 4/8/85).
	1842	Load Decrease 910 MWe to 850 MWe
	1900	Reactor Power 86%
*04/08/85	0700	Reactor Power 89%
04/10/85	0330	Commence TP/1/A/2150/04, Doppler Only Power Coefficient
	0438	4 Load Swings Complete for Doppler 975 MWe
	0745	Reactor Power 91%
	1520	Commence TP/1/A/2650/11, Feedwater Temp Variation Test
	1710	TP/1/A/2650/11 complete, Reactor Power increased 4% to 94%, Turbine Load increased from 1044 MWe to 1105 MWe.
04/11/85	0424	Repeated TP/1/A/2650/11

*NOTE: There was a delay of 36 hrs. while instrumentation personnel reset OTAT and OPAT setpoints based on a Full Power Delta T of 57.4 (instead of 51°F) to allow power to be increased above 88%.

TABLE 2.0-10 (cont.)

CATAWBA 1

MONTHLY UNIT OPERATION SUMMARY

April 1985

<u>Date</u>	<u>Time</u>	<u>Event</u>
04/11/85	0445	Reactor Power stabilized at 92.2%, test complete
	0524	Reactor Power reduced to 90%
	1710	Commenced power escalation to 98%
	2025	Reactor Power at 98%
04/12/85	2200	Reactor Power reduced from 98% to 54% per TP/1/A/2650/10, Large Load Rejection Test.
	2205	Reactor Power reduced to 49% so that only Half Penalty Time will be registered.
	2218	Reactor Power reduced to 41% due to Turbine Runback of unknown origin.
	2245	Reactor Power reduced to 32% by build-in of Xenon (due to 50% load drop).
	2300	Reactor Power 27% and falling. Dilution rate can not keep up with Xenon build-in (7 pcm/min)
	2315	S/G Blowdown isolated to maintain decreasing T_{ave} . Reactor Power at 22%.
	2324	T_{ave} falls below 551°F, Reactor Power at 17%. Xenon still building in.
	2333	Main Feedwater swapped to CA nozzles. Reactor Power 11%.
	2349	Mode 2 Entered.
04/13/85	0010	Inserting Control Rods to prepare for restart. Xenon still building in.
	0020	Mode 3 Entered.
	1313	Commenced Reactor Startup.
	1330	Control Rods withdrawn to top end of Estimated Critical Position (ECP) band. Criticality not achieved.

TABLE 2.0-10 (cont.)

CATAWBA 1

MONTHLY UNIT OPERATION SUMMARY

April 1985

<u>Date</u>	<u>Time</u>	<u>Event</u>
04/13/85	1340	Control Rods fully reinserted. New ECP Calculated.
	1708	Commenced Reactor Startup.
	1720	Control Rods withdrawn to top end of ECP band with criticality not being achieved once again.
	1734	Control Rods fully reinserted.
	1743	Problem with accurate prediction of Xenon worth responsible for the missed ECP's. 1/M Approach to Criticality must be performed.
04/14/85	0008	Commenced Reactor Startup under the Control of PT/1/A/4150/19, 1/M Approach to Criticality.
	0241	Reactor Critical @ 42 steps wd. on Bank D. $C_B = 599$ ppmB. Mode 2 Entered.
	0256	Mode 1 Entered.
	0615	Reactor Power increased to 10%
	0820	Turbine/Generator placed on line. Reactor Power at 15%.
	1520	Reactor Power increased to 48%.
	1900	Reactor Power increased to 66%.
	2335	Power Escalation to 98% commenced.
04/15/85	0040	Reactor Power increased to 76%.
	0153	Reactor Power increased to 86%.
	0218	Reactor Power increased to 89%.
	0350	Reactor Power increased to 94%.
	0420	Commenced load decrease due to high vibration on CF Pump Turbine 1A
	0455	Reactor Power decreased to 90%

TABLE 2.0-10 (cont.)

CATAWBA 1

MONTHLY UNIT OPERATION SUMMARY

April 1985

<u>Date</u>	<u>Time</u>	<u>Event</u>
04/15/85	0522	Reactor Trip due to Lo Lo S/G A Level, caused by faulty Feedwater Regulating Valve Operation. Mode 3 Entered.
	1737	Reactor Startup commenced. Mode 2 Entered.
	1754	Reactor Critical at 102 steps wd. on Control Bank C. $C_B = 560$ ppmB
	1810	Mode 1 Entered.
	1900	Reactor Power increased to 6.7%.
04/16/85	0115	Turbine/Generator placed on line. Reactor Power at 10%
	0307	Turbine/Generator manually tripped due to high vibration. (7.1 mils) on #3 and #8 bearings. Reactor Power decreased from 15% to 7.7%, T_{ave} continues to decrease, however.
	0308	T_{ave} decreases below 551°F. Reactor Power decreasing.
	0312	Mode 2 Entered.
	0318	T_{ave} restored to above 551°F. Reactor Power at 0%.
	0320	Commenced addition of Reactor Makeup Water to prevent shutdown of Unit. Reason for decreasing reactivity discovered to be failure to close NV 238 (Reactor Makeup Water to Blender) following a make-up to the NC System at 0242. Valve was estimated to have been open for = 24 minutes.
	0325	2440 gallons of water have been added. Reactor Power at 5×10^{-11} amps.
	0423	650 more gallons of water added.
	0443	1500 more gallons of water added, Reactor Power increasing.
	0500	Reactor Power increased to 2.5%.
	0552	Mode 1 Entered

TABLE 2.0-10 (cont.)

CATAWBA 1

MONTHLY UNIT OPERATION SUMMARY

April 1985

<u>Date</u>	<u>Time</u>	<u>Event</u>
4/16/85	0740	Reactor Power increased to 7.5%.
	0805	Reactor Power increased to 10%.
	0856	Turbine/Generator placed on line. Reactor Power at 16%.
	0948	Power Escalation from 20% commenced.
	1018	Reactor Power increased to 30%.
	1224	Reactor Power increased to 40%.
	1330	Reactor Power increased to 50%.
	1733	Recommended power escalation from 50%.
	1900	Reactor Power increased to 60%.
04/17/85	0117	Reactor Power increased to 89%.
	0615	Power Escalation secured at 96% due to high inboard and outboard bearing temperatures on CF Pump 1A's Turbine.
	1015	Load on CF Pump 1A decreased to reduce bearing temperatures. Reactor Power increased to 98%.
	1232	Reactor Power decreased from 98% to 83% per TP/1/A/2650/05, Unit Load Transient Test.
	1320	Reactor Power increased from 83% to 101% per TP/1/A/2650/05, Unit Load Transient Test. Rx Power then reduced to 96%.
	1430	Reactor Power increased to 98%.
	1800	Reactor Power decreased from 98% to 83% per TP/1/A/2650/05, Unit Load Transient Test.
	1815	Reactor Power decreased from 83% to 91% per TP/1/A/2650/05, Unit Load Transient Test.
	1900	Reactor Power increased to 98%.

TABLE 2.0-10 (cont.)

MONTHLY UNIT OPERATION SUMMARY

April 1985

<u>Date</u>	<u>Time</u>	<u>Event</u>
04/19/85	1005	Main Generator Breakers 1A and 1B opened per TP/1/A/2650/06, Loss of Electrical Load Test. Reactor Trip from 100% FP due to 2 out of 4 Turbine Control Valves closed (caused by Power Load Unbalance Signal to Turbine). Turbine Trip caused by Reactor trip. Mode 3 Entered.
04/20/85	0047	Mode 4 Entered
	0607	Mode 5 Entered. Unit to remain in Mode 5 for remainder of month for Maintenance outage.

The Core Burnup Calculation for the month of April was based on 17.5 days of power operation which resulted in 44057.81 MWD of depletion. Burnup rates generated from a 90% F.P. Flux Map (FCM/1/01/024 analyzed with SNCUIK theoretical factors) taken on April 8, 1985 were utilized by the Burnup Program. The results of the computer calculation are as follows:

Burnup for Period _____	538.670 MWD/MTU (12.916 EFPD)
Total Energy for Period _____	3607.989 x 10 ⁹ BTU
Cummulative Cycle B/U _____	1165.704 MWD/MTU (27.952 EFPD)
Cummulative Cycle Energy _____	7807.842 x 10 ⁹ BTU

Other Data:

Generator hours on line	<u>383.26</u>
Reactor hours critical	<u>431.80</u>
Unit Electrical Gross (MWH)	<u>339932</u>
Unit Net Generation (MWH)	<u>310437</u>

MONTHLY UNIT OPERATION SUMMARY

May 1985

<u>Date</u>	<u>Time</u>	<u>Event</u>
05/26/85	2214	Commencing Reactor Coolant System Heatup
05/27/85	0624	Entered Mode 4
05/28/85	1700	Entered Mode 3

MONTHLY UNIT OPERATION SUMMARY

June 1985

<u>Date</u>	<u>Time</u>	<u>Event</u>
06/01/85	1027	Entered Mode 4 to repair Safety Injection Pump 1A bearings.
06/08/85	1004	Entered Mode 3.
06/09/85	1020	Entered Mode 2. Reactor Critical at 25 steps wd. on Control Bank D. NC System Boron Conc. at 874 ppm B. Rx power at 10 ⁻⁸ amps.
	1731	Rx power stabilized at 3% F.P.
	1940	Rx power reduced to 10 ⁻⁸ amps to trouble shoot leaking Pzr spray (1NC29).
06/10/85	1855	Rx power increased to 3% F.P.
	2212	Entered Mode 1.
	2230	Rx power increased to 7% F.P.
06/11/85	0005	Rx power increased to 10% F.P.
	0400	Rx power increased to 16% F.P.
	0440	Turbine/Generator placed on line.
	0512	Turbine/Generator tripped during performance of mechanical overspeed test.
	0540	Turbine/Generator placed on line.
	0551	Turbine/Generator tripped manually to verify trip capability.
	0602	Turbine/Generator placed on line. Rx power at 18% F.P.
	1220	Commenced power escalation from 18% F.P.
	1900	Rx power at 38% F.P. and increasing.
06/12/85	0700	Rx power at 60% F.P. and holding.
	1525	Rx power increased to 63% F.P.

MONTHLY UNIT OPERATION SUMMARY

June 1985

<u>Date</u>	<u>Time</u>	<u>Event</u>
06/12/85	2315	Rx power decreased to 50% F.P. in preparation for unit shutdown due to excessive unidentified reactor coolant leakage.
06/13/85	0036	Rx power decreased to 40% F.P. Search for source of reactor coolant leakage in progress.
	0111	Rx power decreased to 30% F.P.
	0230	Rx power decreased to 15% F.P. Main Feedwater Pump tripped on low suction flow due to delayed opening of miniflow recirculation valve as pump discharge pressure was being increased to swap feedwater from main nozzles to aux nozzles. Turbine/Generator tripped as a result.
	0231	Place Main Feedwater Pump back in service. Reactor run back to 9% F.P. occurred.
	0233	Main Feedwater Pump tripped again. Reactor manually tripped per procedure. Mode 3 entered.
	0300	Source of leakage discovered to be Positive Displacement Pump stuffing box.
06/15/85	1312	Reactor startup commenced.
	1328	Entered Mode 2. Reactor critical 15 steps wd. on Bank D. Boron concentration at 862 ppmb. Rx power at 8×10^{-11} amps.
	1530	Entered Mode 1.
	1900	Rx power increased to 7% F.P.
	2352	Rx power at 15% F.P. Turbine/Generator placed on line.
06/16/85	0700	Rx power increased to 25.6% F.P.
	1330	Rx power increased to 40% F.P.

MONTHLY UNIT OPERATION SUMMARY

June 1985

<u>Date</u>	<u>Time</u>	<u>Event</u>
06/16/85	1600	Rx power increased to 50% F.P.
	1616	Reactor tripped due to "NIS HI FLUX RATE P/R RX TRIP", cause unknown.
06/17/85	0236	Rx critical @ 115 steps CTL Bank D @ 750 ppmb, 2166 pcm Xenon 866 pcm Sm.
	0257	Entered Mode 1.
	0310	Manual turbine trip control/intercept valve #5 opened during entry into chest warming.
	0722	Turbine at 1800 rpm. Generator breakers closed.
06/18/85	0945	S/G PORV's opened due to rapid Xenon burnout, rods maintained stable. Holding at = 30%.
	1000	Commencing power increase to = 50%.
	1110	Reactor at = 49% and holding.
	1335	Commencing power increase to = 60%.
	0700	CFPT 1A Inboard Bearing Temperatures High. Taken off line for repair.
	0715	Maintaining = 65%.
06/22/85	0800	Determined 1-AD-2 B1 and B2. "P/R Upper/Lower Det. Hi Flux Dev" Alarm reset needed adjustment.
	0100	CF pump 1B Tripped on low vacuum when isolation to Main Condensor was opened while preparing to bring CF pump 1A back on line. Caused Turbine Trip, Rx Trip. PORV's on S/G C and D opened. Safety relief valve on S/G B opened.
	1450	Rx Critical @ 167 steps withdrawn on Bank D. C _B = 578 ppmb. Mode 2 entered.
	1515	Mode 1 Entered.

TABLE 2.0-12 (cont.)

MONTHLY UNIT OPERATION SUMMARY

June 1985

<u>Date</u>	<u>Time</u>	<u>Event</u>
06/22/84	1600	Turbine at 1800 RPM.
	1637	Generator placed on line. Rx power at = 15%.
	1710	Rx trip due to Low-Low S/G level on S/G C. Rx power was 15.3% prior to trip.
	2315	Mode 2 entered. Control rods withdrawn to top of Estimated Critical Position (ECP) Band. Criticality not achieved.
	2347	Mode 3 entered. Computational error discovered. Recalculated ECP.
06/23/85	0315	Mode 2 entered. Rx critical @ 91 steps withdrawn on Bank D $C_B = 701$ ppm λ .
	0331	Mode 1 entered.
	0730	Rx power at = 25%.
	0930	Rx power at = 54%.
	1630	Rx power at = 66%.
06/24/85	0302	Rx power at = 75%.
	0550	Rx power at = 90%.
	1150	Rx power at = 100%. Commencing 100 consecutive hours at 100% required to be declared commercial.
	1530	Pressurizer level increasing due to valve NI-10 opening as a result of a misplaced jumper during a performance test. Corrective action taken. Maximum Pressurizer level was 78%. (Programmed level is 60%). Power decreased to = 97% during event.
06/28/85	1600	100 Hours at 100% power completed. Commercial operation to be declared @ 0001 on 6/29/85.
	1636	Reducing power to 89% to perform Turbine Control Valve Test.

TABLE 2.0-12 (cont.)

CATAWBA 1

MONTHLY UNIT OPERATION SUMMARY

June 1985

<u>Date</u>	<u>Time</u>	<u>Event</u>
06/28/85	2130	Control Valve Test completed. Increasing power to 100%.
06/29/85	0045	Rx power at 100%. Axial Xenon oscillation in progress.
06/30/85	1947	Using "Bang-Bang" method to try to halt Axial Xenon oscillation which shows no sign of dampening. Pushing Control Bank D to 178 steps from 198 steps.
	2245	Pulling rods back to 198 steps in second step of "Bang-Bang" method to stop Xenon Swing.

MONTHLY UNIT OPERATION SUMMARY

June 1985

The Core Burnup Calculation for the month of June was divided into two parts. The first part of the calculation was for Precommercial Operation and was based upon 13.1 days of power operation through June 28, 1985 which resulted in 32,079.66 MWD of depletion. The burnup calculation used two time steps: an 8.3 day step which used burnup rates generated from a 65% Flux Map (FCM/1/01/026) taken on June 20, 1985; and a 4.8 day step which used burnup rates from a 100% Flux Map (FCM/1/01/027) taken on June 28, 1985. The results of the computer calculations are as follows:

Burnup for Period	_____	392.216 MWD/MTU (9.401 EFPD)
Total Energy for Period	_____	2,627.040 x 10 ⁹ BTU
Cumulative Cycle B/U	_____	1,557.919 MWD/MTU (37.343 EFPD)
Cumulative Cycle Energy	_____	10,434.88 x 10 ⁹ BTU

The second part of the burnup calculation was for Commercial Operation in June. This consisted of 2.0 days of power operation (June 29 and 30) which resulted in 6767.89 MWD of depletion. Burnup rates generated from the 100% Flux Map (FCM/1/01/027) taken on June 28, 1985 were utilized by the Burnup Program. The results of the computer calculation are as follows:

Burnup for Period	_____	82.744 MWD/MTU (1.983 EFPD)
Total Energy for Period	_____	554.213 x 10 ⁹ BTU
Cumulative Cycle B/U	_____	1,640.633 MWD/MTU (39.325 EFPD)
Cumulative Cycle Energy	_____	10,989.10 x 10 ⁹ BTU

Totals for the month of June (including precommercial and commercial operation) are as follows:

Burnup for Period	_____	474.960 MWD/MTU (11.384 EFPD)
Total Energy for Period	_____	3,181.253 x 10 ⁹ BTU
Cumulative Cycle B/U	_____	1640.663 MWD/MTU (39.325 EFPD)
Cumulative Cycle Energy	_____	10,989.10 x 10 ⁹ BTU

Other Data:

Generator hours on line	<u>362.36</u>
Reactor hours critical	<u>462.48</u>
Unit Electrical Gross (MWH)	<u>306662</u>
Unit Net Generation (MWH)	<u>266421</u>
Cumulative Unit Electrical Gross (MWH)	<u>982632</u>

TABLE 2.0-12 (cont)

CATAWBA 1

THERMAL OUTPUT FOR JANUARY

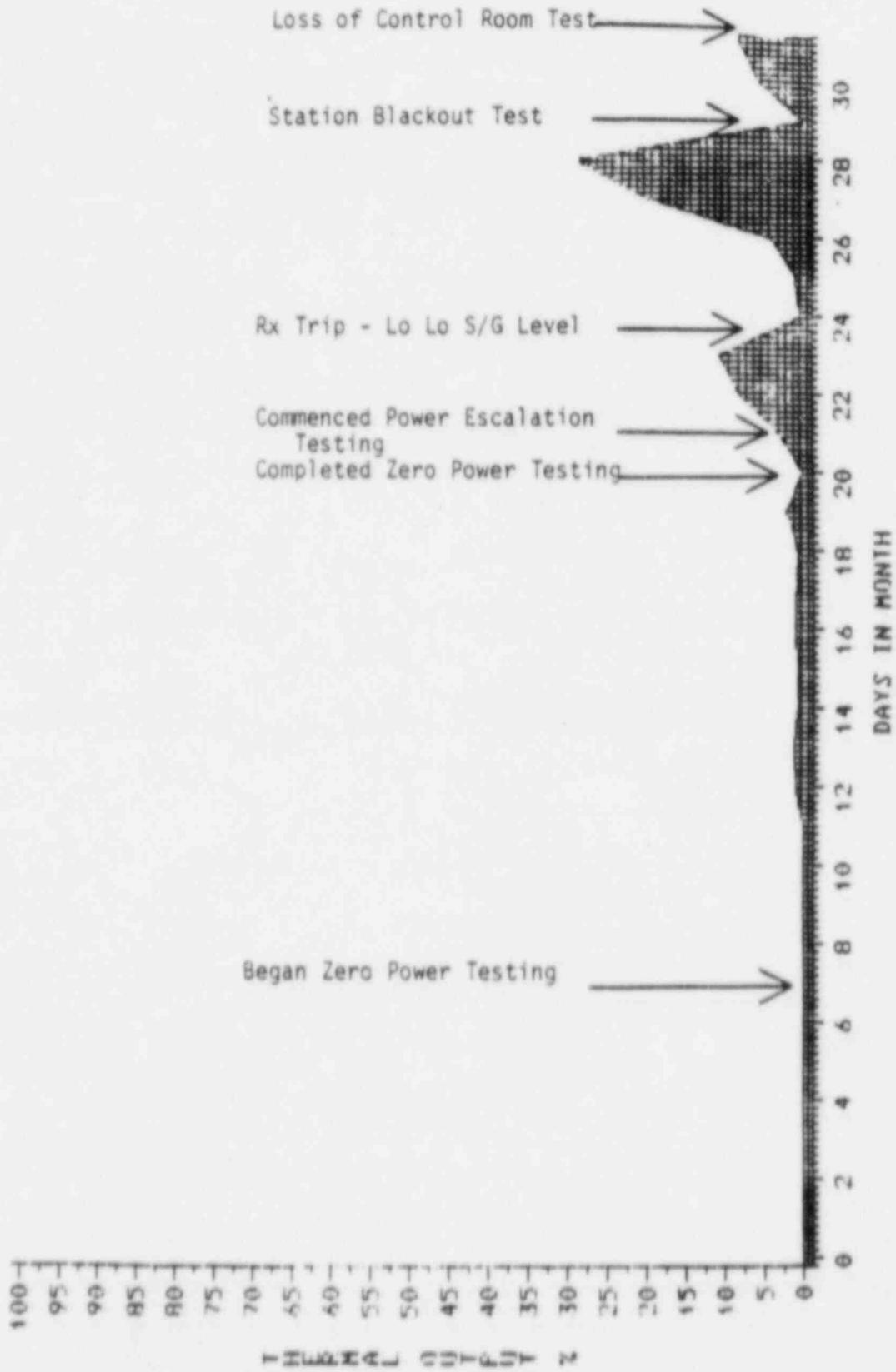


Figure 2.0-1

THERMAL OUTPUT FOR FEBRUARY

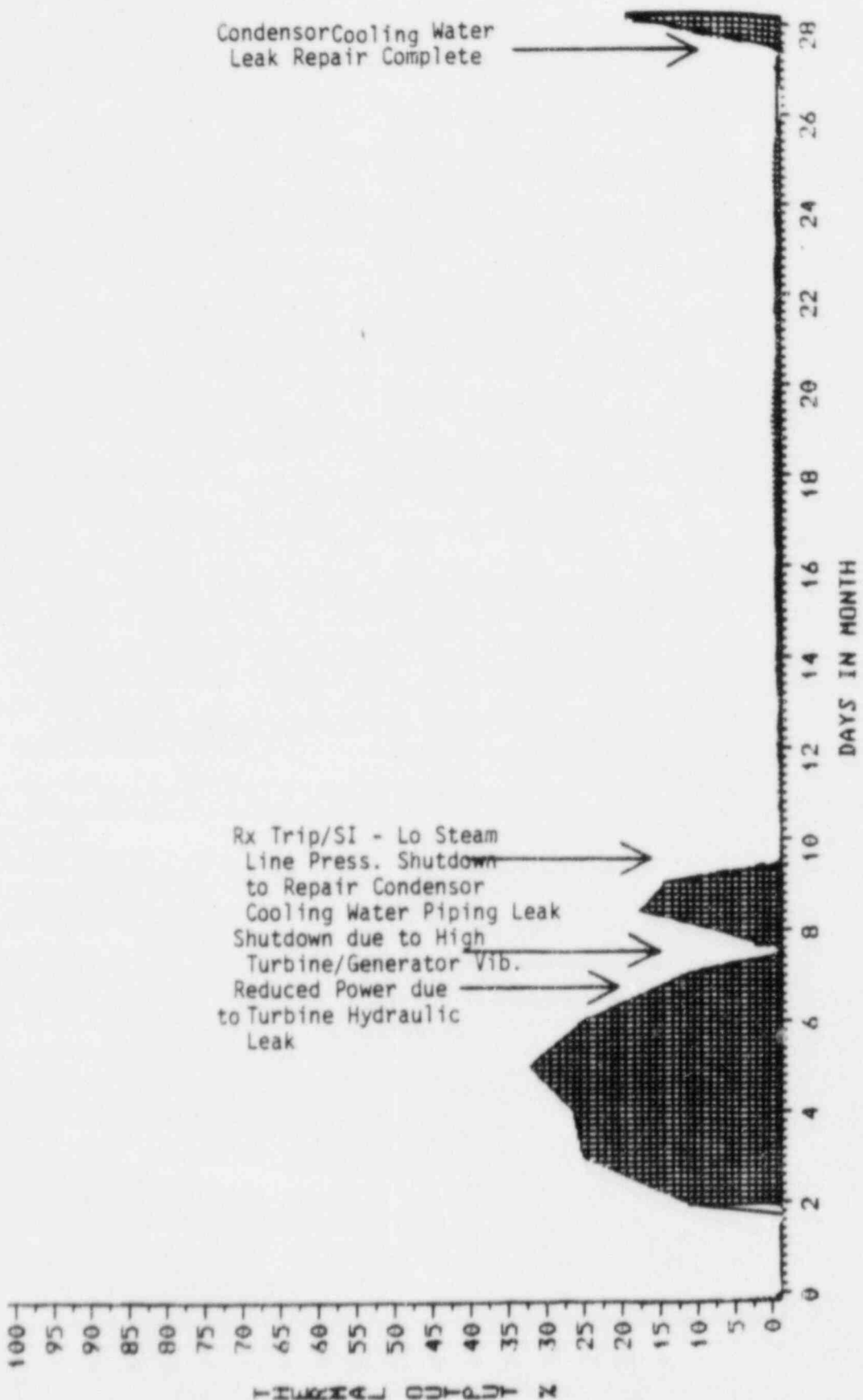


Figure 2.0-2

THERMAL OUTPUT FOR MARCH

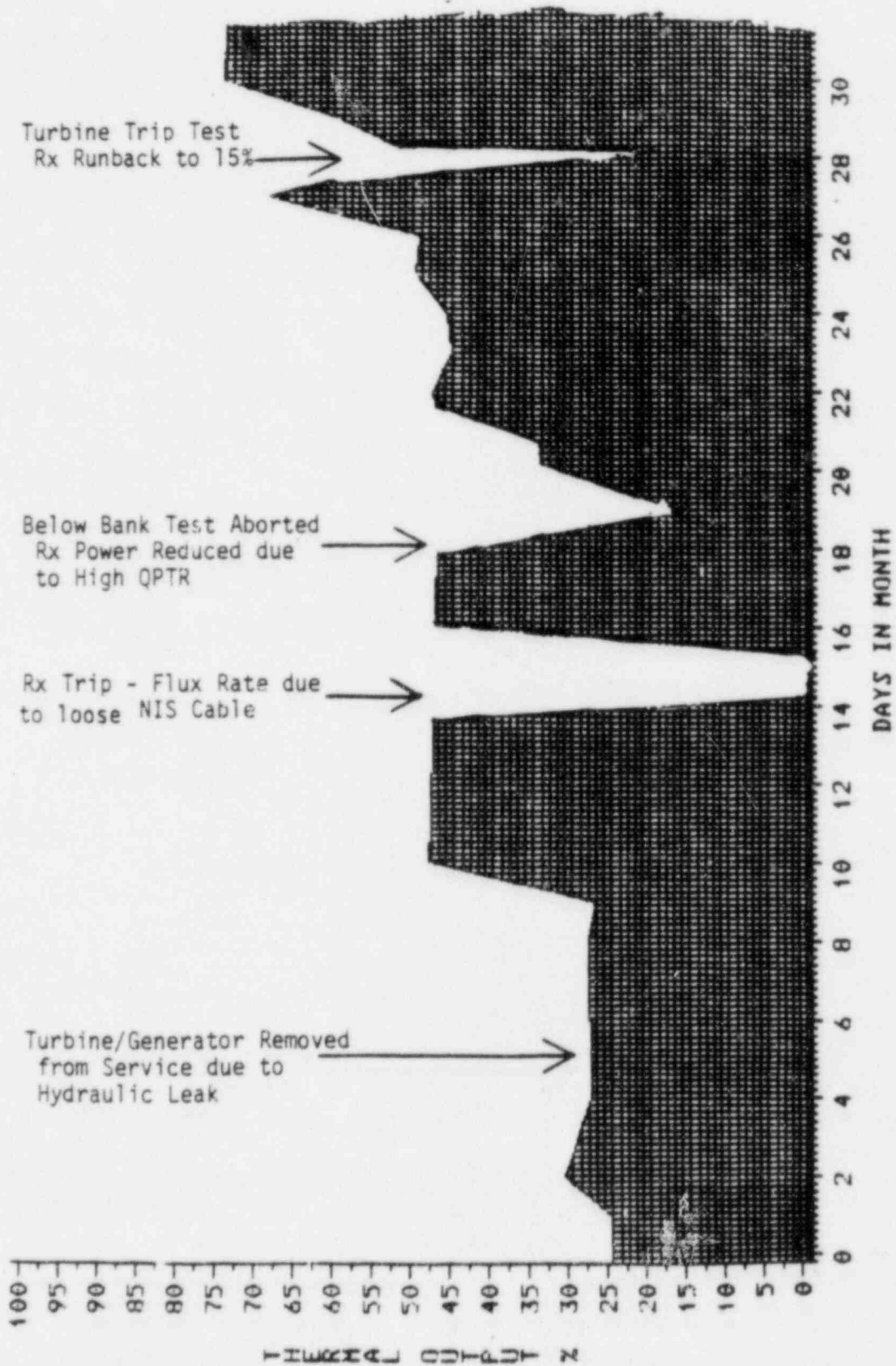


Figure 2.0-3

THERMAL OUTPUT FOR APRIL

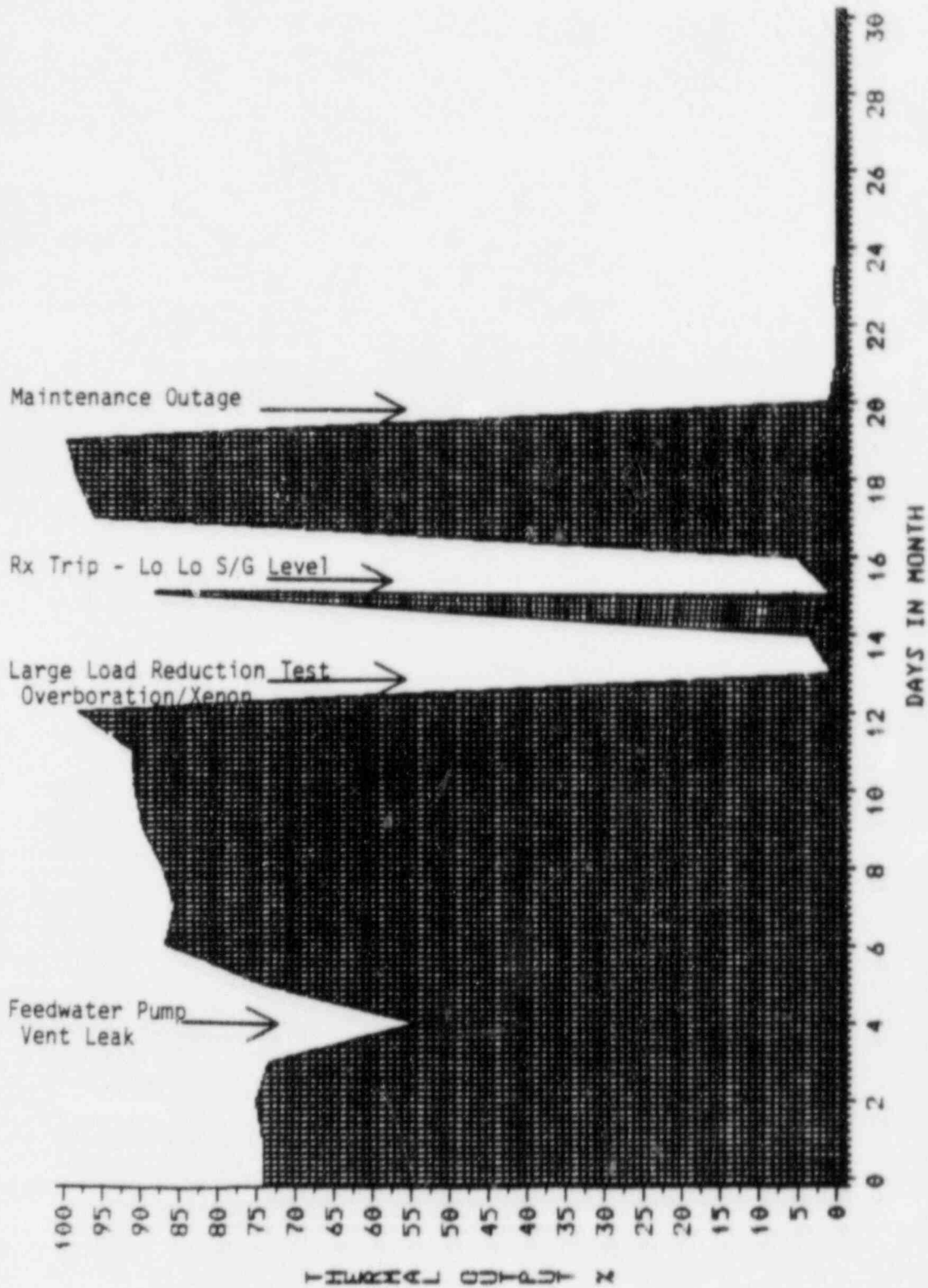


Figure 2.0-4

CATAWBA 1

THERMAL OUTPUT FOR MAY

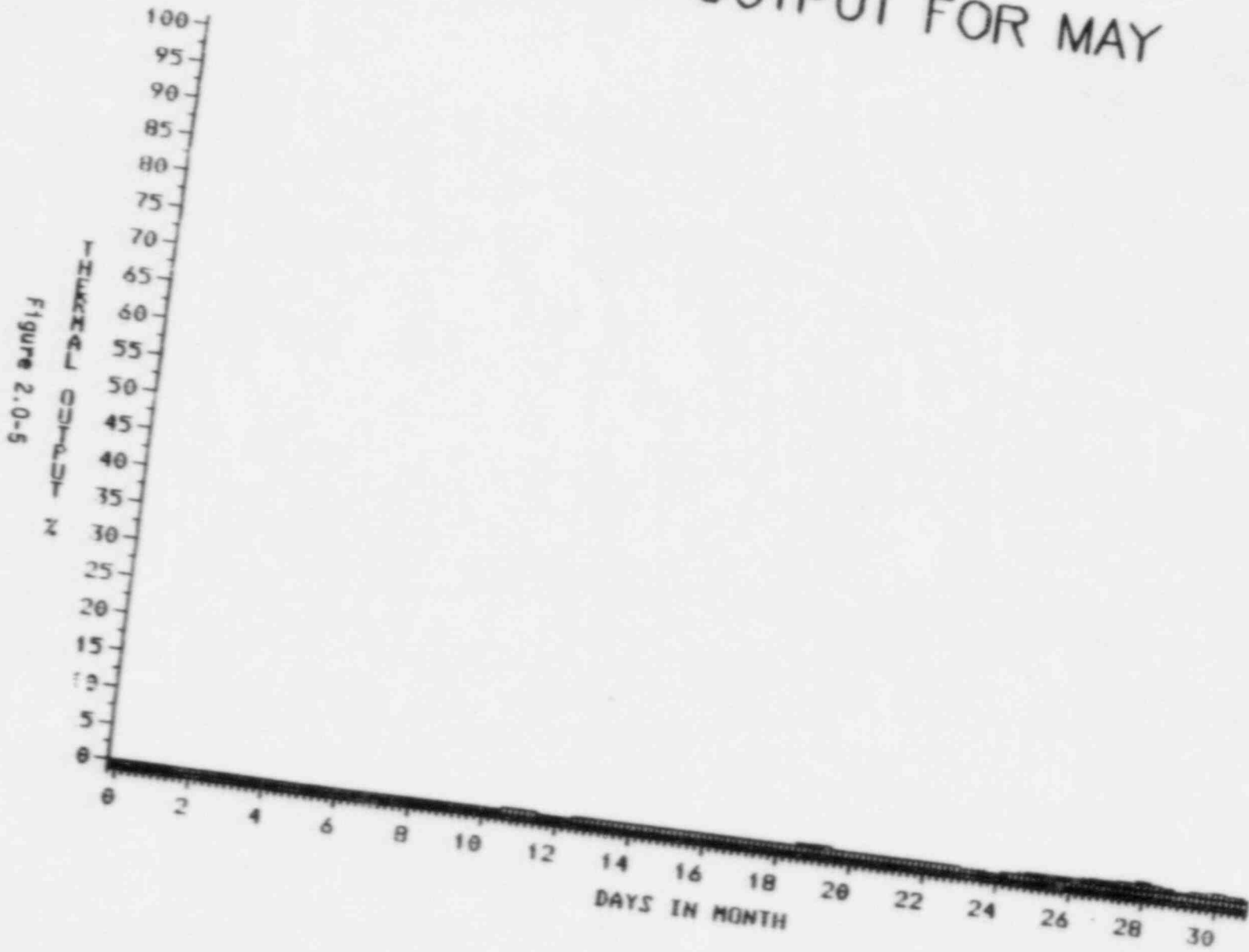


Figure 2.0-5

CATAMBA 1

THERMAL OUTPUT FOR JUNE

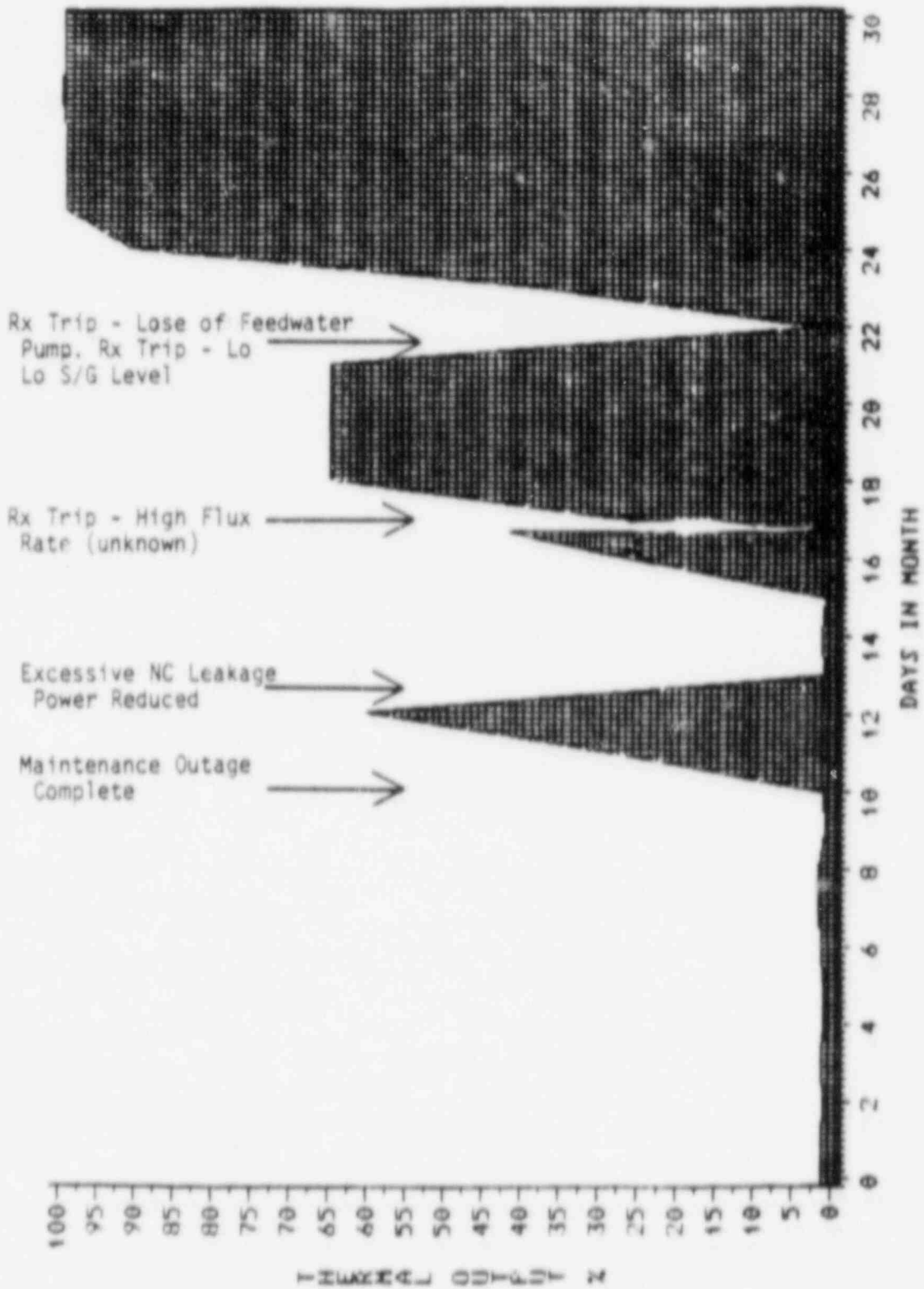


Figure 2.0-6

3.0 INITIAL FUEL LOADING - TP/1/A/2650/01

Date(s) Performed: 7/18/84 - 7/24/84

I. PURPOSE

The purpose of the Initial Fuel Loading procedure was to establish the conditions under which core loading was to be accomplished and to provide a specific sequence of events for installing the fuel assemblies which compose the initial core.

II. METHOD

Each fuel assembly was transferred from the Spent Fuel Pool into the Reactor Building under the direction of the Fuel Handling Supervisor. The Fuel Handling Supervisor was giving directions based upon the loading sequence given in the procedure. As the assembly entered the Reactor Building, the assembly Region Reference Number and insert (i.e. Rod Cluster Control Assembly, Burnable Poison Rod Assembly, thimble plug assembly, source rod assembly) identification number were checked to verify that the proper sequence was being followed.

As each assembly was lowered into the core, three one-minute counts were taken using temporary and permanent neutron detectors. The counts were used to determine an Inverse Count Rate Ratio (ICRR) which was plotted versus the number of assemblies that had been loaded into the core. The plots of ICRR vs. number of assemblies were used to ensure that there was no premature approach to criticality. After the counts were taken for the ICRR and the results verified to be satisfactory, the Performance Test Coordinator would inform the Fuel Handling Supervisor. The Fuel Handling Supervisor would then instruct the operator to disengage the assembly and proceed to the next assembly to be loaded.

After all of the 193 fuel assemblies which compose the initial core had been loaded, PT/1/A/4350/03C, Core Verification, was performed to verify proper loading. Each fuel assembly and associated insert was inspected using an underwater video camera. The location and orientation of all assemblies and inserts were documented as they were inspected. In addition, a videotape was made during the verification. The results were compared to the desired loading pattern and verified correct.

III. RESULTS

The initial core was loaded with no unexpected subcritical multiplication. Significant decreases in ICRR (increases in count rate) occurred as expected, when detectors were "coupled" (i.e. assemblies loaded next to the detectors). During such cases, only the counts from unaffected detectors are used in the ICRR analyses.

Figure 3.0-1 shows the initial core loading sequence. The plots of ICRR vs. number of assemblies are depicted in Figures 3.0-2 through 3.0-6. Table 3.0-1 provides a summary of fuel loading events.

Proper core loading was verified through PT/1/A/4550/03C, Core Verification. Figure 3.0-7 shows the final configuration of fuel assemblies. The assembly insert loading is given in Figure 3.0-8.

IV. CORRECTIVE ACTIONS

A deviation from the prescribed loading sequence occurred when assembly B-17 could not be loaded into Core Location H-09 (Sequence Step 37) due to excessive bow. Fuel assembly B-17 was not placed into the core until assemblies were loaded into core locations adjacent to H-09, creating a "box". B-17 was then loaded (sequence Step 37C) and the remaining assemblies were loaded in the prescribed sequence.

INITIAL FUEL LOADING

FUEL LOADING SUMMARY

<u>DATE</u>	<u>TIME</u>	<u>EVENT</u>
7/18/84	1600	Response check performed on Temporary Detectors
	1900	Response check performed on Source Range Channels (N31 and N32)
7/19/84	0240	Response check repeated on Temporary Detectors
	0335	Response check repeated on N31 and N32
	0400	All prerequisites met. Temporary detectors placed in initial core locations. Background counts obtained.
	0536	First Assembly (C04) is loaded into Core Location L-15. Assembly contains a primary source assembly.
	0601	Second Assembly (C30) also containing a primary source is loaded into location G-01.
	0635	Fuel loading operations suspended due to failure of boron concentration stability. Another sample was taken. Problem shown to be a bad standard.
	0635	A rag was discovered on the lower support plate of the reactor vessel. Fuel loading operations were postponed until the rag could be removed. A complete visual inspection of the lower plate followed removal. The two assemblies which had been loaded were raised to inspect the plate beneath them.
	1639	Third fuel assembly was loaded into core location H-01.
	1716	Fourth fuel assembly was loaded into core location J-01.
	1745	During operation of containment ventilation, dust was blown into the reactor cavity. Fuel loading operations were suspended until Quality Control could perform a cleanliness inspection and clear the reactor vessel for fuel loading.
	1940	Fifth fuel assembly was loaded into core location J-02.
	2010	Sixth fuel assembly was loaded into core location H-02.

TABLE 3.0-1

INITIAL FUEL LOADING

FUEL LOADING SUMMARY (Cont'd.)

<u>DATE</u>	<u>TIME</u>	<u>EVENT</u>
7/19/84	2030	Results of Boron sample taken at 2000 failed stability criteria of ± 20 ppmb of last sample. According to Chemistry personnel, the error in the sampling technique is ± 20 ppmb. The stability criteria was expanded accordingly to ± 40 ppm, with the consent of Westinghouse support personnel (the ± 20 ppmb stability requirement was a Westinghouse recommendation).
	2358	Fuel loading operations resumed with the loading of seventh assembly into core location G-02.
7/20/84	0212	Tenth assembly was loaded into core location F-02. Baseline counts were taken on all detectors. Containment Evacuation alarms for N31 and N32 reset.
	0323	Eleventh fuel assembly was loaded into core location F-03. Counts for ICRR analysis were taken for first time.
	0730	Nineteen (19) assemblies have been loaded. Fuel loading operations suspended to perform a containment air release. Improper operation of the air release system allowed only a slow depressurization of containment. Pressurization of containment was caused by air being used to cool bolts on the reactor coolant pumps.
	1052	Fuel loading resumed with the loading of the twentieth fuel assembly.
	1650	Following the loading of the thirty-fourth fuel assembly, Temporary Detector "C" was moved. Fuel loading operations were suspended for another containment air release. During this release, power to the temporary detectors and the underwater lights were lost when Health Physics air sampler was plugged into the circuit, tripping circuit breaker.
	1936	Fuel handling operations resumed.

TABLE 3.0-1 (Cont'd.)

INITIAL FUEL LOADING

FUEL LOADING SUMMARY

<u>DATE</u>	<u>TIME</u>	<u>EVENT</u>
07/20/84	2010	Assembly B-17, which was to have been the thirty-seventh assembly loaded into the core, could not be loaded into core location H-09 due to excessive bow. Assembly was temporarily stored in the RCCA changing fixture until a four-sided box could be created around Location H-09 and a Westinghouse Fuel Handling Representative was available to advise.
	2350	Resumed Fuel loading operations.
07/21/84	0611	Loaded assembly B-17, which could not be loaded in sequence, was successfully placed into core location H-09. A total of fifty-five fuel assemblies were loaded at this point.
	0811	While loading the fifty-ninth fuel assembly, the containment evacuation alarm was given by N32. The alarm was expected but the procedure called for resetting the alarm setpoint after step sixty was performed, when "coupling" of the detector would be complete. Decision was made to reset alarm setpoint at this time so that the alarm could be cleared before proceeding.
	1149	Fuel loading operations resumed with the loading of the sixtieth assembly. Source range alarm setpoint for N-32 was adjusted again.
	1337	Resumed fuel loading operations.
07/22/84	0231	Ninety-seven (97) fuel assemblies loaded. Reactor Building manipulator crane will not disengage from ninety-eighth assembly due to problems with the Low Load limit switch. Problem was found to be in the Dillon Load cell due to cable stretch during the initial "break-in" period. This problem reoccurred several times over the remainder of fuel loading, but adjustments were made with little interruption of fuel loading operations.
	0547	Fuel loading operations resumed.
07/23/84	0005	One hundred forty-four (144) assemblies loaded.

TABLE 3.0-1 (Cont'd.)

INITIAL FUEL LOADING

FUEL LOADING SUMMARY

<u>DATE</u>	<u>TIME</u>	<u>EVENT</u>
07/23/84	1320	One hundred ninety (190) fuel assemblies placed in core. Temporary detectors A, B and C removed as required to allow final assemblies to be loaded.
	1426	Last assembly placed in core.
07/24/84		PT/1/A/4550/03C, Core Verification was performed. TP/1/A/2650/01, Initial Fuel Loading, completed.

INITIAL FUEL LOADING
CORE LOADING SEQUENCE

- A - Denotes Temporary Detector A (or B or C)
- 1 - Denotes Sequence Step 1 (or 2 through 193)
- Indicates location of primary source
- Indicates an assembly that has previously been loaded into its permanent location
- Location which does not contain an assembly or Temporary Detector

NOTE: Lines with arrows have been drawn to show the movement of an assembly or detector from one location to another; and to indicate the final removal of the temporary detectors and the loading of the last three assemblies.

INITIAL FUEL LOADING
CORE LOADING SEQUENCE

(N31)

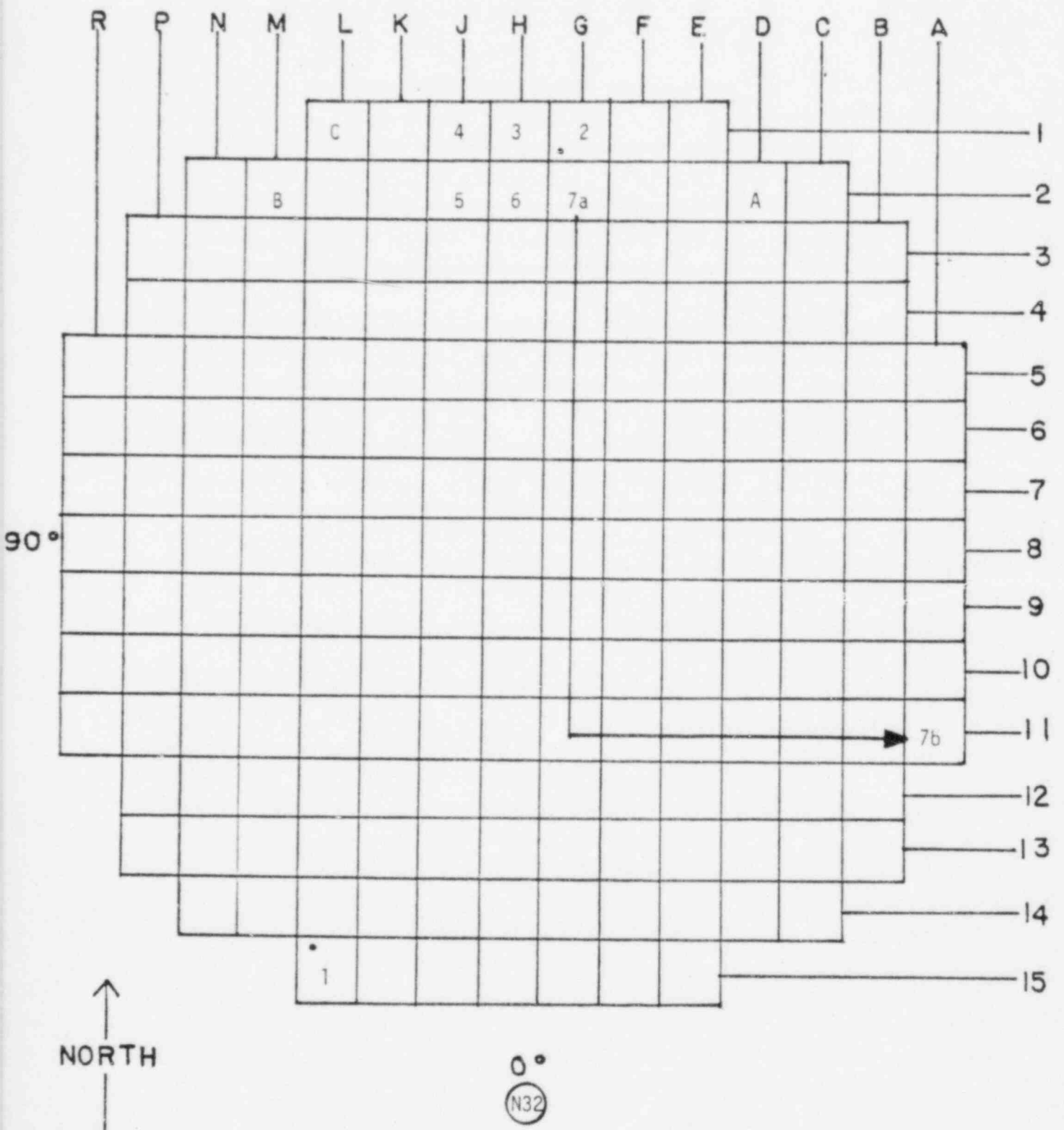
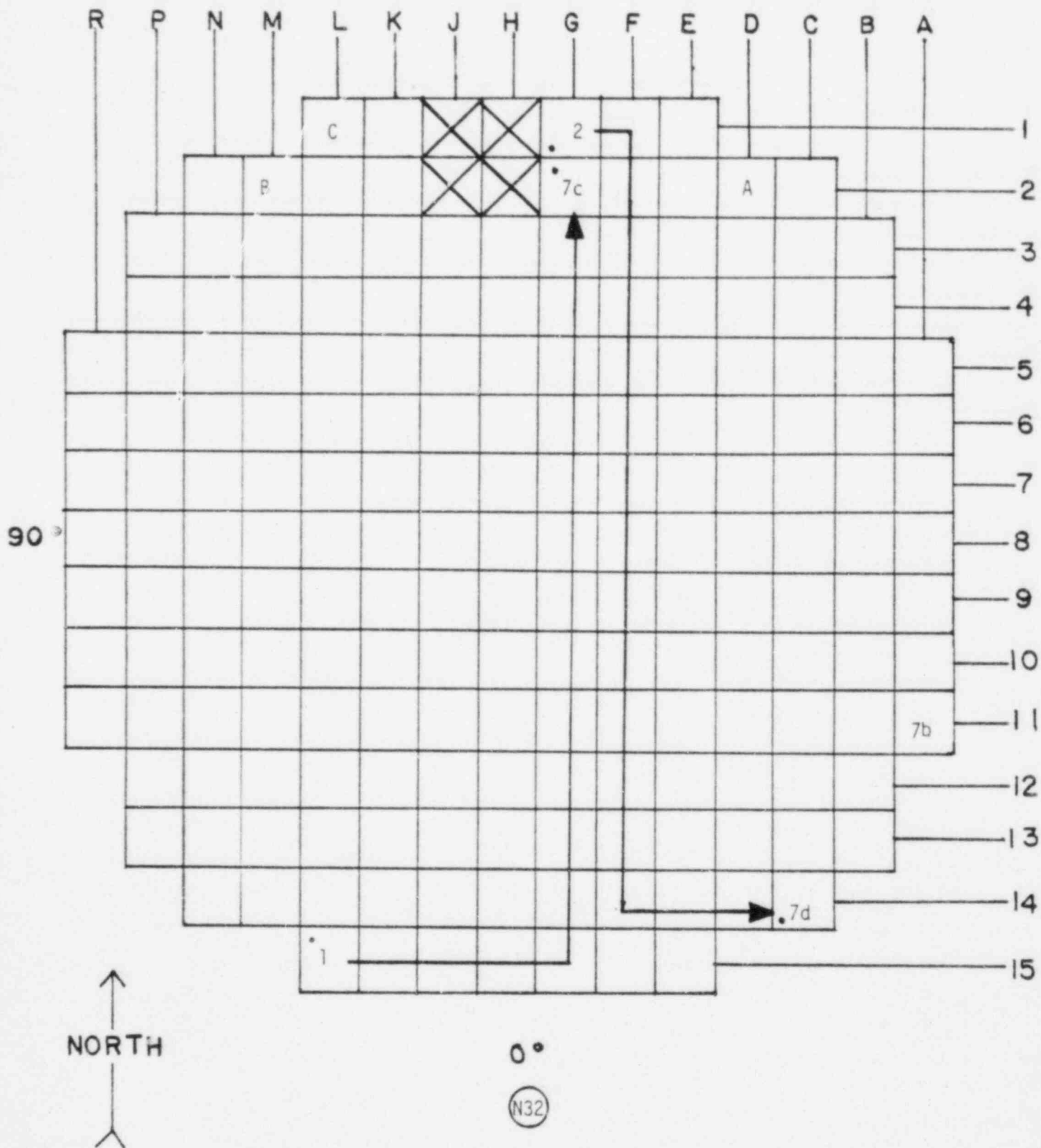


Figure 3.0-1 (continued)

INITIAL FUEL LOADING
CORE LOADING SEQUENCE

(N31)

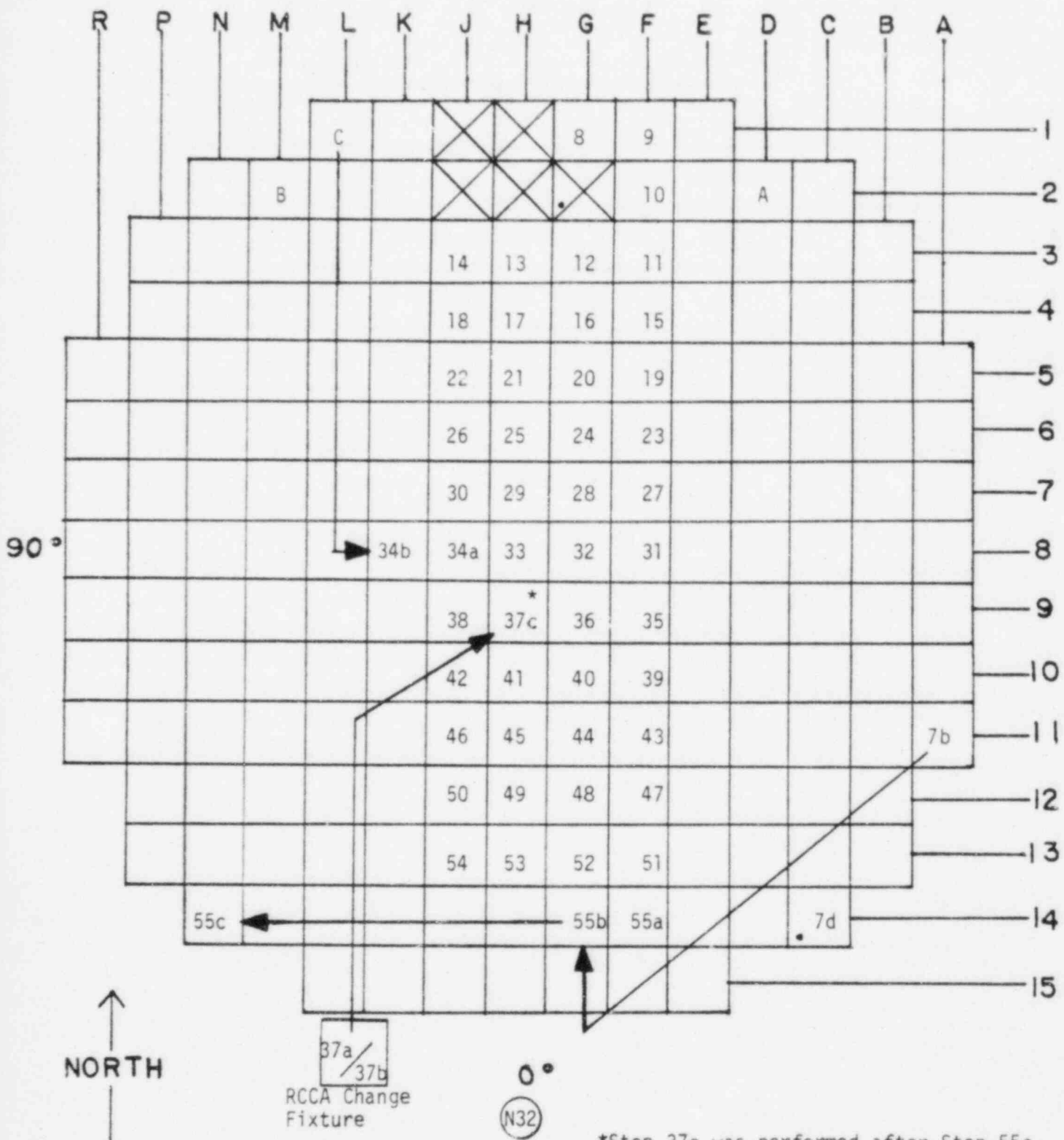


(N32)

Figure 3.0-1 (continued)

INITIAL FUEL LOADING
CORE LOADING SEQUENCE

(N31)



*Step 37c was performed after Step 55c

Figure 3.0-1 (continued)

INITIAL FUEL LOADING
CORE LOADING SEQUENCE

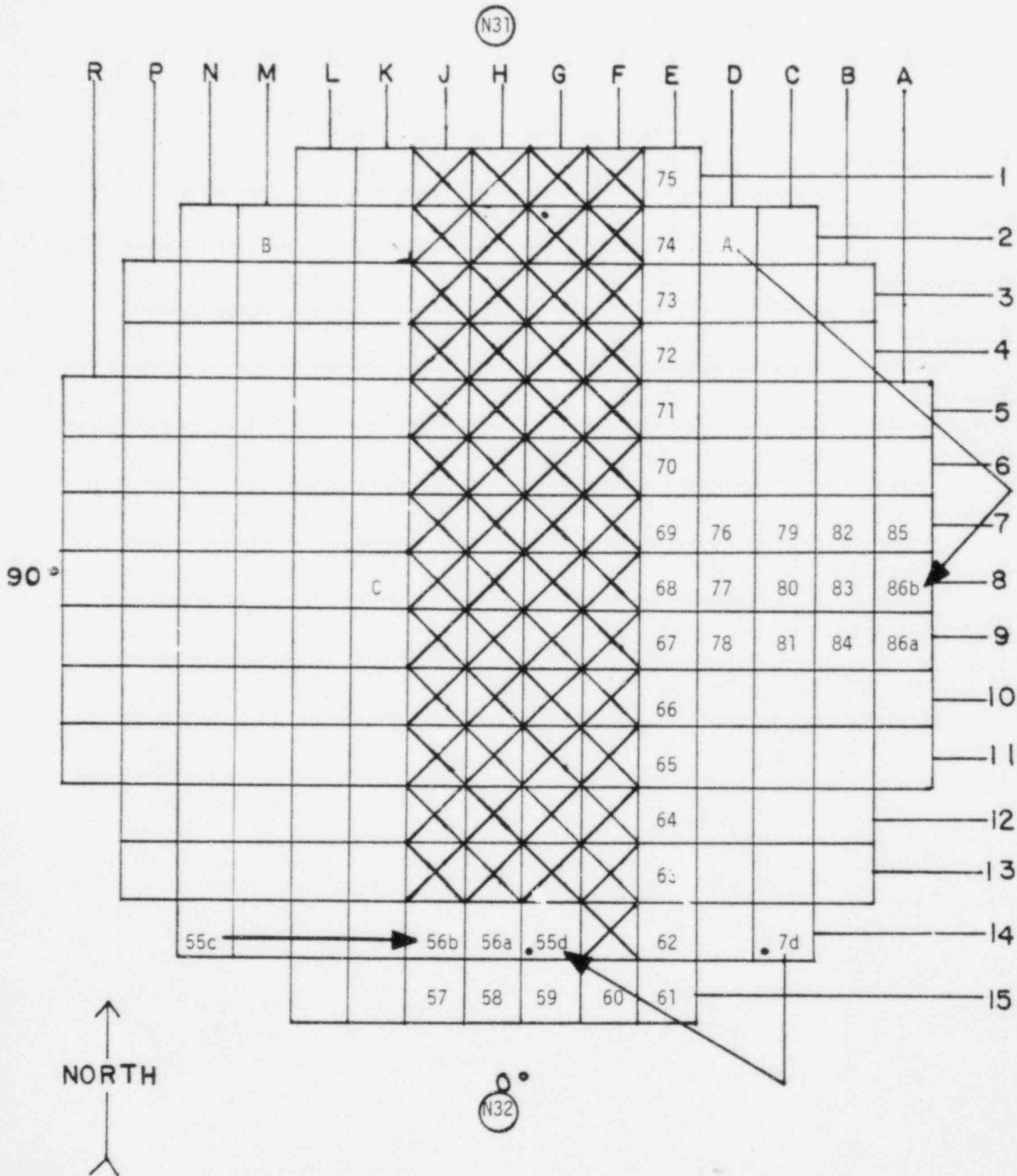


Figure 3.0-1 (continued)

INITIAL FUEL LOADING
CORE LOADING SEQUENCE

(N31)

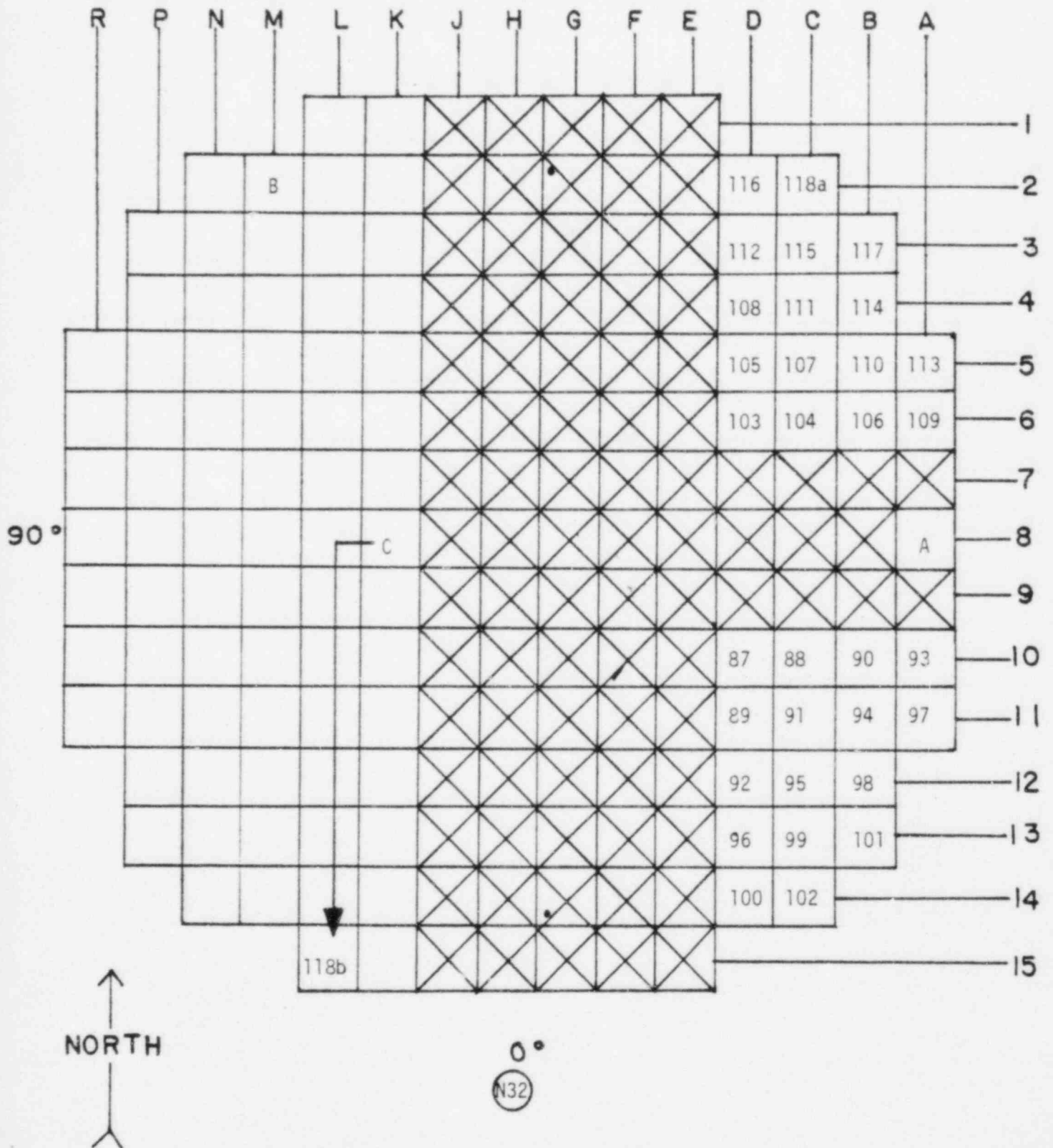


Figure 3.0-1 (continued)

INITIAL FUEL LOADING
CORE LOADING SEQUENCE

(N31)

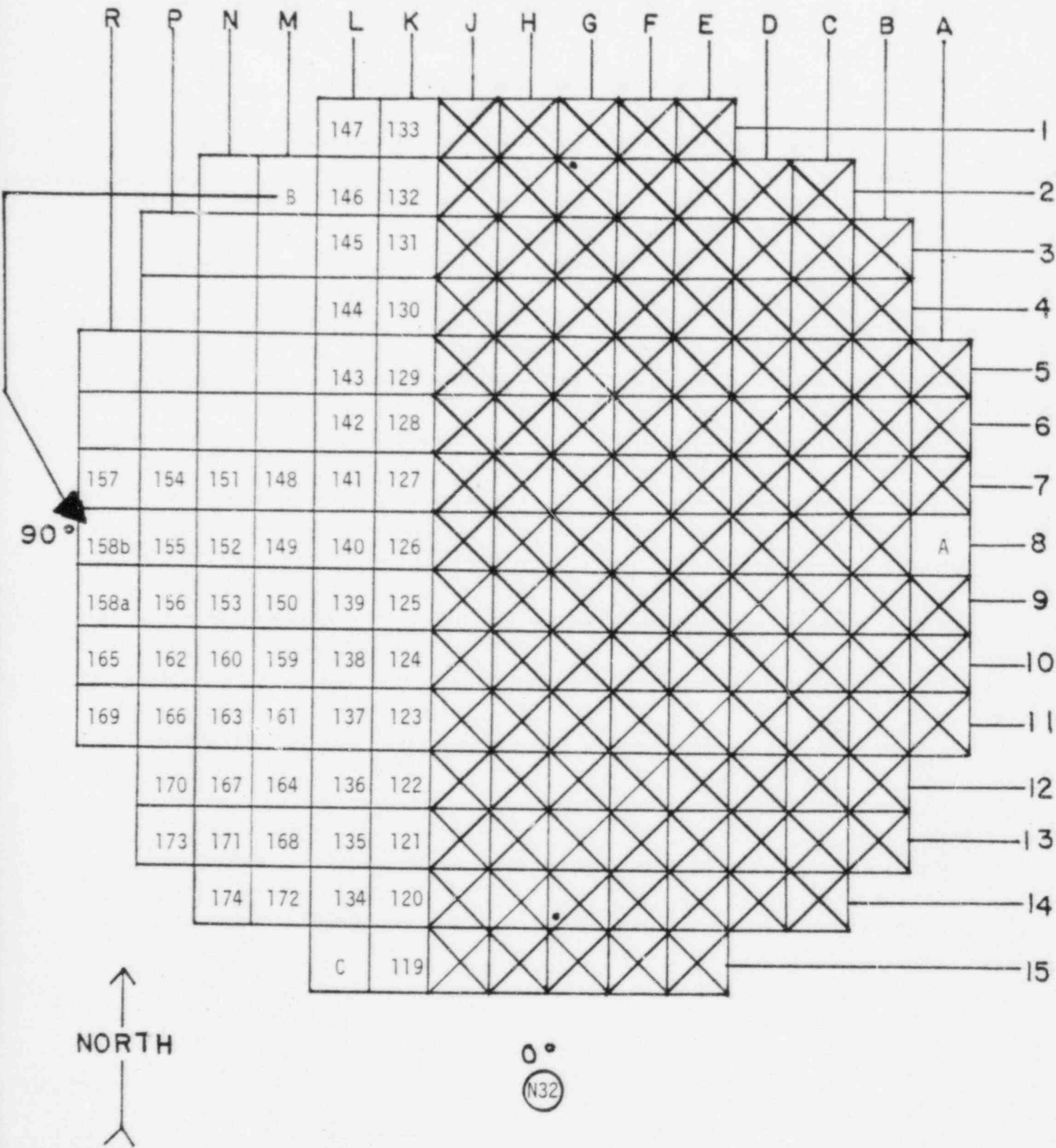


Figure 3.0-1 (continued)

INITIAL FUEL LOADING
CORE LOADING SEQUENCE

(N31)

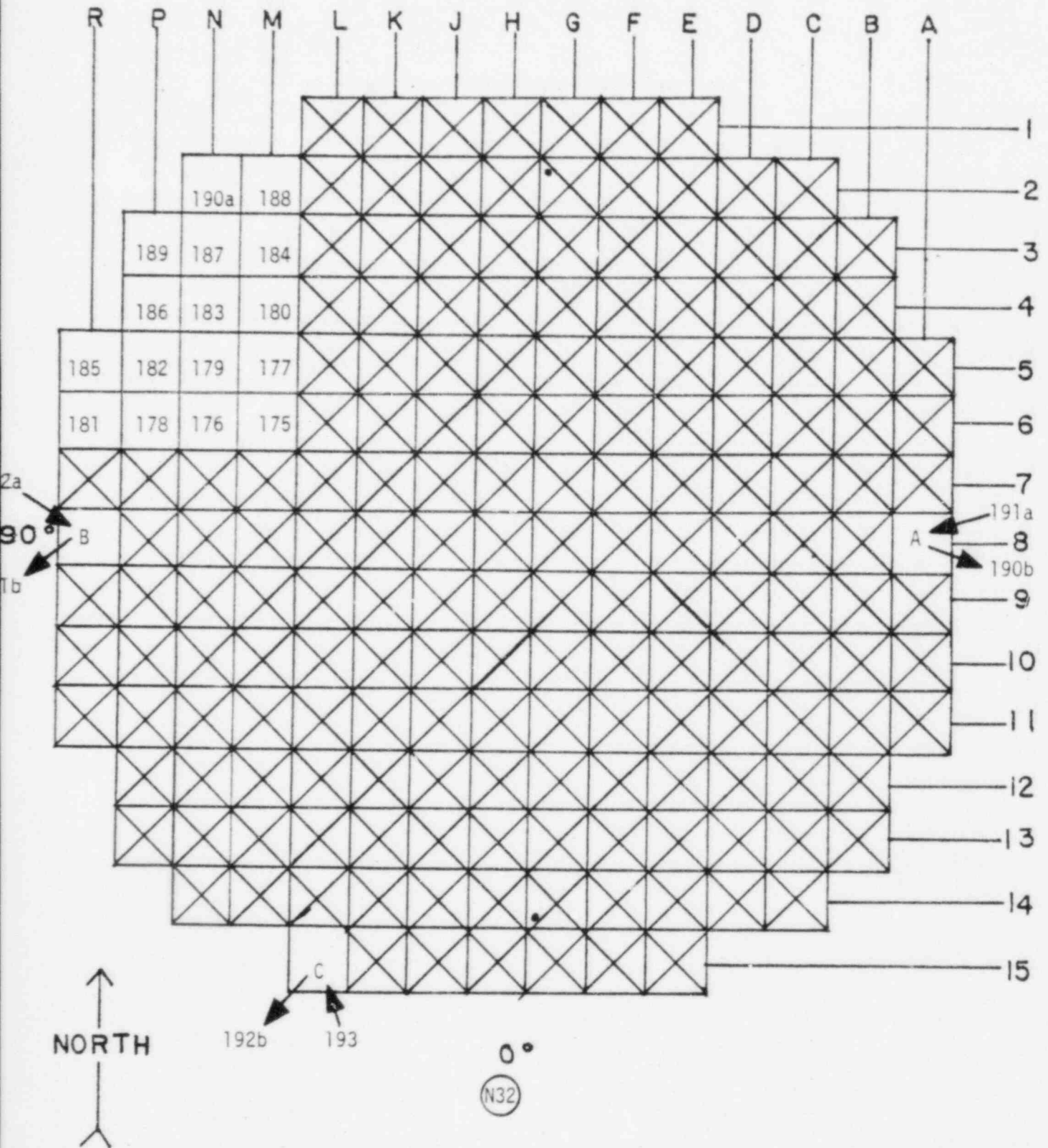


Figure 3.0-1 (continued)

INITIAL FUEL LOADING
ICRR vs. Number of Loaded Assemblies

SOURCE RANGE CHANNEL N31

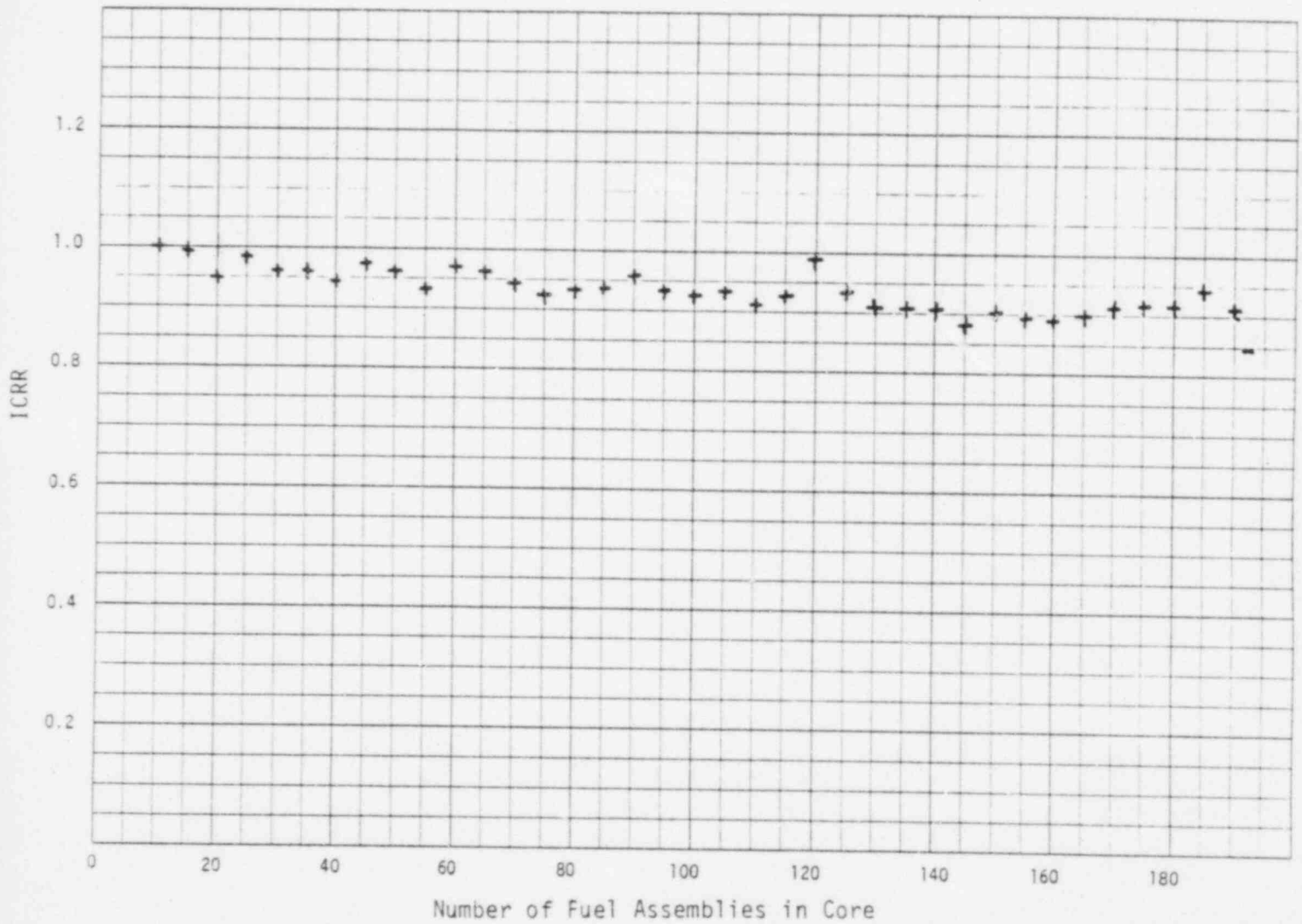


Figure 3.0-2

INITIAL FUEL LOADING

ICRR vs. Number of Loaded Assemblies

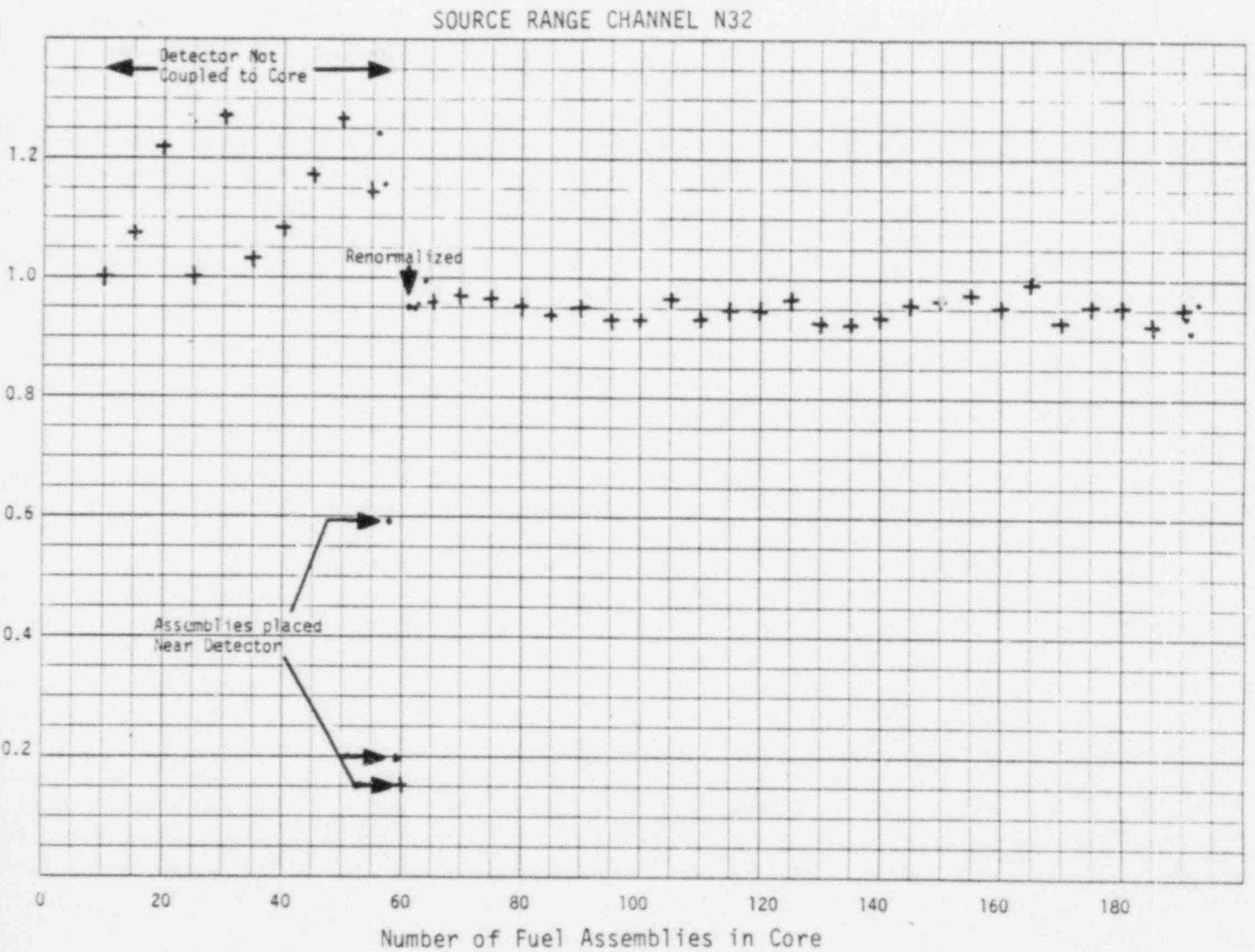


Figure 3.0-3

CATAWBA 1

INITIAL FUEL LOADING

ICCR vs. Number of Loaded Assemblies

TEMPORARY DETECTOR A

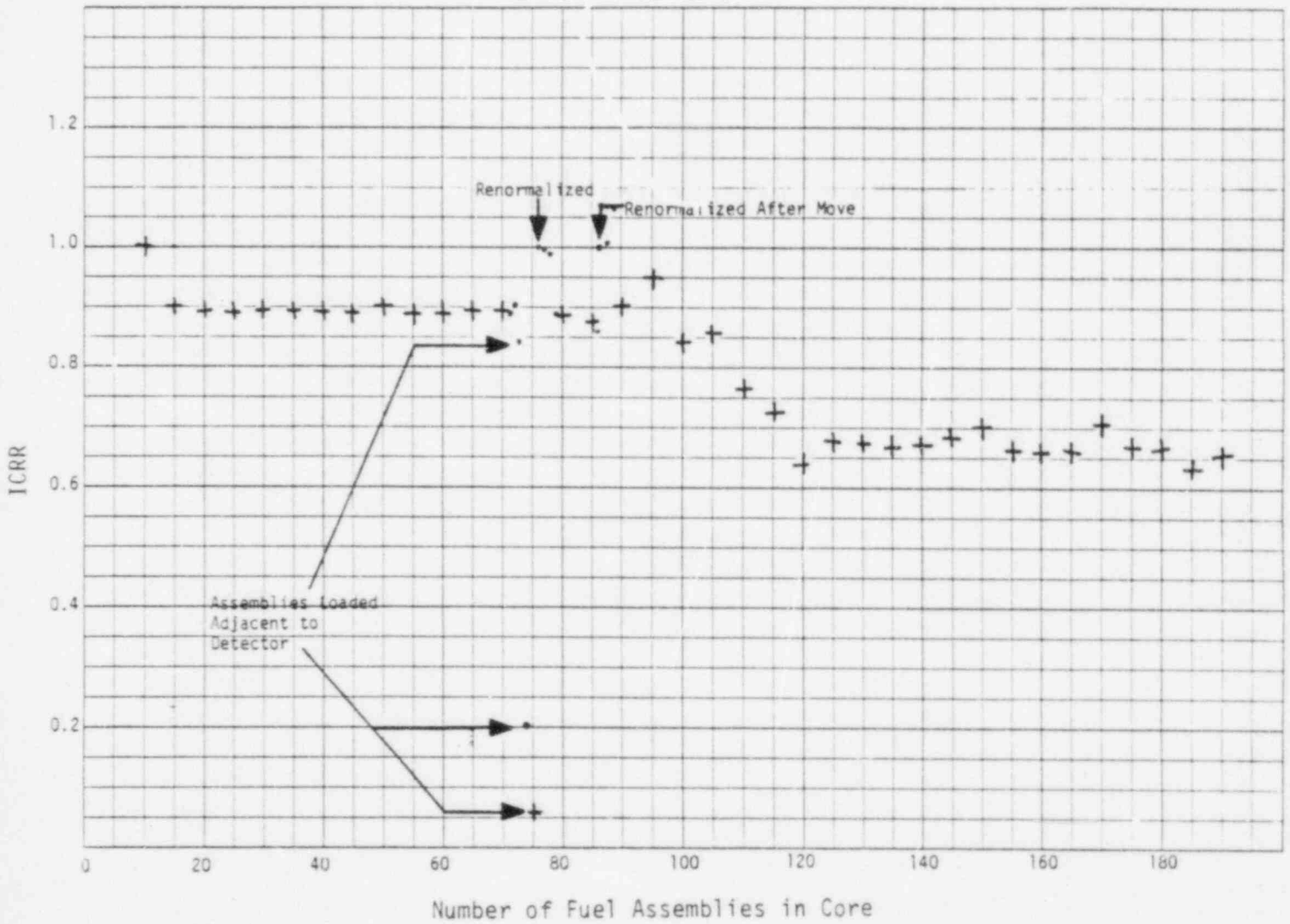


Figure 3.0-4

INITIAL FUEL LOADING

ICRR vs. Number of Loaded Assemblies

TEMPORARY DETECTOR B

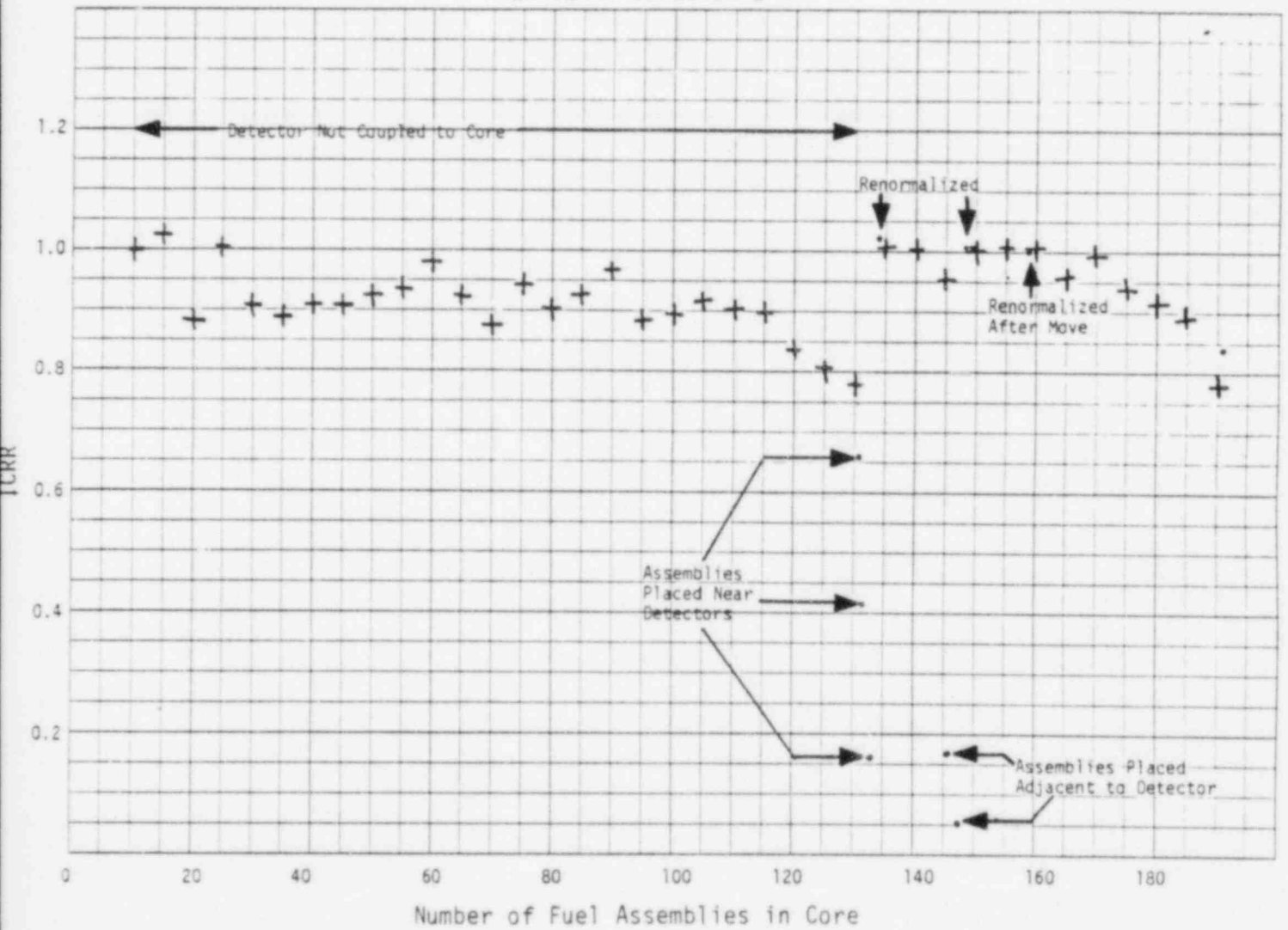


Figure 3.0-5

INITIAL FUEL LOADING

ICRR vs. Number of Loaded Assemblies

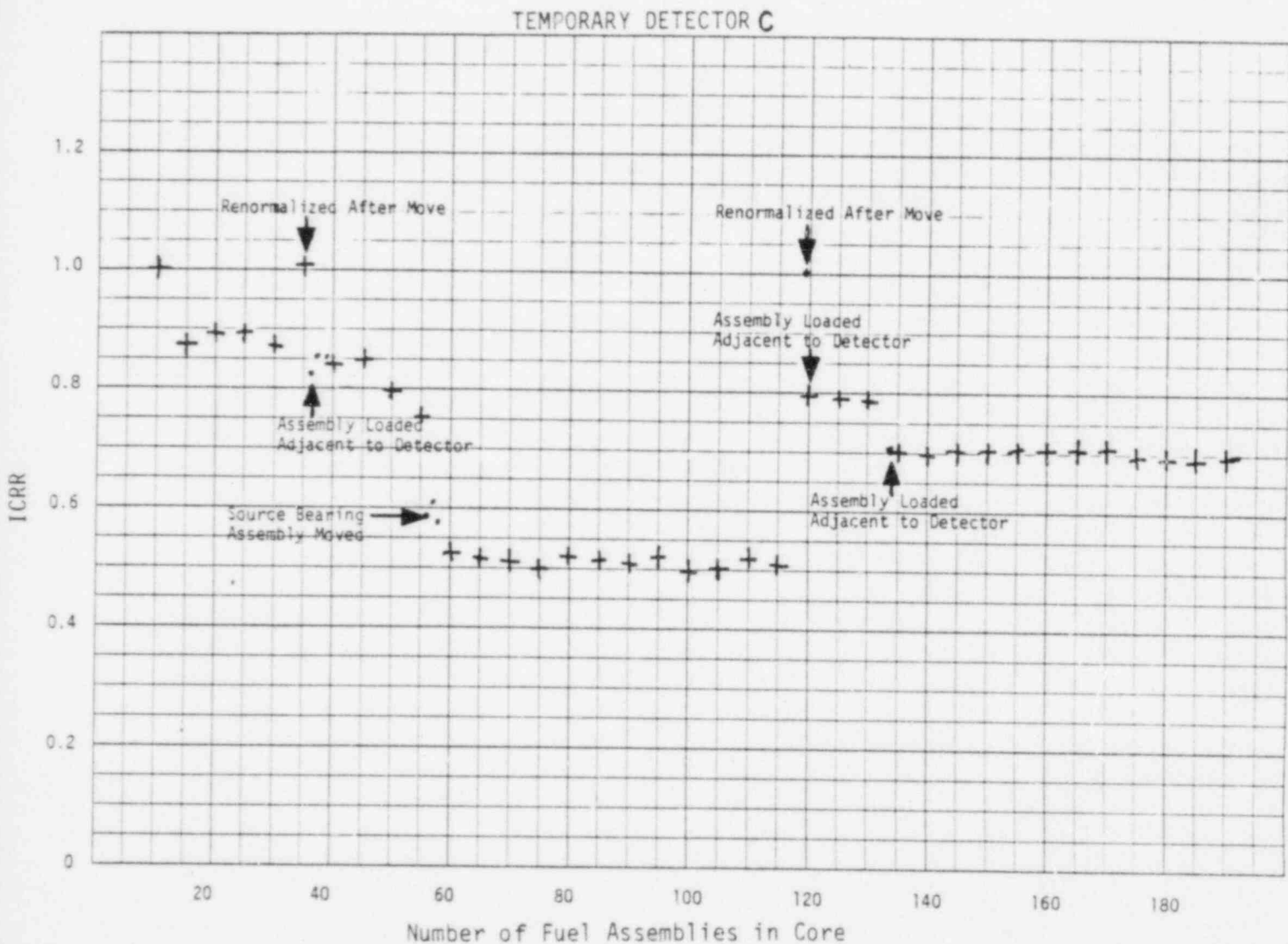
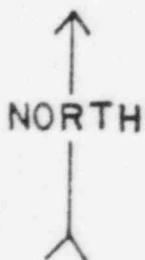
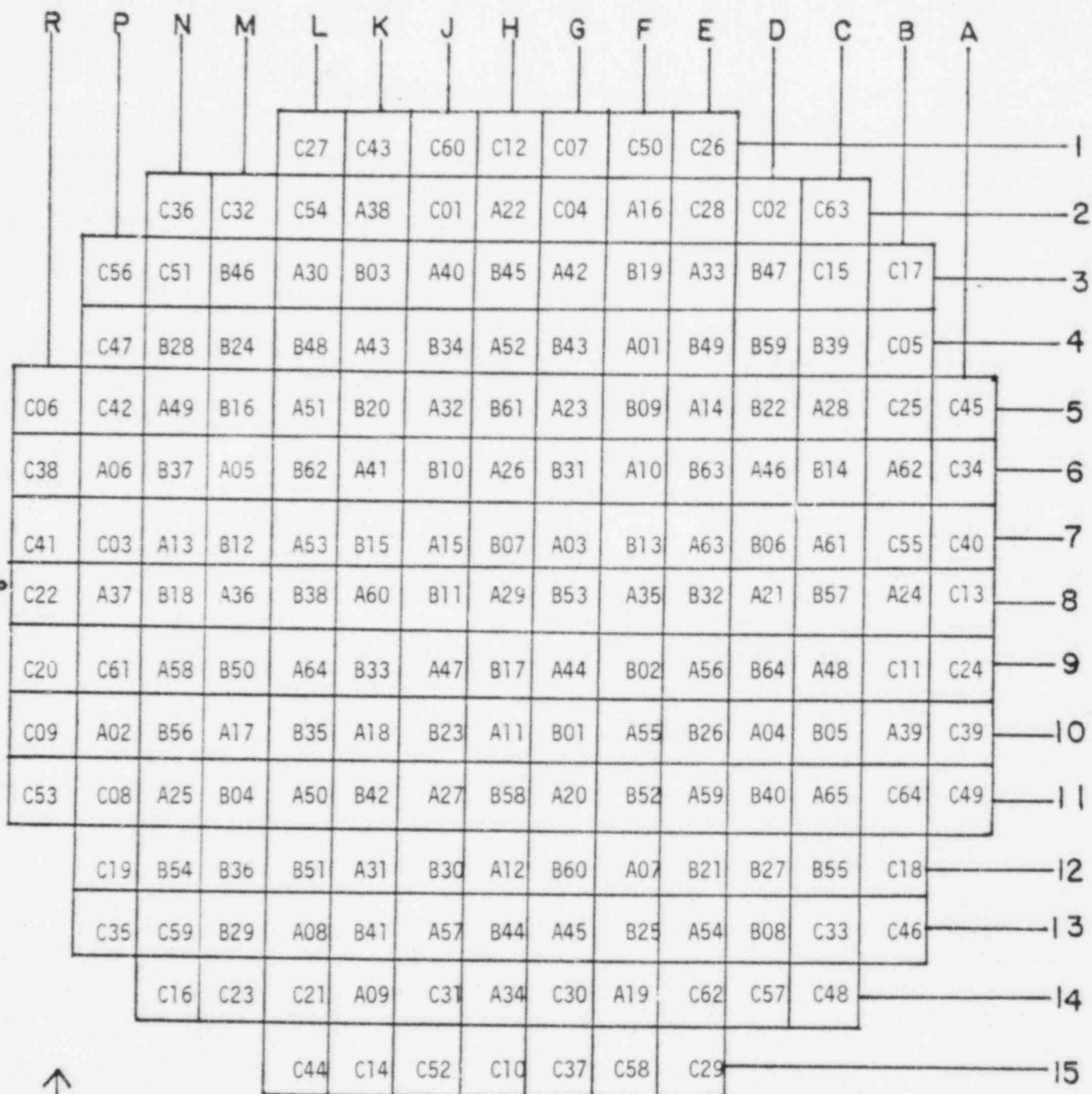


Figure 3.0-6

INITIAL FUEL LOADING
CORE LOADING PATTERN - FUEL ASSEMBLIES

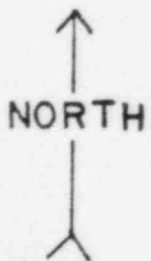
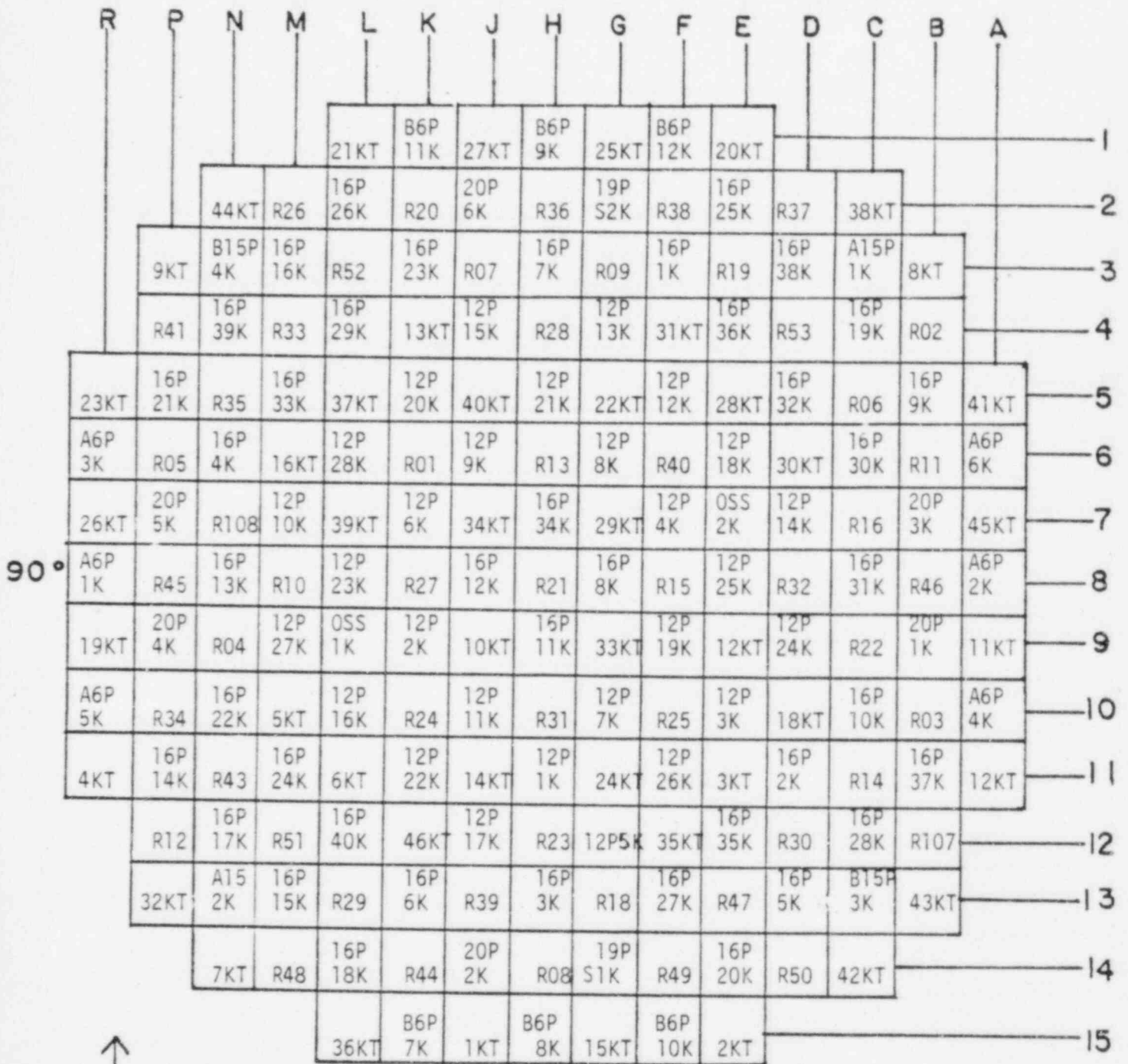


0°

AXX - Region 1 Assembly - 1.60 % U-235
 BXX - Region 2 Assembly - 2.40% U-235
 CXX - Region 3 Assembly - 3.10% U-235

Figure 3.0-7

INITIAL FUEL LOADING
CORE LOADING PATTERN - INSERTS



YYKT - Thimble Plug
 RYY - Rod Cluster Control Assembly
 (A,B) XXPYK - Burnable Poison Rod Assembly with XX poison rods.
 (The A or B designate asymmetric BPRA's)
 OSSYK - Secondary Source Assembly
 19PSYK - Primary Source Assembly containing 19 poison rods and 1 source rod.

Figure 3.0-8

4.0 TESTING PRIOR TO INITIAL CRITICALITY

Following initial fuel loading of Catawba Unit 1, various tests were performed prior to initial criticality. This testing included the following:

- Moveable Incore Detector Functional Test
- Incore Thermocouple and RTD Cross Calibration
- Rod Position Indication Check
- Rod Cluster Control Assembly Drop Time Test
- Rod Control Alignment Test
- Rod Drive Mechanism Timing Test
- Reactor Coolant Flow Test
- Reactor Coolant Flow Coastdown Test
- RTD Bypass Flow Verification
- Pressurizer Functional Test

The tests performed during this period are discussed on the following pages.

4.1 MOVABLE INCORE DETECTOR FUNCTIONAL TEST - TP/1/B/2600/01

Date(s) Performed: 11/20/84 to 12/4/84

I. PURPOSE

The purpose of this test was to demonstrate the operability of the movable detector portion of the Incore Instrumentation (ENA) System. This was achieved by verifying the following:

- A. All drive units are able to drive detectors into all available thimbles in all modes of operation.
- B. All setpoints, limit switches, indicator lights and control circuits function properly.
- C. All drive units exhibit satisfactory motor speed and braking action.
- D. The Leak Detection and Gas Purge sub-systems operate properly.

II. METHOD

Dummy incore detector cable assemblies were installed on the drive units and the system operated from the Control Room console. Each path and mode possible was tested by inserting the detector and observing the various indicators. Limit switches and braking action were checked by determining the distance between the setpoints and the actual stopped position. Motor speed was checked using timed runs between two setpoints. The Leak Detection sub-system was tested by manual actuation of the pressure switch and verifying that the alarm actuated. The Gas Purge sub-system was tested by using an inclined manometer to measure the positive gauge pressure generated during operation.

III. RESULTS

All Acceptance Criteria for this test were met. All drive units were capable of inserting the dummy detector into every available thimble in all modes. All setpoints, limit switches, indicator lights, and control circuits functioned properly. High-speed braking action was within the allowed distance (± 14 inches relative to the setpoint) in all cases. Low speed braking was always well within the ± 2 inch criterion. Typical measured distances for high and low speed braking were 6 and 0.2 inches respectively. Motor speeds were within the allowed range of $12 \text{ ft/min} \pm 0.6 \text{ ft/min}$. Actual measured speeds were all in the range of $12 \text{ ft/min} \pm 0.1 \text{ ft/min}$. The Leak Detection sub-system alarmed when the pressure switch was actuated. The Gas Purge sub-system met its Acceptance Criterion by showing it was capable of generating a positive pressure (relative to atmospheric) within the ten-path enclosures. A pressure of $\sim 0.02'' \text{ H}_2\text{O}$ was measured at a flowrate of $\sim 3\ell/\text{min}$.

IV. CORRECTIVE ACTIONS

Several problems were encountered during this test. The most serious was the failure of the magnetically actuated brakes on two of the drive units. This was apparently due to insufficient force being provided by the compression springs within the brake. This resulted in complete loss of braking action in some cases. Temporary Station Modifications were used to add backup washers beneath the springs to increase their compression. This solved the problem and all brakes met the Acceptance Criteria.

There were numerous instances of misadjusted limit switches and core display light switches. These were adjusted as necessary. Following retesting, all switches met the Acceptance Criteria. Also the position encoder on drive C failed and required replacement.

4.2 INCORE THERMOCOUPLE AND RTD CROSS-CALIBRATION - IP/O/A/3231/01

Date(s) Performed: Sep. 28, 1984; Oct. 3, 23, 1984; Nov. 16 to 17, 21, 1984; and Dec. 4, 6, 1984

I. PURPOSE

The purpose of the cross-calibration was to determine installation errors for each of the Reactor Coolant System Narrow Range Resistance Temperature Detectors (RTD's) and the incore thermocouples.

II METHOD

Reactor Coolant temperature was maintained as stable as possible at a specified temperature plateau (250°F, 350°F, 450°F and 557°F). Resistance readings for all of the RTD's were recorded and incore thermocouple temperature readings were obtained. Resistance data was used to calculate the temperature that each RTD was reading. The results were averaged and the installation error, defined as the deviation of an individual RTD from the average, was determined.

The reading of each incore-thermocouple was subtracted from the average RTD temperature to obtain the correction factor which was input in to the plant computer.

A Heise gauge was used as an independent measure of Reactor Coolant System temperature. The gauge was used to measure the pressure of each main steam line at a given temperature plateau. The corresponding temperature of saturated steam should be the same as the temperature of the reactor coolant system under isothermal conditions.

III. RESULTS

All of the 16 narrow range RTD's were shown to have an installation error of less than 2°F at 557°F, as required. Correction factors when applied to each thermocouple channel gave results consistent with the average RTD temperature.

IV. CORRECTIVE ACTIONS

The test was repeated at 250°F, 350°F and 450°F due to the replacement of two incore thermocouples. The test was repeated several times at 557°F because of questionable results obtained from the calculations. The problem was discovered to be caused by jumpers placed on the test cards in the process cabinets where resistance readings were obtained. These jumpers were part of a Westinghouse modification. Also, the input-shorting straps on the digital multimeter used were not removed. The test was finally completed at 557°F on 12/6/84 after a procedure change had been written to avoid the problems caused by the resistance added by the jumpers.

The results at 250°F, 350°F and 450°F are questionable due to the jumpers installed in the cabinets; however, the test was not repeated at these temperatures, as the data from temperatures other than 557°F was not required to show that Acceptance Criteria were met.

4.3 ROD POSITION INDICATION ALIGNMENT - TP/1/B/2600/04

Date(s) Performed: 11/29/84

I. PURPOSE

To verify that the Digital Rod Position Indication System (DRPI) satisfactorily performs the required indications and alarm functions for each individual rod under hot shutdown conditions.

II. METHOD

- A. Each shutdown bank of rods is fully withdrawn in 24 step increments, then reinserted. The DRPI, group step counter and computer indications are compared.
- B. Each control bank of rods is fully withdrawn in 24 step increments then reinserted. The DRPI, group step counters, computer, Pulse-to-Analog (P/A) converter and control board chart recorder indications are compared.
- C. As each rod is being inserted, the group step counter position indication at which the rod bottom LED illuminates is recorded.

III. RESULTS

All acceptance criteria were met. The maximum deviation of any individual rod position indication when compared to its group step counter was ± 2 steps and well within the acceptance criteria of ± 4 steps. When compared with the appropriate group step counter each rod bottom LED activated at 3 steps ± 1 step which was within the specified tolerance of 3 steps ± 2 steps.

IV. CORRECTIVE ACTION

Mechanism misstepping occurred on various CRDMs. This was remedied by moving rods in/out and/or dropping the various banks, which contained the affected rods, thereby flushing the mechanism. The test was completed satisfactorily.

4.4 ROD CLUSTER CONTROL ASSEMBLY DROP TIME TEST - PHASE 2 - TP/1/B/2600/06

Date(s) Performed: 9/6/84 - 9/11/84, 11/26/84 - 11/29/84

I. PURPOSE

The objectives of the Rod Cluster Control Assembly (RCCA) Drop Time Test were:

- A. To determine the drop time of each RCCA under both conditions of no NC Flow and full NC Flow.
- B. To determine the effectiveness of the dashpot region for decelerating the RCCA during each drop time measurement.

These objectives were completed while the Unit was in a cold shutdown condition and again while in hot standby.

II. METHOD

With the reactor in the cold shutdown condition, each bank of control rods was selected in turn and stepped out to the fully withdrawn position (228 steps). The individual drop time for each rod in the bank was then determined by pulling the selected Control Rod Drive Mechanism's (CRDM) moving and stationary gripper coil fuses in the appropriate IRE System Power Cabinet, and then recording the voltage profile induced by the RCCA drive shaft dropping through the coils of the DRPI detector. Rods for which the initial drop time to dashpot entry differed from the mean drop time (to dashpot entry) by more than two standard deviation units, measured under similar plant conditions, were redropped an additional three times. Proper deceleration through the dashpot region was verified by analysis of the recorded voltage profile. The drop timing was performed under both conditions of no NC Flow and again with full NC Flow. The entire sequence of testing was performed with the unit in cold shutdown and again while in hot standby.

III. RESULTS

All Acceptance Criteria associated with this test were met.

- A. For all combinations of plant conditions (hot or cold, full NC Flow or no Flow), all of the individual full-length shutdown and control rod drop times from the fully withdrawn position were less than or equal to 3.3 seconds from beginning of stationary gripper coil voltage decay to dashpot entry. These results are summarized in Table 4.4-1. Hot Full Flow drop times are detailed in Figure 4.4-1.

- B. The longest and shortest rod drop times for all repeated drops of RCCA's whose initial drop time to dashpot entry differed from the mean by more than two standard deviation units, did not differ by more than 0.04 seconds.
- C. All recorded voltage profiles were consistent in form (but not necessarily amplitude), and demonstrated rod free fall with no abnormalities or evidence of binding. A typical trace is shown on Figure 4.4-2.
- D. Control rod deceleration through the dashpot region, as observed on the recorded voltage profiles, was uniform and consistent for all rods dropped under the same plant conditions.

IV. CORRECTIVE ACTION

During the cold no flow rod drop tests, drop times were inconsistent and many were relatively fast. With thirty-five out of the fifty-three rods dropped, the fastest drop time was 1.08 seconds (H-8) and the slowest was 1.39 seconds (P-6 and H-12). For the data gathered, the mean drop time was 1.31 seconds with a standard deviation (σ) of 0.090 seconds. After careful evaluation, venting and redropping the CRDM's was recommended. After venting H-8, the drop time increased from 1.08 to 1.39 seconds. The remaining CRDM's were then vented and the cold no flow drop tests repeated. The new data resulted in a mean drop time of 1.39 seconds with a σ of only 0.012 seconds.

A noise spike was observed on the cold no flow and, to a lesser extent, on the hot no flow rod drop traces for the RCCA at core location F-2. The noise spike was absent from both the cold full flow and hot full flow traces for F-2. Several additional drops at cold no flow conditions verified that the noise spike was repeatable and consistent in both location and amplitude. It was suggested that the RCCA spider might be nicking a guide tube support plate as it passed through the upper internals of the reactor. The absence of an indication during the full flow runs was attributed to improved alignment of the RCCA caused by high NC flow velocities up through the fuel and guide tube. The indication was not considered significant enough to halt testing for further investigation.

Loose breech guide screws were discovered at several Westinghouse PWR's. In one case, this led to jamming of a CRDM latch assembly such that motion was prevented. This necessitated inspections to the CRDM's at several plants including both Catawba units. The NRC issued IE Information Notice No. 85-14 describing this problem and the inspection results. Fourteen CRDM's were replaced on Catawba 1 following inspection. All RCCA's were then successfully drop time tested under hot full flow conditions on 1/5/85. This was performed using IP/O/A/3220/01, Full Length RCCA Drop Timing. The drop times, shown on Figure 4.4-3, are very similar to the original results.

ROD CLUSTER CONTROL ASSEMBLY
DROP TIME TEST - PHASE 2

ROD DROP TIME DATA SUMMARY

Test Conditions	Max. Drop Time		Min. Drop Time		Mean Drop Time (sec)	Standard Deviation (sec)	Repeated Drops	
	(sec)	Rod#	(sec)	Rod#			Rod#	Max. Dev. (sec)
Cold No Flow	1.42	L-3	1.36	H-14	1.39	0.012	L-3	0.01
		N-5					N-5	0.03
							H-14	0.01
Cold Full Flow	1.81	E-13	1.70	C-11	1.75	0.026	E-13	0.01
Hot Full Flow	1.68	D-14	1.54	L-3	1.58	0.030	M-2	0.01
				K-6			D-14	0.01
				D-11			M-14	0.02
				N-11				
Hot No Flow	1.36	P-4	1.28	H-14	1.31	0.015	P-4	0.02

TABLE 4.4-1

ROD CLUSTER CONTROL ASSEMBLY
DROP TIME TEST - PHASE 2

HOT FULL FLOW DROP TIMES

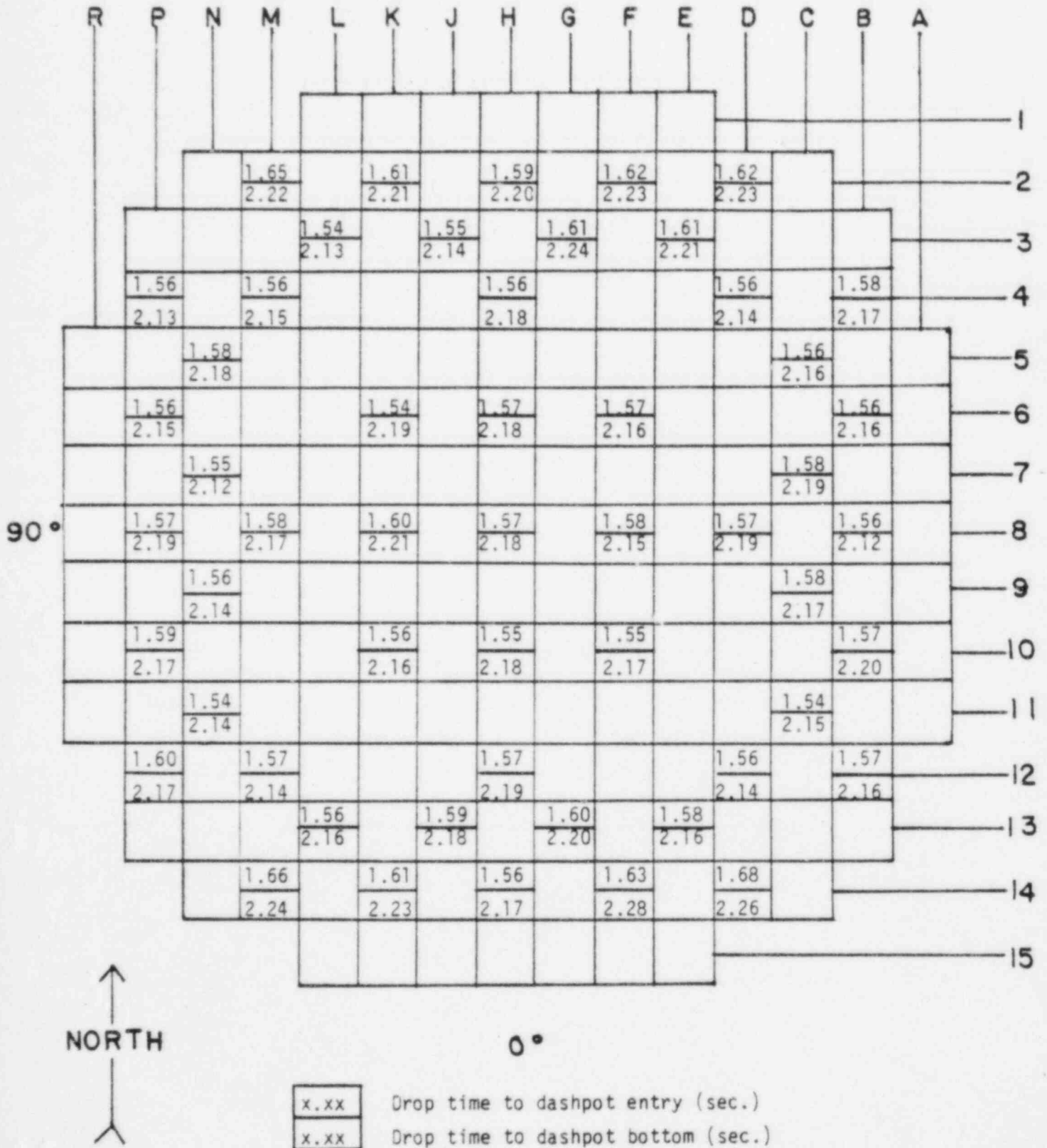


FIGURE 4.4-1

ROD CLUSTER CONTROL ASSEMBLY
DROP TIME TEST - PHASE 2

TYPICAL ROD DROP VOLTAGE TRACE

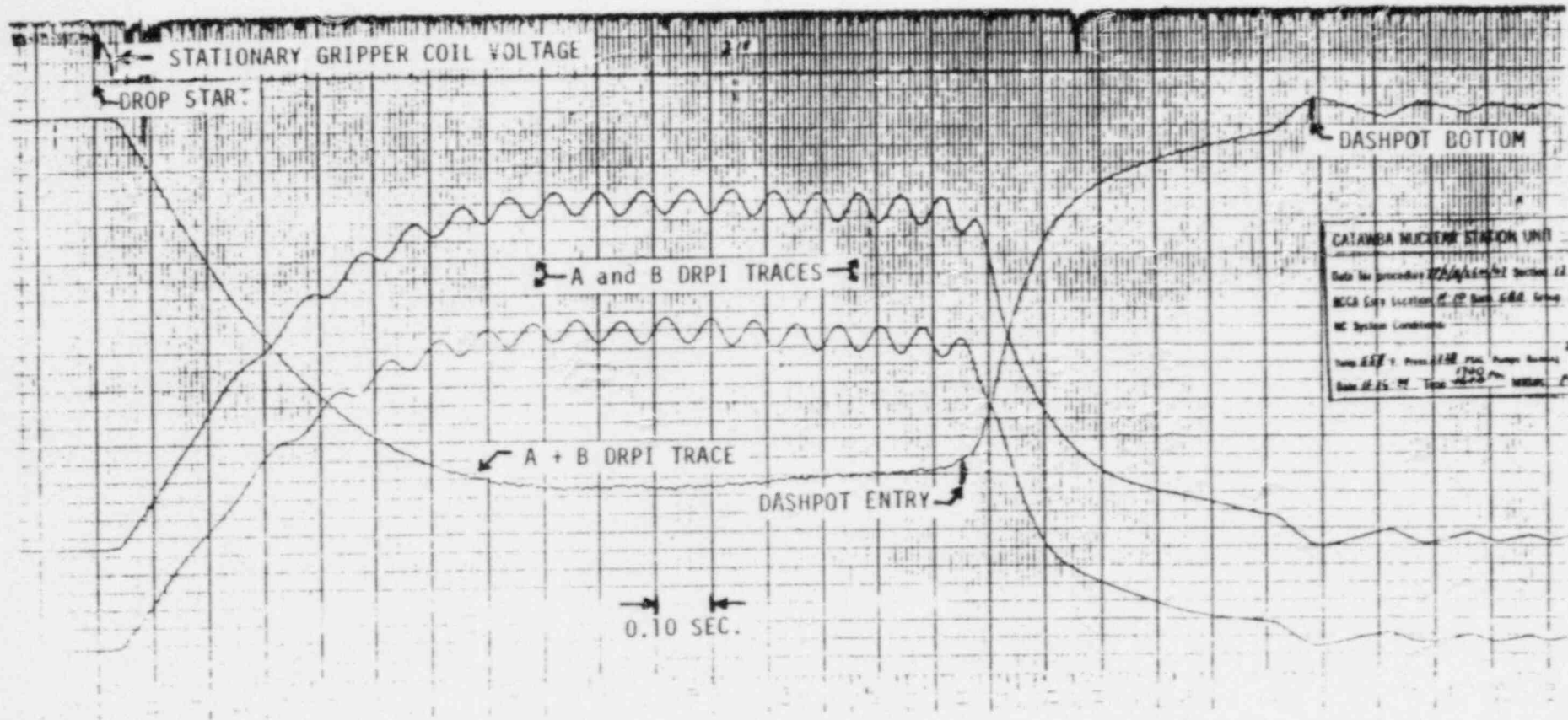


FIGURE 4.4-2

CATAMBA 1

HOT FULL FLOW DROP OF RCCA H-10

ROD CLUSTER CONTROL ASSEMBLY
DROP TIME TEST - PHASE 2

HOT FULL FLOW DROP TIMES (RETEST)

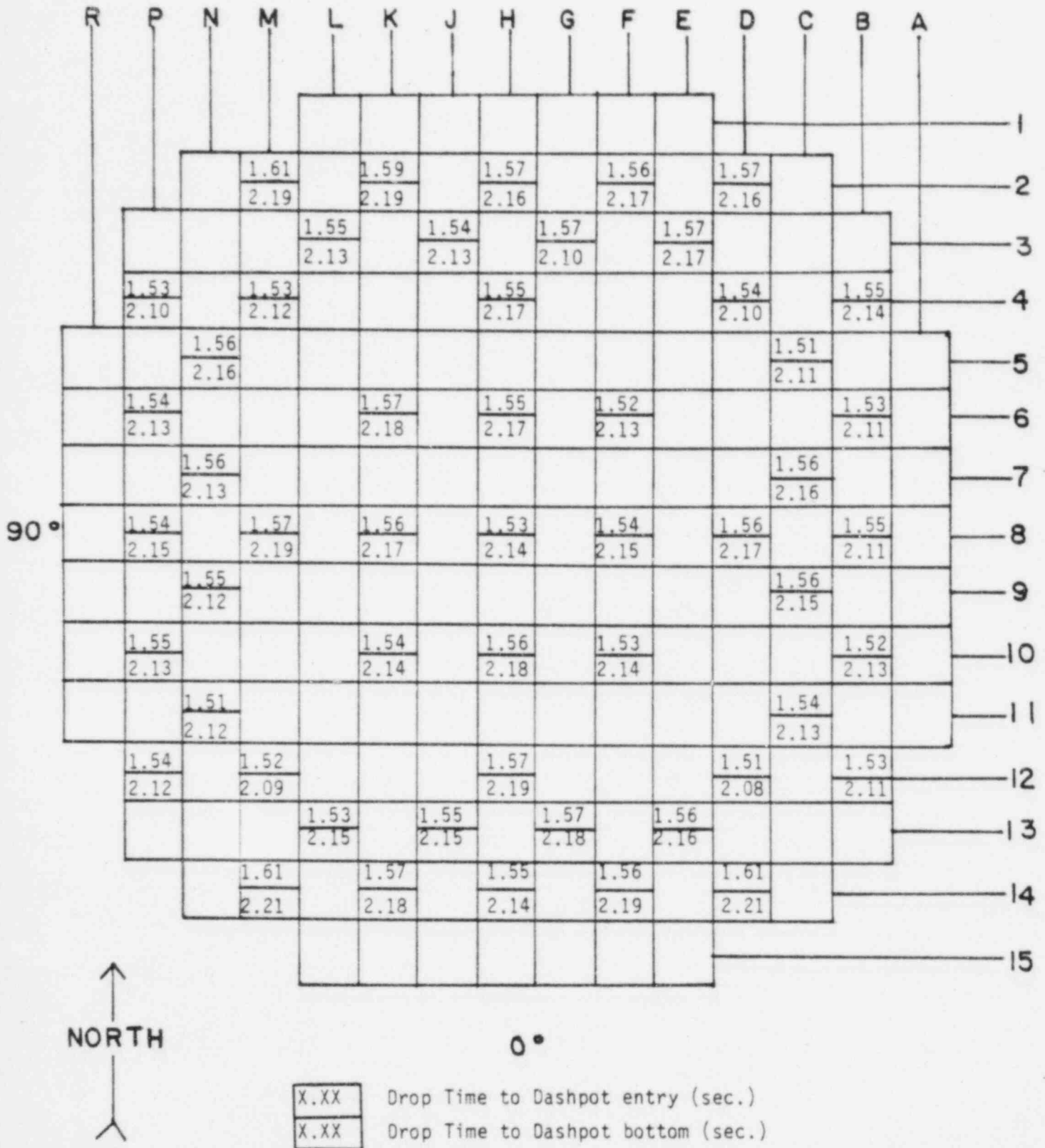


FIGURE 4.4-3

CATAWBA 1

4.5 ROD CONTROL SYSTEM ALIGNMENT TEST - TP/1/B/2600/05

Date(s) Performed: Aug. 16, 17, 18 and 22, 1984; Nov. 8 and 9, 1984;
and Nov. 30, 1984

I. PURPOSE

The purpose of the Rod Control System Alignment Test was to assure proper connection, identification and continuity of Rod Control System cabling to Control Rod Drive Mechanisms (CRDM's) while in the cold shutdown condition prior to initial operation of the CRDM's; and to adjust Rod Control system bank overlap setpoints and demonstrate proper system control and indication while in the hot standby condition, prior to initial criticality.

II. TEST METHOD

Proper connection identification and continuity of Rod control system cabling to the CRDM's was verified through continuity and resistance checks while the reactor was in cold shutdown, prior to operation of the CRDM's.

Bank overlap setpoint adjustment was performed by operating Control Banks A through D in the manual overlap mode while observing indications of rod position. Proper system control and indication was demonstrated by operating the rod control system in various manual modes while observing indications and alarms.

III. RESULTS

All CRDM coil stack resistance readings were within allowable limits. Results are given in Table 4.5-1. All insulation resistance measurements were acceptable ($> 10 \text{ M}\Omega$). Proper connection and identification of CRDM cabling was verified.

In the Hot Standby condition, the Digital Rod Position Indication (DRPI) display, the P/A Converter Display and the Bank Step Counters were demonstrated to be in agreement at 48 steps withdrawn and at full insertion for each bank. The bank overlap control operated as desired to withdraw the control banks in the sequence shown in Figure 4.5-1. Associated alarms were observed and verified to operate as designed.

IV. CORRECTIVE ACTIONS

In the initial run of the procedure at Cold shutdown, the insulation resistance requirements were not met for the coil stacks at Core Location P-4 and P-8. Investigation of the failure revealed that

the coils at P-4 had been wetted by boric acid which had trickled down the rod-travel housing from its vent plug which had leaked during Hot Functional Testing. The failures at P-8 were caused by the damaged P-4 coils as they share lift coil and moveable gripper coil returns in the power cabinet. The coil stacks at P-4 were replaced and P-4 and P-8 were retested. All Acceptance Criteria were met.

All of the CRDM coil stacks were replaced during a subsequent outage. The test was performed again. All acceptance criteria for the cold shutdown portion of this test were met. (The results given in Table 4.5-1 are from this retest).

During performance of the Hot Standby portion of the test, some misstepping occurred for mechanisms in Control Banks B and D and Shutdown Bank A. Testing was halted for about 6 hours while these mechanisms were "exercised". No misstepping occurred afterwards and all criteria were met.

ROD CONTROL SYSTEM ALIGNMENT TEST

CRDM RESISTANCE MEASUREMENT RESULTS

<u>BANK</u>	<u>CONTROL ROD CORE LOCATION</u>	<u>STATIONARY GRIPPER COIL RESISTANCE, Ω</u>	<u>LIFT COIL RESISTANCE, Ω</u>	<u>MOVEABLE GRIPPER COIL RESISTANCE, Ω</u>
Shutdown A	D-2	8.806	1.368	8.793
Shutdown A	B-4	8.737	1.362	8.719
Shutdown A	M-2	8.808	1.358	8.824
Shutdown A	B-12	8.863	1.343	8.806
Shutdown A	P-4	9.111	1.344	8.774
Shutdown A	D-14	8.969	1.343	8.813
Shutdown A	P-12	9.052	1.354	9.064
Shutdown A	M-14	8.888	1.374	9.123
Shutdown B	J-13	9.092	1.380	9.143
Shutdown B	N-9	9.074	1.394	8.995
Shutdown B	N-7	8.864	1.375	8.786
Shutdown B	G-13	8.934	1.358	8.912
Shutdown B	J-3	8.898	1.346	8.896
Shutdown B	C-9	8.849	1.371	8.843
Shutdown B	C-7	9.070	1.357	8.997
Shutdown B	G-3	8.966	1.346	8.804
Shutdown C	E-3	8.872	1.352	8.800
Shutdown C	C-11	8.784	1.353	8.774
Shutdown C	N-5	8.950	1.363	8.848
Shutdown C	L-13	8.885	1.366	9.012
Shutdown D	N-11	8.862	1.368	9.053
Shutdown D	E-13	8.919	1.351	8.880

NOTE: Acceptable resistance for gripper coils is $8.38 \leq R \leq 9.45$, where R is resistance in Ohms. For Lift Coil, Acceptable Resistance is $1.28 \leq R \leq 1.45$.

TABLE 4.5-1

ROD CONTROL SYSTEM ALIGNMENT TEST

CRDM RESISTANCE MEASUREMENT RESULTS

<u>BANK</u>	<u>CONTROL ROD CORE LOCATION</u>	<u>STATIONARY GRIPPER COIL RESISTANCE, Ω</u>	<u>LIFT COIL RESISTANCE, Ω</u>	<u>MOVEABLE GRIPPER COIL RESISTANCE, Ω</u>
Shutdown D	L-3	8.813	1.345	8.804
Shutdown D	C-5	8.995	1.340	8.895
Shutdown E	D-8	8.821	1.364	8.810
Shutdown E	H-4	8.924	1.354	8.802
Shutdown E	H-12	9.186	1.364	9.027
Shutdown E	M-8	8.949	1.353	8.951
Control A	H-10	9.128	1.356	8.908
Control A	K-8	8.996	1.351	8.887
Control A	H-6	8.878	1.355	8.963
Control A	F-8	8.872	1.366	8.893
Control B	B-6	8.890	1.371	8.871
Control B	B-10	8.853	1.354	8.851
Control B	F-2	8.780	1.347	8.749
Control B	K-2	9.160	1.358	8.859
Control B	F-14	8.953	1.355	8.827
Control B	P-6	8.960	1.373	9.126
Control B	K-14	9.099	1.354	8.954
Control B	P-10	8.772	1.373	8.990
Control C	P-8	9.225	1.399	9.008
Control C	H-14	9.241	1.370	8.924
Control C	K-10	9.086	1.361	9.077

NOTE: Acceptable resistance for gripper coils is $8.38 \leq R \leq 9.45$, where R is resistance in Ohms. For Lift Coil, Acceptable Resistance is $1.28 \leq R \leq 1.45$.

TABLE 4.5-1 (Cont'd.)

ROD CONTROL SYSTEM ALIGNMENT TEST
CRDM RESISTANCE MEASUREMENT RESULTS

<u>BANK</u>	<u>CONTROL ROD CORE LOCATION</u>	<u>STATIONARY GRIPPER COIL RESISTANCE, Ω</u>	<u>LIFT COIL RESISTANCE, Ω</u>	<u>MOVEABLE GRIPPER COIL RESISTANCE, Ω</u>
Control C	F-10	8.848	1.357	8.874
Control C	K-6	8.616	1.352	8.983
Control C	H-2	8.806	1.350	8.749
Control C	B-8	8.802	1.344	8.801
Control C	F-6	8.817	1.352	9.040
Control D	D-12	8.958	1.351	9.022
Control D	D-4	8.825	1.349	9.004
Control D	H-8	8.935	1.337	8.911
Control D	M-4	8.786	1.354	8.801
Control D	M-12	9.067	1.363	9.113

NOTE: Acceptable resistance for gripper coils is $8.38 \leq R \leq 9.45$, where R is resistance in Ohms. For Lift Coil, Acceptable Resistance is $1.28 \leq R \leq 1.45$.

TABLE 4.5-1 (Cont'd.)

ROD CONTROL SYSTEM ALIGNMENT TEST
CONTROL BANK OVERLAP TEST SEQUENCE

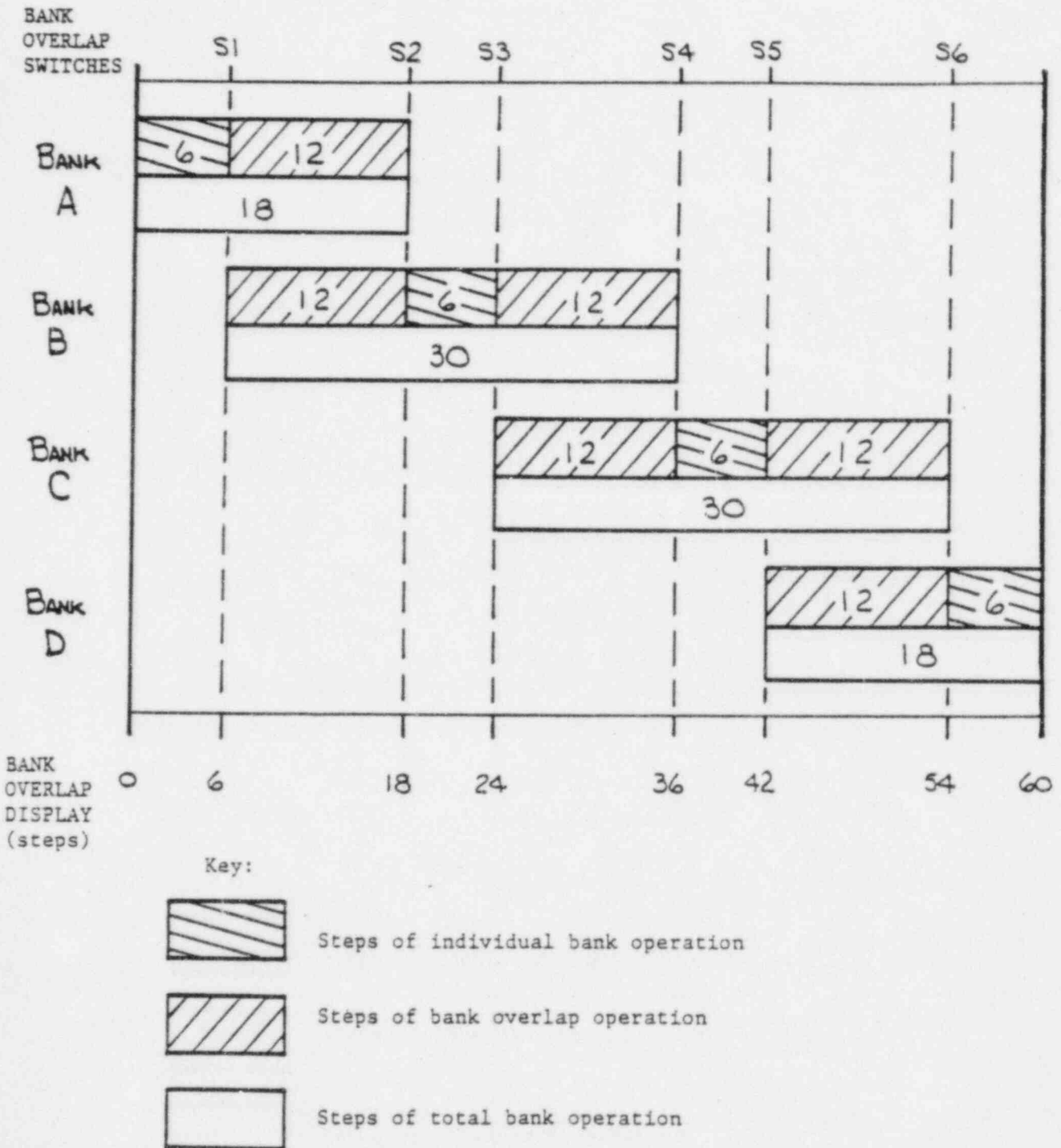


FIGURE 4.5-1

4.6 ROD CLUSTER CONTROL ASSEMBLY DROP TIME TEST - PHASE 1 - TP/1/B/2600/06

Date(s) Performed: 9/1/84 - 9/5/84, 11/24/84 - 11/26/84

I. PURPOSE

- A. To perform an operational check of each Full Length Control Rod Drive Mechanism (CRDM) with a Rod Cluster Control Assembly (RCCA) attached prior to initial use of the mechanisms under both cold and hot plant conditions.
- B. To verify proper slave cyclers timing for each Power Cabinet.
- C. To verify proper latching and releasing of the stationary and moving grippers of each CRDM.

II. METHOD

With the reactor in the cold shutdown condition, the timing of each slave cycler was checked. Each full length control rod drive mechanism (CRDM) was manually operated with a rod cluster control assembly (RCCA) attached to check the latching and unlatching of each gripper. Each CRDM was checked again with the reactor in the hot standby condition.

III. RESULTS

The following acceptance criteria were satisfied:

- A. Proper slave cycler timing was verified by lift, moveable gripper and stationary gripper coil current traces which demonstrated the proper timing sequence for rods out (up) and rods in (down) motion as required to ensure that rod misstepping does not occur.
- B. Prior to plant heatup beyond Mode 5, the lift, moveable gripper and stationary gripper coil current and sound signal traces for each CRDM demonstrated proper operation of the mechanism at cold plant conditions by conforming to the normal oscillograph traces in the Westinghouse Startup Document. These traces present the acceptable "Pull-in Time" (defined as the time from full current application to the coil to the gripper reaching its final position in the Withdraw Mode) and "Drop-out Time" (defined as the time from deenergization of the coil to the gripper reaching its final position in the Insert Mode) for each of the CRDM's.

Figures 4.6-1 through 4.6-6 summarize these results and compare them to the average behavior of the CRDM's in eight other Westinghouse plants.

- C. Prior to plant startup beyond Mode 3, the lift, moveable gripper and stationary gripper coil current and sound signal traces for each CRDM demonstrated proper operation of the mechanism at hot plant conditions by conforming to the normal oscillograph traces in the Westinghouse Startup Document (CNM-1201.13-038).

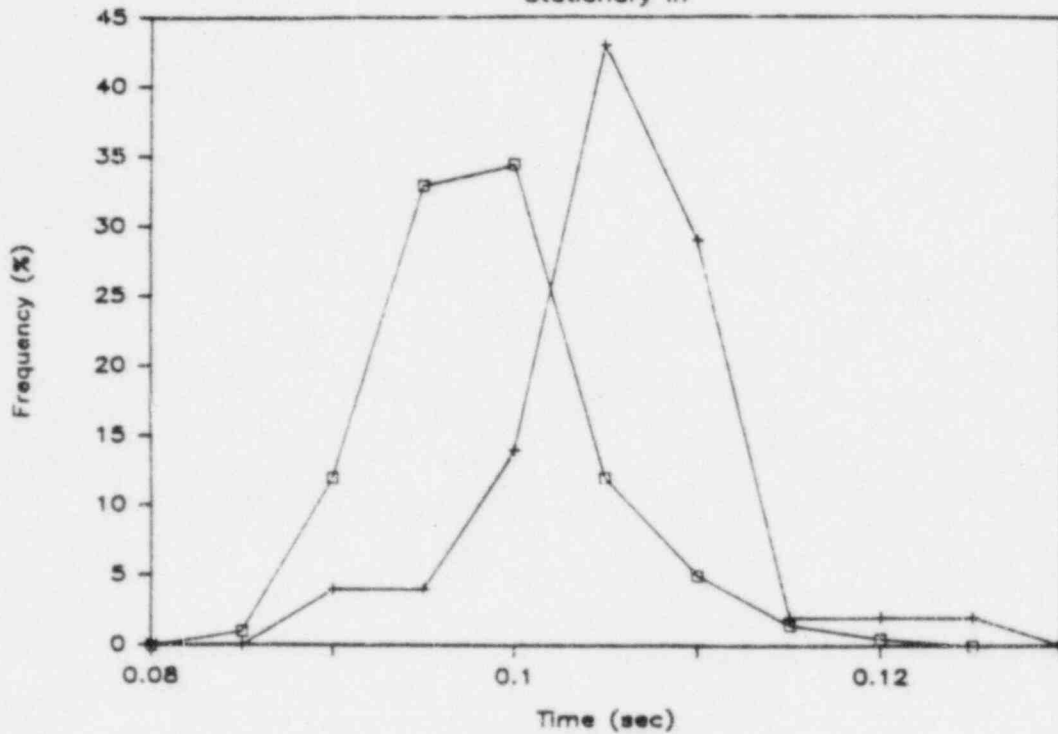
IV. CORRECTIVE ACTIONS

All CRDM timing traces demonstrated proper operation of the mechanisms with no evidence of misstepping. However, during the Mode 5 portion of the test, a problem was discovered with CRDMs at core locations J-3, J-13, H-10 and H-14 which resulted in reverse rod motion. Investigation revealed that the stationary gripper (SG) and moving gripper (MG) coil leadwires were rolled within the CRDMs. After correcting the wiring discrepancy, the four mechanisms were verified to be operating properly. Following the completion of the Mode 5 Tests, an unscheduled outage occurred during which all of the CRDM coil stacks except B-4 were removed and replaced with new coil stacks. However, proper operation of the new coil stacks was verified by the successful completion of the Mode 3 Tests. Based upon these test results, the IRE system is accepted for continued operation.

RCCA DROP TIME TEST
 PHASE 1
 Stationary Gripper Results

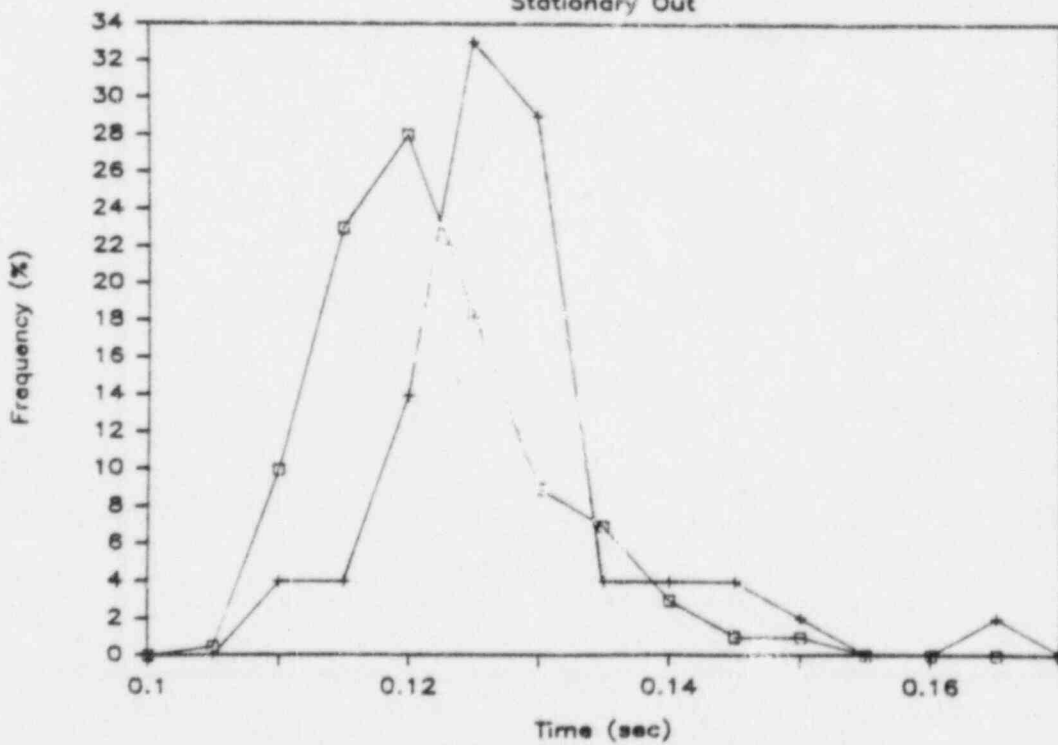
INSERT

Stationary In



INSERT

Stationary Out



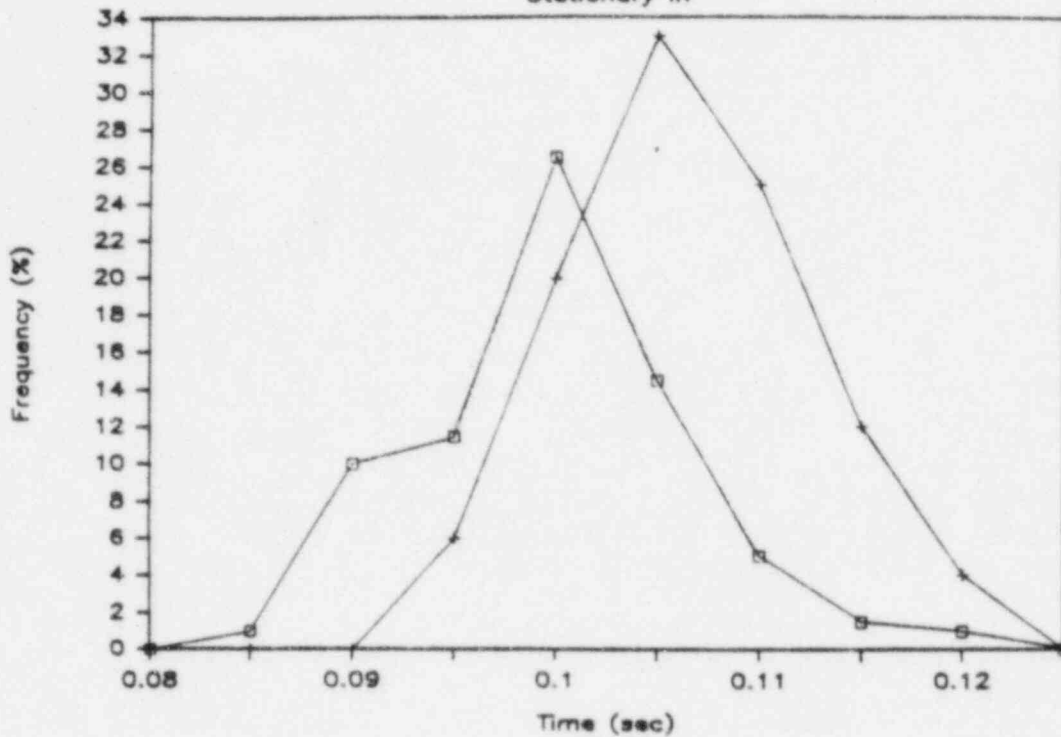
- Summarized Data from 8 other Plants
- + Catawba 1 Data

Figure 4.6-1

RCCA DROP TIME TEST
 PHASE 1
 Stationary Gripper Results

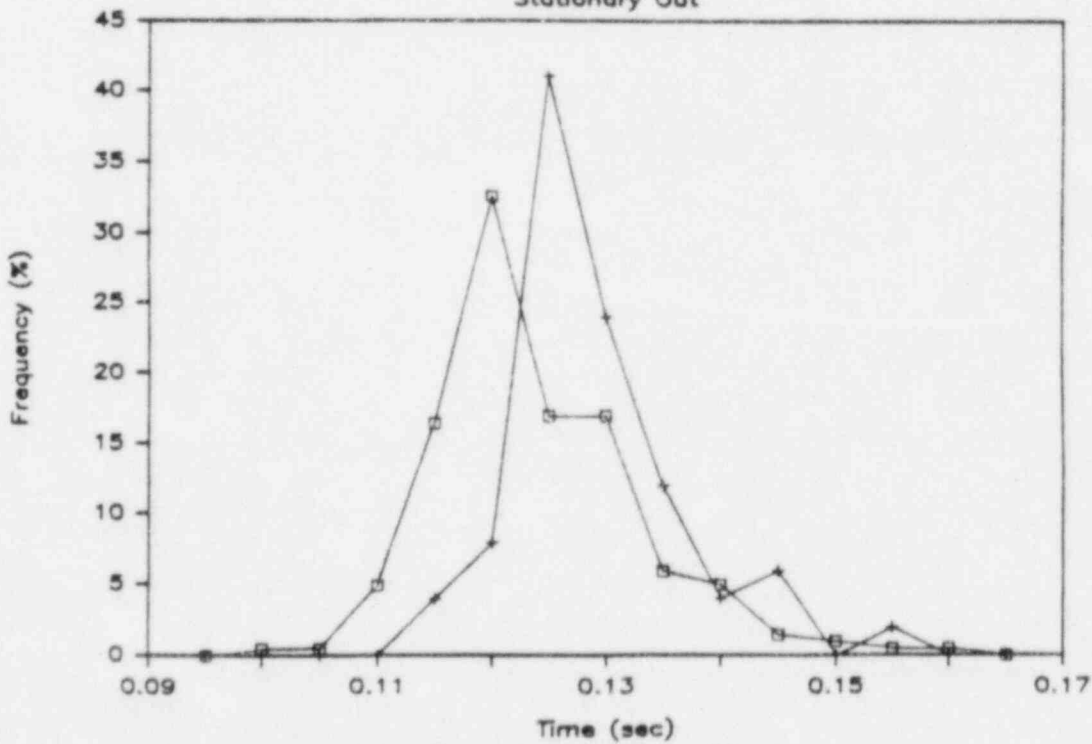
WITHDRAW

Stationary In



WITHDRAW

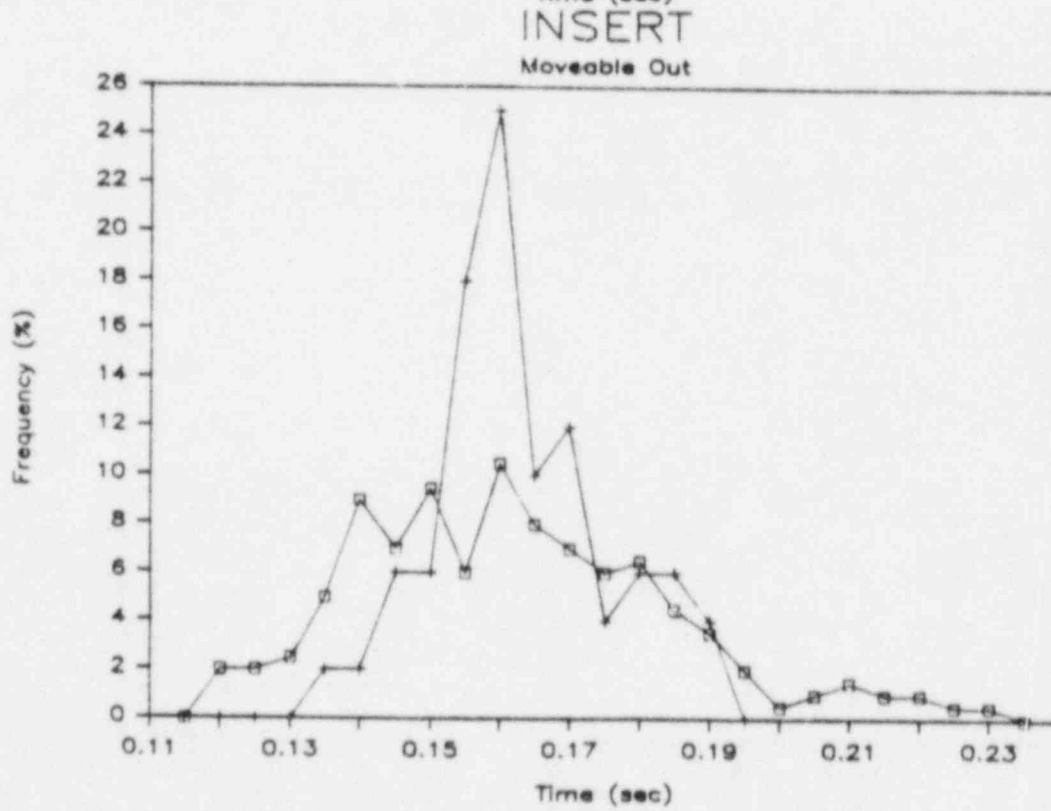
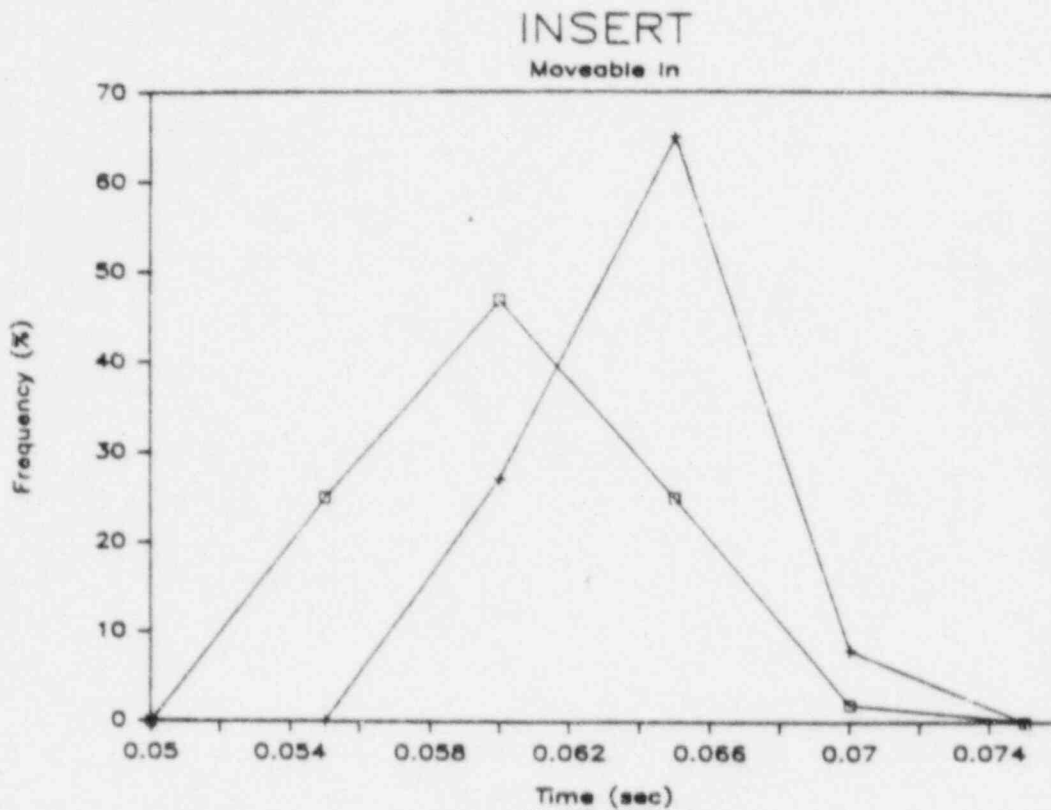
Stationary Out



- Summarized Data from 8 other Plants
- + Catawba 1 Data

Figure 4.6-2

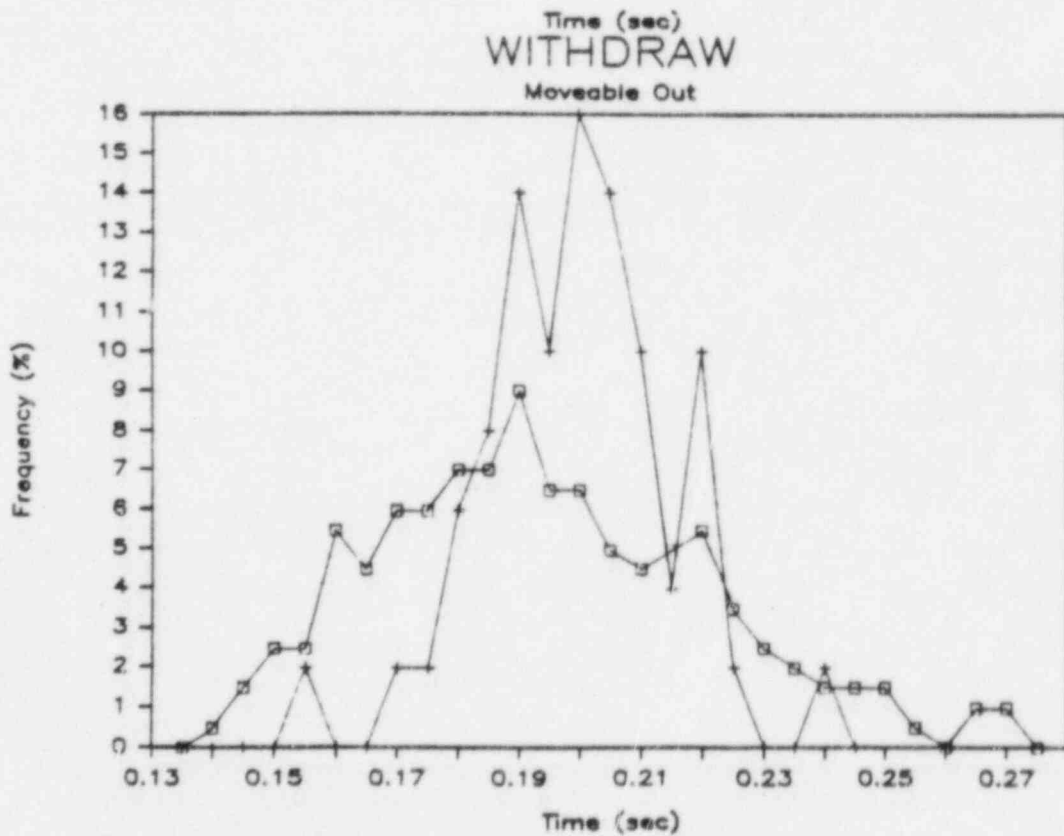
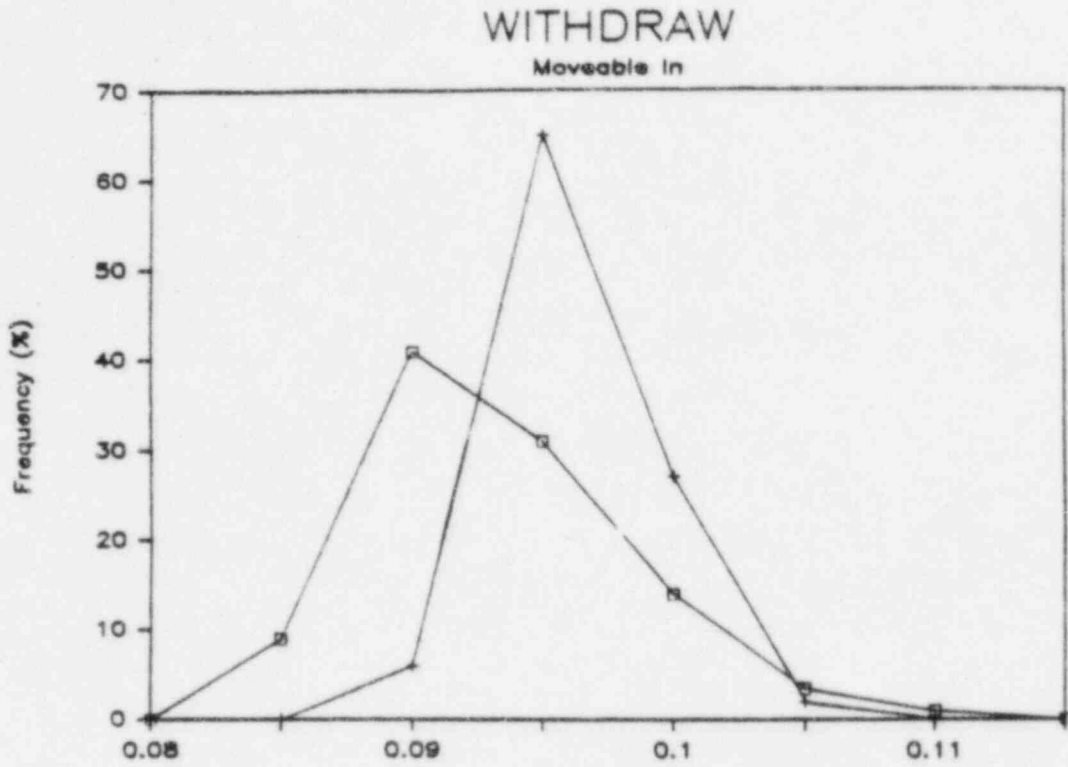
RCCA DROP TIME TEST
 PHASE 1
 Moveable Gripper Results



□ Summarized Data from 8 other Plants
 + Catawba 1 Data

Figure 4.6-3

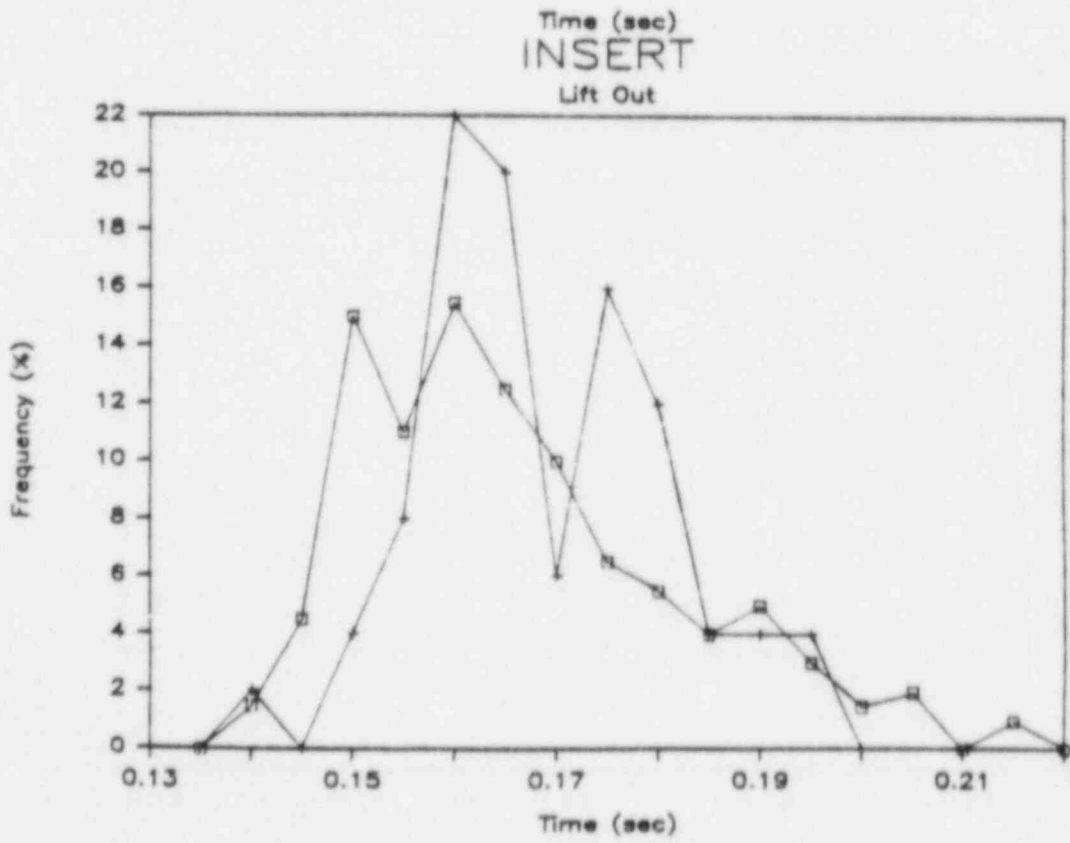
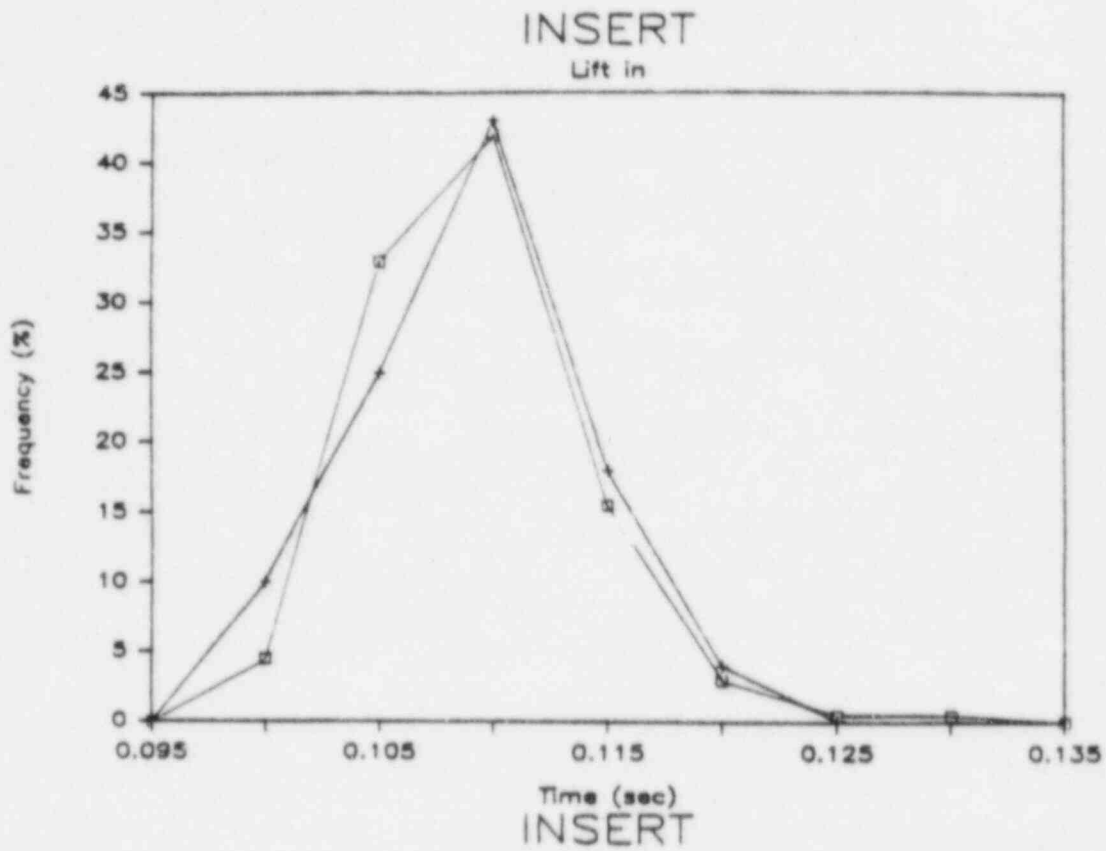
RCCA DROP TIME TEST
 PHASE 1
 Moveable Gripper Results



- Summarized Data from 8 other Plants
- + Catawba 1 Data

Figure 4.6-4

RCCA DROP TIME TEST
 PHASE I
 Lift Coil Results

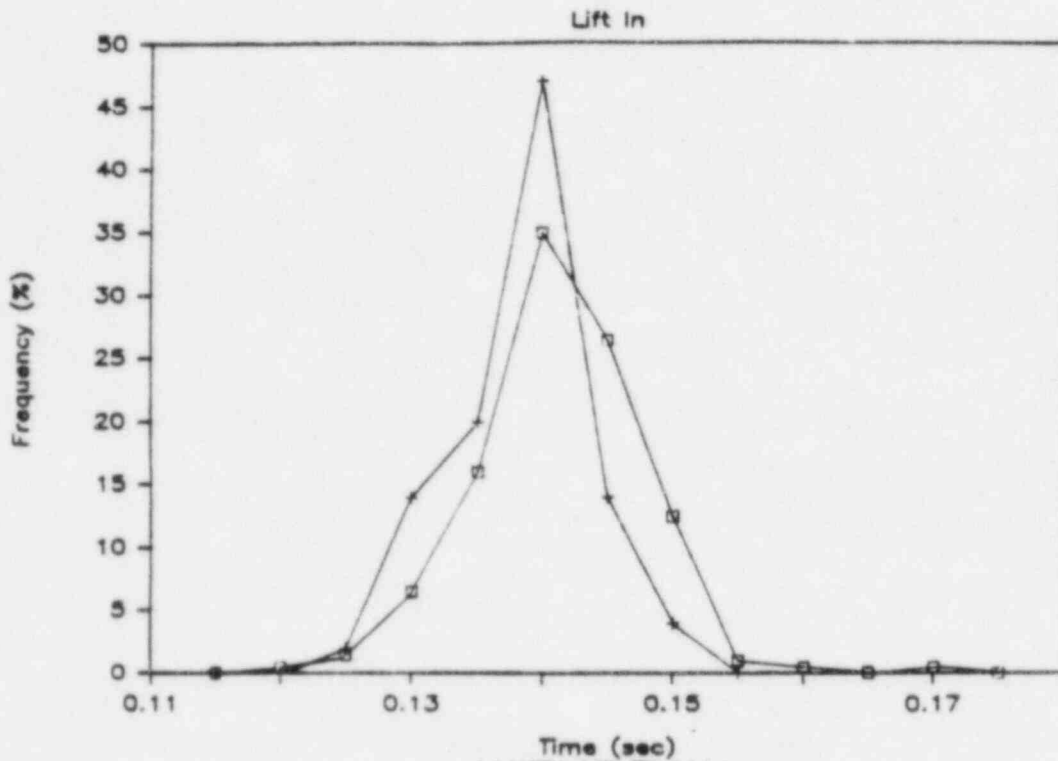


□ Summarized Data from 8 other Plants
 + Catawba 1 Data

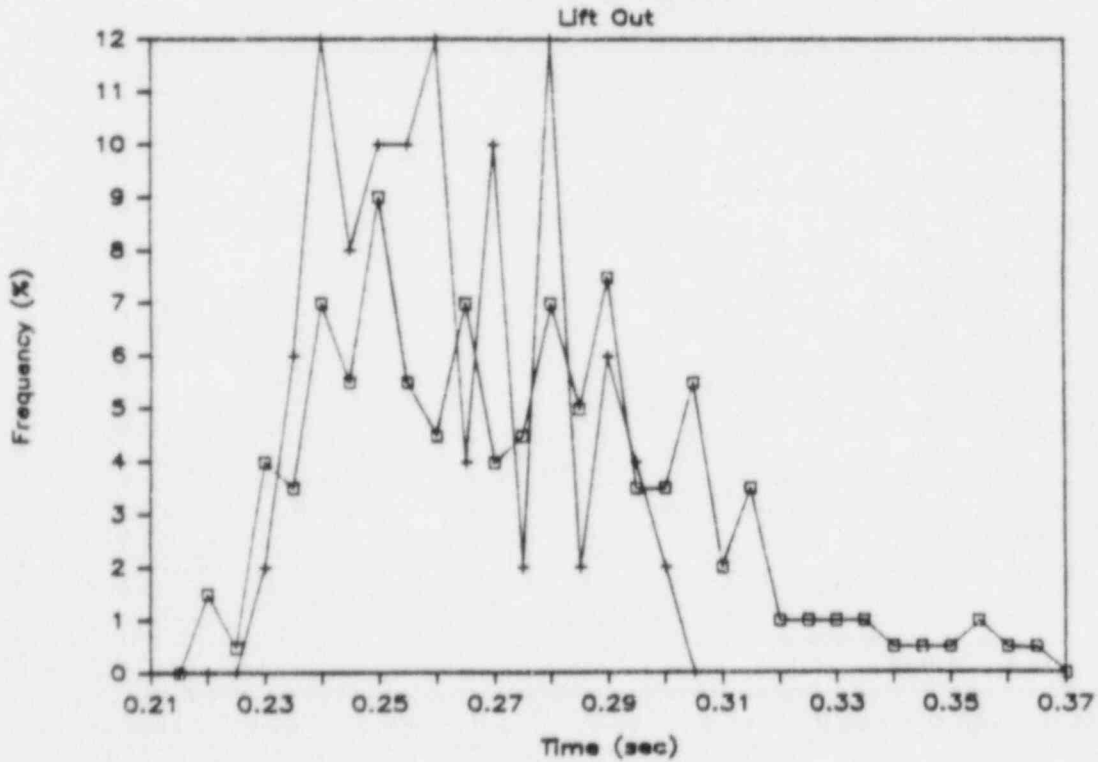
Figure 4.6-5

RCCA DROP TIME TEST
 PHASE 1
 Lift Coil Results

WITHDRAW



WITHDRAW



- Summarized Data from 8 other Plants
- + Catawba 1 Data

Figure 4.6-6

4.7 REACTOR COOLANT FLOW TEST - TP/1/A/2150/01

Date(s) Performed: 11/21/84 - 12/4/84

I. PURPOSE

The Reactor Coolant Flow Test was performed on the following dates:

<u>Date</u>	<u>Activity</u>
11/21/84, 11/22/84	Initial Flow Data Obtained
11/27/84	Initial Data Analyzed
11/30/84	Retest Data Obtained and Analyzed
12/3/84, 12/4/84	Westinghouse approval of results and Normalization of Control Room Flow Gauges
12/17/84	NRC petitioned for change to FSAR Test Abstract to accept measured results
1/14/85	NRC approval of FSAR Abstract Change

The purpose of this test was:

- A. To verify that the elbow tap indication of Reactor Coolant Flow is in excess of the Thermal Design Value of 387,600 gpm.
- B. To verify that the indicated flow is not in excess of the Mechanical Design value of 420,000 gpm.
- C. To ensure that the Control Room Flow Gauges are adjusted (based on this test's data) to indicate $100\% \pm 2\%$ with all Reactor Coolant pumps operating.
- D. To ensure that the adjusted Control Room Flow Gauges indicate $0\% \pm 2\%$ when all pumps are off.

II. METHOD

The flow rates for each Reactor Coolant Loop were derived by first obtaining the output voltage (0-10 volts) from the Elbow Tap differential pressure transmitters with Digital Voltmeters at the 7300 Process Cabinets. These voltages were converted to equivalent differential pressures (inches H₂O). A Westinghouse supplied curve was then applied to convert these differential pressures to flow rates (in gpm).

The summation of the four Reactor Coolant Loop flows was used to verify that the flow rate acceptance criteria were met. The differential pressure data obtained during the test was used to assist instrumentation personnel in normalizing the Control Room Flow Gauges to 100% (at full flow) and 0% (at no flow). Calculation and input of the "K" constants to the OAC to allow conversion of direct Elbow Tap d/p to units of "Millions lb_m per Hour" was also performed per the test data.

III. RESULTS

Initially the flow rate acceptance criteria was met. A summary of the four loops' data is as follows:

Loop	Average Transmitter Voltage	Average Zero Shift Voltage	Corrected Voltage	Flow Rate (GPM)
A	8.814	+0.897	7.917	101,200
B	8.132	+1.021	7.111	96,067
C	9.153	+0.891	8.262	103,267
D	8.214	+1.051	7.163	96,233

Total Flow = 396,767 GPM

This value for total flow was within the original acceptance criteria of > 387,600 GPM and < 420,000 GPM. The validity of the values of "Zero Shift Voltage" was questioned, however, by Station instrumentation personnel. The method by which this Zero Shift had first been measured simply involved opening the manifold equalization valve on the d/p cell and measuring the corresponding "zero" voltage from the transmitter. Instrumentation personnel contended that the voltages obtained by this method were far in excess of reasonable values.

The Zero Shift voltages were therefore remeasured, this time with all Reactor Coolant Pumps off, to obtain the true Zero Shift of the flow transmitters. Subsequent results are as follows:

Loop	Average Transmitter Voltage	Average Zero Shift Voltage	Corrected Voltage	Flow Rate (GPM)
A	8.814	-0.115	8.929	107,000
B	8.132	-0.106	8.238	103,333
C	9.153	-0.082	9.235	109,333
D	8.214	-0.126	8.340	103,666

Total Flow = 423,332 GPM

The slightly negative values of Zero Shift indicate that a vacuum exists on the d/p cell for each loop under no flow conditions due to the peculiar tubing arrangement from the Elbow Taps to the cells themselves.

Once the transmitter voltages were corrected with these new Zero Shifts a value of total flow in excess of the Mechanical Design Limit originally specified by the acceptance criteria was obtained. This necessitated a request to Westinghouse personnel for an evaluation of these results for acceptability.

The resulting Westinghouse evaluation concluded that due to the fact that the Elbow Tap Measurement technique incurs errors of $\pm 10\%$ the Mechanical Design Flow Limit was not, in fact, exceeded. A change was made to the procedure allowing an upper limit on the flow rate acceptance criteria of 462,000 GPM (10% greater than 420,000 GPM). Hence, the measured flow rate of 423,332 GPM was well within the acceptance band.

Following the recalibration of the flow transmitters and the normalization of the Control Room Flow Gauges the acceptance criteria for "full flow" ($100\% \pm 2\%$) and "no flow" ($0\% \pm 2\%$) Control Room indication were met. The results of these surveys are as follows:

<u>Gauge</u>	<u>4 Pump Operation Indication</u>	<u>No Pump Operation Indication</u>
1NCP5000	100%	0%
1NCP5010	100%	0%
1NCP5020	100%	0%
1NCP5030	100%	0%
1NCP5040	99%	0%
1NCP5050	99.5%	0%
1NCP5060	100%	0%
1NCP5070	100%	0%
1NCP5080	100%	0%
1NCP5090	100%	0%
1NCP5100	100.5%	0%
1NCP5110	99.5%	0%

IV. CORRECTIVE ACTIONS

The slight vacuum which is drawn on the Reactor Coolant Loop d/p cells when they are supposedly "Zeroed" is the result of somewhat deficient tubing from the Loop Elbow Taps. This deficiency does not impact the normalization of the flow gauges either prior to Startup or during Power Escalation when the Reactor Coolant Flow is calculated by precision secondary side calorimetric measurement. Retubing shall therefore not be undertaken.

Some leakage was observed on several of the manifold and d/p cell fittings. A Work Request was generated to have these leaks repaired coincident with the recalibration of the flow transmitters (for the normalization of the Control Room Gauges).

Three occurrences caused delays in the completion of this test. The first was the reacquisition of the Zero Shift data and the reanalysis which was performed with it. This caused a delay of three days until proper system conditions (all Reactor Coolant Pumps off) were present.

The second was a wait of three days for Westinghouse to approve the flow rate eventually derived.

The final one was a wait of approximately six weeks for the NRC to review and approve the measured results and permit a change to the FSAR to incorporate them.

4.8 REACTOR COOLANT FLOW COASTDOWN TEST - TP/1/A/2150/02

Date(s) Performed: 11/28/84 - 12/3/84

I. PURPOSE

The Reactor Coolant Flow Coastdown Test was performed on the following days:

<u>DATE</u>	<u>ACTIVITY</u>
11/28/84	Signed off prerequisites and set chart recorders
11/29/84	Obtained all test data
11/30/84	Analyzed test data
12/03/84	Received approval by Westinghouse for results not meeting Acceptance Criteria
1/14/84	Received approval by NRC to revise FSAR Chapters 14 and 15

The objectives of the Reactor Coolant Flow Coastdown Test were as follows:

A. For the 1 out of 4 Pump Coastdown to verify:

1. That the measured Low Flow Delay Time is ≤ 2.43 sec.
2. That the measured Undervoltage Trip Delay Time is ≤ 1.5 sec.
3. That the measured Underfrequency Trip Delay Time is ≤ 0.6 sec.
4. That all points on the Faulted Loop Flow Coastdown Curve are above the corresponding points on the predicted curve on Figure 15.3.1-1 of the FSAR.
5. That all points on the Total Core Flow Coastdown Curve are above the corresponding points on the predicted curve on Figure 15.3.1-1 of the FSAR.

B. For the 4 out of 4 Pump Coastdown to verify:

1. That the slope of the Inverse Measured Flow Coastdown Curve is ≤ 0.0851 , between 3 and 10 seconds after start of coastdown.
2. That all points on the Total Core Flow Coastdown Curve are above the corresponding points on the predicted curve on Figure 15.3.2-1 of the FSAR.

II. METHOD

A. 1 out of 4 Coastdown

A Gould chart recorder was wired to the 4 Process Cabinet cards carrying the Channel 1 Elbow Tap d/p transmitter signals to trend all 4 loops' flow rates.

4 of the Gould recorder's channels were wired to contacts in the Reactor Coolant Pump Power Monitor Panels to provide indication of the pump breaker positions.

A Visicorder was wired to the Power Monitor Panel for the pump to be tripped (D Pump) and the 3 Process Cabinet cards carrying the signals of all 3 Low Flow Bistables. In addition, A and B Reactor Trip Breaker Switchgears were wired to the Visicorder to indicate the reactor trip (due to low flow).

Once these test recorders were set up the P-8 permissive was simulated (indicating reactor power \geq 48%) by defeating the Source Range Detector Block in the SSPS Cabinets and feeding a test signal of 48% power to two of the four Power Range Detectors.

With this permissive simulated, a reactor trip is attainable via the tripping of one Reactor Coolant Pump.

Reactor Coolant Pump D was manually tripped by Control Room pushbutton to initiate the test. The Visicorder traces were analyzed to measure the trip delay times and the Gould Recorder traces were analyzed to create Flow coastdown Curves for comparison to those in the FSAR.

B. 4 out of 4 Coastdown

For this section of the test, the Gould Chart Recorder was set up the same way as it was for the 1 out of 4 Coastdown. The Visicorder was not used for this section.

The test was initiated by simultaneously tripping all 4 Reactor Coolant Pumps. This was accomplished by first simulating a P-7 permissive (\geq 10%) in the same manner the P-8 permissive was obtained in the 1 out of 4 test and pulling the Underfrequency circuit fuse in one of the Reactor Coolant Pump Power Monitor Panels and then turning the key in another to the "Underfrequency Test" position. This created a loss of all 4 NC Pumps due to the simulated loss of 2 pumps on underfrequency.

The traces for Reactor Coolant Flow on the Gould Recorder were analyzed to obtain a Total Core Flow Coastdown Curve for comparison with the predicted curve in the FSAR.

III. RESULTS

- A. For the 1 out of 4 Pump Coastdown, the measured delay response time results compared with the respective acceptance criteria as follows:

<u>PARAMETER</u>	<u>MEASURED VALUE</u>	<u>ACCEPTANCE CRITERIA</u>
Low Flow Delay Time	2.26 sec	≤ 2.43 sec
Undervoltage Trip Delay Time	0.86 sec	≤ 1.5 sec
Undervoltage Trip Delay Time	0.36 sec	≤ 0.6 sec

The Flow Coastdown Curve (total core and faulted loop) data measured (at 1 sec intervals) was the following:

<u>TIME (Sec)</u>	<u>TOTAL CORE (Measured)</u>	<u>TOTAL CORE (FSAR)</u>	<u>FAULTED LOOP (Measured)</u>	<u>FAULTED LOOP (FSAR)</u>
0	1.0	1.0	1.0	1.0
0.65	0.9935	0.9914	0.9309	0.9543
1.65	0.9708	0.9728	0.8485	0.8529
2.65	0.9551	0.9564	0.7659	0.7626
3.65	0.9376	0.9406	0.7024	0.6806
4.65	0.9260	0.9271	0.6429	0.6091
5.65	0.9114	0.9151	0.5773	0.5428
6.65	0.9059	0.9037	0.5416	0.4839
7.65	0.8946	0.8917	0.4899	0.4275
8.65	0.8783	0.8742	0.4320	0.3608
9.65	0.8737	0.8570	0.4000	0.2927

* These results are presented graphically on Figures 4.8-1 and 4.8-2.

For the most part, these curves passed the acceptance criteria. There were, however, a few points on each which were not acceptable. This failure was conveyed to Westinghouse for resolution.

- B. For the 4 out of 4 Pump Coastdown, the slope of the Inverse Measured Flow Curve proved to be too severe to pass the acceptance criteria of ≤ 0.0851 between 3 and 10 seconds into the coastdown (0.0981 was the measured slope). This failure is linked directly to the failure of the 4 out of 4 total core flow coastdown curve to meet its acceptance criteria. The results of this curve compared to its acceptance curve are as follows:

<u>TIME</u> <u>(Sec)</u>	<u>TOTAL CORE FLOW</u> <u>(Measured)</u>	<u>TOTAL CORE FLOW</u> <u>(FSAR)</u>
0	1.0	1.0
0.65	0.9284	0.9623
1.65	0.8556	0.8892
2.65	0.7933	0.8290
3.65	0.7376	0.7747
4.65	0.6900	0.7290
5.65	0.6436	0.6904
6.65	0.6080	0.6520
7.65	0.5732	0.6166
8.65	0.5395	0.5844
9.65	0.5134	0.5565

* These results are presented graphically on Figure 4.8-3.

The 4 out of 4 Coastdown data, since it is more limiting case from a safety analysis standpoint, was transmitted to Westinghouse for re-analysis. Westinghouse approved this data on the basis that the Design Limit DNBR was not reached during the complete loss of flow test. This analysis also made the partial loss of flow data acceptable.

The coastdown capability of the Reactor Coolant System was verified to be acceptable per Westinghouse analysis of the data obtained by this test. The Low Flow Delay time (or time from loss of flow until control rods are released to fall into the core) was measured and verified to be acceptable.

IV. CORRECTIVE ACTIONS

The only component which did not perform as expected was the contact device in the Reactor Coolant Pump Power Monitor Panel which was used to monitor the opening of the pump breakers. It was discovered that this device did not indicate an open breaker until the Undervoltage Delay time through the Reactor Protection System had elapsed. This proved to be no problem for the data analysis, though as the Undervoltage Delay was simply included in the calculation of the Low Flow Delay time. Also, the starting point of the coastdown for both the partial and total loss of flow cases was set back the amount of time of the Undervoltage Delay.

Acceptance Criteria as discussed in FSAR Chapter 14 was not initially met. Following re-analysis by Westinghouse, FSAR Chapter 14 abstract and Chapter 15 Section 15.3.2.2 were submitted to NRR for review and approval on 12/17/84. These changes were approved on 1/14/85.

NC FLOW COASTDOWN TEST

1/4 Coasting Down
Total Core

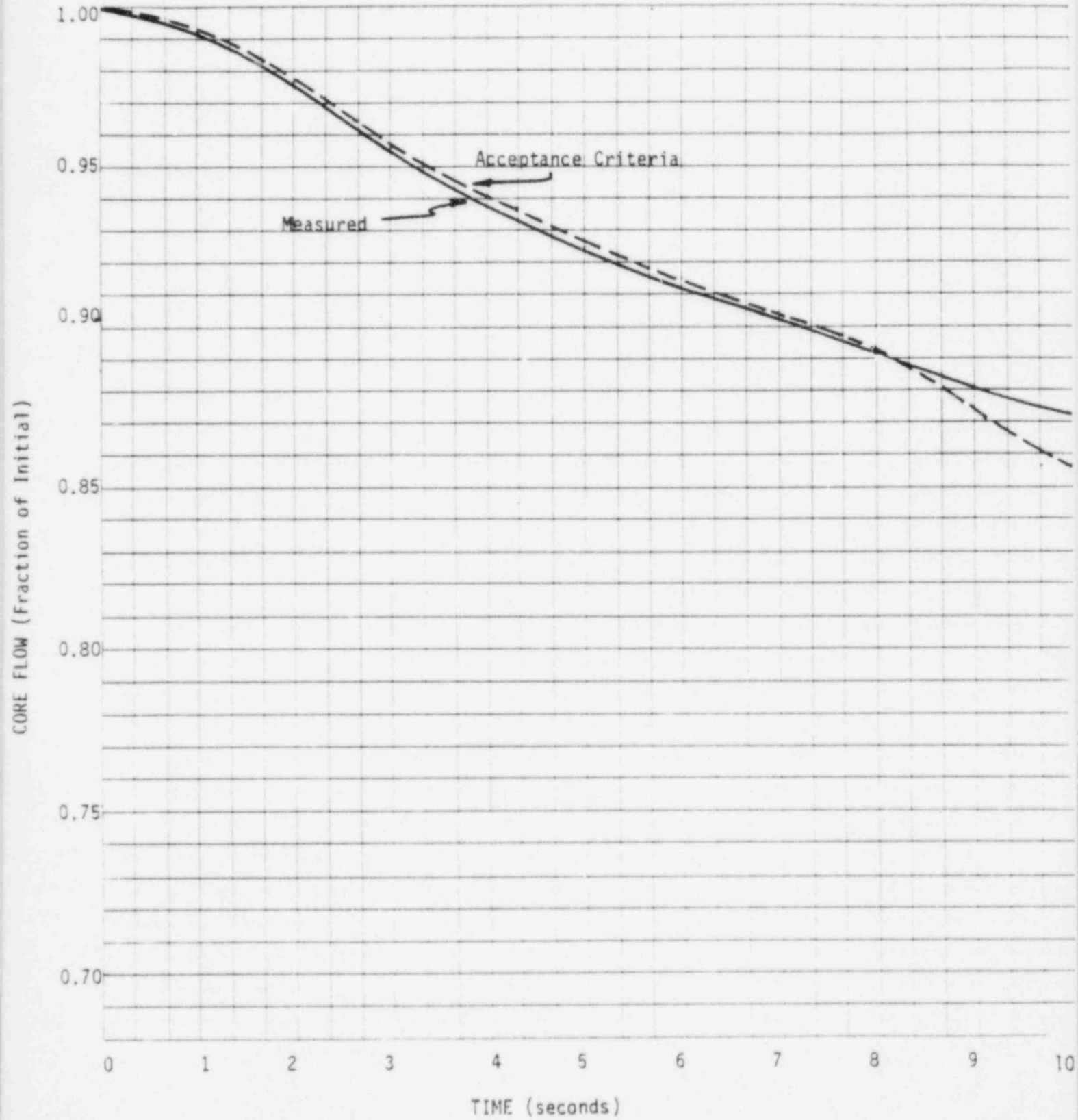


Figure 4.8-1

NC FLOW COASTDOWN TEST

1/4 Coasting Down
Faulted Loop

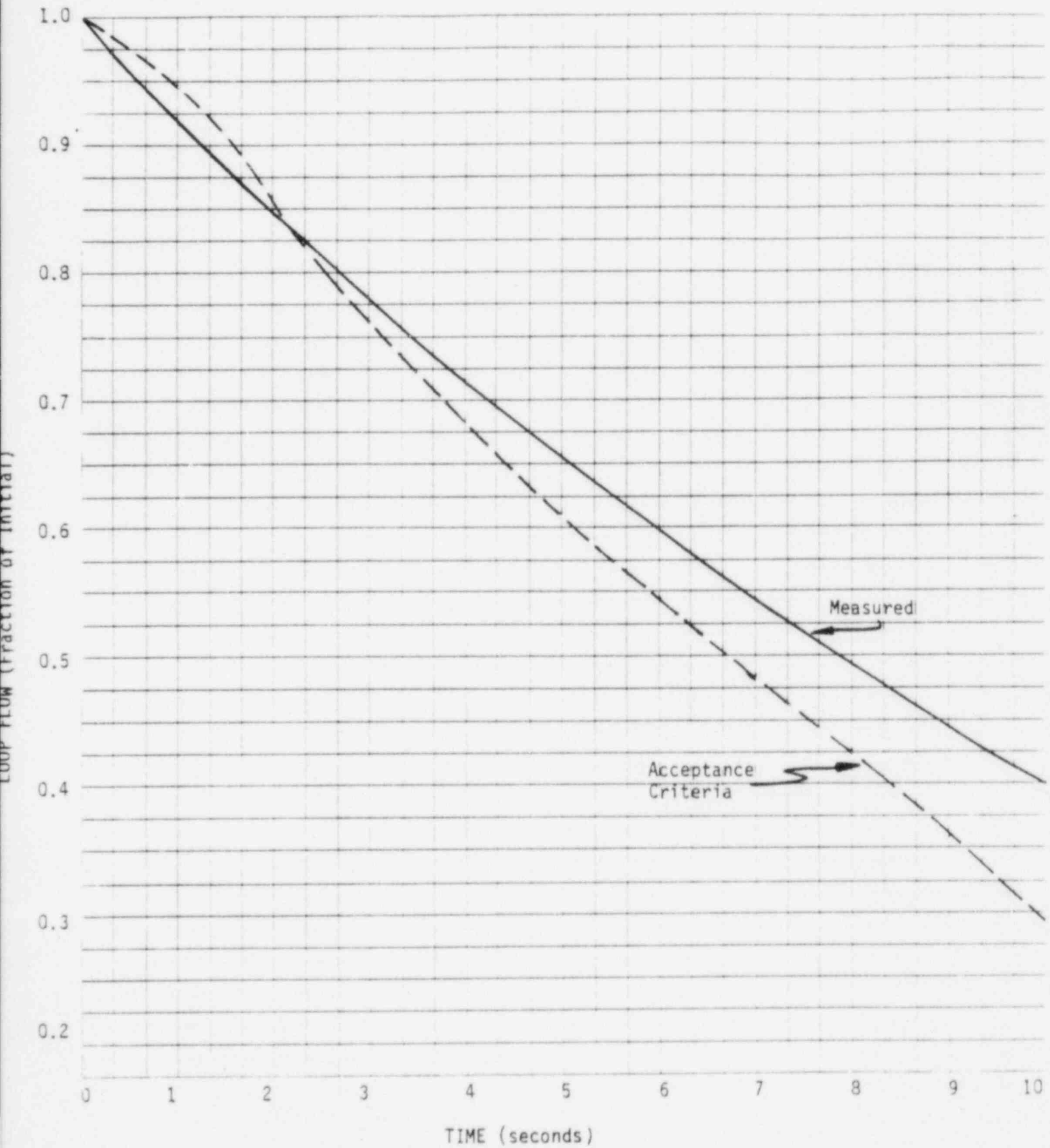


Figure 4.8.2

NC FLOW COASTDOWN TEST
4/4 Coasting Down

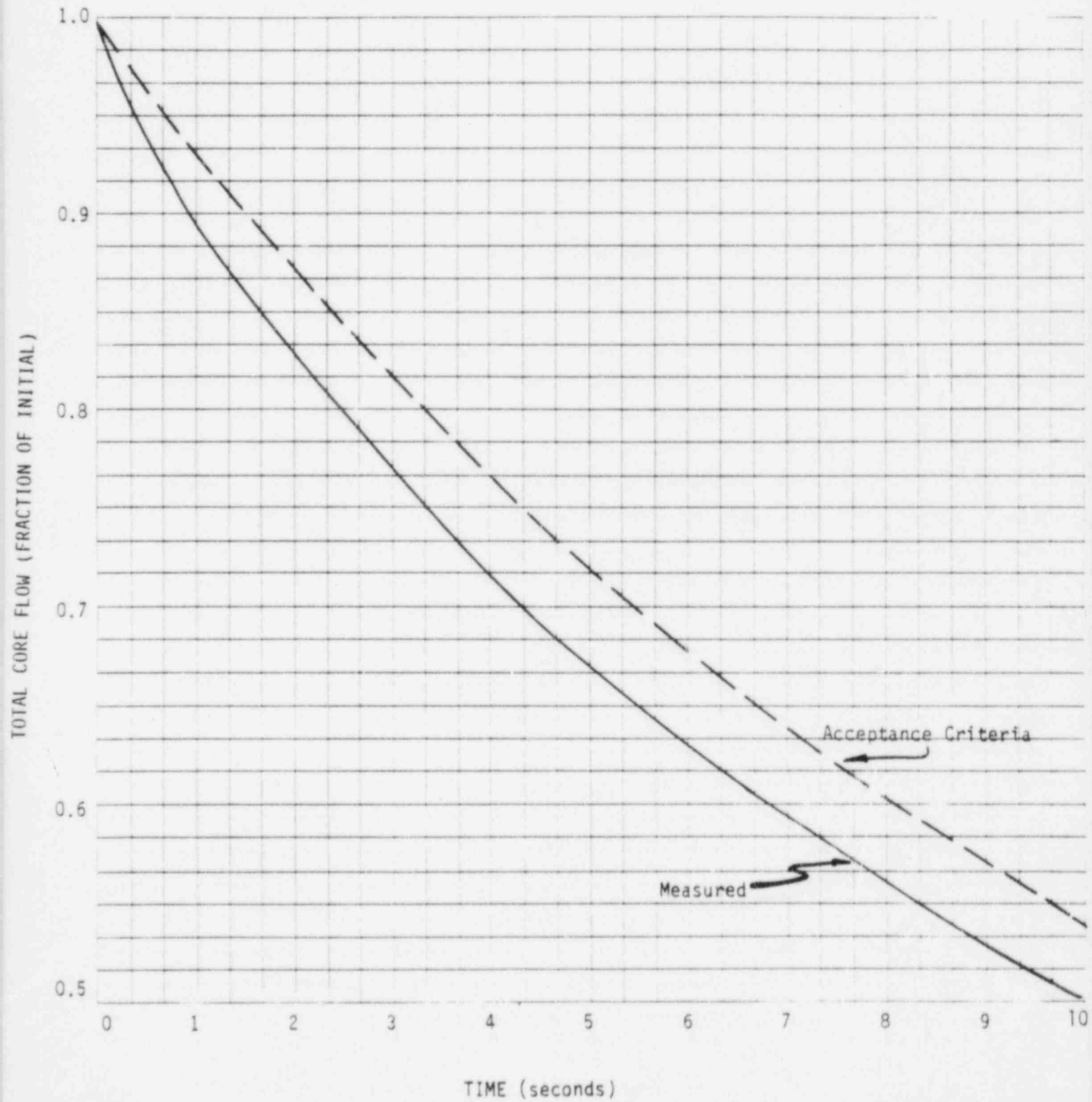


Figure 4.8.3

4.9 RTD BYPASS FLOW VERIFICATION - TP/1/A/2600/08

Date(s) Performed: 6/8/83-6/9/83; 11/21/84-11/22/84; 12/2/84-12/3/84

I. PURPOSE

The objectives of the RTD Bypass Flow Verification were:

- A. To determine the necessary flowrate to achieve the design objective for NC coolant transport time in each RTD bypass loop.
- B. To ensure the flowrate in each RTD bypass loop and NC coolant transport line is acceptable.
- C. To establish the low flow alarm setpoint for the total bypass flow in each loop and to verify for each loop that the alarm activates at the assigned setpoint.

II. METHOD

The first part of the test was conducted prior to NC piping insulation. The RTD bypass lines were measured from the bypass connections on the respective hot and cold legs to the centerline of the last RTD on the RTD manifold. For each of the three 1" lines on each hot leg, the values for each run of pipe are summed yielding a single value of 1" measured distance.

Once the measurements are complete, the values are used to calculate the necessary flowrate to achieve a coolant transport time of 1.0 second, with separate calculations for each hot and cold leg for each loop.

After fuel loading, flowrates were recorded for the bypass lines and verified to be greater than the flowrates calculated earlier. Low flow alarm setpoints were determined and checked by throttling the Hot Leg or Cold Leg isolation valve for each loop.

III. RESULTS

The results of the piping and flow rate measurements are shown in Table 4.9-1. All flow rates were greater than the flow rate required for a transport time of one second. Initially, Loop A cold leg flowrate was not sufficient to meet the minimum acceptable flow, but the orifice in the line was rebored to allow adequate flow. Lo-flow alarm setpoints were adjusted to 90% of measured flow. Alarms were tested and all were verified to actuate at $90 \pm 2\%$ of measured flow.

IV. CORRECTIVE ACTION

In the initial flow measurement, performed on 11/21 through 11/22/84, Loop A Cold Leg failed the flow rate criterion. The orifice in the Loop A Cold Leg to RTD Manifold line was bored to increase flow rate. A retest was subsequently performed on 12/2 through 12/3/84 to ensure acceptable flow in Loop A.

RTD BYPASS FLOW VERIFICATION
RESULTS OF RTD BYPASS FLOW VERIFICATION

	TOTAL VOLUME (ft ³)	MINIMUM FLOWRATE	MEASURED FLOWRATE	CALCULATED ALARM SETPOINT	ALARM ACTUATION SETPOINT
Loop A	N/A	N/A	297.0	267 ± 5.9	263
Hot Leg	0.1692	75.95	133.5	N/A	N/A
Cold Leg	0.2232	100.2	163.5	N/A	N/A
Loop B	N/A	N/A	300.	270 ± 6.0	265
Hot Leg	0.1963	88.12	118.2	N/A	N/A
Cold Leg	0.2199	98.70	181.8	N/A	N/A
Loop C	N/A	N/A	300.	270 ± 6.0	268
Hot Leg	0.1810	81.24	144.7	N/A	N/A
Cold Leg	0.2890	129.7	155.3	N/A	N/A
Loop D	N/A	N/A	290.0	261 ± 5.8	264
Hot Leg	0.1672	75.03	107.7	N/A	N/A
Cold Leg	0.3362	150.9	182.3	N/A	N/A

NOTE: All flow rates are given in Gallons per Minute.

4.10 PRESSURIZER FUNCTIONAL TEST - TP/1/A/2150/13

Date(s) Performed: 11/22/84 - 11/24/84

I. PURPOSE

The objectives of the Pressurizer Functional Test were:

- A. To verify and adjust if necessary, the desired position of the continuous spray flow valves (1NC28 and 1NC30) such that a $< 125^{\circ}\text{F } \Delta\text{T}$ exists between the pressurizer spray and steam temperatures.
- B. To use the data gathered above and provide IAE with setpoints for the Pressurizer Spray Line Low Temperature Alarms (part of the 7300 Process Control System).
- C. To demonstrate the effectiveness of the pressurizer heaters by testing their ability to pressurize the NC system.
- D. To demonstrate the effectiveness of the pressurizer spray system by testing their ability to depressurize the NC system under both normal and abnormal conditions (e.g. less than 4 NC pumps operating, only one operable spray valve, etc.).
- E. To verify proper response time and low pressure interlock operation of the pressurizer PORV's.

II. METHOD

With the pressurizer at hot, no load conditions, the continuous pressurizer spray flow valves 1NC28 and 1NC30 were closed while the pressurizer spray control valves (1NC27 and 1NC29) were fully closed. Data was gathered after spray line temperatures had stabilized. This allowed verification that a less than $125^{\circ}\text{F } \Delta\text{T}$ existed between the spray and pressurizer steam temperature. The data was used to calculate the setpoints for the Pressurizer Spray Line Low Temperature Alarms in the 7300 Process Control System. A Work Request was issued to IAE who performed the alarm calibration.

The pressurizer heaters' ability to pressurize the NC system was verified by closing all spray valves and manually energizing all the heaters. Pressure versus time was recorded and plotted for comparison to Westinghouse typical response data. The ability of the spray control valves to depressurize the NC system was verified using various pump/spray valve combinations representing both normal and abnormal situations. In the case with all pumps on and both spray valves open, pressure versus time data was recorded and plotted for comparison to a Westinghouse curve showing the typical expected response.

Each of the pressurizer PORV's were response time tested from the closed to open positions. In each case, the pressure was allowed to decrease while the PORV was open until the Lo Pressure PORV Interlock automatically closed the PORV. Data was collected to determine response time and the pressure at which the interlock actuated.

XII. RESULTS

Spray line temperatures were measured and compared to the pressurizer steam temperature with both 1NC28 and 1NC30 fully closed. ΔT 's of 97.1 and 104.6°F respectively were calculated for each spray line. This met the Acceptance Criterion of < 125°F. The Pressurizer Spray Line Low Temperature Alarms were set, using the above results, to 520.3°F and 512.8°F for the two spray lines. This is 35°F below the equilibrium temperature and meets the Acceptance Criterion of 35 ± 15°F.

The pressurization response due to the heaters matched the Westinghouse expected response curve well. Similar results occurred during the depressurization test with all pumps on and both spray valves open. Figures 4.10-1 and 4.10-2 show these results. The Acceptance Criteria that the data be obtained and compared to the Westinghouse curves were met.

The results of the spray effectiveness tests are shown on Table 4.10-1. Effective depressurization was demonstrated in all the tests. In the test with only NC Pump A on the spray was least effective. This was due to backflow through leaky spray valves on the B loop. The Acceptance Criteria for demonstrating ability to depressurize in both normal and abnormal situations was met in all cases.

PORV response times and low pressure interlock results are shown on Table 4.10-2. All PORV's failed the initial Acceptance Criterion of < 2.0 seconds response time. An NSM (Nuclear Station Modification) had already been initiated which would replace the undersized valve components. A re-analysis was performed and a new response time of < 3.8 seconds was developed. All three PORV's met this new criterion. The pressures at which the low pressure interlock actuated for each of the PORV's were all within the Acceptance Criterion range of 2185 psig ± 13 psi.

IV. CORRECTIVE ACTION

Several problems were encountered during the performance of this test. It appears that one or more valves were leaking by during this test. During the first portion of the test (adjustment of the continuous spray flow valves) it was found that the spray line temperatures were unaffected by the position of 1NC28 and 1NC30.

This indicated that either these valves or the main spray valves were leaking by. Nothing was found when the reach rods and operators for the continuous spray flow valves were checked. During the spray effectiveness testing there were indications of backflow past 1NC29 or 1NC30. This was evidenced by the slower depressurization rates measured during the tests with NCP B off. These occurrences of valves leaking by were determined to not be of great concern as the valves were still capable of performing adequately.

The slow response time of the PORV's was due to undersized components in the PORV actuating systems. These will be modified. A re-analysis of the Acceptance Criteria for response time allowed the PORV's to be declared acceptable. This also necessitated a revision to the FSAR test abstract to incorporate the change in acceptable PORV response times. This revision, submitted on 12/17/84, was approved by the NRC on 1/14/85.

The Pressurizer Pressure Master Controller was found to be inoperable in the Auto mode during the heater effectiveness test. This was discovered after two failed test runs. A bad card in the 7300 Process Control cabinets was replaced and the heaters retested successfully.

PRESSURIZER FUNCTIONAL TEST
 SPRAY EFFECTIVENESS RESULTS

TEST CONDITIONS		DEPRESSURIZATION	TIME
<u>NCP's On</u>	<u>Spray Valves Open</u>	<u>(PSI)</u>	<u>(Min.)</u>
A, B, C, D	1NC27, 1NC29	262	2:00
A, C, D	1NC27, 1NC29	239	6:50
B, C, D	1NC27, 1NC29	229	2:40
A	1NC27, 1NC29	104	16:20
B	1NC27, 1NC29	239	4:55
A, B, C, D	1NC27	245	2:50
A, B, C, D	1NC29	243	2:30

NOTE: NC Pressure was ~2230 psig at the start of each depressurization.

PRESSURIZER FUNCTIONAL TEST
PORV TEST RESULTS

	<u>1NC32B</u>	<u>1NC34A</u>	<u>1NC36B</u>
Response time - Closed to Open	3.4 sec.	3.6 sec.	3.4 sec.
Pressure when Lo Pressure Interlock Actuated	2189 psig	2179 psig	2172 psig

Acceptable Pressure Range
for Interlock Actuation

2172 psig - 2198 psig

PRESSURIZER FUNCTIONAL TEST
HEATER PRESSURIZATION CURVE

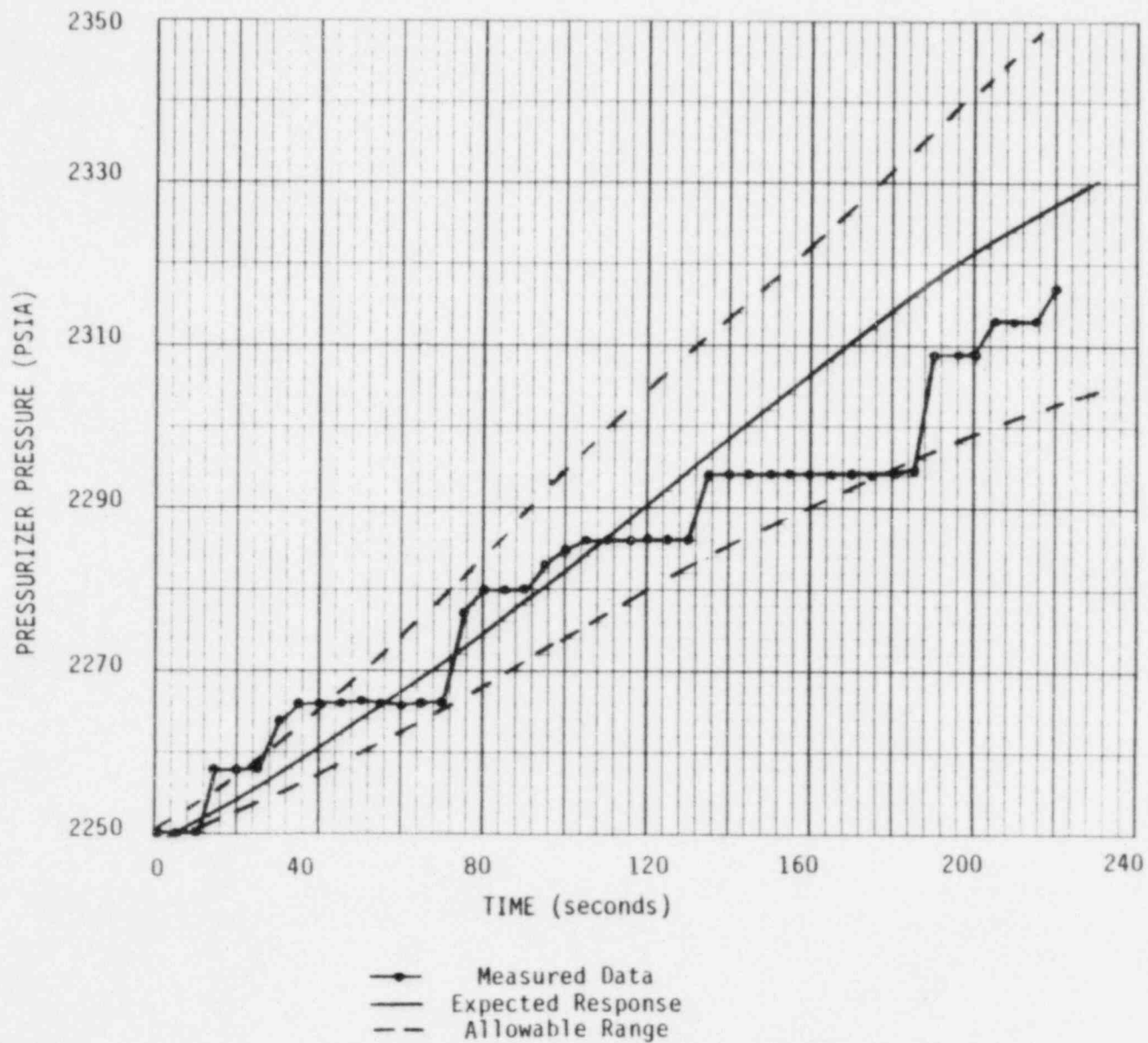


Figure 4.10-1

PRESSURIZER FUNCTIONAL TEST
SPRAY DEPRESSURIZATION CURVE

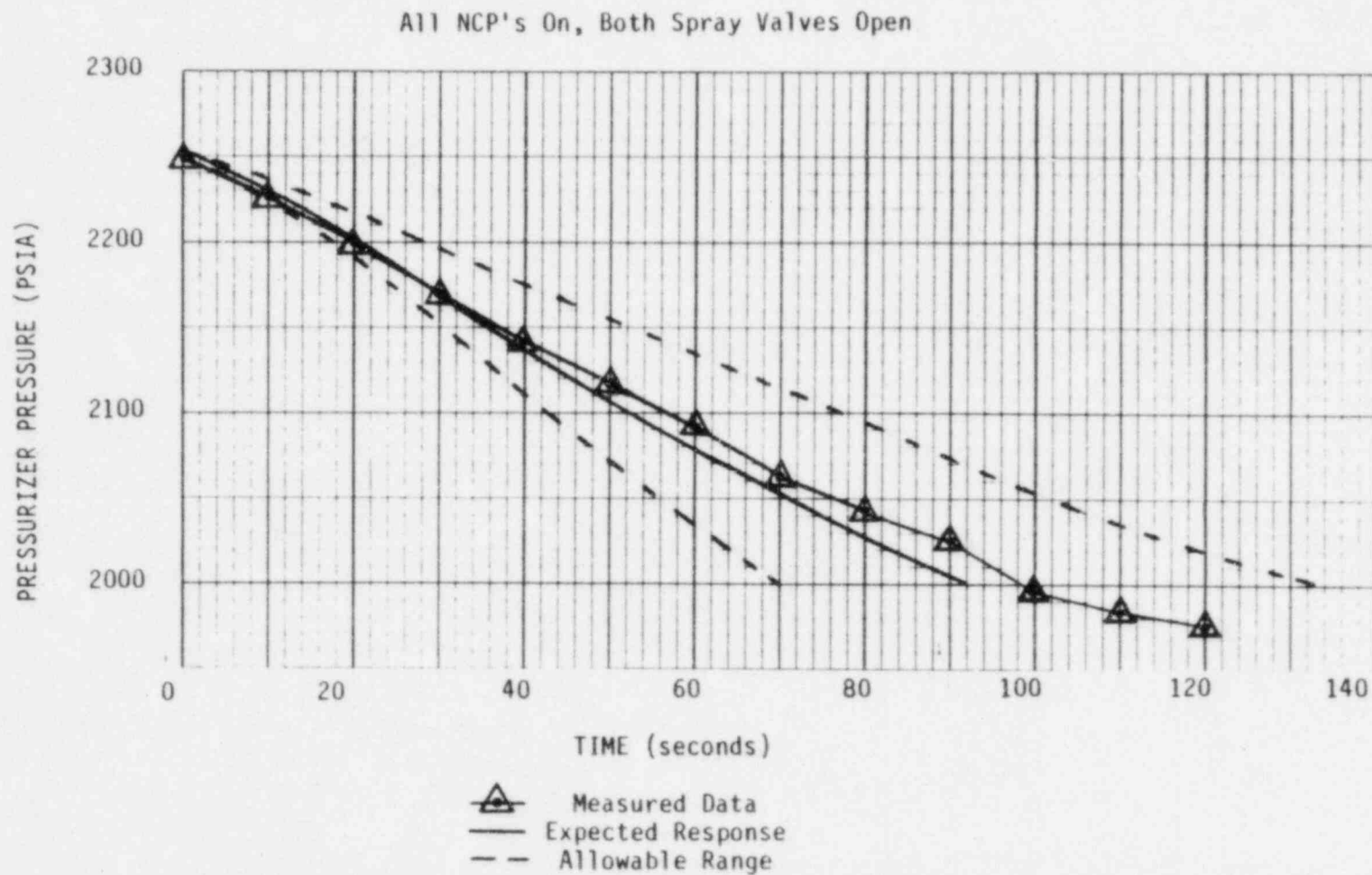


Figure 4.10-2

5.0 1/M APPROACH TO CRITICALITY - PT/1/A/4150/19

Date(s) Performed: 1/6/85 - 1/7/85

I. PURPOSE

The objectives of this test were:

- A. To ensure criticality is achieved in a safe and orderly manner.
- B. To verify that the critical boron concentration is within ± 50 ppm of the predicted value.

II. METHOD

After establishing baseline counts, the shutdown banks were withdrawn in normal sequence in ≤ 50 step intervals with the inverse count rate ratio (ICRR) plotted at each interval.

With the shutdown banks fully withdrawn, the control banks were withdrawn in normal sequence and normal overlap in ≤ 50 step intervals plotting ICRR at each interval. The control banks were withdrawn in this manner until Bank D was at 156 steps wd ± 2 steps. Then, baseline counts for boron dilution were established.

The Reactor Coolant System was deborated at a rate of < 60 gpm. Boron samples were taken in 15-minute intervals. ICRR was plotted at each interval as a function of boron concentration, time and water addition.

Boron dilution continued until ICRR ≈ 0.2 at which time the dilution was terminated and the NC System allowed to mix. New baseline counts were then established. Boron dilution continued by batch water addition at a rate of ≤ 30 gpm until ICRR ≈ 0.2 , then the NC System was allowed to mix until criticality was reached.

III. RESULTS

All acceptance criteria for this test were met. Initial criticality was achieved in a controlled manner with Control Bank D @ 141 steps wd. The Critical Boron Concentration was 969 ppmB. The Acceptance Criteria for Bank D @ 141 steps wd of between 884 ppmB and 984 ppmB was satisfied.

IV. CORRECTIVE ACTIONS

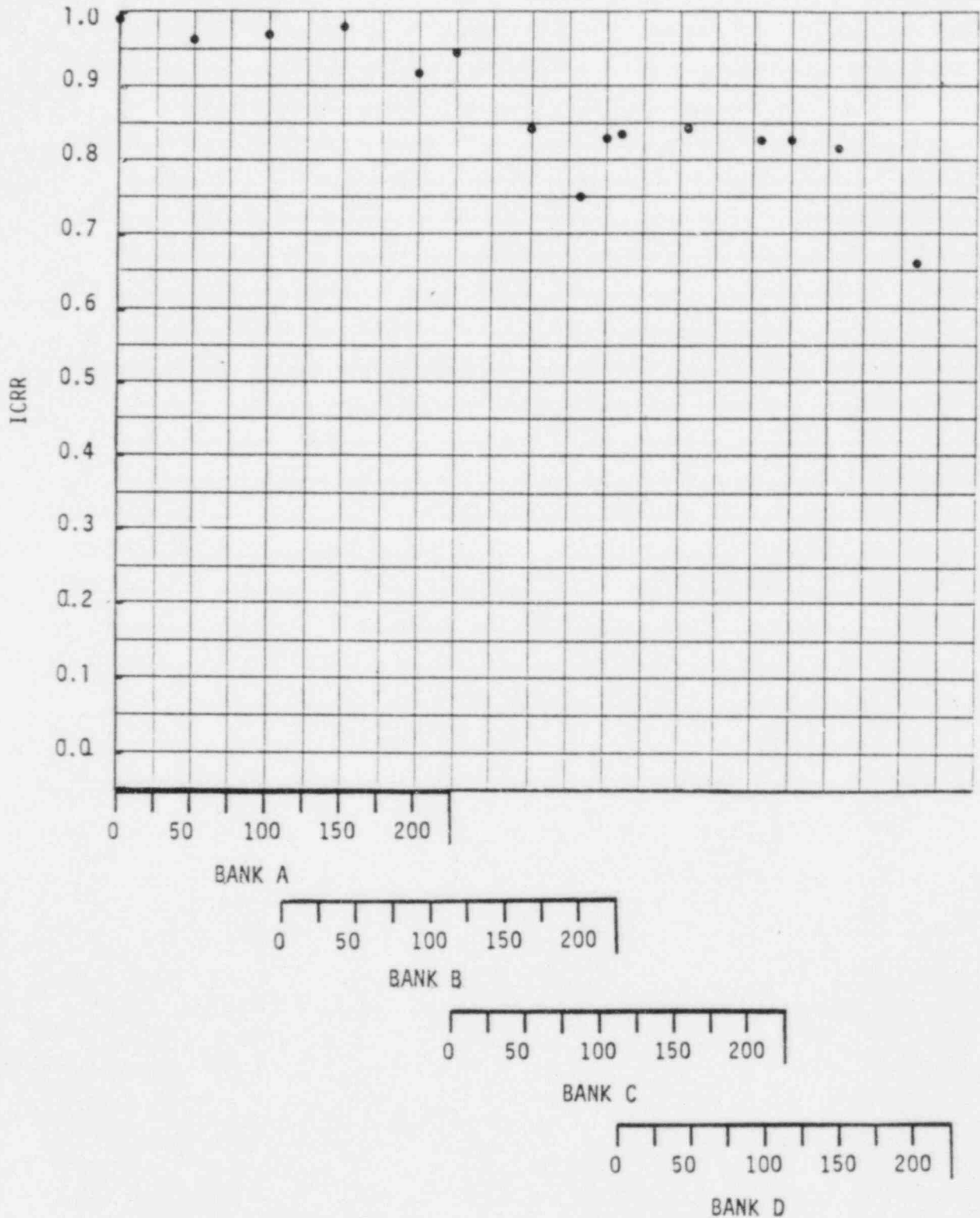
During withdrawal of Shutdown Banks D & E, one rod misstepped such that the remaining rods were not fully withdrawn (≈ 224 steps). These rod banks were exercised to verify that the rod was in fact misaligned. The rods were realigned per normal control rod operating procedures during the first mixing hold. Approach to criticality was then resumed.

Improper overlap between Control Banks A and B also occurred during rod withdrawal. The Overlap Counters in the Control Rod Logic cabinet were reset. Control Rod withdrawal was continued with no further overlap deviations observed.

The DRPI for Shutdown Bank B indicated only 222 steps wd when the bank was in fact fully withdrawn. The bank was completely reinserted and withdrawn again to correct this problem.

INITIAL CRITICALITY
ICRR VERSUS
CONTROL BANK POSITION

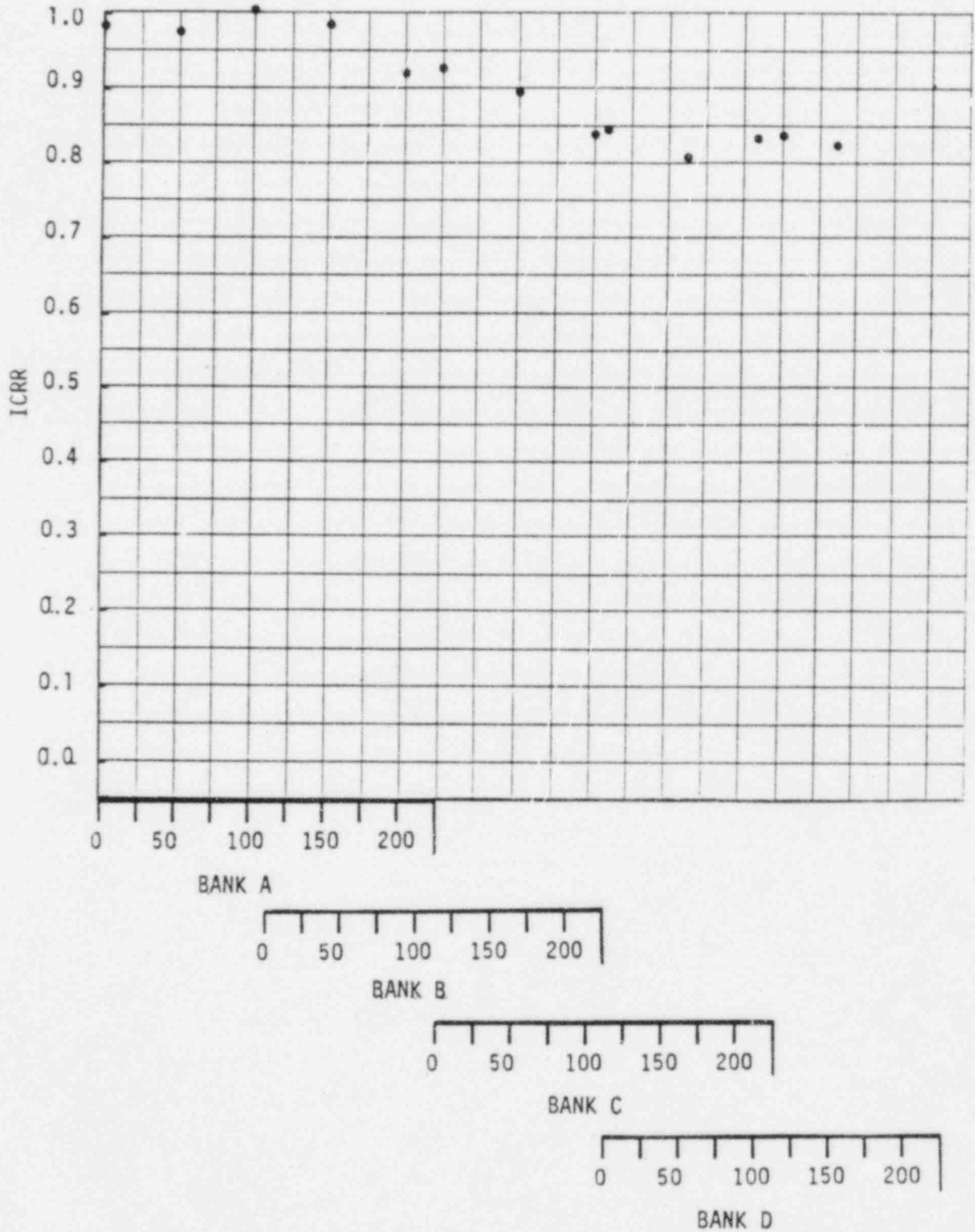
N31



STEPS WITHDRAWN
Figure 5.0-1

INITIAL CRITICALITY
ICRR VERSUS
CONTROL BANK POSITION

N32



STEPS WITHDRAWN

Figure 5.0-2

INITIAL CRITICALITY
ICRR VERSUS DILUTION

N31

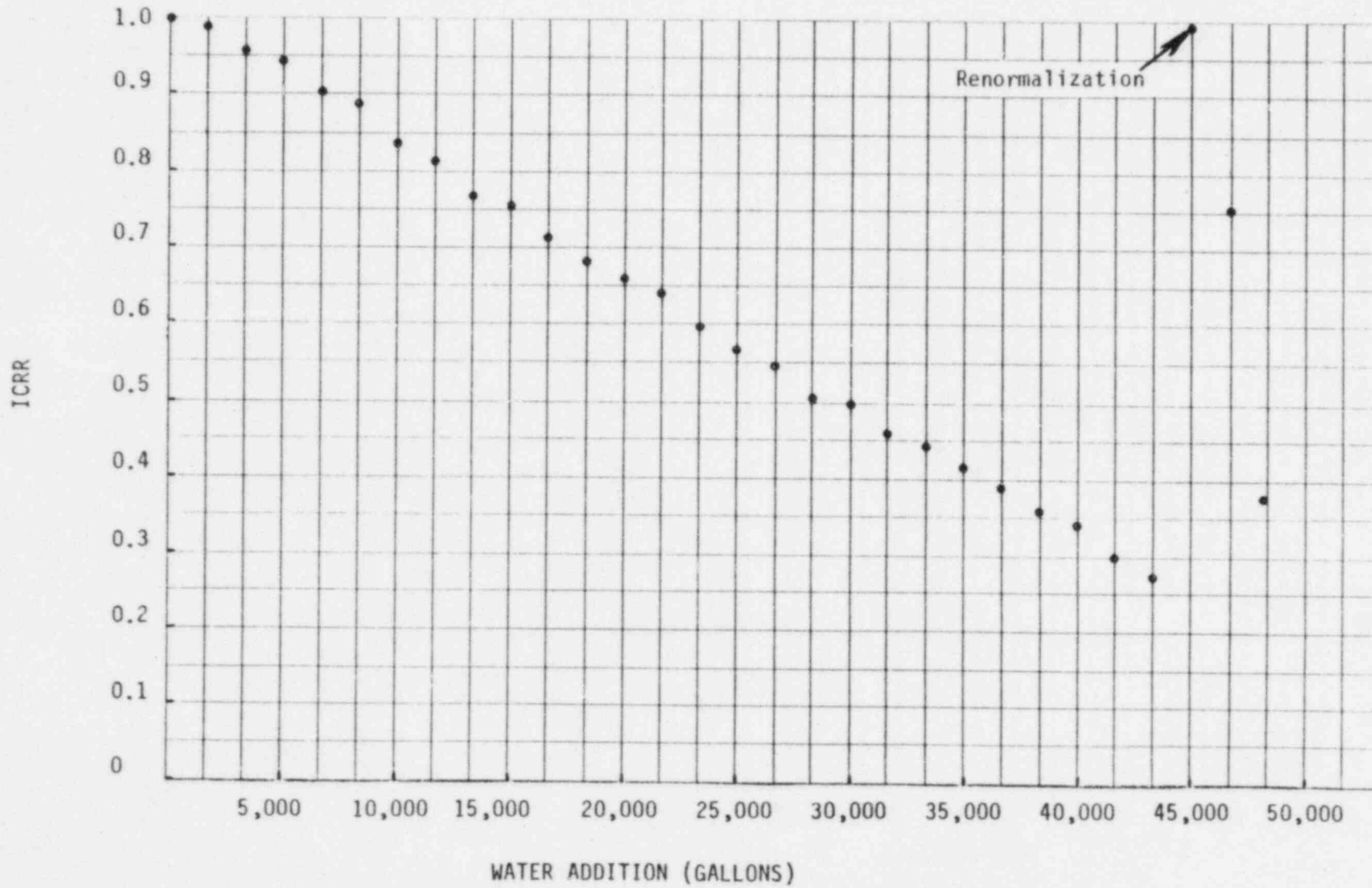


Figure 5.0-3

INITIAL CRITICALITY
ICRR VERSUS DILUTION
N32

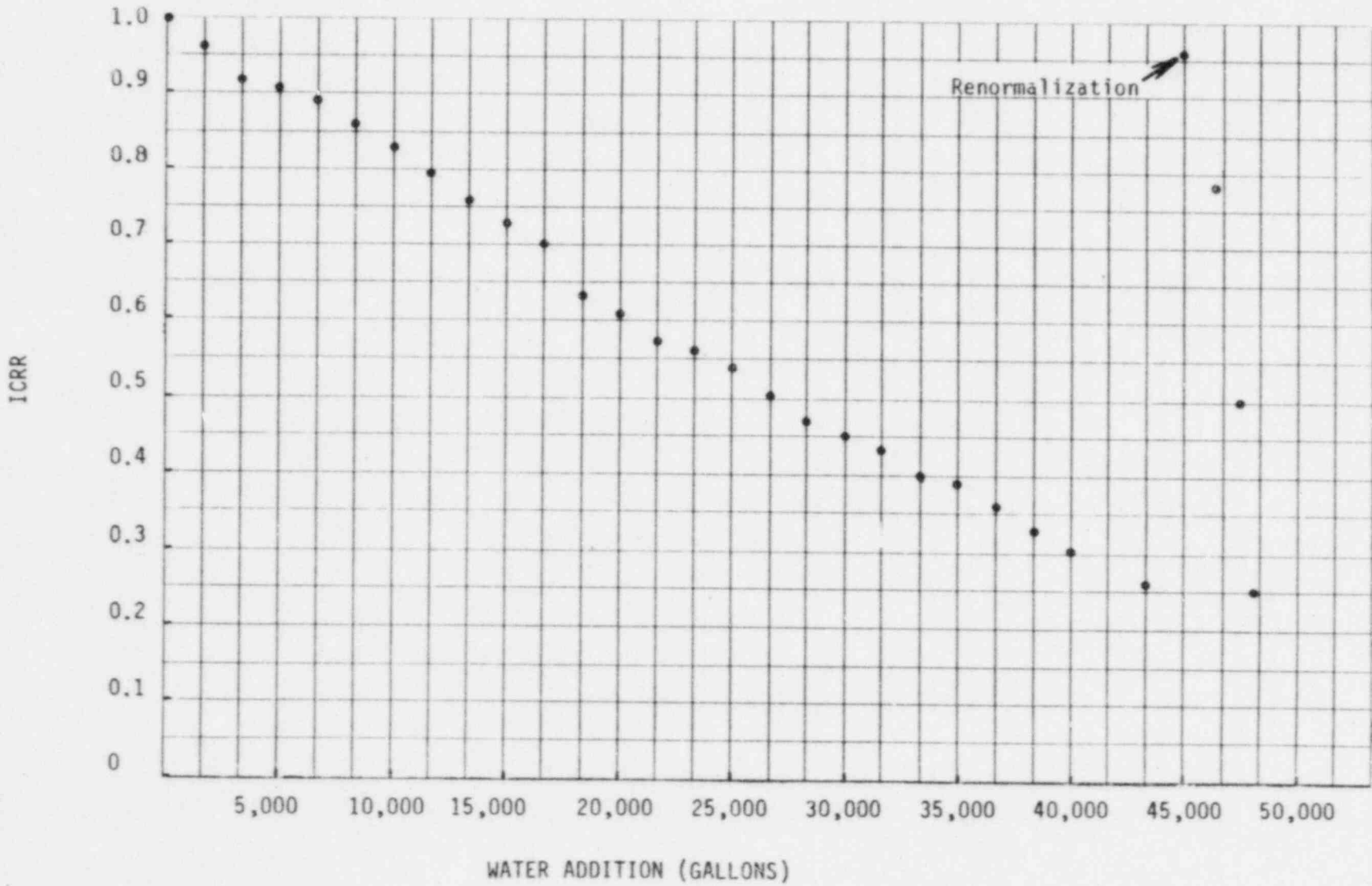


Figure 5.0-4

CATAWBA 1

INITIAL CRITICALITY
ICRR VERSUS TIME

N31

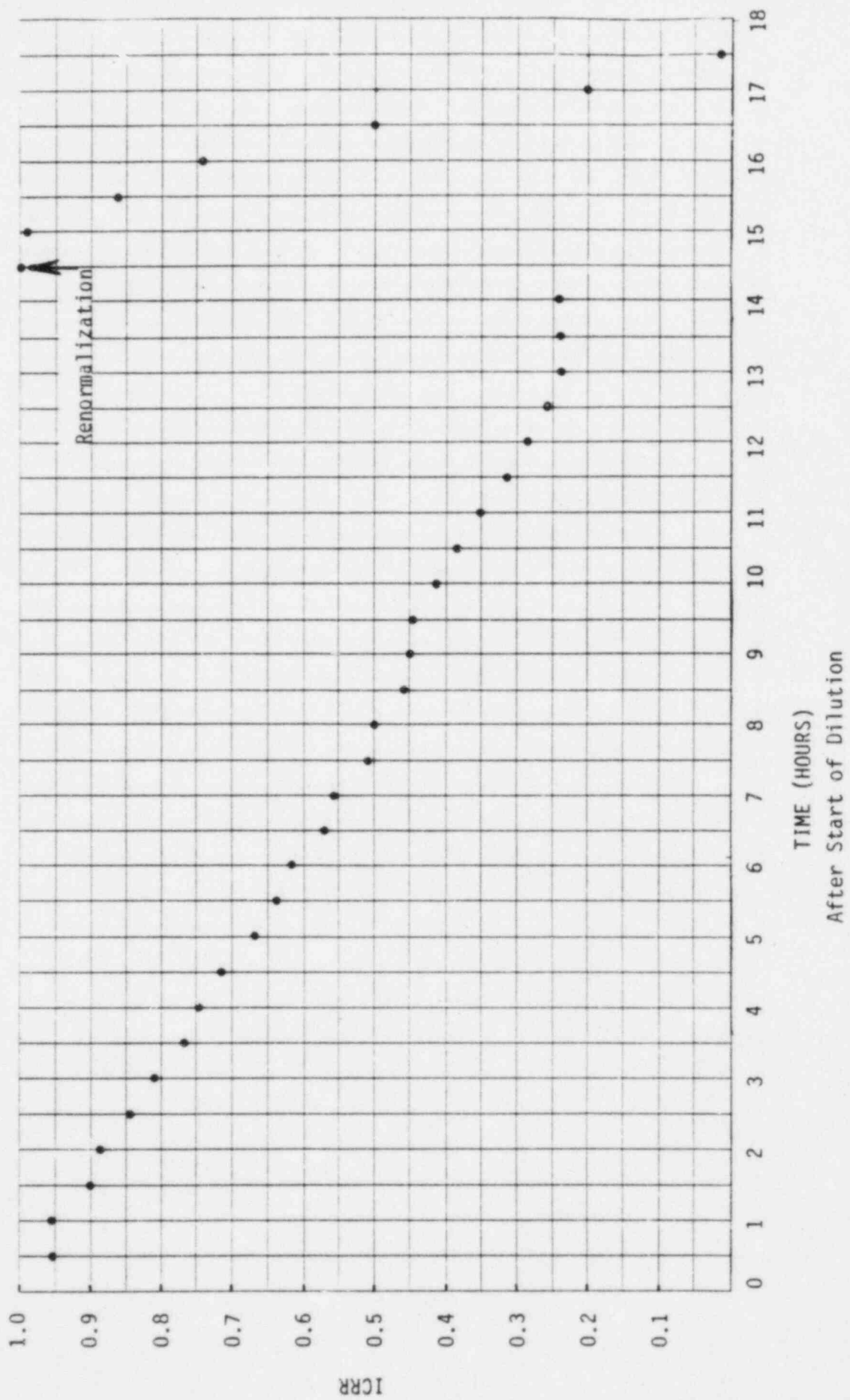


Figure 5.0-5

INITIAL CRITICALITY
ICRR VERSUS TIME

N32

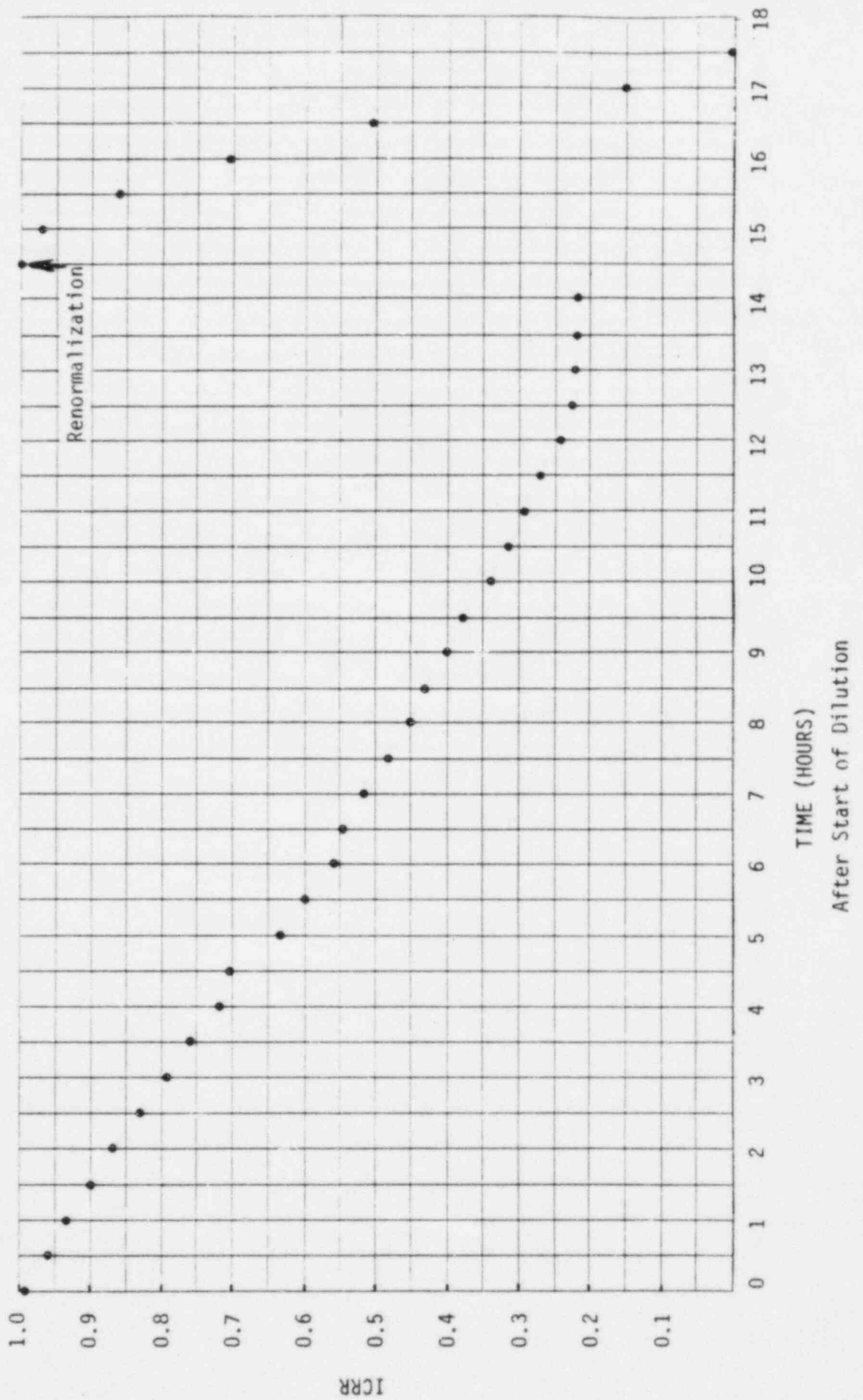


Figure 5.0-6

6.0 ZERO POWER PHYSICS CONTROLLING PROCEDURE - TP/1/A/2100/02

Date(s) Performed: 1/17/85 - 1/20/85

I. PURPOSE

- A. To verify adequate overlap between the Intermediate Range and Source Range Instrumentation.
- B. To determine the flux level for the onset of nuclear heat.
- C. To establish a testing range for and direct the sequence of numerous test procedures conducted for the purpose of measuring core physics characteristics. Among these tests were:
 - 1. PT/1/A/4150/10, Boron Endpoint Measurement
 - 2. PT/1/A/4150/12A, Isothermal Temperature Coefficient of Reactivity Measurement
 - 3. PT/1/A/4150/05, Core Power Distribution
 - 4. PT/1/A/4150/11A, Control Rod Worth Measurement by Boration/Dilution
 - 5. PT/1/A/4150/24, Stuck Rod Worth Measurement
 - 6. TP/1/A/2150/06A, Pseudo Rod Ejection Test
- D. To determine the Hot Zero Power differential boron worth.
- E. To direct the performance of TP/1/A/2650/13, Natural Circulation Verification Test.
- F. To perform a checkout of the reactivity computer.

II. METHOD

- A. To verify proper overlap between the Intermediate (N35, N36) and Source (N31, N32) Range detectors, the flux level was slowly increased at a rate of ≤ 0.25 DPM by control rod movement. The flux level was stabilized by control rod movement when the Intermediate Range read $\approx 1 \times 10^{-11}$ amps and $\approx 1 \times 10^{-10}$ amps and Source Range data collected.

- B. To determine the onset of nuclear heat, the flux level was slowly increased at a rate of ≤ 0.25 DPM by control rod movement. The flux level was allowed to increase until evidence of nuclear heat was detected. Nuclear heat was determined by an increase in average NC temperature accompanied by a decay of the reactivity trace. The range for performing subsequent reactivity measurements was determined such that the upper end of the testing decade was a factor of $(1/10)^{\frac{1}{2}}$ below the point of nuclear heat.
- C. 1. For the Method for the Boron Endpoint Measurement, refer to Section 6.1.
2. For the Method for the Isothermal Temperature Coefficient, refer to Section 6.2.
3. For the Method for the Core Power Distribution, refer to Section 6.3.
4. For the Method for the Control Rod Worth Measurement by Boration/Dilution, refer to Section 6.4.
5. For the Method for the Stuck Rod Worth Measurement, refer to Section 6.5.
6. For the Method for the Pseudo Rod Ejection Test, refer to Section 6.6.
- D. Results of the Boron Endpoint Measurements were used in conjunction with the results of the Rod Worth Measurements to determine the differential boron worth.
- E. For the Method for the Natural Circulation Verification, refer to Section 7.0.
- F. During reactivity physics measurements, the core reactivity was monitored via an analog and/or digital reactivity computer. Each computer provided a solution to the delay neutron precursor decay rate equation for the six groups of delayed neutrons. Refer to Table 6.0-1 for the data specific to Catawba 1. Each computer received a signal via a picoammeter from a power range channel.

The initial setup and checkout of the analog computer required adjusting of amplifier potentiometers corresponding to the properties of the delay neutron precursor groups. The initial setup and checkout of the digital computer (IBM 9000) was performed by verifying proper calibration of the computers analog to digital converters.

Electronic checkout of each computer was performed on a daily basis by performing an exponential test. Positive and negative exponential test signals were generated by the analog computer. The resulting reactivity solutions were compared to vendor reactivity predictions.

Dynamic checkout of the analog reactivity computers was performed by withdrawing/inserting the control rods a reactivity of approximately ± 50 pcm. From the measured doubling/halving times, the vendor predicted reactivity was determined and compared to the computer solutions.

III. RESULTS

- A. Overlap between the source and intermediate range channels was \geq one decade (as seen on the intermediate range). Refer to Table 6.0-2 for measured data.
- B. The point of onset of nuclear heat met the range for performing reactivity measurements were identified. Refer to Table 6.0-3 for measured data.
- C.
 1. For the Results for the Boron Endpoint Measurement, refer to Section 6.1.
 2. For the Results for the Isothermal Temperature Coefficient, refer to Section 6.2.
 3. For the Results for the Core Power Distribution, refer to Section 6.3.
 4. For the Results for the Control Rod Worth Measurement by Boration/Dilution, refer to Section 6.4.
 5. For the Results for the Stuck Rod Worth Measurement, refer to Section 6.5.
 6. For the Results for the Pseudo Rod Ejection Test, refer to Section 6.6.
- D. The measured differential boron worth (pcm/ppmB) over the Control Bank worth was acceptable. Refer to Figure 6.0-1 for measured data.

<u>Vendor Prediction</u>	<u>Duke Prediction</u>	<u>Measured</u>
-12.98 \pm 1.3	-12.29	-12.96

- E. For results refer to Section 7.0, Natural Circulation Verification.
- F. The reactivity computer solutions for reactivity were within $\pm 4\%$ of the design reactivity values for the daily operation checks. The analog computer solution for reactivity was within $\pm 4\%$ of the design reactivity values. Refer to Table 6.0-4 for the measured values.

IV. CORRECTIVE ACTIONS

The reactivity computers were not giving reliable results using N44. The input signal was changed to N42.

Refer to Section 6.1 through 6.6 and 7.0 for corrective action for individual zero power tests.

DELAYED NEUTRON DATA
BEGINNING OF LIFE, HOT ZERO POWER

Group	$\bar{\beta}_i$	λ_i (sec ⁻¹)
1	0.000216	0.0125
2	0.001460	0.0308
3	0.001346	0.1150
4	0.002800	0.3105
5	0.000941	1.2363
6	0.000320	3.3163

Total Delayed Neutron Fraction, $\bar{\beta} = 0.007083$

Prompt Neutron Lifetime, ℓ^* (μ sec) = 22.06

Importance Function, $\bar{I} = 0.970$

Effective Delayed Neutron Fraction, $\bar{\beta}_{eff} = 0.006871$

ZPPT - NUCLEAR INSTRUMENTATION SYSTEM

OVERLAP DATA

Indication Location	Source Range (Counts per Second)		Intermediate Range (Amps)	
	N31	N32	N35	N36
1. Control Room	550	550	1.0E-11	1.5E-11
OAC	493.5	608.3	1.17E-11	1.15E-11
2. Control Room	2.0E4	2.0E4	1.0E-10	2.0E-10
OAC	2.0E4	2.6E4	1.0E-10	1.45E-10
1- Flux at 1×10^{-11} amps				
2- Flux at 1×10^{-10} amps				

TABLE 6.0-2

ZPPT - NUCLEAR HEAT DETERMINATION

	<u>Picoammeter</u> (Amps)	<u>N35</u> (Amps)	<u>N36</u> (Amps)
Start #1	1.4×10^{-7}	5.65×10^{-8}	5.78×10^{-8}
Nuclear Heat	3.5×10^{-6}	1.22×10^{-6}	1.95×10^{-6}
Start #2	1.79×10^{-6}	5.50×10^{-7}	7.48×10^{-7}
Nuclear Heat	4.36×10^{-6}	1.34×10^{-6}	1.58×10^{-6}

Upper limit for reactivity measurements = 1.11×10^{-6} amps
(as seen on picoammeter)

Actual testing decade: 1×10^{-7} amps to 1×10^{-6} amps
(as seen on picoammeter)

ZPPT - ANALOG REACTIVITY COMPUTER
DYNAMIC CHECKOUT

Doubling Time (sec)	Period (sec)	Measured Reactivity (pcm)	Theoretical Reactivity (pcm)	(1) Percent Error(%)
112.77	162.7	42.5	42.95	1.05
118.23	170.6	41.5	41.18	-0.78
119.32	172.2	42.0	40.88	-2.74
135.48	195.5	37.5	36.65	-2.32
Halving Time (sec)	Period (sec)	Measured Reactivity (pcm)	Theoretical Reactivity (pcm)	(1) Percent Error(%)
150.97	217.8	-44.0	-46.93	6.24
169.65	244.8	-39.0	-40.69	4.15
163.76	236.3	-40.0	-42.44	5.75
170.72	246.31	-38.5	-40.37	4.63
Average:				2.0

$$1 - \% \text{ Error} = \frac{\text{Theoretical} - \text{Measured}}{\text{Theoretical}} * 100\%$$

ZPPT - DIFFERENTIAL BORON WORTH

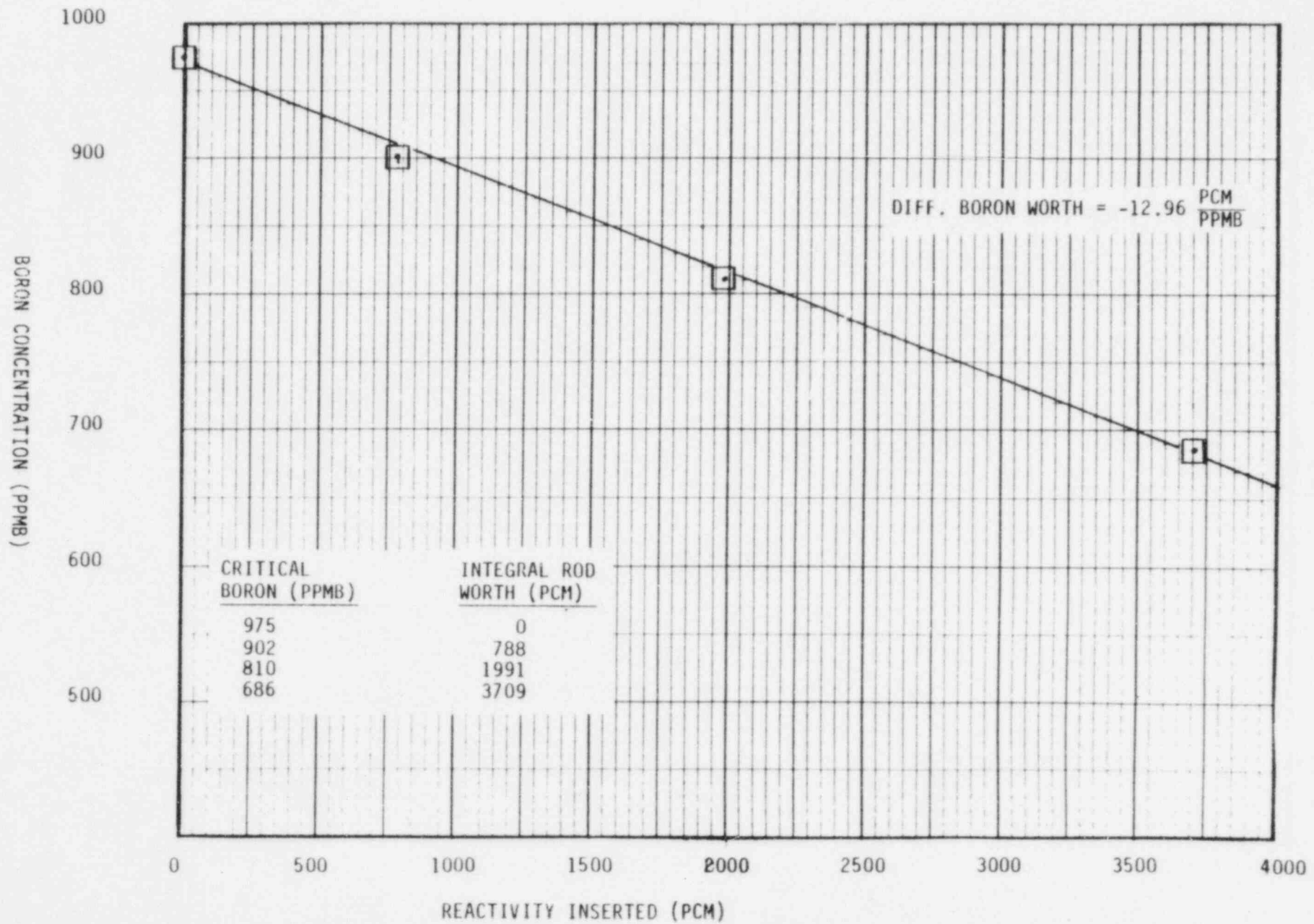


Figure 6.0-1

6.1 BORON ENDPOINT MEASUREMENT - PT/1/A/4150/10

Date(s) Performed: 1/9/85 (ARO), 1/11/85 (D-in)
1/12/85 (C&D-in), 1/13/85 (A,B,C&D-in)

I. PURPOSE

The purpose of the Boron Endpoint Measurement was to determine the critical boron concentration at specified rod configurations. The rod configurations for which a boron endpoint was determined are: All Rods Out (ARO); Control Bank D at 0 steps, and all other banks at 228 steps (D-in), Control Banks C and D as 0 steps and all other banks at 228 steps (C&D-in); all Control Banks at 0 and all shutdown banks at 228 steps (A,B,C,&D-in).

II. METHOD

Critical conditions were established with rod banks near the desired configuration (One rod bank would be just slightly inserted or withdrawn). The boron concentration was measured, then the appropriate bank would be withdrawn or inserted to achieve the specified configuration. The reactivity worth of this move was measured and converted to an equivalent boron concentration, which was added to or subtracted from the measured boron concentration. The process was repeated as necessary to ensure confidence in the results.

III. RESULTS

The results of the four boron endpoint measurements performed are tabulated in Table 6.1-1, along with design predictions. The only acceptance criteria associated with the measurement was that the ARO critical boron concentration was 946 ± 50 ppm. The measured ARO critical boron concentration was 975 ppm. Figure 6.1-1 shows the reactivity trace for a typical bank withdrawal (or insertion) used to calculate the boron endpoint.

IV. CORRECTIVE ACTION

No corrective actions were required.

BORON ENDPOINT MEASUREMENT
HOT ZERO POWER CRITICAL BORON CONCENTRATIONS

Rod Configuration	Westinghouse Prediction	Duke Prediction	Measured	% Difference Meas. - West.	% Difference Meas. - Duke
ARO	946	957	975	-2.97	-1.85
Control D-in	884	897	902	-2.00	-0.55
Control C&D-in	784	816	810	-3.21	0.74
Control A,B, C&D-in	645	693	686	-5.98	1.02

TABLE 6.1-1

BORON ENDPOINT MEASUREMENT

TYPICAL REACTIVITY TRACE

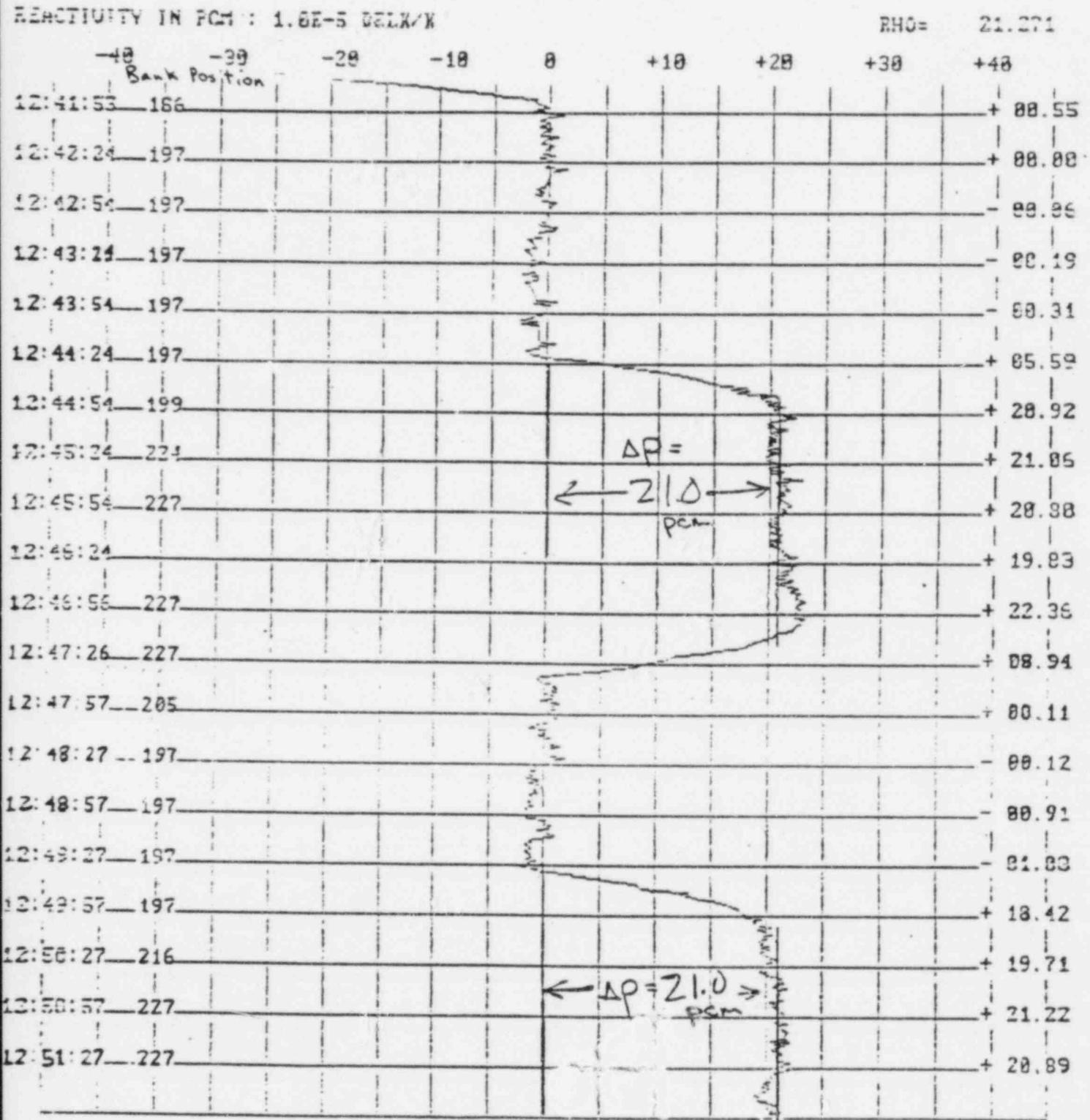


FIGURE 6.1-1

CATAWBA 1

6.2 ISOTHERMAL TEMPERATURE COEFFICIENT OF REACTIVITY MEASUREMENT
PT/1/A/4150/12A

Date(s) Performed: 1/9/85, 1/11/85 - 1/12/85, 1/12/85

I. PURPOSE

The purpose of this test was to measure the Isothermal Temperature Coefficient (ITC) and from this data derive a value for the Moderator Temperature Coefficient (MTC) for the following conditions:

- A. All Control Rods Withdrawn (ARO).
- B. Control Bank D Inserted (CD IN).
- C. Control Banks C and D Inserted (CD + CC IN).

II. METHOD

A heatup and cooldown of the NC System of $\approx 10^{\circ}\text{F}/\text{hour}$ was initiated and the resulting reactivity change vs. temperature recorded on a X-Y plotter. As soon as sufficient data had been obtained (after $\approx 3^{\circ}\text{F}$ change) the heatup/cooldown was stopped. The isothermal temperature coefficient was then determined from the average slope of the reactivity vs. temperature plots. The value of the moderator temperature coefficient was then calculated by subtracting out the effect of the doppler coefficient from the isothermal temperature coefficient measurement.

III. RESULTS

All Acceptance Criteria were met. The measured ITC agreed with the Westinghouse predictions. The MTC was negative for all cases. See Table 6.2-1 for the measured values of the ITC and MTC for each case. See Figure 6.2-1 for typical reactivity trace. Due to the MTC being close to zero, the test was performed with Control Bank D inserted (CD IN) and with Control Bank C and D inserted (CD + CC IN) in order to establish administrative rod withdrawal limits.

IV. CORRECTIVE ACTIONS

None required.

ISOTHERMAL TEMPERATURE COEFFICIENT OF REACTIVITY MEASUREMENT

ITC AND MTC RESULTS

ITC Results

	<u>Westinghouse Acceptance Criteria</u>	<u>Duke Prediction</u>	<u>Measured Value</u>
ARO ITC	-1.62 ± 3.0 pcm/°F	-3.93 pcm/°F	-1.745 pcm/°F
CD IN ITC	-2.71 ± 3 pcm/°F	-5.98 pcm/°F	-2.77 pcm/°F
CD + CC IN ITC	-7.28 ± 3.0 pcm/°F	-11.50 pcm/°F	-8.01 pcm/°F

MTC Results

	<u>Tech Spec 3.1.1.3.a</u>	<u>Measured Value</u>	
ARO MTC	< 0 pcm/°F	-0.015 pcm/°F*	
	<u>Westinghouse Prediction</u>	<u>Duke Prediction</u>	<u>Measured Value</u>
CD IN MTC	-0.97 pcm/°F	-4.78 pcm/°F	-1.04 pcm/°F*
CD + CC IN MTC	-5.55 pcm/°F	-9.92 pcm/°F	-6.28 pcm/°F*

*Based on Westinghouse Doppler Coefficient prediction (-1.73 pcm/°F)

ISOTHERMAL TEMPERATURE COEFFICIENT OF REACTIVITY MEASUREMENT
TYPICAL REACTIVITY TRACE

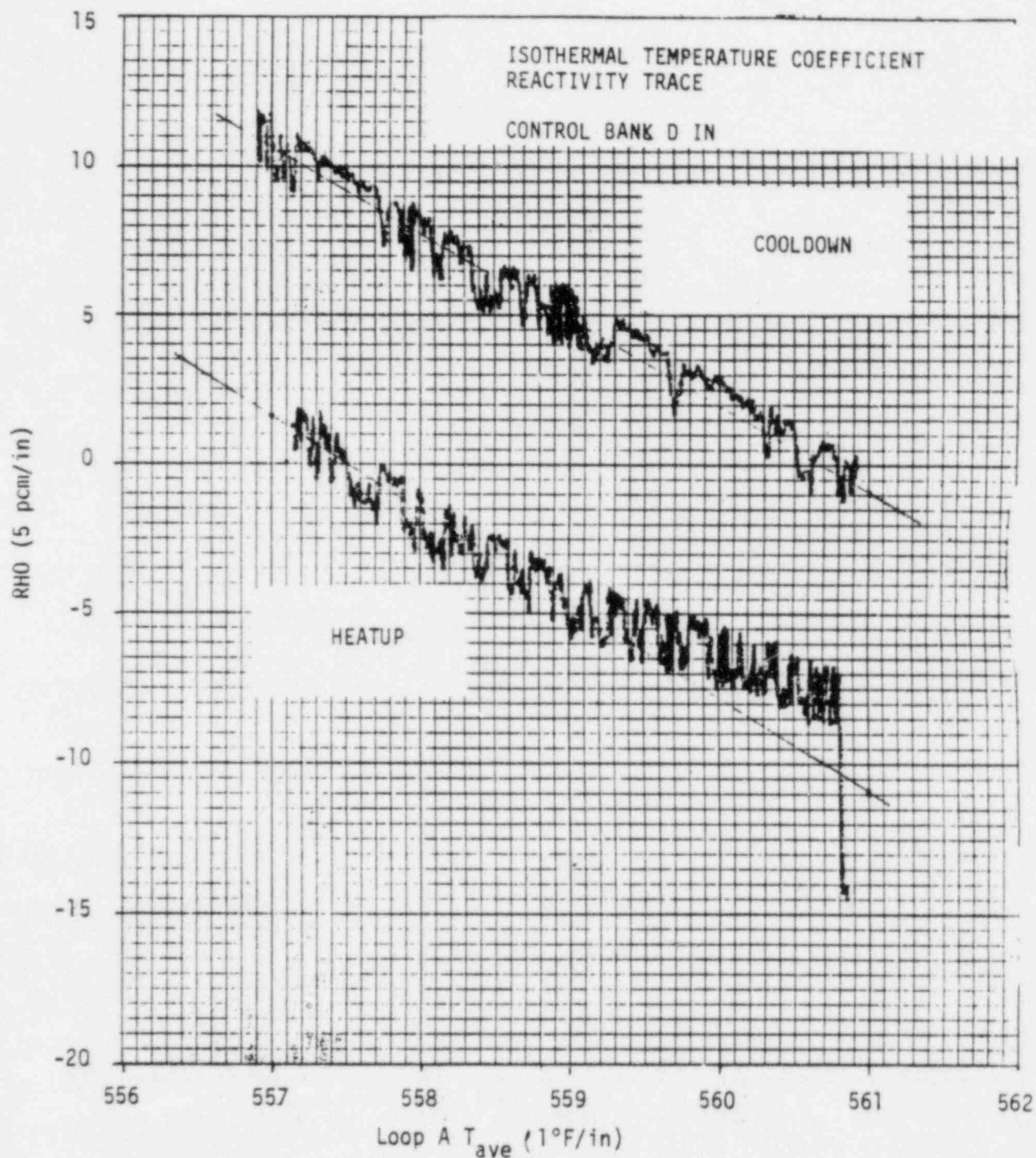


Figure 6.2-1

6.3 CORE POWER DISTRIBUTION - PT/1/A/4150/05

Date(s) Performed: 1/10/85 - 1/11/85

I. PURPOSE

- A. To obtain and analyze the reactor core power distribution at approximately hot zero power conditions (< 5% RTP).
- B. To verify that the measured power distribution is maintained within the limits of Technical Specifications and/or design predictions.

II. METHOD

Reactor power was increased to a level < 5% RTP. Control Bank D was positioned to > 200 steps withdrawn.

A full core flux map (FCM/1/01/001) was obtained utilizing the moveable incore detector (ENA) system. During the mapping process, pertinent plant data was collected by the OAC. The moveable detector data was input into the SNC-CORE computer program. The results of the program were then compared to appropriate power distribution Tech Specs and design predictions.

A second full core flux map (FCM/1/01/002) was obtained and analyzed as described above, with Control Bank D and = 0 steps withdrawn, to confirm results of FCM/1/01/001.

III. RESULTS

Refer to Tables 6.3-1 and 6.3-2 and Figures 6.3-1 through 6.3-4 for measured data.

- A. Measured values for $F_Q(z)$ were less than the Tech Spec value of 4.64 for Mode 1 operation.
- B. Measured values of $R(F_{\Delta H}^N)$ were less than the Tech Spec value of 1.0 for Mode 1 operation.
- C. Measured values of $F_{xy}(z)$ were less than the Tech Spec value of 2.0375 (FCM/1/01/001) and 2.2306 (FCM/1/01/002) for Mode 1 operation.
- D. The difference of the predicted to measured $F_{\Delta H}^N$ failed the vendor $\pm 10\%$ criteria.
- E. Measured Quadrant Power Tilt Ratios (QPTR) exceeded the Tech Spec value of < 1.02 for Mode 1 operation > 50% RTP.

IV. CORRECTIVE ACTIONS

FCM/1/01/002 was performed to confirm results of FCM/1/01/001.

Duke Power and Westinghouse performed a review of items D and E above. It was concluded the core peaking factors might become limited > 50% RTP. Additional core power measurements would be required at interim power levels (20%, 30%, 50%) to confirm that the measured values are not greater than those measured as zero power (Refer to Section 8.1 for more discussion).

Duke Power and Westinghouse reviewed the following items for potential sources of the higher than anticipated QPTR:

- A. Poor Measurement
- B. Temperature/Flow Variation
- C. Misloaded Core
- D. Misaligned Control Rods
- E. Enrichment Variability
- F. Modeling of Instrumented Assemblies and Burnable Poisons

The root cause the high QPTR could not be determined.

ZPPT - CORE POWER DISTRIBUTION

RESULTS

Test Date:	1/10/85
Map ID:	FCM/1/01/001
Power Level:	= 3%
Boron Concentration:	977 ppmB
Rod Position:	Control Bank D-213 Steps Withdrawn
Measured R:	0.8876
Measured NC Flow:	422,840 gpm
Maximum Measured F_{xy}^* :	1.8556 Axial Location 51 Core Location C-13
Maximum F_Q :	2.7761 Axial Location 31 Core Location C-13
Maximum F_Z :	1.4868 Axial Location 31
Maximum $F_{\Delta H}^N$:	1.7163 Core Location C-13
Maximum $F_{\Delta H}^N$ Error (from predicted):	-11.8% Core Location B-13
Total Core Axial Offset:	-0.553%
Maximum Quadrant Power Tilt Ratio:	1.07178 in Quadrant 2 Top Half

*In locations unexcluded by Technical Specifications

ZPPT - CORE POWER DISTRIBUTION

RESULTS

Test Date:	1/11/85
Map ID:	FCM/1/01/002
Power Level:	= 3.5%
Boron Concentration:	902 ppm
Rod Position:	Control Bank C-218 Steps Withdrawn Control Bank D-0 Steps Withdrawn
Measured R:	0.9307
Measured NC Flow:	422,881 gpm
Maximum Measured F_{xy}^* :	1.7937 Axial Location 51 Core Location J-14
Maximum F_Q :	2.8384 Axial Location 30 Core Location G-14
Maximum F_z :	1.4612 Axial Location 25
Maximum $F_{\Delta H}^N$:	1.7877 Core Location G-14
Maximum $F_{\Delta H}^N$ Error (from predicted):	+13.9% Core Location R-08
Total Core Axial Offset:	-1.827%
Maximum Quadrant Power Tilt Ratio:	1.0542 in Quadrant 2 Top Half

*In locations unexcluded by Technical Specifications

ZPPT - CORE POWER DISTRIBUTION
 FCM/1/01/001
 AXIAL OFFSETS AND INCORE TILTS

AXIAL OFFSET RATIOS FROM ASSEMBLAGE POWERS

TOTAL CORE AXIAL OFFSET = -0.00553

AXIAL OFFSET FOR QUADRANTS ^{1 2} AND FOR QUADRANTS ^{8 6}
_{4 3} ₇

0.00114 -0.00352 -0.01592 -0.00452 -0.00179
 -0.01922 -0.00132 -0.00015

AXIAL OFFSET BY OCTANTS ^{1 2 3}
_{8 7 6} _{5 4}

-0.00009 0.00233 -0.01101 0.00393
 -0.03206 -0.00603 0.00553 -0.00783

TILTING FACTORS FROM ASSEMBLAGE POWERS

TILTING FACTORS FOR QUADRANTS ^{1 2} AND FOR QUADRANTS ^{8 6} FOR ENTIRE CORE HEIGHT
_{4 3} ₇

0.99377 1.06962 0.96792 1.03871 1.04458
 0.94553 0.99107 0.94879

TILTING FACTORS FOR OCTANTS ^{1 2 3} FOR ENTIRE CORE HEIGHT
_{8 7 6} _{5 4}

0.97730 1.01024 1.08717 1.07207
 0.95853 0.93253 0.96506 1.01709

TILTING FACTORS FOR QUADRANTS ^{1 2} AND FOR QUADRANTS ^{8 6} FOR TOP HALF, NORMALIZED
_{4 3} ₇

1.00044 1.07178 0.95781 1.03976 1.04851
 0.93251 0.99527 0.95393

TILTING FACTORS FOR QUADRANTS ^{1 2} AND FOR QUADRANTS ^{8 6} FOR BOT. HALF, NORMALIZED
_{4 3} ₇

0.98718 1.06748 0.97792 1.03767 1.04070
 0.95841 0.98693 0.94372

ZPPT - CORE POWER DISTRIBUTION
 FCM/1/01/001 ASSEMBLY F_{ΔH} AND F_Q

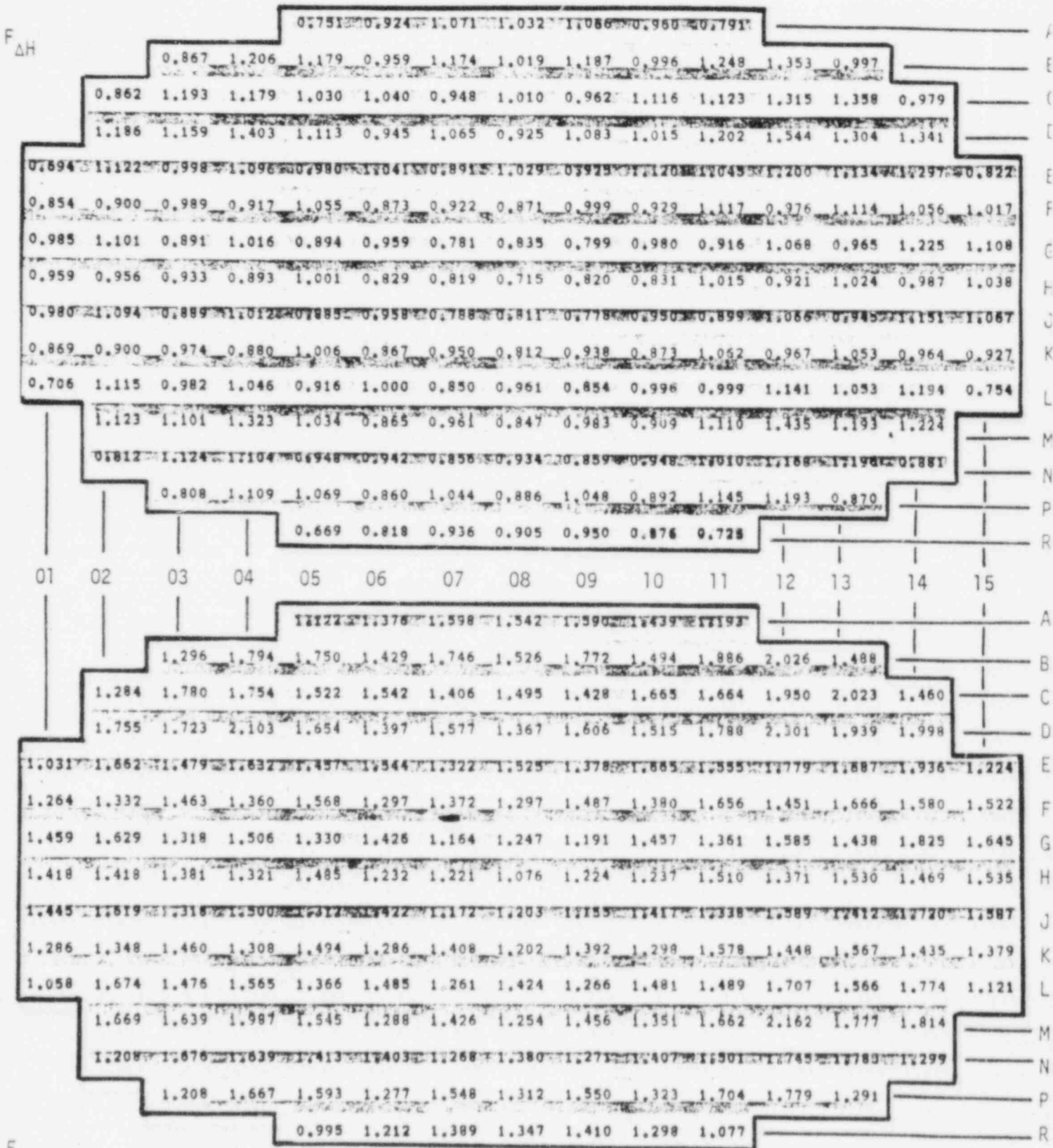


Figure 6.3-2

CATAWBA 1

ZPPT - CORE POWER DISTRIBUTION
 FCM/1/01/002
 AXIAL OFFSETS AND INCORE TILTS

AXIAL OFFSET RATIOS FROM ASSEMBLAGE POWERS

TOTAL CORE AXIAL OFFSET = -0.01827

AXIAL OFFSET FOR QUADRANTS

AND FOR QUADRANTS

1	2	3	4	5	6	7	8
4	3						
-0.01494	-0.01764					-0.01222	-0.02371
-0.01462	-0.02589					-0.01631	-0.02090

AXIAL OFFSET BY OCTANTS

1	2	3	4	5	6	7	8
-0.01822	-0.01178	-0.01264				-0.02256	
-0.01441	-0.01483	-0.02695				-0.02491	

TILTING FACTORS FROM ASSEMBLAGE POWERS

TILTING FACTORS FOR QUADRANTS

AND FOR QUADRANTS

FOR ENTIRE CORE HEIGHT

1	2	3	4	5	6	7	8
0.99582	1.05354					0.98035	1.03058
0.96404	0.98659					0.94476	1.04431

TILTING FACTORS FOR OCTANTS

FOR ENTIRE CORE HEIGHT

1	2	3	4	5	6	7	8
0.97632	1.01532	1.04683				1.06125	
0.98438	0.94270	0.99451				1.02738	

TILTING FACTORS FOR QUADRANTS

AND FOR QUADRANTS

FOR TOP HALF, NORMALIZED

1	2	3	4	5	6	7	8
0.99920	1.05422					0.98232	1.03693
0.96763	0.97894					0.94223	1.03852

TILTING FACTORS FOR QUADRANTS

AND FOR QUADRANTS

FOR BOT. HALF, NORMALIZED

1	2	3	4	5	6	7	8
0.99256	1.05288					0.98746	1.02445
0.96058	0.99397					0.94719	1.04990

ZPPT - CORE POWER DISTRIBUTION
 FCM/1/01/002 ASSEMBLY F_{ΔH} AND F_Q

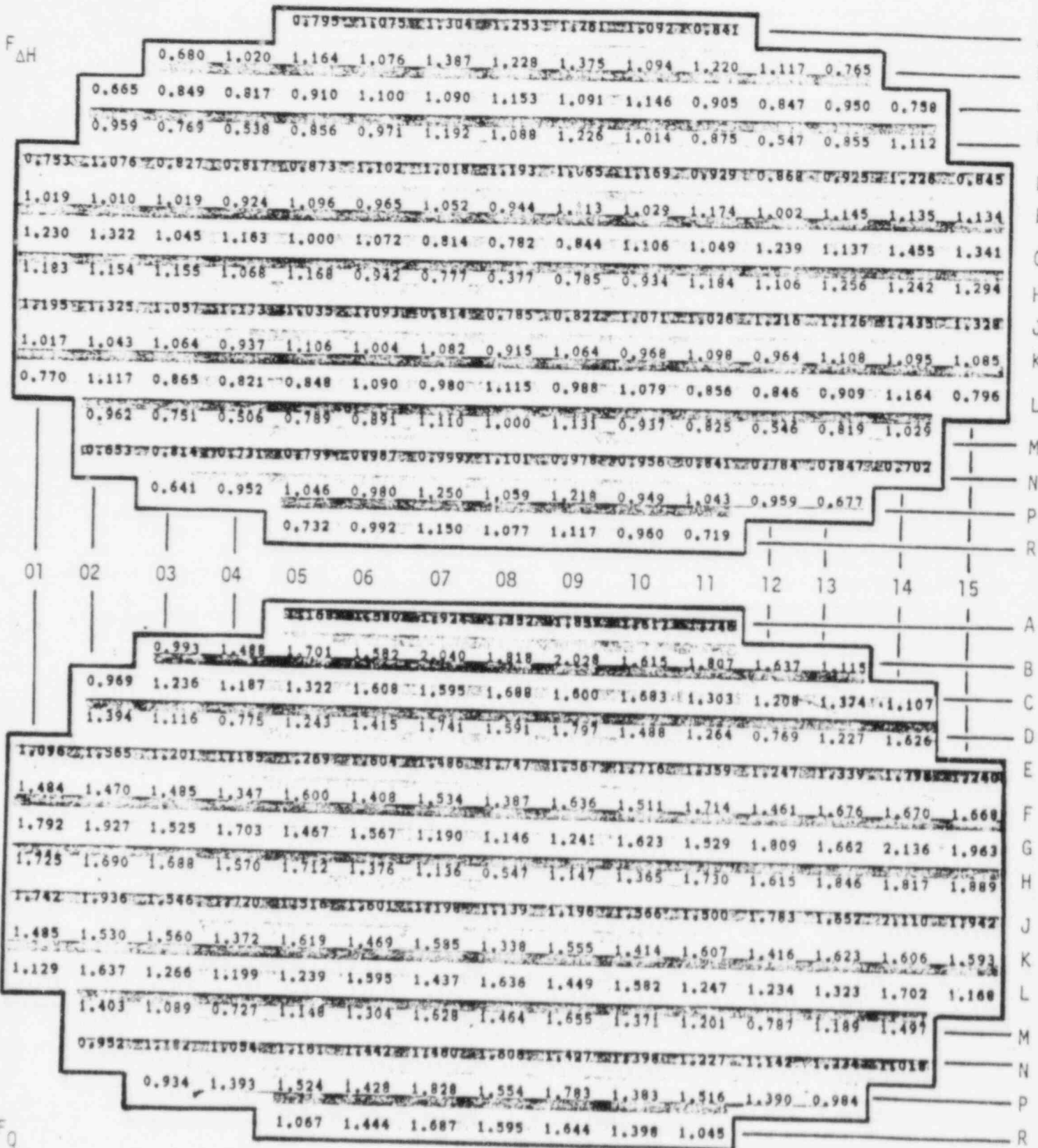


Figure 6.3-4

CATAWBA 1

6.4 CONTROL ROD WORTH MEASUREMENT BY BORATION/DILUTION - PT/1/A/4150/11A

Date(s) Performed: 1/10/85 - 1/18/85

I. PURPOSE

The objective of these tests was to measure the differential and integral worths of the various Control and Shutdown Banks both individually and in overlap.

II. METHOD

Control Banks A, B, C and D and Shutdown Banks C, D, and E were each individually tested. In each case, demineralized water was used to continuously dilute the NC System boron concentration. The bank being tested was inserted in discrete increments to compensate for the change in reactivity. The change in reactivity due to each incremental insertion was measured by the Reactivity Computer. Differential and integral bank worth was then calculated. Boration and Control Bank withdrawal were used when measuring the worth of the Control Banks moving in overlap.

III. RESULTS

Table 6.4-1 summarizes the test results. All Acceptance Criteria were met. Each individual bank worth was within $\pm 10\%$ of the Westinghouse supplied predicted value. The overlap measurement was within $\pm 4\%$ of the sum of the four individual measured control bank worths. Figures 6.4-1 through 6.4-8 show the integral and differential worth curves for each of the tests. Figure 6.4-9 shows several typical reactivity changes caused by incremental rod insertion during dilution.

IV. CORRECTIVE ACTION

None

CONTROL ROD WORTH MEASUREMENT BY BORATION/DILUTION

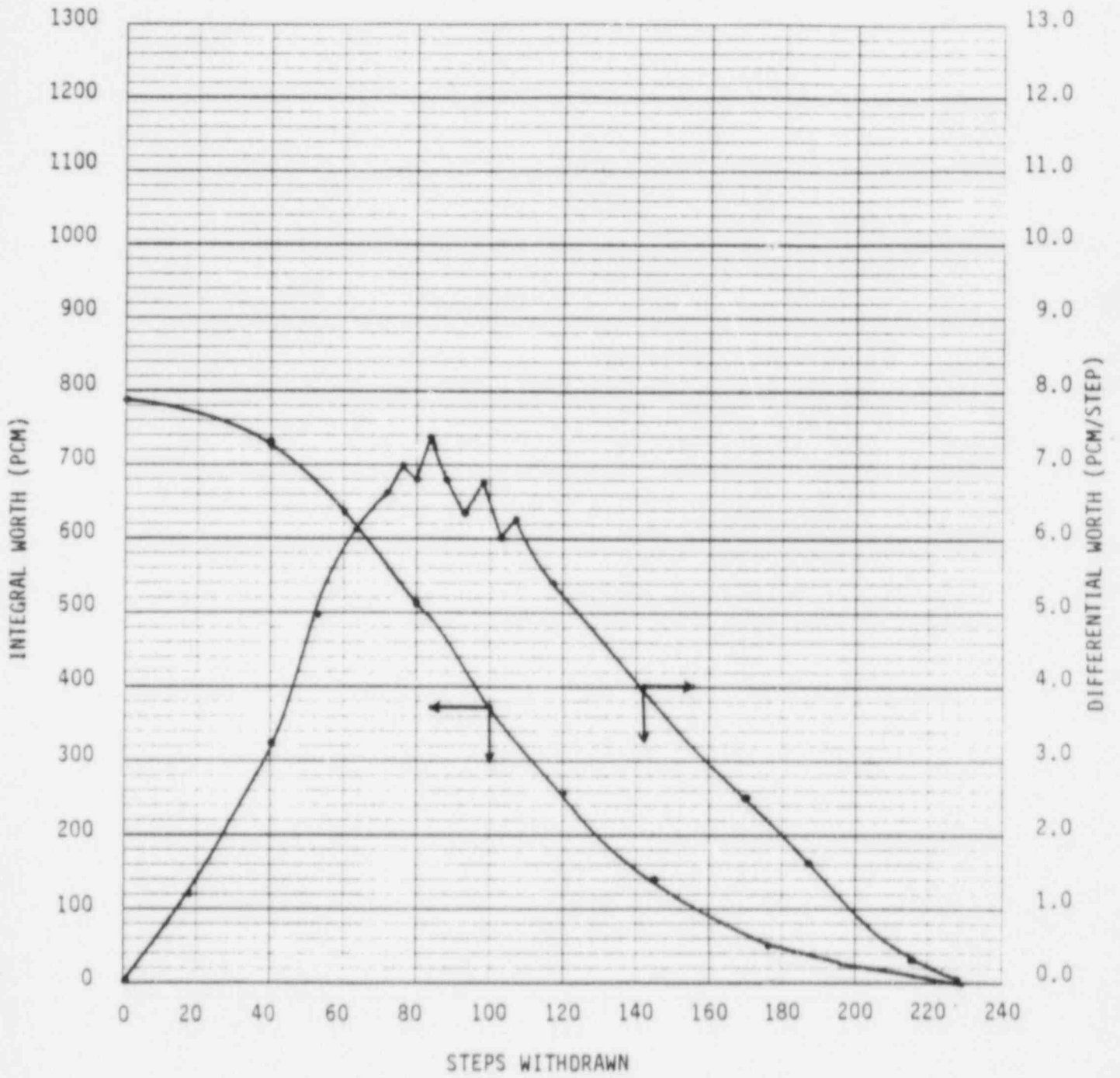
HZP INTEGRAL BANK WORTHS

Bank Tested	Measured Worth (pcm)	W Acceptance Criteria (pcm)	Duke Predicted Worth (pcm)
Control D	788	810 ± 81	739
Control C	1203	1270 ± 127	974
Control B	1171	1270 ± 127	1195
Control A	548	560 ± 56	331
Shutdown E	461	490 ± 49	307
Shutdown D	772	760 ± 76	561
Shutdown C	1099	1160 ± 116	934
Control Banks in Overlap	3778	3710 ± 148	3217

TABLE 6.4-1

CONTROL ROD WORTH MEASUREMENT
BY BORATION/DILUTION

INTEGRAL AND DIFFERENTIAL WORTH
OF CONTROL BANK D

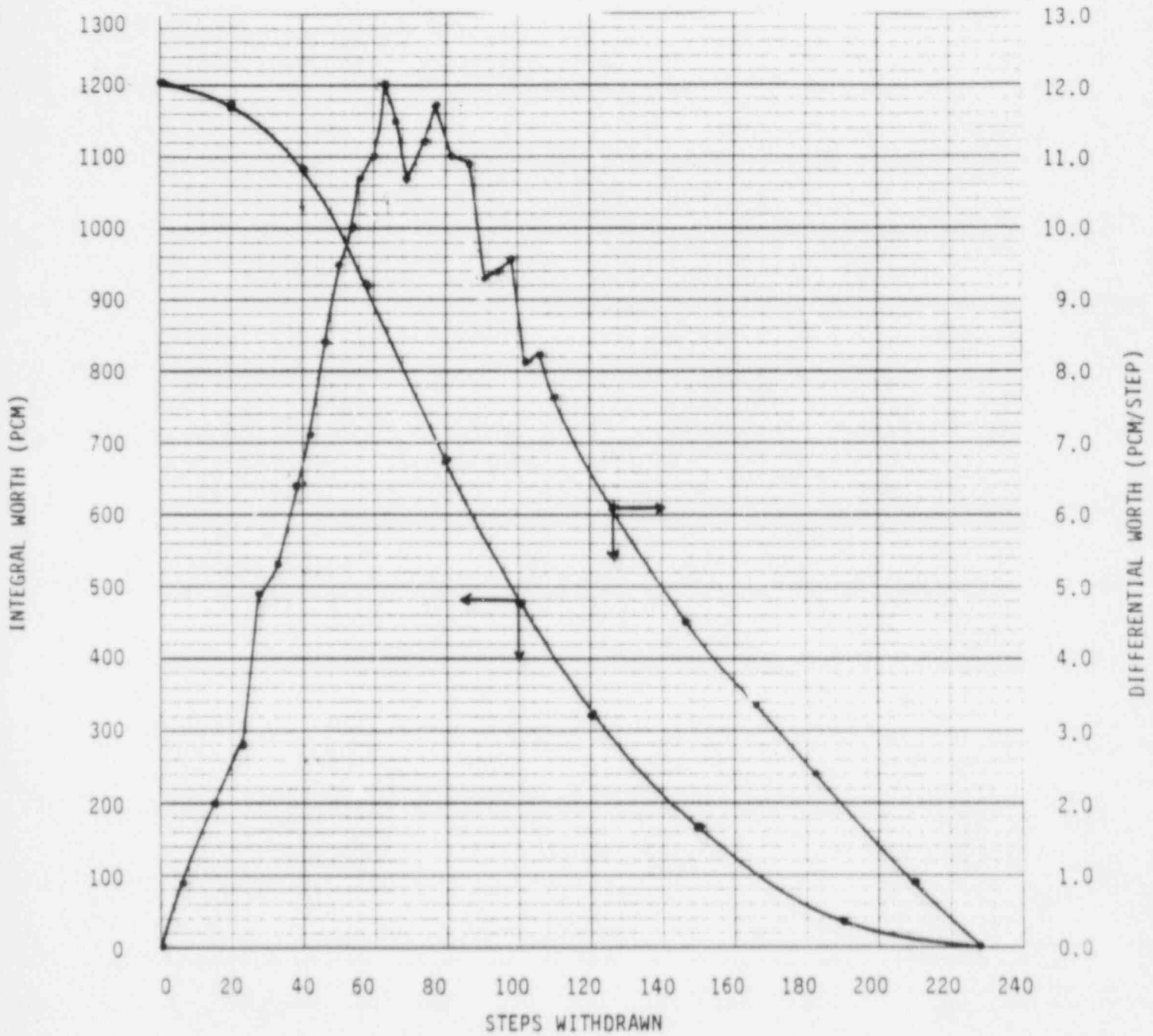


Note: All other Banks @ 228 steps withdrawn

Figure 6.4-1

CONTROL ROD WORTH MEASUREMENT
BY BORATION/DILUTION

INTEGRAL AND DIFFERENTIAL WORTH
OF CONTROL BANK C

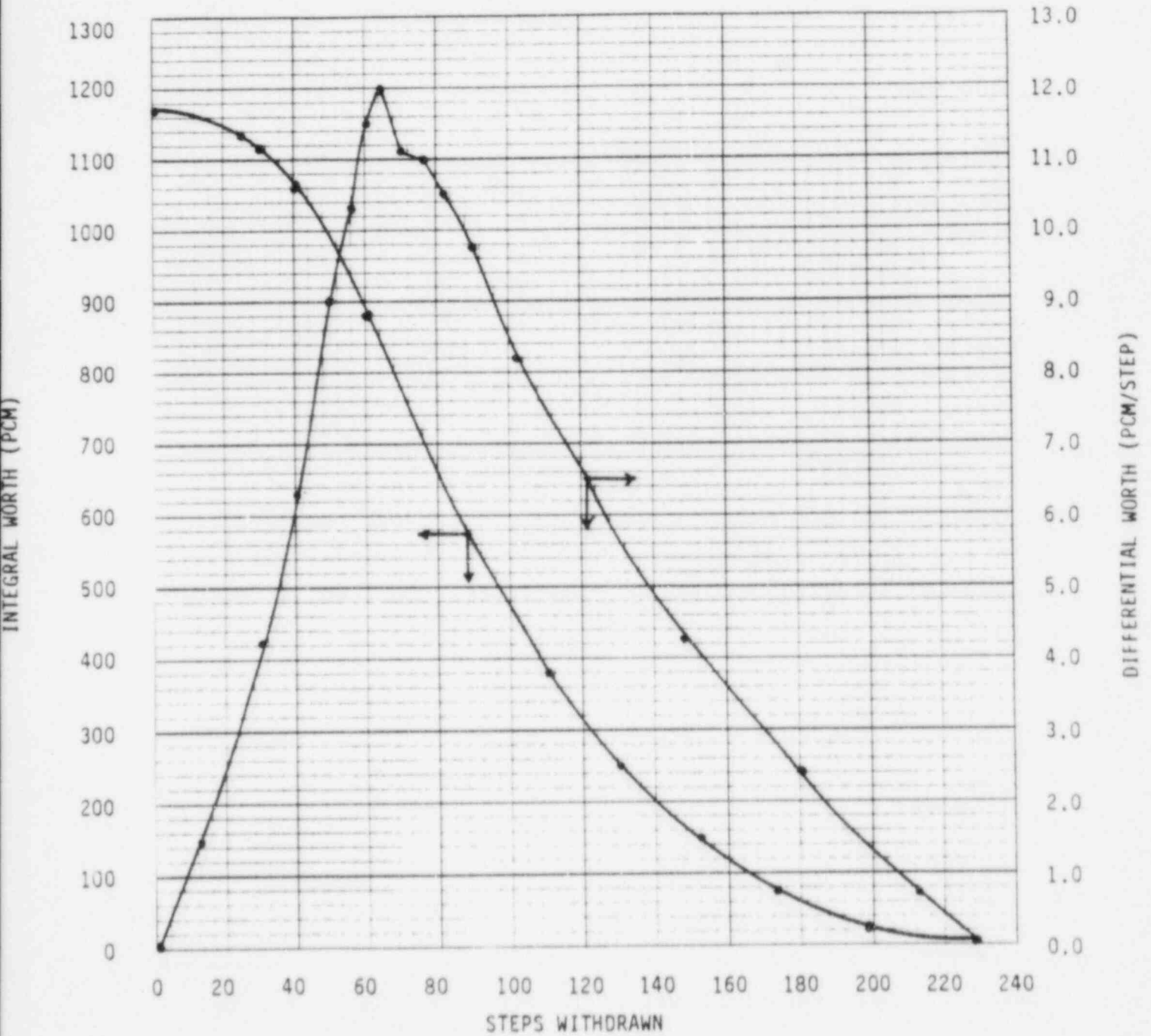


Note: ARO except Control Bank D @ 37 steps withdrawn

Figure 6.4-2

CONTROL ROD WORTH MEASUREMENT
BY BORATION/DILUTION

INTEGRAL AND DIFFERENTIAL WORTH
OF CONTROL BANK B

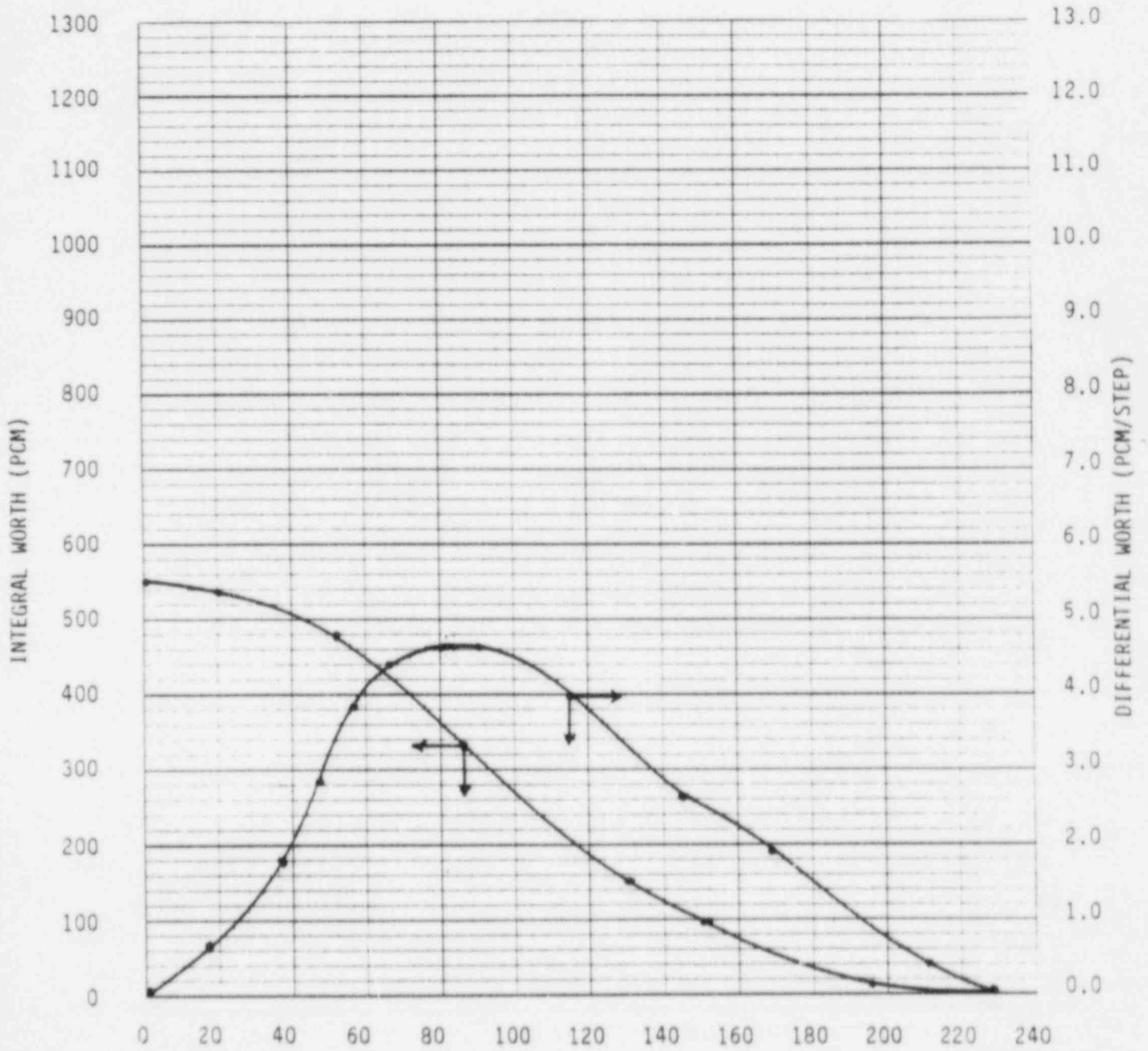


Note: ARQ except Control Banks C and D @ 0 steps withdrawn

Figure 6.4-3

CONTROL ROD WORTH MEASUREMENT
BY BORATION/DILUTION

INTEGRAL AND DIFFERENTIAL WORTH
OF CONTROL BANK A



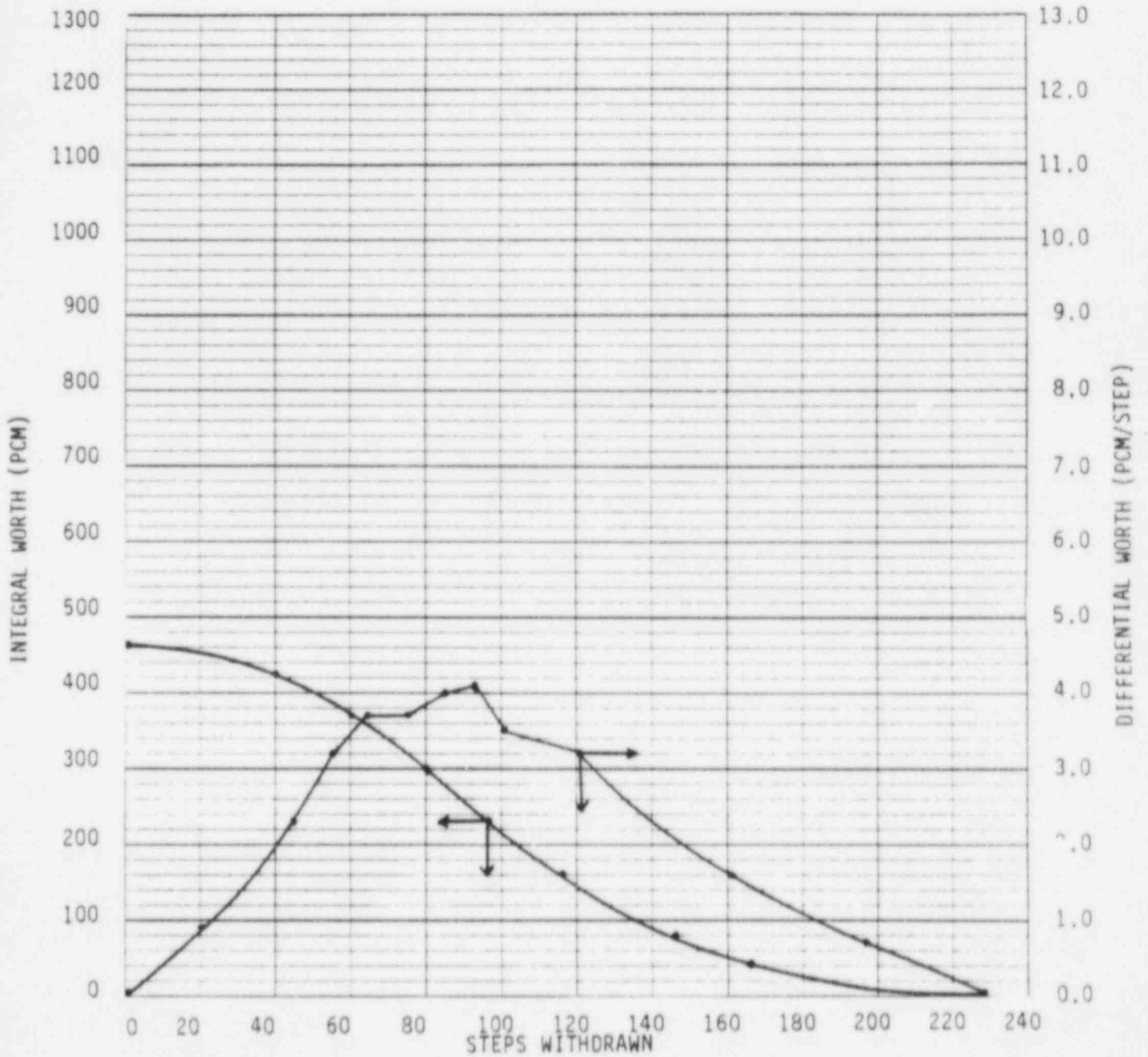
Note: ARO except Control Banks B, C and D @ 0 steps withdrawn

Figure 6.4-4

CATAWBA 1

CONTROL ROD WORTH MEASUREMENT
BY BORATION/DILUTION

INTEGRAL AND DIFFERENTIAL WORTH
OF SHUTDOWN BANK E

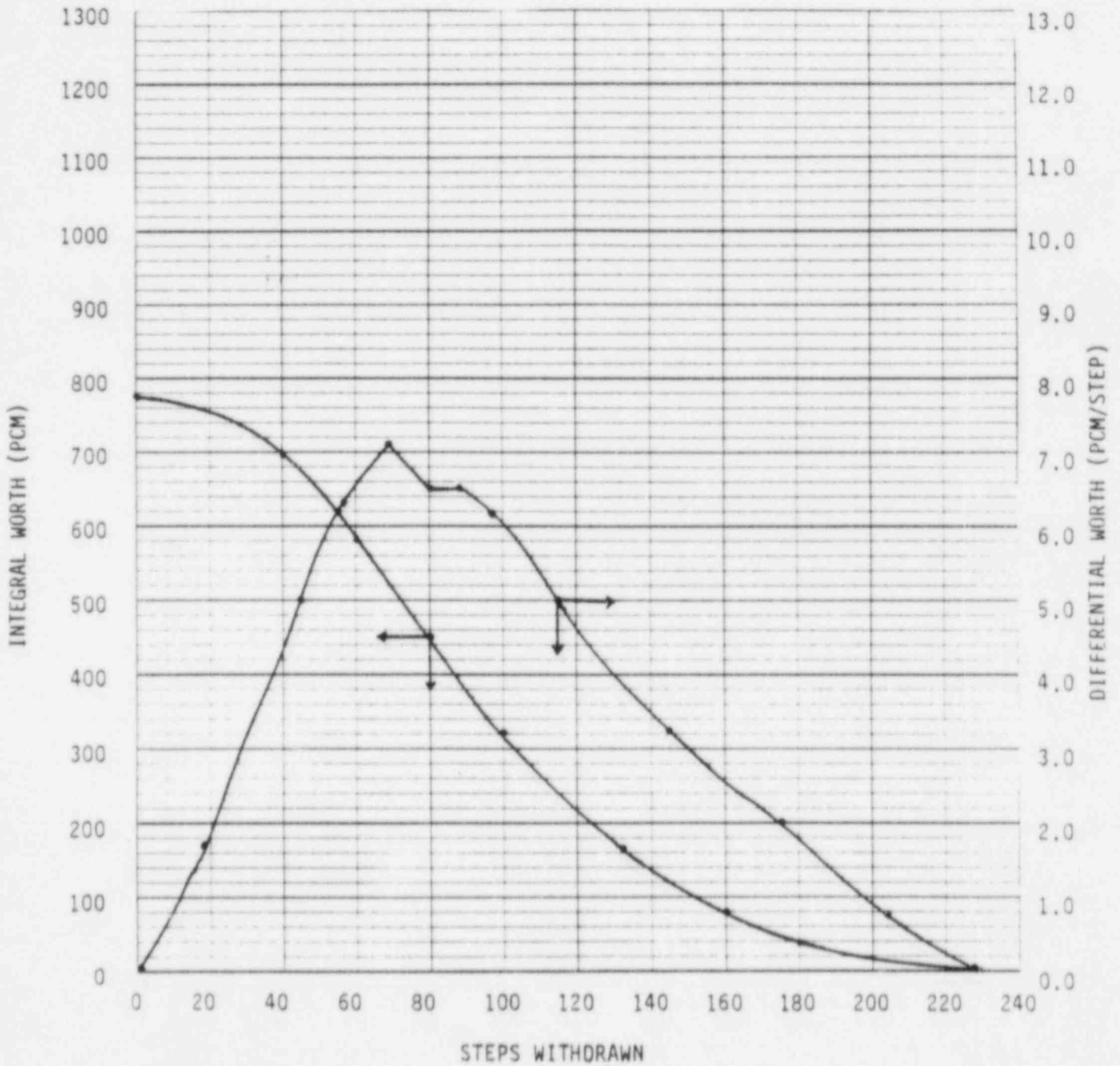


Note: All Control Banks @ 0 steps withdrawn.
Shutdown Banks A,B,C and D @ 228 steps withdrawn.

Figure 6.4-5

CONTROL ROD WORTH MEASUREMENT
BY BORATION/DILUTION

INTEGRAL AND DIFFERENTIAL WORTH
OF SHUTDOWN BANK D



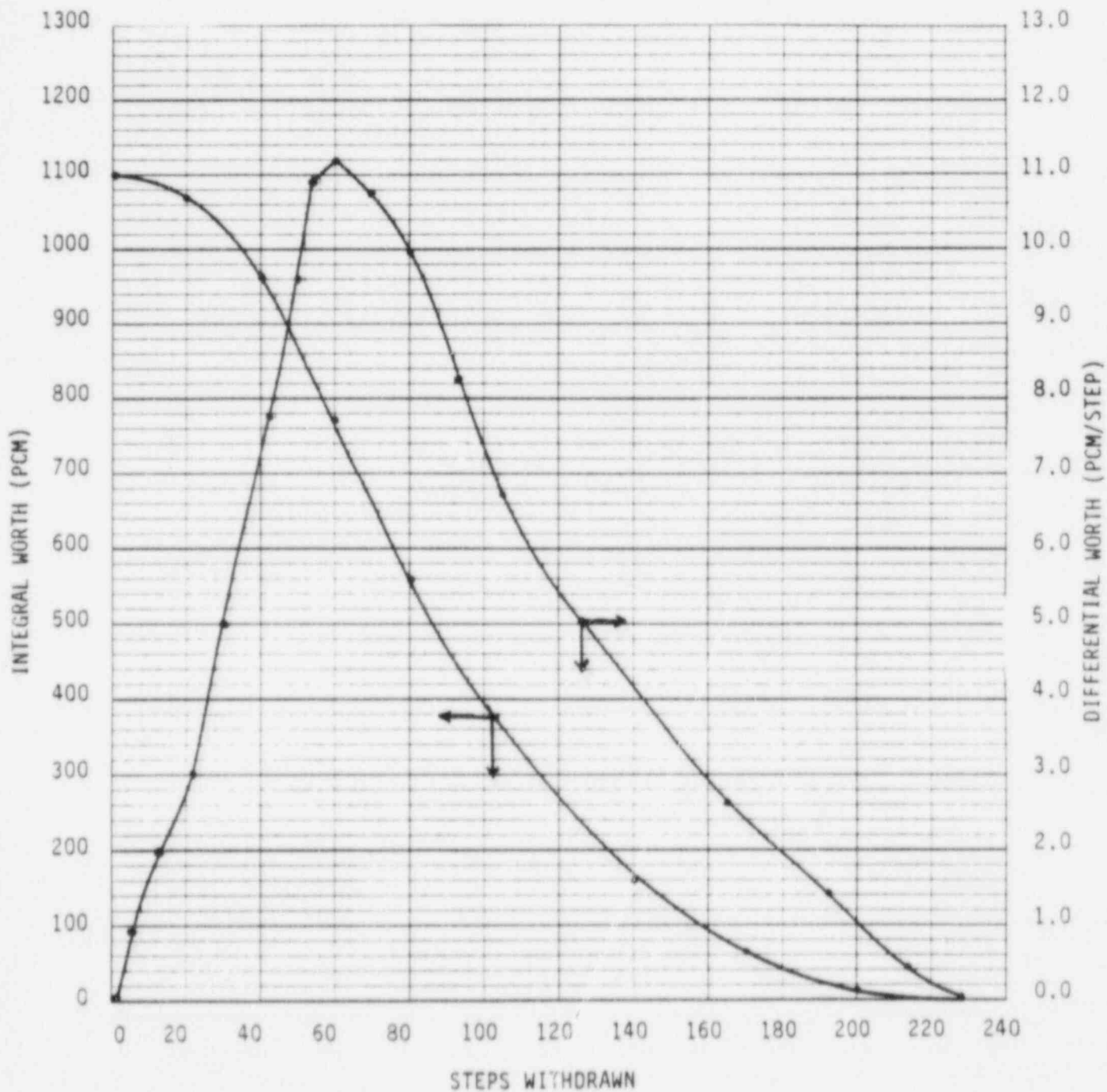
Note: All Control and Shutdown Bank E @ 0 steps withdrawn
Shutdown Banks A,B and C @ 228 steps withdrawn

Figure 6.4-6

CATAWBA 1

CONTROL ROD WORTH MEASUREMENT
BY BORATION/DILUTION

INTEGRAL AND DIFFERENTIAL WORTH
OF SHUTDOWN BANK C



Note: ARI except Shutdown Banks D and E @ 228 steps withdrawn

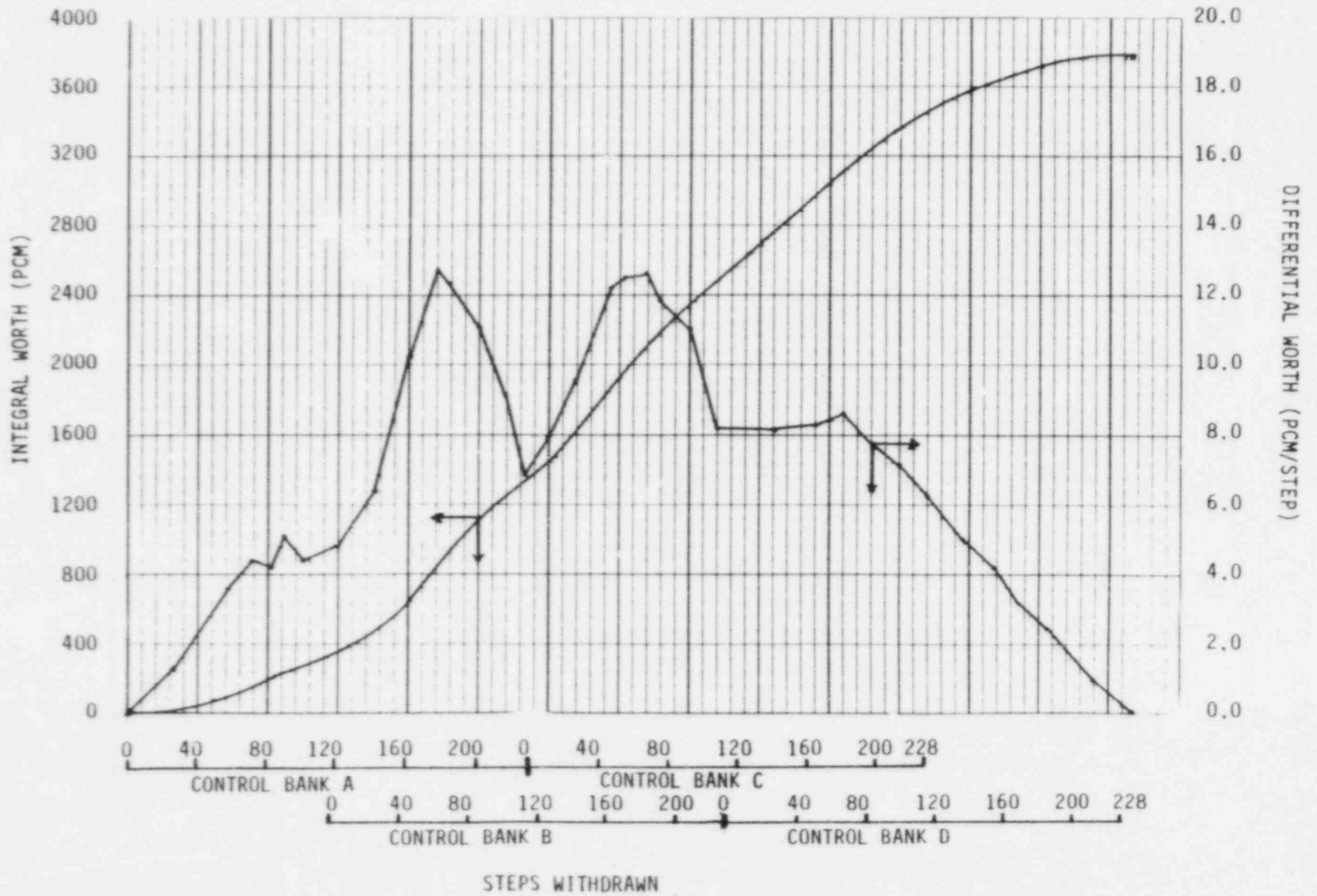
Figure 6.4-7

CONTROL ROD WORTH MEASUREMENT BY BORATION/DILUTION

INTEGRAL AND DIFFERENTIAL WORTH OF CONTROL BANKS IN OVERLAP

Figure 6.4-8

CATAMBA 1



CONTROL ROD WORTH MEASUREMENT BY BORATION/DILUTION
TYPICAL REACTIVITY TRACE

DILUTION OF CONTROL BANK A

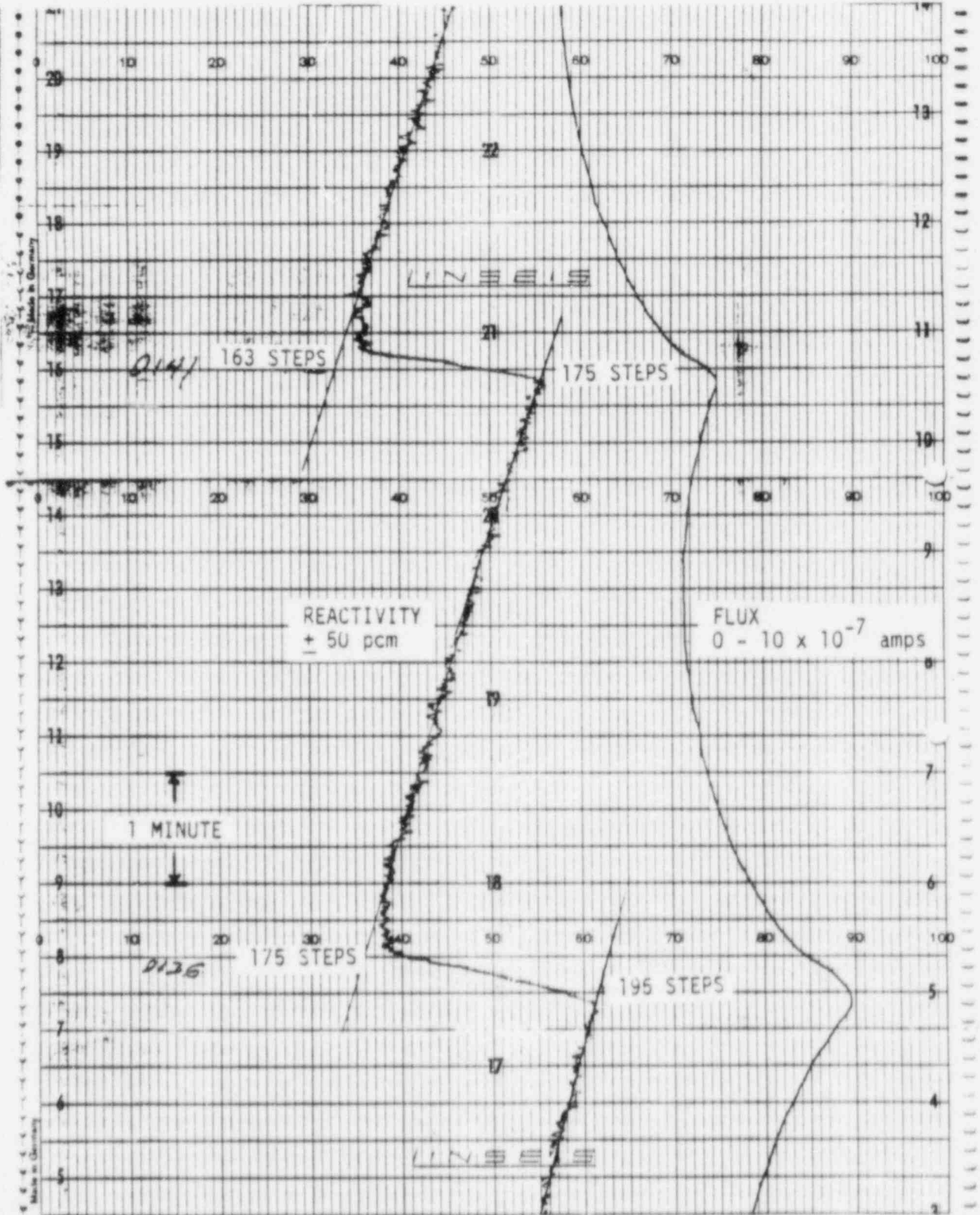


FIGURE 6.4-9

6.5 STUCK ROD WORTH MEASUREMENT - PT/1/A/4150/24

Date(s) Performed: 1/14/85 - 1/15/85

I. PURPOSE

This test was performed to verify that the Control Rod insertion limits specified in the Technical Specification provides adequate (1.3% $\Delta K/K$) shutdown margin with the most reactive rod (RCCA H14) stuck out of the core.

II. TEST METHOD

With all rod banks except for shutdown Banks B and A fully inserted RCCA H14 is withdrawn while compensating for the reactivity insertion with Shutdown Bank B. After RCCA H14 was fully withdrawn, a deboration was started to fully insert Shutdown Bank B and to insert Shutdown Bank A to approximately 30 steps withdrawn.

A boron endpoint measurement was to have been performed on Shutdown Bank A. However, criticality was established with Shutdown Bank A at 0 steps and the critical boron concentration was measured. The measured concentration was used in the calculations.

As soon as the measurement had been made, the reactor was manually tripped to trip RCCA H14 into the core. A boration of the reactor coolant system was initiated to achieve a boron concentration high enough to require all Shutdown Banks withdrawn for criticality.

The critical boron concentration prior to RCCA H14 withdrawal was used with the critical boron concentration for all-rods-in except H14 (N-1 configuration) to calculate the reactivity worth of the H14 withdrawal - Shutdown Banks A and B insertion. This was added to the sum of the measured worths of the Control Banks and Shutdown Banks C, D and E to calculate the worth of all rods minus the most reactive rod (H14).

III. RESULTS

The N-1 rod worth was calculated to be 7413.5 pcm, compared to the criterion of 6920 ± 692 pcm. The N-1 critical boron concentration was measured to be 402 ppmb, which was within the criterion of 377 ± 60 ppmb. The shutdown margin available at the HZP insertion limits was calculated to be 1.58% $\Delta K/K$. The results are summarized in Table 6.5-1.

IV. CORRECTIVE ACTION

No corrective action was required.

STUCK ROD WORTH MEASUREMENT

SUMMARY OF RESULTS

Critical Boron Concentration for Shutdown Banks A & B out, all other rods in	508 ppmb
Critical Boron Concentration for N-1 configuration	(-) 402 ppmb
Change in Critical Boron Concentration	106 ppmb
Differential Boron Worth (measured)	(x) 12.938 pcm/ppmb
Reactivity Worth of Change in Boron Concentration	1371.4 pcm
Sum of Worth of Control Banks, Shutdown Banks C, D, and E (measured)	(+) 6042.1 pcm
Worth of all rods minus the most reactive (H14)	7413.5 pcm
Worth of all rods above zero power insertion limits	(-) 580.4 pcm
Shutdown Margin at Zero power insertion limits	1583.1 pcm or 1.58% $\Delta K/K$

	Acceptance Criteria	Results
N-1 Critical Boron Concentration	377 \pm 60 ppmb	402 ppmb
N-1 Worth	6920 \pm 692 pcm	7413.5 pcm
Shutdown Margin	1.3% $\Delta K/K$	1.58% $\Delta K/K$

TABLE 6.5-1

CATAWBA 1

6.6 PSEUDO ROD EJECTION TEST (ZERO POWER) - TP/1/A/2150/06A

Date(s) Performed: 1/16/85 - 1/19/85

I. PURPOSE

The objectives of the Pseudo Rod Ejection Test performed at reactor power levels of zero and 3% F.P. were the following:

- A. To measure the reactivity worth of the highest worth inserted rod to verify that the postulated worth used in the rod ejection accident analysis is conservative.
- B. To verify that core peaking factors are within limits by a base case flux map taken with the Control Banks at their Hot Zero Power Insertion Limit.
- C. To verify that the maximum total Heat Flux Hot Channel Factor (F_Q) does not exceed 10.58 with the highest worth rod fully withdrawn by taking a flux map with rods at the HZP Insertion Limit and with rod D-12 fully withdrawn.
- D. To verify that the Rod Deviation Alarm actuates when the unit RCCA is moved to a position ≥ 12 steps from its bank's position.

II. METHOD

Reactor power was increased to a level of $\approx 3\%$ F.P. and a full core flux map was obtained with the Control Banks at their HZP Insertion Limit (Control Bank B @ 186 steps wd., Control Bank C @ 71 steps wd., and Control Bank D @ 0 steps wd.). The flux level was then reduced to bring reactor power to 0% F.P. and Control Bank D was positioned at 5 steps wd.

The lift coils for all Control Bank D rods except for rod D-12 were de-energized and the rod was withdrawn as the reactor coolant system was borated for reactivity compensation. A reactivity trace was obtained during this withdrawal to measure the integral worth of the rod.

As rod D-12 was withdrawn to a position above Bank D the actuation of the Rod Deviation Alarm was noted.

Once the rod was fully withdrawn flux level was again increased to raise reactor power to $\approx 3\%$ F.P. A full core flux map was obtained in this configuration.

Following reduction of reactor power to 0% F.P. the rod was inserted as the reactor coolant system was diluted for reactivity compensation. A reactivity trace was again obtained to measure the integral worth of the rod.

III. RESULTS

All Acceptance Criteria for the test were met. These are summarized as follows:

- A. The measured core peaking factors for the rods at the HZP Insertion Limits, F_Q and $F_{\Delta H}$, were within Tech Spec limits. The R_1 factor, based on $F_{\Delta H}$, was less than 1.0 as required by Tech Specs. These results are recorded on Table 6.6-1. The results of the "by assembly" calculations of F_Q and $F_{\Delta H}$ are presented on Figures 6.6-1 and 6.6-2.
- B. Only the Heat Flux Hot Channel factor was evaluated for the flux map taken with D-12 ejected. The measured maximum F_Q was below the limit specified by the safety analysis. The peaking factor results from the flux map are presented on Table 6.6-1 and Figures 6.6-1 and 6.6-2.
- C. The reactivity traces obtained during the withdrawal and subsequent reinsertion of rod D-12 yielded integral reactivity worths below the value used in the safety analysis (780 pcm). Portions of the reactivity traces are displayed on Figure 6.6-3 along with the measurement results. In addition to the measurement by reactivity computer trace analysis the worth of D-12 was calculated using the critical reactor coolant system boron concentrations with D-12 aligned with Bank D (@ 5 steps wd.) and fully withdrawn. A differential boron worth was calculated using this data and a rod worth was derived using the two boron endpoints. This value is noted on Figure 6.6-3.
- D. The Rod Deviation Alarm which is designed to actuate when a unit RCCA is detected to be ≥ 12 steps away from its bank position was verified to function properly. The alarm was received when Rod D-12 was withdrawn to 11.5 steps above Control Bank D.

IV. CORRECTIVE ACTIONS

None required.

PSEUDO ROD EJECTION TEST (ZERO POWER)
 SUMMARY OF HOT CHANNEL FACTORS FROM SNC CORE PROGRAM

FLUX MAP I.D.	ROD D-12 CONFIG.	MAXIMUM MEASURED F_Q	TECH SPEC LIMIT	CORE LOCATION	MAXIMUM MEASURED $F_{\Delta H}$	CALCULATED R_1
				AXIAL LOCATION		
FCM/1/01/003	ALIGNED WITH BANK D	3,4240	4.64	G-14	1.7366	0,9042
				15		
FCM/1/01/004	EJECTED	8.0279	4.64*	C-13	4.9525	2.5785**
				25		

*NOTE: Tech Spec Limit for F_Q not applicable for Ejected Rod Case. Test acceptance criterion is 10.58 as set forth by Westinghouse Safety Review Criterion in DCP/DDP-SU-3.1.1

**NOTE: Calculated R_1 based on Measured $F_{\Delta H}$ analysis not required for the Ejected Rod Case.

Table 6.6-1

CATAMBA 1

PSEUDO ROD EJECTION TEST (ZERO POWER)
 MEASURED ASSEMBLY $F_{\Delta H}$ VALUES - BASE CASE vs EJECTED CASE

Base Case Map

FCM/1/01/003

	01	02	03	04	05	06	07	08	09	10	11	12	13	14	15
A															
B															
C															
D															
E															
F															
G															
H															
J															
K															
L															
M															
N															
P															
R															

FCM/1/01/004

Ejected Rod Map

	01	02	03	04	05	06	07	08	09	10	11	12	13	14	15
A															
B															
C															
D															
E															
F															
G															
H															
J															
K															
L															
M															
N															
P															
R															

Figure 6.6-1

PSEUDO ROD EJECTION TEST (ZERO POWER)
 MEASURED ASSEMBLY F_Q VALUES - BASE CASE vs EJECTED CASE

Base Case Map

FCM/1/01/003

	1.434	2.080	2.192	1.974	2.442	2.115	2.413	2.012	2.340	2.249	1.585			
	1.372	1.738	1.617	1.741	2.007	1.889	2.047	1.966	2.136	1.831	1.798	1.937	1.530	
	1.898	1.518	1.027	1.604	1.772	2.138	2.021	2.246	1.885	1.711	1.188	1.734	2.125	
	1.360	2.013	1.583	1.530	1.603	1.975	1.781	2.128	1.923	1.128	1.176	2.070	1.784	1.295
	1.794	1.829	1.898	1.684	1.952	1.672	1.798	1.628	1.975	1.850	2.128	1.813	2.085	2.080
	2.161	2.353	1.877	2.059	1.732	1.872	1.411	1.372	1.525	1.984	1.866	2.216	2.010	2.584
	2.036	2.047	2.047	1.884	2.060	1.687	1.362	0.650	1.381	1.647	2.108	1.952	2.165	2.109
	1.989	2.333	1.889	1.075	1.827	1.814	1.423	1.331	1.905	1.847	1.811	2.176	1.988	2.282
	1.793	1.939	1.978	1.775	1.991	1.725	1.868	1.577	1.848	1.700	1.971	1.754	2.015	1.939
	1.439	2.123	1.644	1.558	1.600	1.949	1.704	1.941	1.751	1.943	1.555	1.566	1.738	2.231
	1.879	1.471	0.974	1.497	1.625	1.940	1.760	2.013	1.700	1.500	1.070	1.583	2.011	
	1.286	1.522	1.441	1.548	1.843	1.787	1.965	1.738	1.789	1.804	1.508	1.828	1.408	
		1.306	1.927	2.011	1.767	2.205	1.649	2.148	1.718	2.019	1.876	1.325		
				1.342	1.728	1.982	1.852	1.946	1.782	1.388				
01	02	03	04	05	06	07	08	09	10	11	12	13	14	15
				1.227	1.898	2.347	2.864	3.438	3.502	3.149				
		0.879	1.332	1.573	1.648	2.532	2.353	3.480	3.453	5.185	5.828	4.356		
	0.741	1.101	1.051	1.211	1.662	1.865	2.322	2.731	3.767	4.532	5.734	6.703	4.215	
	1.014	0.863	0.616	1.071	1.364	1.978	2.110	3.021	3.122	4.652	6.695	5.532	5.401	
	0.566	0.916	0.761	0.808	0.980	1.249	1.821	2.184	2.104	3.142	3.398	4.581	4.384	5.088
	0.698	0.702	0.806	0.786	1.080	1.043	1.424	1.547	2.197	2.214	3.209	3.505	3.997	3.570
	0.751	0.873	0.707	0.897	0.874	1.065	0.946	1.176	1.487	2.270	2.312	3.062	2.708	3.544
	0.735	0.659	0.724	0.722	0.904	0.783	0.766	0.451	1.189	1.546	2.219	2.086	2.303	2.212
	0.686	0.778	0.813	0.798	0.888	0.807	0.878	0.789	0.898	1.149	1.864	2.028	1.814	2.462
	0.592	0.562	0.589	0.591	0.705	0.623	0.805	0.778	1.054	1.049	1.447	1.412	1.747	1.671
	0.426	0.601	0.497	0.503	0.534	0.698	0.668	0.880	0.832	1.025	0.969	1.112	1.297	1.705
		0.591	0.469	0.312	0.488	0.540	0.734	0.694	0.869	0.783	0.794	0.612	1.074	1.413
		0.348	0.516	0.248	0.278	0.372	0.408	0.574	0.678	0.795	0.774	0.818	1.084	0.964
		0.380	0.538	0.616	0.580	0.752	0.636	0.850	0.706	0.967	0.982	0.732		
FCM/1/01/004				0.416	0.577	0.652	0.668	0.720	0.772	0.619				

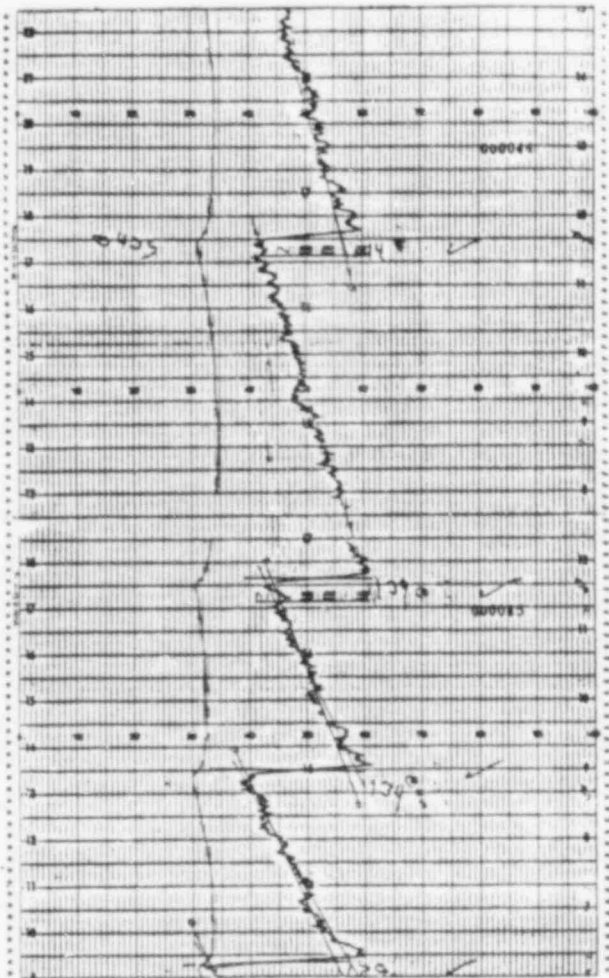
Ejected Rod Map

Figure 6.6-2

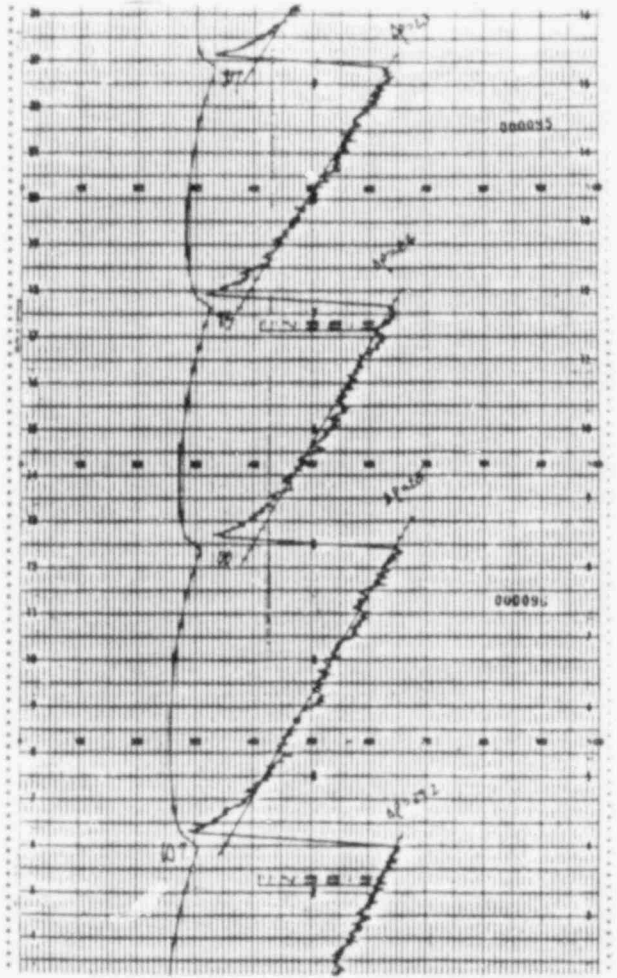
CATAWBA 1

PSEUDO ROD EJECTION TEST (ZERO POWER)
 ROD D-12 REACTIVITY WORTH MEASUREMENTS

ROD WITHDRAWAL DURING BORATION
 (typical trace)



ROD INSERTION DURING DILUTION
 (typical trace)



Measured Integral Worth = 694.50 pcm

Worth Corrected for 10%
 Uncertainty = 763.95 pcm

Measured Integral Worth = 429.60 pcm

Worth Corrected for 10%
 Uncertainty = 472.56 pcm

Average Measured Integral Worth = 618.26 pcm

D-12 Worth by Boron Endpoints = 707.70 pcm

Postulated Worth for Accident Analysis = 780.00 pcm

7.0 NATURAL CIRCULATION VERIFICATION - TP/1/A/2650/13

Date(s) Performed: 1/19/85 - 1/20/85

I. PURPOSE

The objectives of the Natural Circulation Verification Test were:

- A. To demonstrate the ability of the NSSS to remove heat via natural circulation of the primary coolant.
- B. To verify that pressurizer pressure and level controls can respond automatically to a loss of forced circulation and can maintain NC pressure within acceptable limits.
- C. To verify that adequate control over steam generator level and feedwater flow can be maintained under natural circulation conditions.
- D. To provide operator training and experience in unit operation under natural circulation conditions and the use of appropriate emergency procedures.

II. METHOD

With the Reactor held at ~ 3% Rated Thermal Power all four NC pumps were simultaneously tripped. The establishment of natural circulation was verified by observation of W/R NC loop temperatures and core exit thermocouples. Stable natural circulation conditions were maintained for the 30 minute minimum period while data was gathered. Recovery was then initiated by inserting the Control Banks to decrease Reactor power and restarting the NC pumps. In order to perform this test the OPAT and OTAT trip functions were defeated and UHI isolation valves closed and gagged throughout the test.

III. RESULTS

All Acceptance Criteria for this test were successfully met as described below:

- A. No Reactor trip or Safety Injection occurred or was required. Pertinent parameters remained within all safety guidelines specified within the procedure.
- B. Establishment of steady state natural circulation was verified by the simultaneous achievement of the following conditions over a 30 minute time interval (see Figure 7.0-1):
 1. The highest calculated loop ΔT remained $\leq 65^{\circ}\text{F}$. During the test, a maximum ΔT of 29.2°F was observed.

2. The highest loop ΔT stabilized such that a $\leq \pm 3^\circ\text{F}$ change occurred during the test interval. Only a 0.7°F maximum change was observed.
3. The ΔT between the highest incore T/C reading and the lowest W/R T_{COLD} reading remained $\leq 65^\circ\text{F}$. The highest ΔT observed during the test was only 34°F .
4. The ΔT between the highest incore T/C reading and the lowest W/R T_{COLD} reading stabilized such that a $\leq \pm 3^\circ\text{F}$ change occurred during the test interval. Only a 1.6°F maximum change was observed.
5. The NC subcooling margin remained $\geq 25^\circ\text{F}$. Minimum margin observed was 47.2°F .
6. Pressurizer pressure remained at 2235 ± 25 psig. The highest and lowest pressures observed were 2244 and 2239 psig respectively.
7. Pressurizer level remained above the level recorded when the NC pumps were tripped. The lowest level observed during natural circulation was 35% whereas the level recorded at the time of tripping the NC pumps was 28%.
8. All Steam Generator levels remain above 25% (N/R). The lowest level observed during the test interval was 37%.

IV. CORRECTIVE ACTION

During recovery from natural circulation T_{AVE} dropped below the Tech Spec required minimum temperature of 541°F . This excessive cooldown following the insertion of the Control Banks was due to the inability to restart NC Pump A because of a high standpipe level indication. The Unit was taken to Mode 3 per the applicable Tech Spec action item. Criticality was re-established ~ 12 hours later.

NATURAL CIRCULATION VERIFICATION TEST
TYPICAL NC TEMPERATURE TREND

NC LOOP C

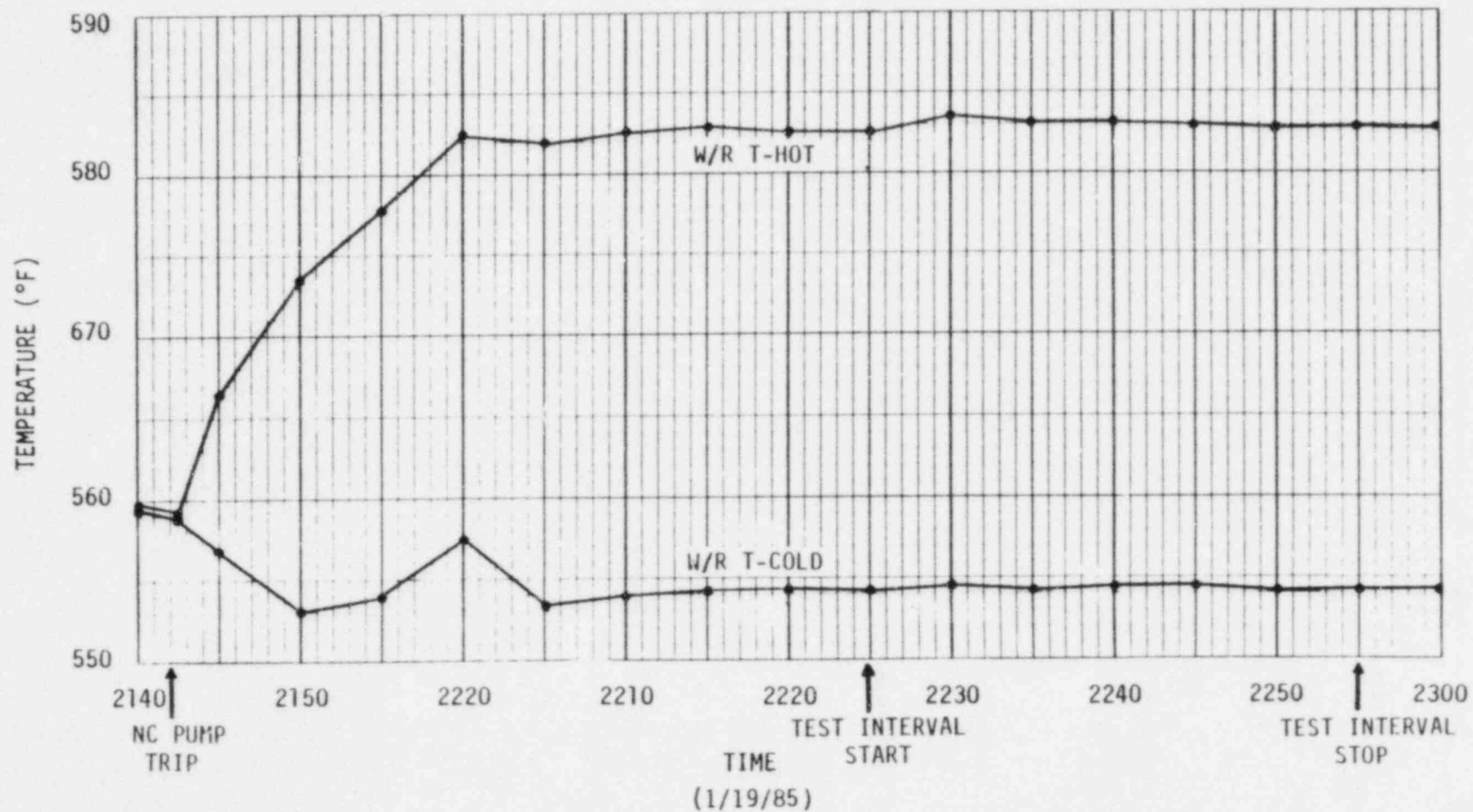


Figure 7.0-1

8.0 POWER ESCALATION TESTING - CORE PERFORMANCE/PLANT PERFORMANCE

The core performance/plant response areas of the power escalation testing program was set up to gather data in areas of core physics as well as plant response due to induced transients. This data was used to verify proper system design and operation. This testing included the following:

- Core Power Distribution
- Unit Load Steady State
- Doppler Only Power Coefficient Verification
- Pseudo Ejected Rod Test
- Unit Load Transient Test
- Unit Loss of Electrical Load Test
- Turbine Trip Test
- Station Blackout Test
- Preliminary Incore and Nuclear Instrumentation Systems Correlation
- Below Bank Rod Test
- Calorimetric NC Flow Measurement
- Large Load Reduction Test

These tests are described on the following pages.

8.1 CORE POWER DISTRIBUTION - PT/1/A/4150/05

Date(s) Performed:	3/5/85	30% PWR
	3/12/85 - 3/13/85	50% PWR
	4/2/85 - 4/3/85	75% PWR
	4/8/85 - 4/9/85	90% PWR
	4/12/85	100% PWR

I. PURPOSE

The purpose of the Core Power Distribution Test was as follows:

- A. To obtain and analyze core power distribution for various control rod configurations and power levels.
- B. To verify that the actual core power distribution is maintained within the power distribution limits of the CNS Tech Specs and/or Westinghouse predictions.

II. METHODS

At each of the 30%, 50%, 75%, 90% and 100% power levels a Full Core Flux map was obtained. SNC-Core software was used to analyze the data from each full core flux map.

III. RESULTS

A. 30% Power Level Plateau

- (1) All Acceptance Criteria for Incore Quadrant tilt, heat flux hot channel factor, total Reactor Coolant (NC) Flow, and percent error in the relative assembly powers were met.
- (2) The F_{xy} limit at Rated Thermal Power (RTP) was exceeded, requiring additional F_{xy} surveillance during power escalation.
- (3) The Quadrant tilt was greater than desired but less than at zero power. Based on this decreasing tilt Westinghouse recommended escalation to 50% power.
- (4) See Table 8.1-1 for flux map results at 30% power.

B. 50% Power Level Plateau

- (1) The Acceptance Criteria for the Percent Error in the relative assembly powers ($F_{\Delta H}^N$) was violated as follows:

<u>Assembly Location</u>	<u>$F_{\Delta H}^N$</u>	<u>Actual % Error</u>	<u>% Error Allowed</u>
B-13	1.226	-11.4	± 10

- (2) All other Acceptance Criteria were met.

(3) The Quadrant tilt was higher than expected but less than at previous power levels. Westinghouse reviewed the data to date and recommended continued power escalation.

(4) See Table 8.1-2 for flux map results at 50% power.

C. 75% Power Level Plateau

(1) All acceptance criteria were met.

(2) The F_{xy} limit at RTP was exceeded requiring additional F_{xy} surveillance during power escalation.

(3) See Table 8.1-3 for flux map results at 75% power.

D. 90% Power Level Plateau

(1) All Acceptance Criteria were met.

(2) The F_{xy} limit at RTP was exceeded requiring additional F_{xy} surveillance during power escalation.

(3) See Table 8.1-4 for flux map results at 90% power.

E. 100% Power Level Plateau

(1) All Acceptance Criteria were met.

(2) The F_{xy} limit at RTP was exceeded requiring additional F_{xy} surveillance during power escalation.

(3) See Table 8.1-5 for flux map results at 100% power.

IV. CORRECTIVE ACTIONS

Due to the F_{xy} limit at Rated Thermal Power being exceeded at the 30%, 75%, 90% and 100% power levels, additional F_{xy} surveillance was conducted as required by the CNS Tech Specs.

CORE POWER DISTRIBUTION
CORE POWER DISTRIBUTION RESULTS
30% POWER TEST

Test Date: 3/5/85
 Map ID: FCM/1/01/006
 Power Level: 30%
 Boron Concentration: 773 ppm
 Rod Position: Control Bank D 162 Steps withdrawn
 Measured NC Flow: 417,460 gpm
 Measured R: 0.7749
 Maximum Measured F_{xy}^* : 1.6334 Axial Location 13 Core Location D-12
 Maximum F_Q : 2.2859 Axial Location 25 Core Location D-12
 Maximum F_Z : 1.4427 Axial Location 25
 Maximum $F_{\Delta H}^N$: 1.3939 Core Location G-14
 Maximum $F_{\Delta H}^N$ Error
 (from predicted): +8.62% Core Location P-04
 Total Core Axial Offset: -11.75%

Quadrant Power Tilt Ratios (Normalized):

	<u>Top Half of Core</u>	<u>Bottom Half of Core</u>
Quadrant 1:	1.00240	0.99955
Quadrant 2:	1.02126	1.02828
Quadrant 3:	0.98046	0.99480
Quadrant 4:	0.99588	0.97737

*In locations unexcluded by Technical Specifications.

CORE POWER DISTRIBUTION

CORE POWER DISTRIBUTION RESULTS
50% POWER TEST

Test Date: 3/12/85 - 3/13/85
 Map ID: FCM/1/01/010
 Power Level: 49%
 Boron Concentration: 709 ppm
 Rod Position: Control Bank D 212 Steps Withdrawn
 Measured NC Flow: 417,025 gpm
 Measured R: 0.8426
 Maximum Measured F_{xy}^* : 1.6217 Axial Location 51 Core Location B-12
 Maximum F_Q : 2.1083 Axial Location 32 Core Location C-13
 Maximum F_Z : 1.3610 Axial Location 33
 Maximum $F_{\Delta H}^N$: 1.4485 Core Location B-12
 Maximum $F_{\Delta H}^N$ Error
 (from predicted): -11.39% Core Location B-11
 Total Core Axial Offset: +4.34%

Quadrant Power Tilt Ratios (Normalized):

	<u>Top Half of Core</u>	<u>Bottom Half of Core</u>
Quadrant 1:	1.01300	1.00165
Quadrant 2:	1.01955	1.02242
Quadrant 3:	0.98392	0.99387
Quadrant 4:	0.98354	0.98205

*In locations unexcluded by Technical Specifications.

CORE POWER DISTRIBUTION

CORE POWER DISTRIBUTION RESULTS
75% POWER TEST

Test Date: 4/2/85 - 4/3/85
 Map ID: FCM/1/01/019
 Power Level: 75%
 Boron Concentration: 641 ppm
 Rod Position: Control Bank D 160 Steps Withdrawn
 Measured NC Flow: 404,470 gpm
 Measured R: 0.8901
 Maximum Measured F_{xy}^* : 1.7365 Axial Location 51 Core Location G-14
 Maximum F_Q : 2.3474 Axial Location 22 Core Location D-12
 Maximum F_Z : 1.5138 Axial Location 23
 Maximum $F_{\Delta H}^N$: 1.4314 Core Location G-14
 Maximum $F_{\Delta H}^N$ Error
 (from predicted): -6.32% Core Location F-15
 Total Core Axial Offset: 21.994%

Quadrant Power Tilt Ratios (Normalized):

	<u>Top Half of Core</u>	<u>Bottom Half of Core</u>
Quadrant 1:	0.99680	0.99784
Quadrant 2:	1.01599	1.03031
Quadrant 3:	0.99048	0.99098
Quadrant 4:	0.99673	0.98088

*In locations unexcluded by Technical Specifications.

CORE POWER DISTRIBUTION

CORE POWER DISTRIBUTION RESULTS
90% POWER TEST

Test Date: 4/8/85 - 4/9/85
 Map ID: FCM/1/01/024
 Power Level: 90%
 Boron Concentration: 637 ppm
 Rod Position: Control Bank D 217 Steps Withdrawn
 Measured NC Flow: 397,684 gpm
 Measured R: 0.905
 Maximum Measured F_{xy}^* : 1.5800 Axial Location 13 Core Location D-12
 Maximum F_Q : 2.0915 Axial Location 25 Core Location D-12
 Maximum F_Z : 1.3728 Axial Location 25
 Maximum $F_{\Delta H}^N$: 1.391 Core Location G-14
 Maximum $F_{\Delta H}^N$ Error
 (from predicted): -6.5% Core Location P-04
 Total Core Axial Offset: -7.875%

Quadrant Power Tilt Ratios (Normalized):

	<u>Top Half of Core</u>	<u>Bottom Half of Core</u>
Quadrant 1:	0.99961	0.99683
Quadrant 2:	1.01720	1.02907
Quadrant 3:	0.98475	0.99320
Quadrant 4:	0.99844	0.98089

*In locations unexcluded by Technical Specifications.

CORE POWER DISTRIBUTION

CORE POWER DISTRIBUTION RESULTS
100% POWER TEST

Test Date: 4/12/85
 Map ID: FCM/1/01/025
 Power Level: 100%
 Boron Concentration: 618 ppm
 Rod Position: Control Bank D 207 Steps Withdrawn
 Measured NC Flow: 397,090 gpm
 Measured R: 0.9535
 Maximum Measured F_{xy}^* : 1.5835 Axial Location 13 Core Location D-12
 Maximum F_Q : 2.0926 Axial Location 25 Core Location P-11
 Maximum F_{Δ} : 1.3581 Axial Location 25
 Maximum $F_{\Delta H}^N$: 1.4235 Core Location P-11
 Maximum $F_{\Delta H}^N$ Error
 (from predicted): -9.08% Core Location D-14
 Total Core Axial Offset: -7.709%

Quadrant Power Tilt Ratios (Normalized):

	<u>Top Half of Core</u>	<u>Bottom Half of Core</u>
Quadrant 1:	0.99426	0.99427
Quadrant 2:	1.02055	1.03070
Quadrant 3:	0.98731	0.988779
Quadrant 4:	0.99788	0.98724

*In locations unexcluded by Technical Specifications.

8.2 UNIT LOAD STEADY STATE - PT/1/B/4150/16

Date(s) Performed:	1/19/85 - 1/21/85, 5/30/85	0% F.P.
	1/22/85 - 1/26/85, 6/11/85	20% F.P.
	2/5/85, 3/8/85 - 3/9/85	30% F.P.
	3/17/85 - 3/22/85, 6/12/85	50% F.P.
	3/31/85 - 4/1/85, 6/24/85	75% F.P.
	4/8/85 - 4/9/85	90% F.P.
	4/12/85, 6/26/85, 8/8/85	100% F.P.

I. PURPOSE

- A. To measure important primary and secondary side steady state parameters during power escalation and compare them to design predictions.
- B. To demonstrate the capability of major plant control systems for maintaining normal stable conditions.
- C. To perform a cross-check verification of feedwater and main steam flow indications.
- D. To project full power NC ΔT and T-AVG from 75% F.P.
- E. To check the full power NC ΔT and T-AVG at 100% F.P.
- F. To gather data for adjusting the T-AVG program (T-REF) to achieve design Steam Generator outlet pressure.
- G. To gather data on turbine impulse pressure versus power for adjusting reactor control and turbine control systems.
- H. To gather Loose Parts Monitoring System baseline data.

II. METHOD

At each test plateau data was obtained for various primary and secondary side parameters using the OAC. These parameters included NC temperatures, PZR level, S/G level and pressure, CF flow and temperature, and turbine impulse pressure. The data was plotted at each power level and compared to the expected response (see Figures 8.2-1 through 8.2-8). Also a comparison of steam vs. feedwater flows for each S/G was made at each power level to determine if steam flow indication adjustment was necessary.

Full power T-AVG and ΔT was extrapolated at 75% F.P. and checked at 100% F.P. This information was used to adjust plant control systems and to verify that full load T-AVG would not exceed the design limit. Similarly, full power S/G outlet pressure was extrapolated while at 50 and 75% F.P. and checked at 100% F.P. This provided information necessary to adjust the T-AVG (T-REF) program.

Loose Parts Noise Monitoring System baseline data was recorded at each power level.

III. RESULTS

All Acceptance Criteria for this test were met as follows:

- A. The NSSS displayed stable behavior in all cases.
- B. The recorded parameters followed design predictions well in most cases. At 20 and 30% several parameters fell slightly outside of their allowance bands. These deviations were evaluated and found to be due to instrumentation problems or operation under manual rod control at a T-AVG below T-REF. At 50% and higher power levels all measured parameters fell within their allowance bands. The Acceptance Criterion was revised such that it was only applicable to data obtained at or above 50% F.P. Below this level the allowance bands serve as guidelines only.
- C. A comparison between feedwater and steam flows was made at each power level. This data was transmitted to IAE who performed the necessary instrument calibrations.
- D. The extrapolated and measured values of full power T-AVG and ΔT were transmitted to IAE. At full power a T-AVG of ≈ 590.4 °F and a ΔT of ≈ 57.4 °F existed. This was within the design limit of 592.5 °F T-AVG.
- E. Initially, full power S/G outlet pressures were found to be ≈ 40 psi too high. This necessitated a reduction in full power T-AVG to 586.0 °F to achieve an outlet pressure within ± 10 psi of the design value (1000 psia).
- F. Turbine impulse pressure was found to be trending acceptably and required no adjustment.
- G. At each power level at least 5 minutes of Loose Parts Noise Monitoring System data was recorded from each channel for use as baseline data.

IV. CORRECTIVE ACTION

Feedwater and steam flow instrumentation required repeated adjustment due to difficulties encountered with their calibration. This also contributed to large mismatches during the steam flow/feedwater flow comparisons.

High full power S/G outlet pressure resulted from a combination of too high a full power T-AVG being input into the control systems and a lack of significant S/G tube fouling. Steam pressures were found to be ≈ 40 psi higher than the design full power value. The full power T-AVG was reduced to 586.0 °F to compensate for the high pressure.

UNIT LOAD STEADY STATE

AVERAGE NC T-HOT, T-COLD AND T-AVG VS. POWER

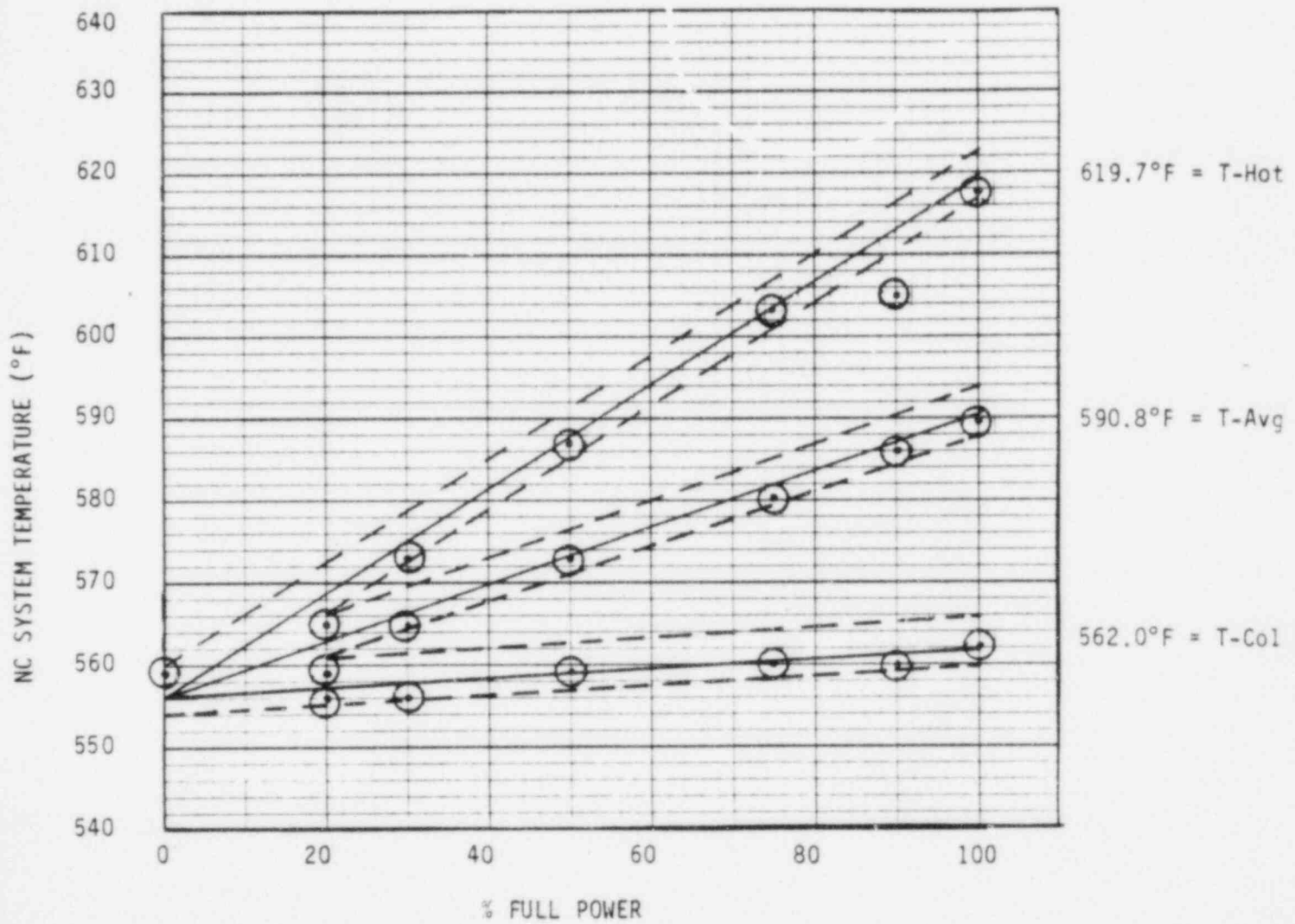


Figure 8.2-1

CATAWBA 1

UNIT LOAD STEADY STATE

AVERAGE NC ΔT VS. POWER

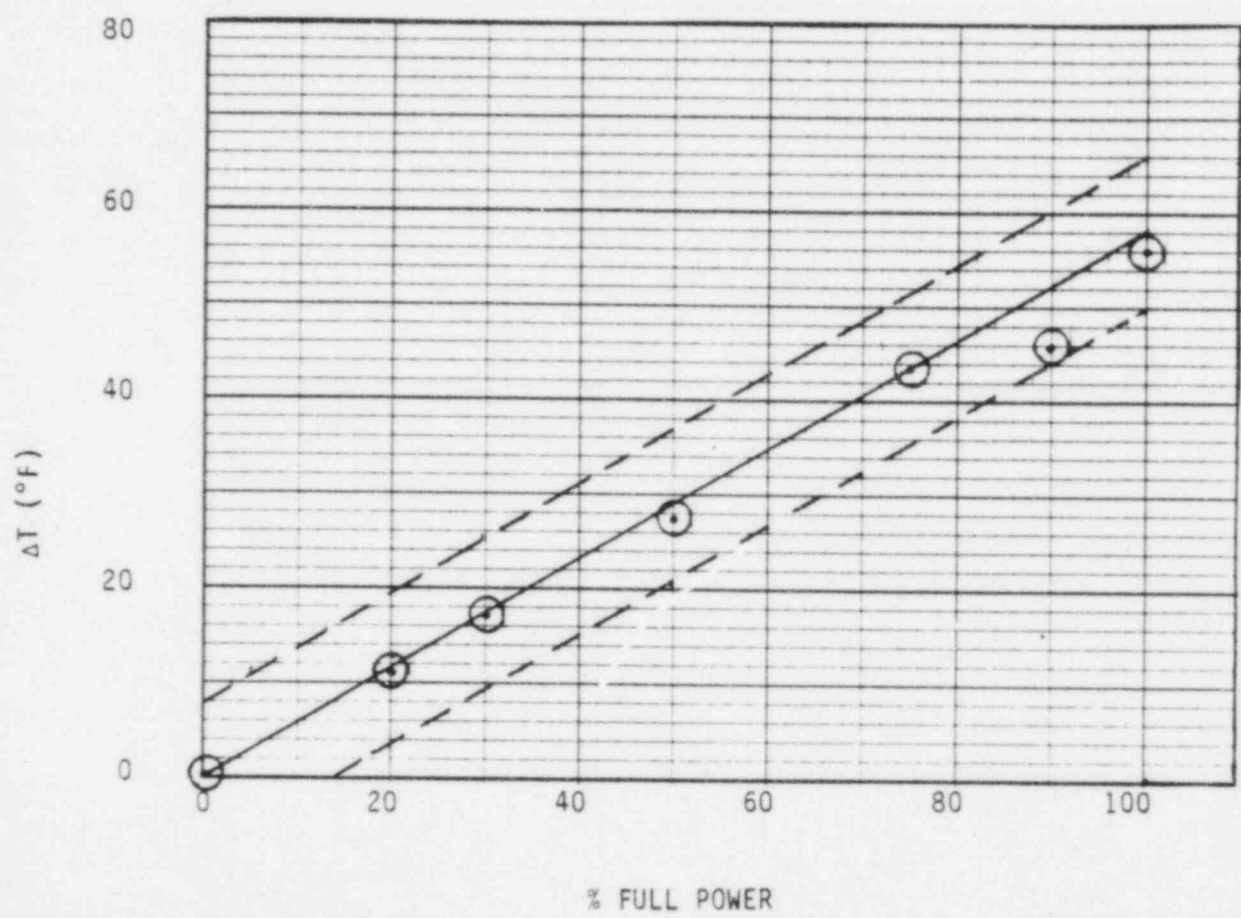


Figure 8.2-2

CATAWBA 1

UNIT LOAD STEADY STATE
PRESSURIZER LEVEL VS. POWER

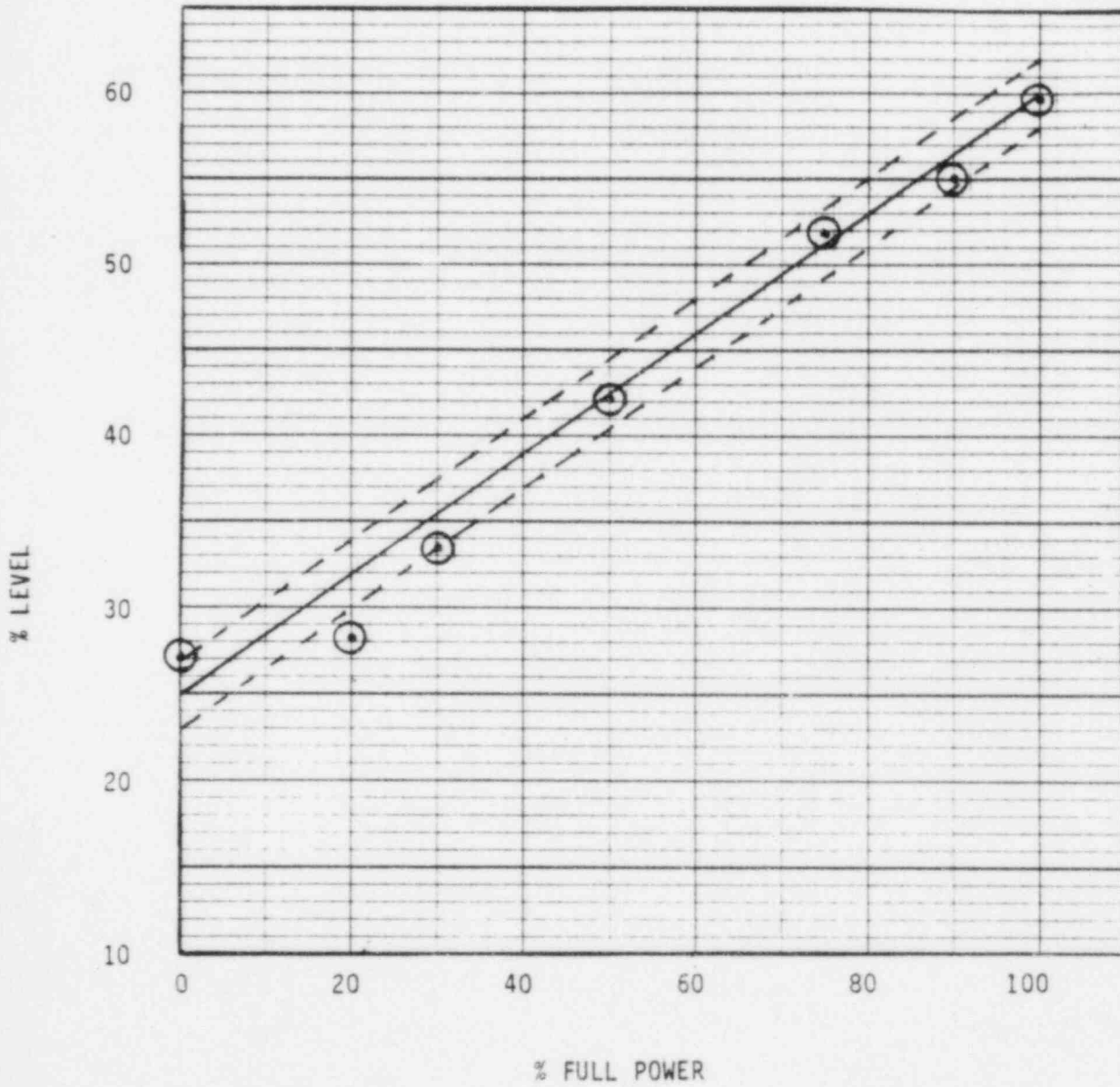


Figure 8.2-3

CATAWBA 1

UNIT LOAD STEADY STATE

AVERAGE MAIN STEAM HEADER PRESSURE VS. POWER

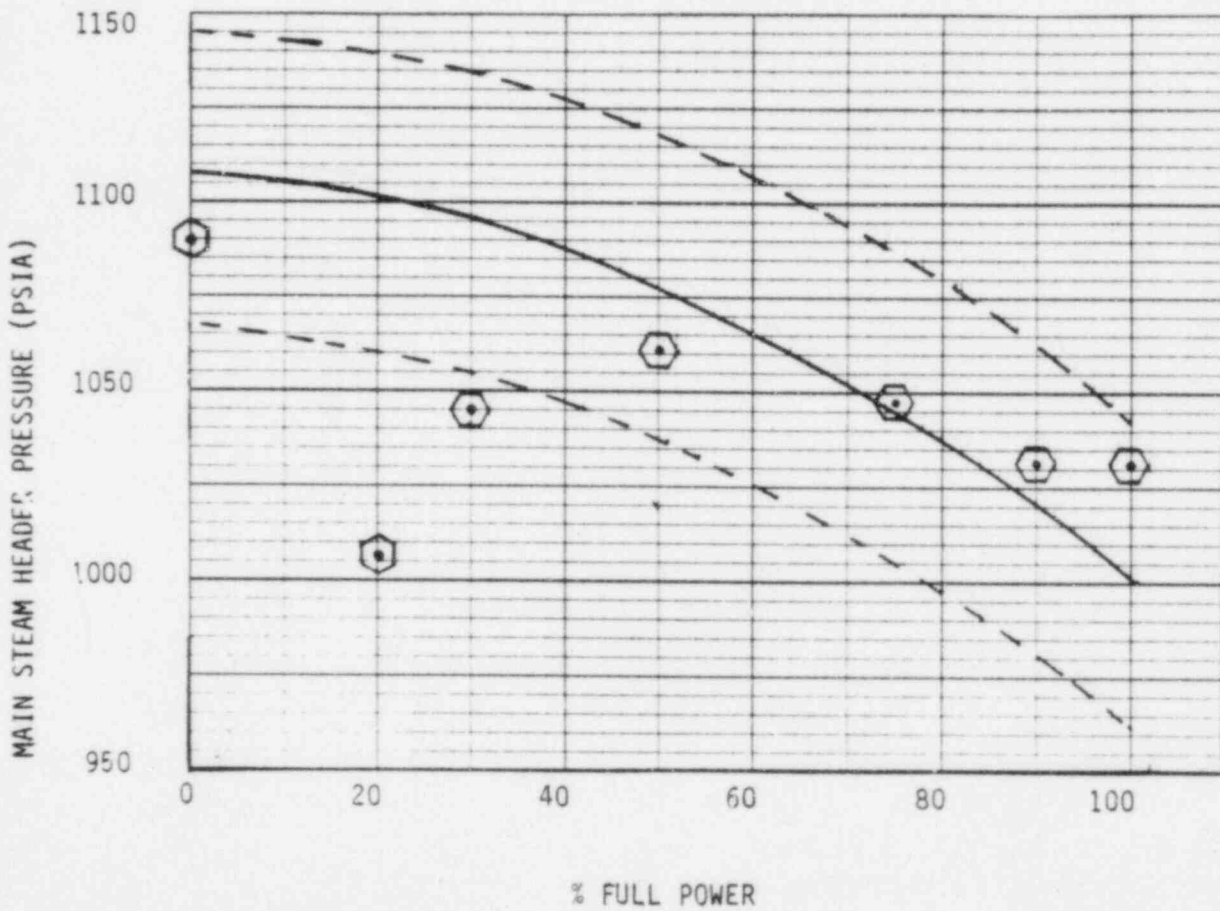


Figure 8.2-4

CATAWBA 1

UNIT LOAD STEADY STATE

AVERAGE S/G LEVEL VS. POWER

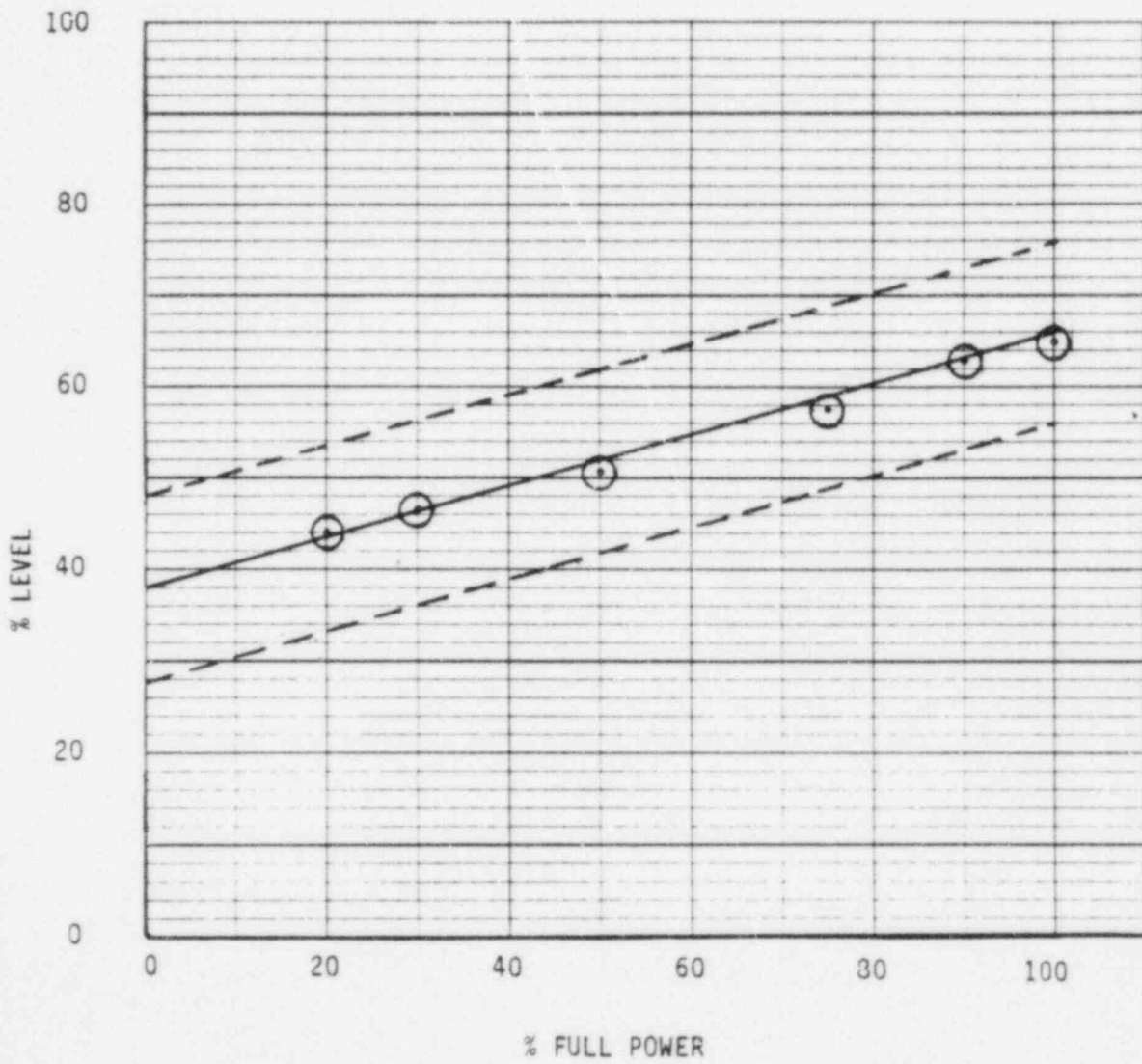


Figure 8.2-5

CATAWBA 1

UNIT LOAD STEADY STATE

AVERAGE FEEDWATER FLOW VS. POWER

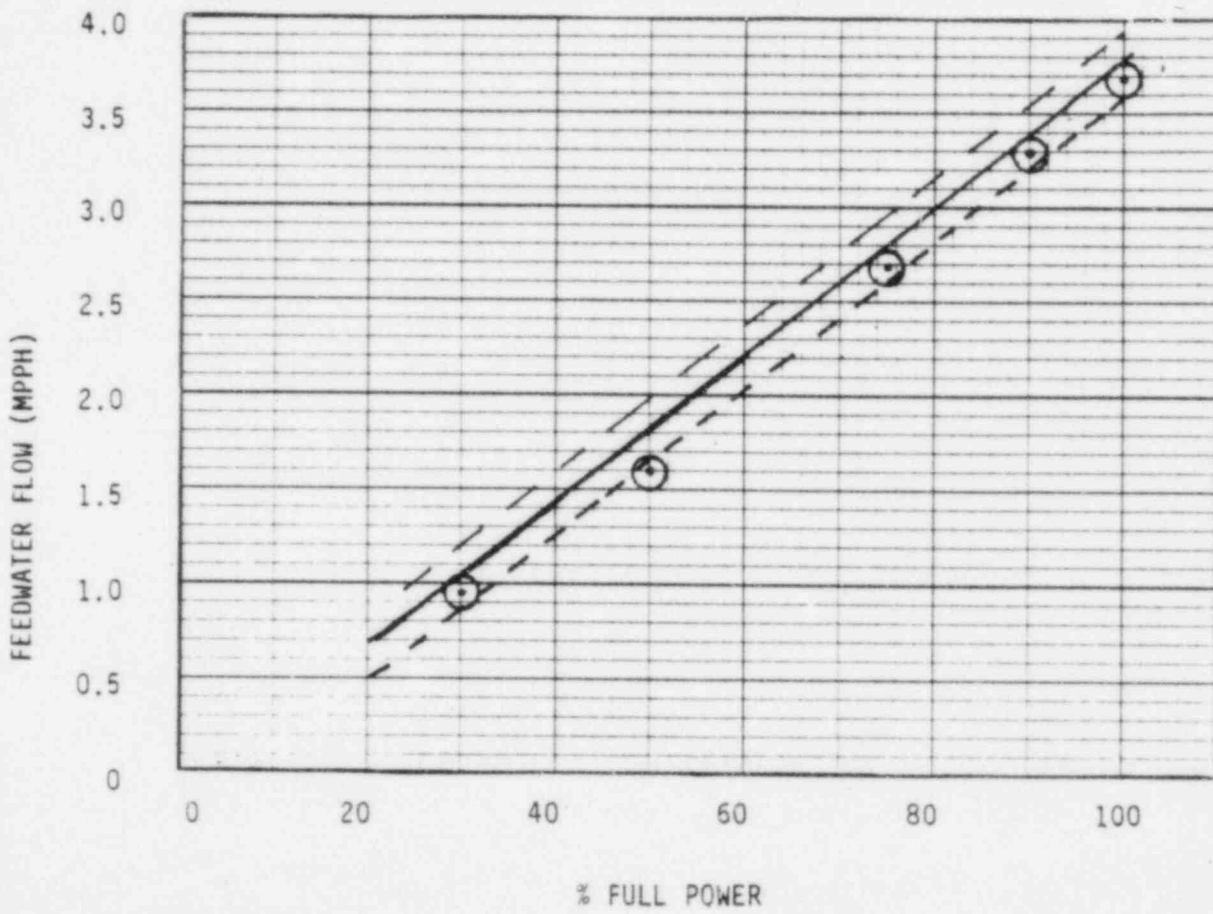


Figure 8.2-6

CATAWBA 1

UNIT LOAD STEADY STATE

AVERAGE FEEDWATER TEMPERATURE VS. POWER

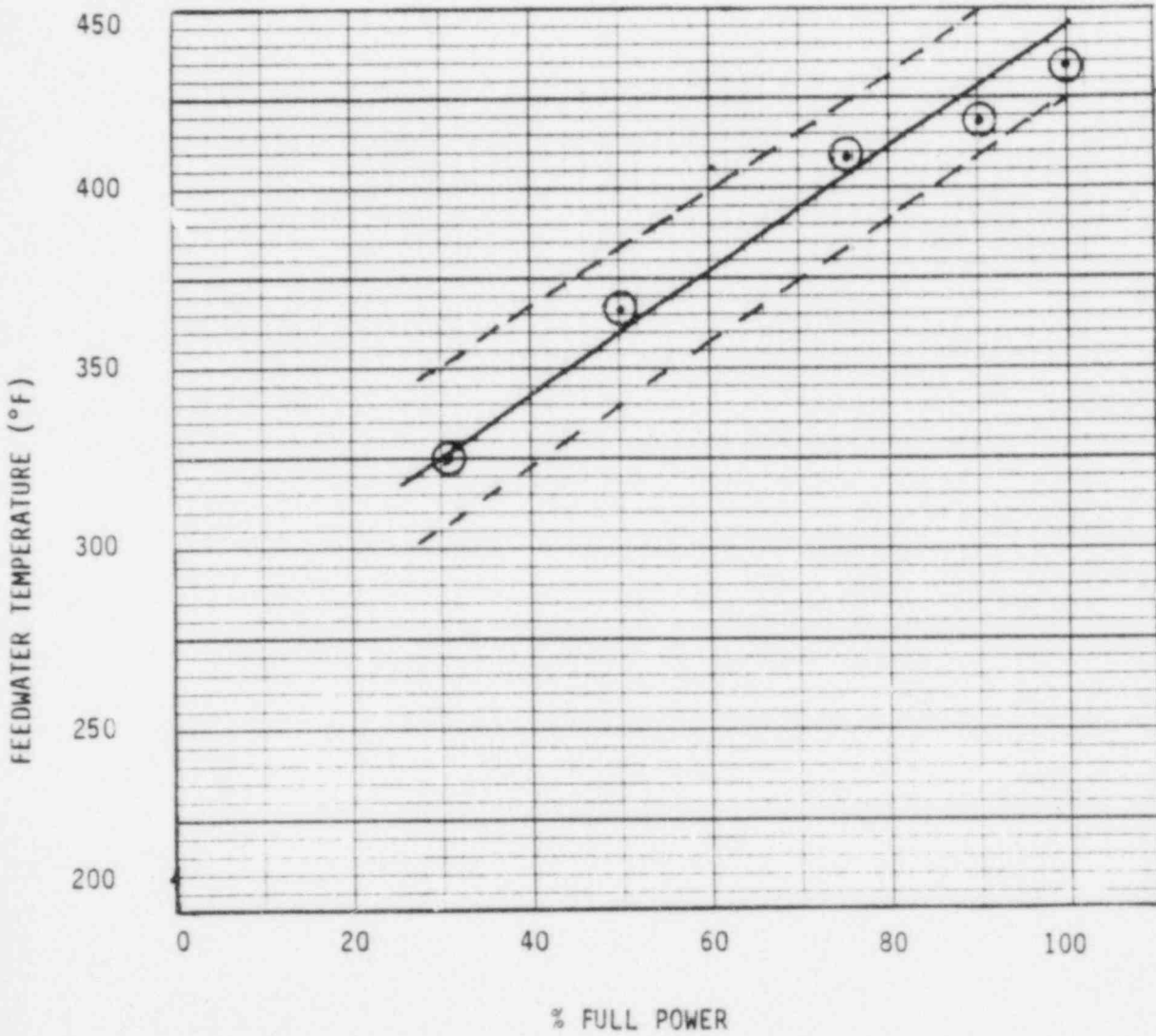


Figure 8.2-7

CATAWBA 1

UNIT LOAD STEADY STATE

TURBINE IMPULSE PRESSURE VS. POWER

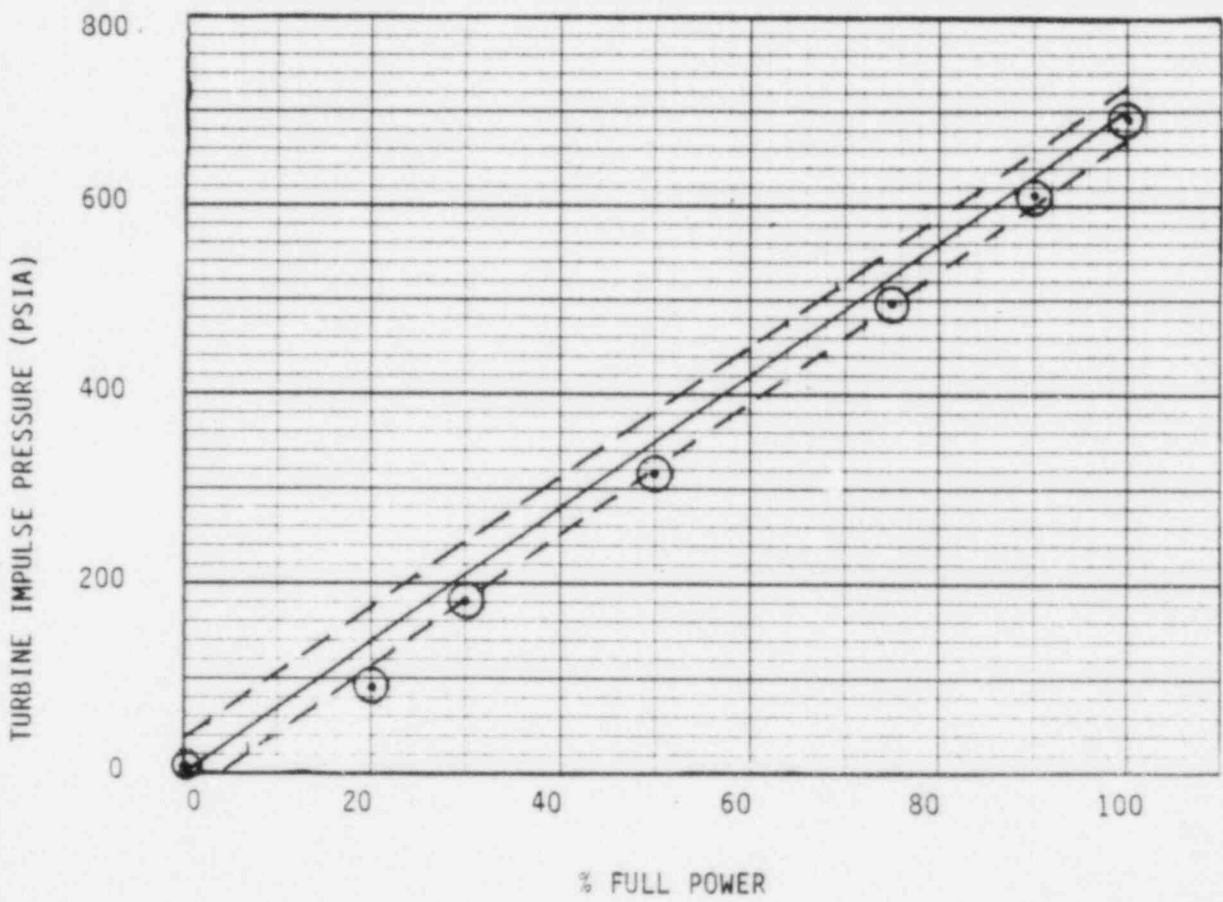


Figure 8.2-8

CATAWBA 1

8.3 DOPPLER ONLY POWER COEFFICIENT VERIFICATION - TP/1/A/2150/04

Date(s) Performed:	3/6/85 - 3/7/85	30% Full Power
	3/11/85 - 3/12/85	50% Full Power
	3/30/85 - 3/31/85	75% Full Power
	4/10/85	90% Full Power

I. PURPOSE

The objective of the Doppler Only Power Coefficient Verification test was to verify the nuclear design predictions of the doppler only power coefficient.

II. METHOD

At each of the power levels of 30%, 50%, 75% and 90% of Full Power, step increases/decreases of 3-4% in power were made while data on Reactivity, NC Temperature and power was collected by the Operator Aid Computer. The data collected was used to calculate a doppler only verification factor (C^m) which was compared to the predicted value of the doppler only power coefficient C^P .

III. RESULTS

The results of this test are listed in Table 8.3-1. The Acceptance Criterion requiring that the average measured value of the doppler only power coefficient be within ± 0.5 °F/%PWR of the average predicted value was met.

IV. CORRECTIVE ACTION

None required.

DOPPLER ONLY POWER COEFFICIENT VERIFICATION

RESULTS OF DOPPLER ONLY POWER COEFFICIENT VERIFICATION

POWER	C^m ($^{\circ}F/\%PWR$)	C^p ($^{\circ}F/\%PWR$)	$C^m - C^p$ ($^{\circ}F/\%PWR$)
$\approx 30\%$	-2.771	-2.753	0.018
$\approx 50\%$	-2.182	-2.149	0.078
$\approx 75\%$	-1.460	-1.460	0.00
$\approx 90\%$	-1.056	-1.165	0.109

Acceptance Criterion: $C^m = C^p \pm 0.5$ ($^{\circ}F/\%PWR$)

8.4 PSEUDO ROD EJECTION TEST (AT POWER) - TP/1/A/2150/06B

Date(s) Performed: 3/5/85 - 3/6/85

I. PURPOSE

The purpose of this test was:

- A. To measure the worth of the ejected rod and verify that it is less than that assumed in the FSAR Chapter 15 Accident Analysis.
- B. To measure the hot channel factors with the rod ejected and verify that the FSAR Chapter 15 assumptions were conservative.
- C. To demonstrate that the incore and excore detector systems are capable of detecting a misaligned rod.
- D. To measure core peaking factors with the rod 15 inches misaligned and verify that Tech Spec Limits are not exceeded.

II. METHOD

With Reactor power at approximately 30% power and Control Bank D at its full power insertion limit, the most reactive RCCA in the Bank (D-12) is withdrawn while flux and reactivity traces are taken.

Incore and excore detector data and incore thermocouple data are taken during RCCA withdrawal. A full core power distribution map was taken when the RCCA was fully withdrawn. The results of this map were used to verify that core peaking factors remained within Technical Specification Limits and below the values assumed in FSAR Chapter 15. The RCCA was diluted back into position with the remainder of Bank D.

III. RESULTS

- A. The measured ejected rod worth was 20.6 pcm, compared to 34 pcm predicted by Westinghouse. This was much less than the 230 pcm assumed in the FSAR Chapter 15 Accident Analysis.
- B. The maximum measured hot channel factor ($F_Q^T(Z)$) with the rod ejected was 2.4552. The FSAR Chapter 15 assumption is 5.90. A summary of the full core power distribution maps is given in Table 8.4-3.
- C. A perceptible change in power distribution as monitored by the Excore Power Range detectors and the incore thermocouples occurred with RCCA D-12 12 steps misaligned. The change became more obvious as the RCCA continued to be withdrawn. The Trace Pair Analysis performed with the moveable incore detector system also detected the rod misaligned at 12 steps misaligned. A summary of the Excore detector and incore thermocouples results are given in Table 8.4-1. Table 8.4-2 summarizes the Trace Pair Analyses performed.

- D. The core peaking factors measured with RCCA D-12 fully ejected (= 40 inches misaligned) were used to verify that Technical Specifications would not be violated with a rod 15 inches misaligned. The lowest margin of measured F_Q^T (Z) to Tech Spec limit was 0.8484 for a measured F_Q^T (Z) of 2.4475 compared to the limit of 4.5540. The measured reactor coolant flow rate was sufficient based upon the maximum enthalpy rise hot channel factor ($F_{\Delta H}$) of 1.707. Table 8.4-3 summarizes the results of the Flux maps, and Figures 8.4-1 and 8.4-2 compare the power distribution before the ejected rod to the power distribution after the rod was ejected.

IV. CORRECTIVE ACTION

None Required.

PSEUDO ROD EJECTION TEST (AT POWER)

QUADRANT POWER TILT RATIOS DURING ROD EJECTION

RCCA D-12 Position (Steps Misaligned)	Excore Detector Tilts				Thermocouple Tilts			
	Quad 1 N43	Quad 2* N42	Quad 3 N44	Quad 4 N41	Quad 1	Quad 2*	Quad 3	Quad 4
0	1.011	1.025	0.972	0.922	1.007	1.095	0.932	0.928
12	0.999	1.060	0.964	0.976	1.007	1.100	0.922	0.930
25	0.992	1.096	0.955	0.957	0.998	1.150	0.910	0.915
40	0.983	1.132	0.946	0.940	0.990	1.170	0.905	0.910
66**	0.978	1.149	0.941	0.931	0.978	1.217	0.885	0.904

*Quadrant 2 contains RCCA D-12

**RCCA D-12 is fully withdrawn

PSEUDO ROD EJECTION TEST (AT POWER)

SUMMARY OF TRACE PAIR ANALYSES

Trace Pair Analysis is normally performed to verify Control Rod Position when rod position indication is not fully operable or is suspect. The procedure for performing the analysis is OP/O/A/6150/12, Control Rod Position Verification.

Results:

D12 Position (Steps Misaligned)	Peaks*		Approximate % Difference**		
	E11	L05	Top	Middle	Bottom
12	16	16	10	5	1.5
25	15	16	40	15	4
40	15	17	55	15	4

*For the Trace Pair analysis, the core is divided into 30 axial locations, with location 30 as the top. Thus, a peak at 15 denotes an axial peak in the flux signal occurring in the middle of the core. If the peaks in the Trace Pair differ by more than 1 axial data point, a misaligned rod is assumed to exist. The core locations used in the Trace Pair Analysis (E11 and L05 in this case) are specified by procedure and consist of a location near the RCCA (E11) and a symmetric location (L05).

**Percent difference is the difference between the flux signal for location E11 and the signal at L05 for a specific axial point. For the purpose of illustration the 30 axial points were condensed to 3 regions. A difference of more than 5% for any one of the 30 axial points indicates a misaligned rod.

PSEUDO ROD EJECTION TEST (AT POWER)

FLUX MAPS SUMMARY

FCM/1/01/006 (with Control Bank D at 162):

Peaking Factors	Core Location		Axial Location*	
Horizontal Planar, $F_{xy} = 1.6459$	G14		51	
Axial, $F_z = 1.4427$	D12		25	
Total Heat Flux, $F_Q^T = 2.2859$	D12		25	
Nuclear Heat Flux, $F_Q^N = 2.114$	D12		25	
Enthalphy Rise, $F_{\Delta H} = 1.3939$	G14		N/A	
	<u>Quad 1</u>	<u>Quad 2</u>	<u>Quad 3</u>	<u>Quad 4</u>
Axial Offset:	-0.11609	-0.12087	-0.12464	-0.10823
Upper Tilt:	1.00240	1.02126	0.98046	0.99588
Lower Tilt:	0.99955	1.02828	0.99480	0.97737

FCM/1/01/007 (with RCCA D12 fully withdrawn, rest of Control Bank D at 168):

Peaking Factor	Core Location		Axial Location*	
Horizontal Planar, $F_{xy} = 2.2322$	C13		51	
Axial, $F_z = 1.3887$	C13		32	
Total Heat Flux, $F_Q^T = 2.4552$	C13		39	
Nuclear Heat Flux, $F_Q^N = 2.270$	C13		39	
Enthalphy Rise, $F_{\Delta H} = 1.707$	C13		N/A	
	<u>Quad 1</u>	<u>Quad 2</u>	<u>Quad 3</u>	<u>Quad 4</u>
Axial Offset:	-0.01726	0.4909	-0.01654	-0.02405
Upper Tilt:	0.96146	1.18609	0.94877	0.90368
Lower Tilt:	0.99542	1.07530	0.98088	0.94840

*For Flux Mapping purposes, there are 61 axial locations, with 61 representing the top of the core.

PSEUDO ROD EJECTION TEST (AT POWER)

F_Q^N BEFORE AND AFTER ROD EJECTION

Control Bank D at HFP Insertion Limits (FCM/1/01/006)

	01	02	03	04	05	06	07	08	09	10	11	12	13	14	15
A					1.548	1.817	1.796	1.876	1.717	1.785	1.555				
B			1.635	1.839	1.923	1.399	1.946	1.435	1.903	1.374	1.931	1.971	1.737		
C		1.675	1.975	1.783	1.515	1.671	1.435	1.607	1.409	1.711	1.566	1.912	2.083	1.737	
D		1.930	1.826	2.018	1.787	1.480	1.708	1.424	1.724	1.497	1.886	2.114	1.880	1.954	
E	1.407	1.819	1.490	1.787	1.535	1.727	1.452	1.718	1.483	1.790	1.571	1.886	1.553	1.866	1.384
F	1.648	1.272	1.623	1.473	1.758	1.456	1.680	1.440	1.736	1.500	1.781	1.472	1.709	1.469	1.908
G	1.670	1.857	1.379	1.705	1.465	1.712	1.380	1.563	1.402	1.692	1.446	1.702	1.419	1.961	1.772
H	1.874	1.398	1.576	1.439	1.721	1.437	1.566	1.278	1.549	1.417	1.704	1.475	1.630	1.397	1.831
J	1.758	1.893	1.378	1.691	1.470	1.722	1.393	1.545	1.374	1.706	1.464	1.719	1.415	1.898	1.719
K	1.786	1.375	1.621	1.435	1.739	1.478	1.714	1.416	1.690	1.454	1.747	1.477	1.668	1.365	1.738
L	1.521	1.889	1.489	1.803	1.496	1.731	1.435	1.662	1.420	1.715	1.523	1.837	1.512	1.876	1.481
M		1.837	1.773	2.015	1.764	1.394	1.669	1.377	1.650	1.444	1.820	2.018	1.801	1.902	
N		1.833	1.941	1.807	1.427	1.538	1.341	1.952	1.310	1.514	1.923	1.744	1.932	1.705	
P		1.619	1.788	1.777	1.294	1.804	1.331	1.771	1.300	1.829	1.855	1.660			
R				1.414	1.668	1.641	1.747	1.641	1.705	1.473					

RCCA D12 Fully Withdrawn, Remainder of Bank D at HFP Insertion Limits (FCM/1/01/007)

	01	02	03	04	05	06	07	08	09	10	11	12	13	14	15
A					1.518	1.784	1.800	1.935	1.792	1.873	1.639				
B			1.599	1.755	1.883	1.372	1.932	1.466	1.950	1.444	2.036	2.149	1.948		
C		1.585	1.868	1.687	1.483	1.630	1.399	1.597	1.423	1.770	1.633	2.023	2.270	1.948	
D		1.813	1.714	1.893	1.679	1.438	1.688	1.431	1.749	1.544	1.966	2.222	2.009	2.116	
E	1.293	1.691	1.400	1.678	1.428	1.672	1.423	1.713	1.512	1.843	1.637	1.974	1.637	2.071	1.667
F	1.515	1.169	1.526	1.370	1.636	1.391	1.628	1.416	1.739	1.543	1.876	1.972	1.776	1.506	1.954
G	1.527	1.689	1.251	1.563	1.364	1.612	1.331	1.529	1.379	1.653	1.560	1.702	1.441	1.980	1.772
H	1.696	1.265	1.433	1.304	1.593	1.349	1.444	1.205	1.697	1.367	1.656	1.826	1.632	1.389	1.807
J	1.588	1.712	1.249	1.534	1.346	1.594	1.306	1.470	1.328	1.643	1.410	1.675	1.397	1.885	1.695
K	1.612	1.241	1.462	1.320	1.582	1.391	1.580	1.341	1.598	1.379	1.665	1.419	1.608	1.326	1.690
L	1.375	1.704	1.342	1.857	1.376	1.572	1.294	1.499	1.332	1.597	1.447	1.760	1.460	1.820	1.439
M		1.712	1.627	1.881	1.615	1.271	1.905	1.252	1.823	1.354	1.724	1.920	1.725	1.832	
N	1.522	1.781	1.634	1.302	1.411	1.223	1.419	1.206	1.401	1.347	1.649	1.827	1.644		
P		1.484	1.621	2.629	1.197	1.671	1.224	1.640	1.196	1.664	1.721	1.581			
R				1.280	1.555	1.565	1.618	1.519	1.561	1.340					

FIGURE 8.4-1

PSEUDO ROD EJECTION TEST (AT POWER)

$F_{\Delta H}^N$ BEFORE AND AFTER ROD EJECTION

Control Bank D at HFP Insertion Limits (FCM/1/01/006)

	01	02	03	04	05	06	07	08	09	10	11	12	13	14	15
A					1.084	1.298	1.280	1.339	1.226	1.261	1.042				
B			1.112	1.263	1.329	0.981	1.381	1.072	1.753	0.962	1.327	1.376	1.163		
C		1.134	1.309	1.171	1.017	1.156	1.009	1.141	0.997	1.187	1.056	1.259	1.375	1.163	
D		1.212	1.195	1.231	1.175	1.016	1.199	1.013	1.211	1.035	1.240	1.295	1.230	1.320	
E	0.993	1.266	1.005	1.176	1.033	1.204	1.024	1.213	1.034	1.240	1.063	1.243	1.046	1.360	1.109
F	1.179	0.904	1.135	1.017	1.220	1.072	1.186	1.008	1.211	1.048	1.239	1.018	1.185	1.030	1.351
G	1.200	1.326	0.976	1.197	1.032	1.202	0.960	1.075	0.972	1.185	1.016	1.196	1.005	1.394	1.259
H	1.343	1.003	1.120	1.015	1.214	1.006	1.077	0.783	1.063	0.989	1.200	1.010	1.158	0.995	1.300
J	1.255	1.348	0.976	1.185	1.033	1.210	0.971	1.064	0.955	1.190	1.026	1.210	1.001	1.345	1.219
K	1.263	0.962	1.124	0.997	1.211	1.037	1.207	0.996	1.193	1.015	1.214	1.071	1.161	0.964	1.238
L	1.062	1.302	0.995	1.185	1.013	1.205	1.011	1.174	1.001	1.184	1.037	1.209	1.020	1.305	1.041
M		1.274	1.172	1.229	1.102	0.966	1.173	0.977	1.159	0.998	1.203	1.239	1.189	1.306	
N		1.122	1.295	1.187	0.967	1.072	0.951	1.108	0.928	1.047	0.971	1.154	1.290	1.169	
P		1.105	1.222	1.246	0.908	1.286	0.950	1.255	0.912	1.261	1.264	1.134			
R					0.997	1.178	1.168	1.243	1.165	1.205	1.029				

RCCA D12 Fully Withdrawn, Remainder of Bank D at HFP Insertion Limits (FCM/1/01/007)

	01	02	03	04	05	06	07	08	09	10	11	12	13	14	15
A					1.078	1.298	1.318	1.412	1.307	1.373	1.205				
B			1.094	1.256	1.315	0.981	1.411	1.078	1.450	1.069	1.504	1.611	1.468		
C		1.108	1.284	1.157	0.995	1.154	1.021	1.175	1.063	1.307	1.218	1.515	1.707	1.469	
D		1.284	1.166	1.185	1.172	0.995	1.208	1.050	1.292	1.137	1.466	1.655	1.493	1.591	
E	0.942	1.221	0.982	1.150	0.990	1.189	1.032	1.259	1.105	1.359	1.212	1.468	1.221	1.542	1.236
F	1.116	0.857	1.109	0.962	1.167	0.982	1.175	1.026	1.276	1.120	1.369	1.154	1.315	1.114	1.431
G	1.126	1.235	0.906	1.122	0.978	1.158	0.941	1.082	0.997	1.253	1.115	1.268	1.070	1.463	1.311
H	1.241	0.926	1.029	0.941	1.143	0.962	1.045	0.750	1.067	1.006	1.248	1.036	1.200	1.028	1.843
J	1.164	1.249	0.901	1.097	0.961	1.136	0.925	1.045	0.931	1.179	1.032	1.219	1.013	1.363	1.236
K	1.174	0.893	1.045	0.948	1.132	0.951	1.130	0.962	1.133	0.987	1.199	1.004	1.178	0.980	1.250
L	0.985	1.206	0.920	1.112	0.956	1.124	0.934	1.096	0.953	1.136	0.999	1.174	1.017	1.309	1.038
M		1.232	1.117	1.148	0.991	0.902	1.085	0.911	1.093	0.949	1.137	1.177	1.171	1.307	
N		1.081	1.233	1.121	0.910	1.010	0.886	1.028	0.874	0.998	0.933	1.110	1.250	1.177	
P		1.048	1.152	1.187	0.863	1.117	0.891	1.193	0.864	1.188	1.204	1.106			
R					0.950	1.135	1.130	1.178	1.106	1.136	0.967				

FIGURE 8.4-2

8.5 UNIT LOAD TRANSIENT TEST - TP/1/A/2650/05

Date(s) Performed:	3/7/85	30% PWR
	3/16/85	50% PWR
	4/3/85 - 4/5/85	75% PWR
	4/17/85	100% PWR

I. PURPOSE

The purpose of the Unit Load Transient Test was to demonstrate proper plant response (in particular, automatic control system response) to 10% step load increases and decreases at various power levels. These load changes were initiated by adjusting the Turbine Generator (T/G) load.

II. METHOD

From stable conditions with control systems in automatic at a power level appropriate to the current testing plateau (30%, 50%, 75% or 100% Full Power) a 10% step load decrease was initiated. This was accomplished by reducing the T/G load limiter setting until the predetermined load corresponding to the 10% power reduction was reached. Automatic control systems were allowed to bring the plant to equilibrium. The OAC Transient Monitor was manually frozen and data dumped to floppy disks. Once stability was re-established, a 10% load increase was initiated with load increased to its initial value. Data was then obtained again from the OAC Transient Monitor. Both load changes were performed with sufficient speed to approximate step load changes. Plots were made using the Transient Monitor data and printouts obtained from the OAC Alarm Summary program in order to verify that the Acceptance Criteria were met.

III. RESULTS

All Acceptance Criteria were met as detailed below:

- A. Reactor and Turbine never tripped during any test.
- B. Safety Injection never occurred during any test.
- C. At each power level, a test was eventually performed successfully in that no manual operator intervention was required during the transient or recovery. The 30% and 50% tests were successfully performed on the first attempt. At 75%, CF Regulating Valve instability required the operators to assume manual control of S/G feed flow. This occurred during 4 attempts at the load decrease. Following control adjustments by IAE, the fifth attempt was successful. At the 100% plateau, manual operator action was required during the load increase. This was due to S/G feedwater flow control problems. IAE adjustment allowed the retest to be successfully completed.
- D. Pressurizer PORV's never lifted during any test.

- E. Pressurizer code safety valves never lifted during any test.
- F. Steam Generator PORV's never lifted during any test.
- G. Steam Generator safety valves never lifted during any test.
- H. The monitored plant parameters did not indicate any sustained or divergent oscillations except as noted in C above. At each power level, the final test was successful in this respect.
- I. Nuclear Power never over or under shot the desired load change by more than the 3% maximum. Over/Undershoots at each power level are listed in Table 8.5-1.
- J. Plots of the OAC Transient Monitor data are given in the following figures.
 - 1. 30% Full Power - Figure 8.5-1 to Figure 8.5-10
 - 2. 50% Full Power - Figure 8.5-11 to Figure 8.5-20
 - 3. 75% Full Power - Figure 8.5-21 to Figure 8.5-30
 - 4. 100% Full Power - Figure 8.5-31 to Figure 8.5-40

IV. CORRECTIVE ACTIONS

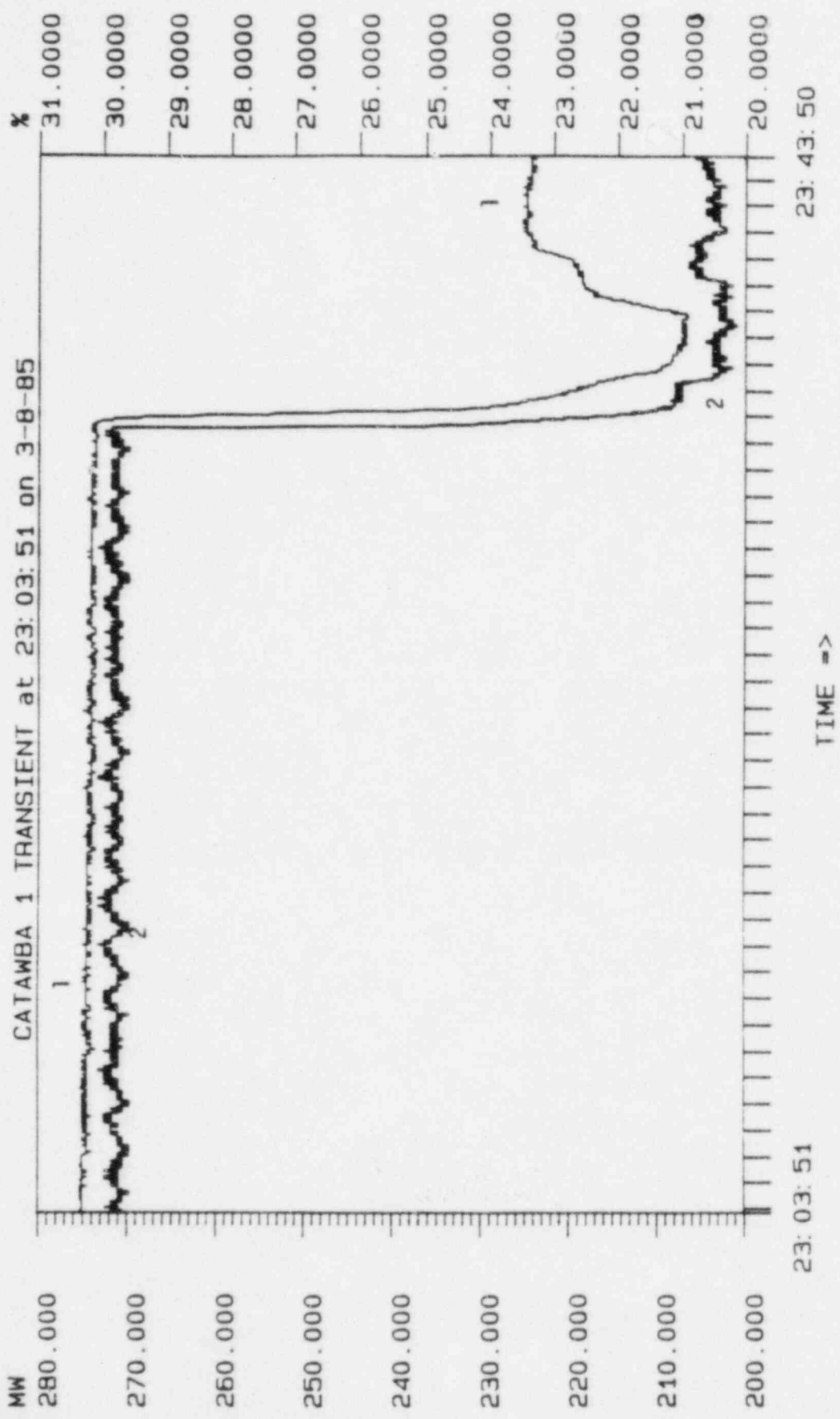
None required except as described in Section III.C.

UNIT LOAD TRANSIENT TEST
NUCLEAR POWER OVER/UNDERSHOOTS

Test Plateau	Maximum Over/Undershoot	
	Load Decrease	Load Increase
30%	2.4%	2.1%
50%	2.6%	1.8%
75%	0.1%	0.0%
100%	1.8%	0.0%

TABLE 8.5-1

UNIT LOAD TRANSIENT TEST - 30% DECREASE
PRIMARY AND SECONDARY POWER TREND



23:03:51

23:43:50

	MAX	MIN	PID	MESSAGE
> 1	30.3600	20.8800	A0758	POWER RANGE AVG LEVEL QUADRANT 1 (N-43)
< 2	273.223	200.878	A0697	GENERATOR MEGAWATTS

< Manual Freeze >

Figure 8.5-1

UNIT LOAD TRANSIENT TEST - 30% DECREASE
NC AVERAGE TEMPERATURE TREND

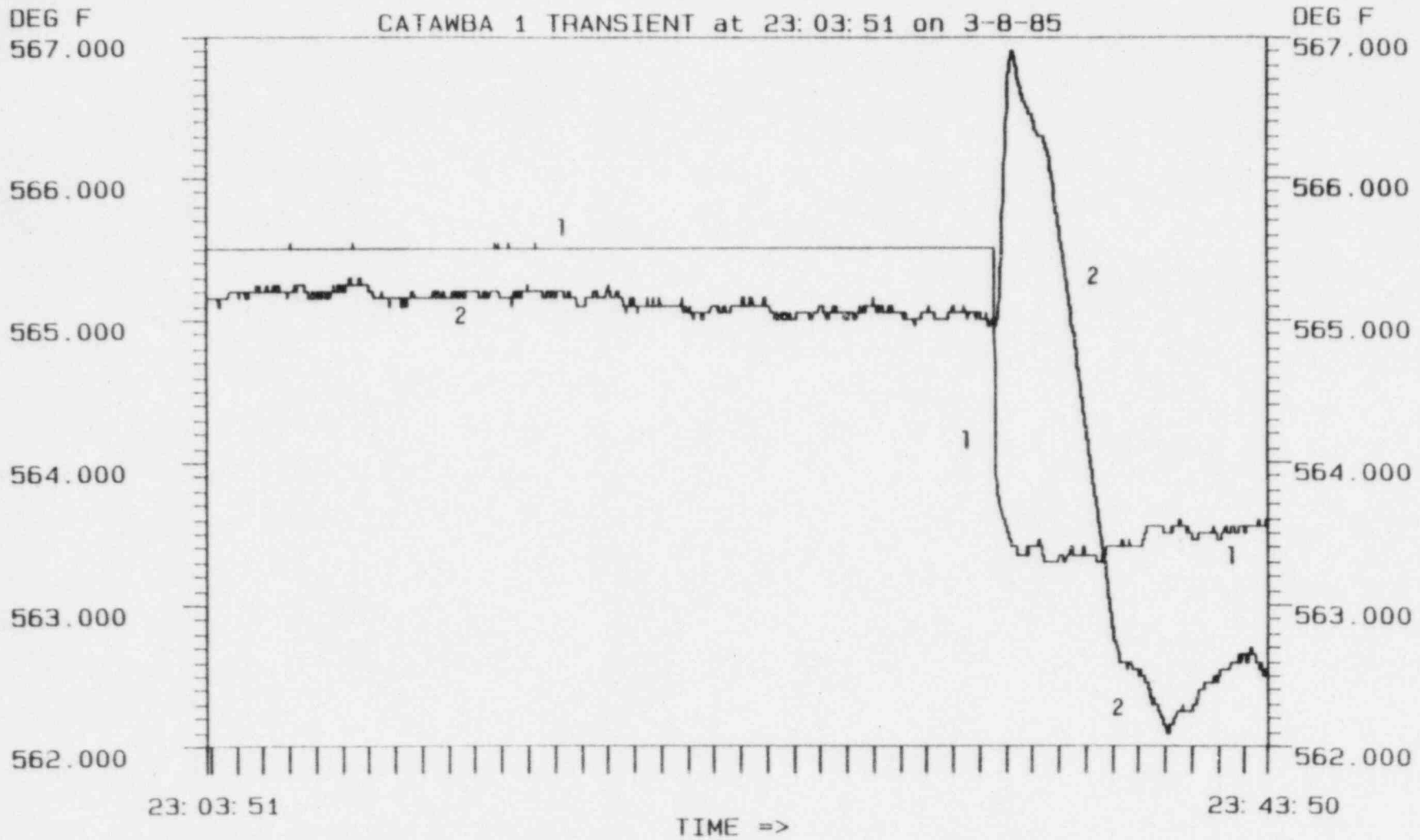


Figure 8.5+2

CATAWBA 1

	MAX	MIN	PID	MESSAGE
1	565.550	563.300	A0725	T-REF
2	566.900	562.100	A0701	NC LOOP HIGHEST T-AVG < Manual Freeze >

UNIT LOAD TRANSIENT TEST - 30% DECREASE
TYPICAL NC LOOP TEMPERATURE TREND

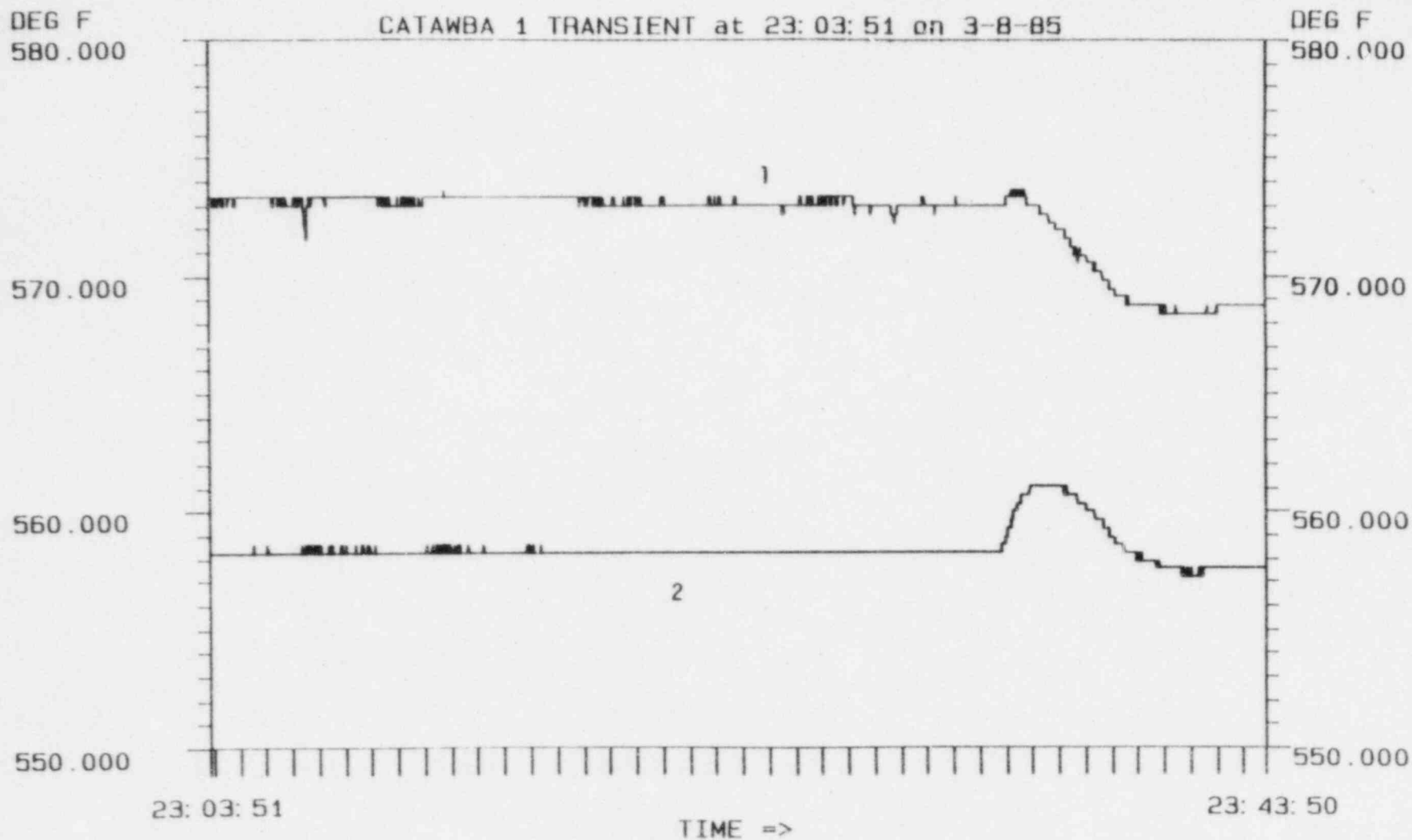


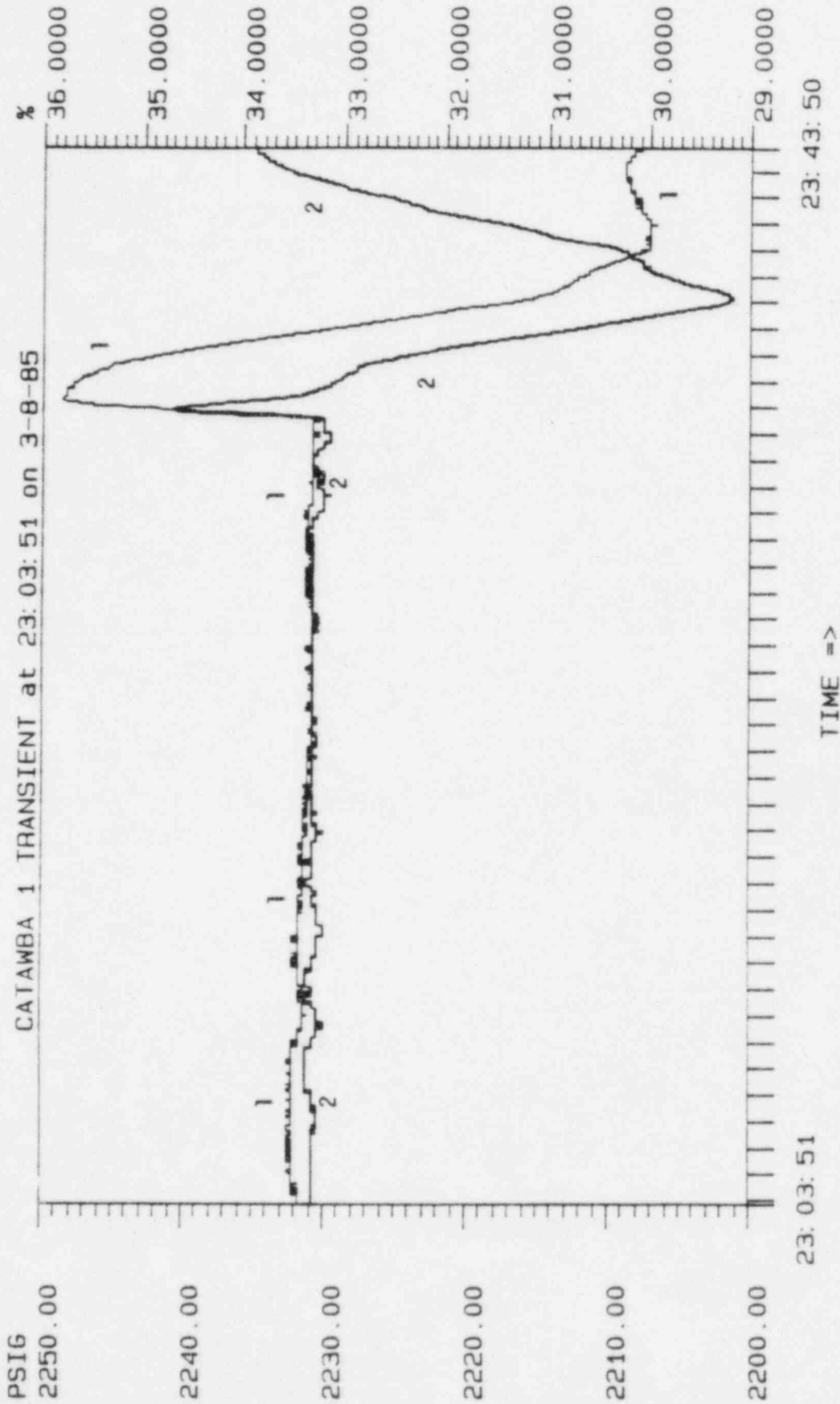
Figure 8.5-3

CATAWBA 1

	MAX	MIN	PID	MESSAGE
1	573.650	568.400	A0668	NC LOOP A WIDE RANGE HOT LEG TEMP
2	561.050	557.200	A0700	NC LOOP A WIDE RANGE COLD LEG TEMP

< Manual Freeze >

UNIT LOAD TRANSIENT TEST - 30% DECREASE
 PRESSURIZER PRESSURE AND LEVEL TREND



	MAX	MIN	PID	MESSAGE
> 1	35.8000	29.9500	A0707	PZR LEVEL CH I
< 2	2240.80	2201.20	A0713	PZR PRESS CH 1

< Manual Freeze >

Figure 8.5-4

UNIT LOAD TRANSIENT TEST
TYPICAL S/G PRESSURE AND LEVEL TREND

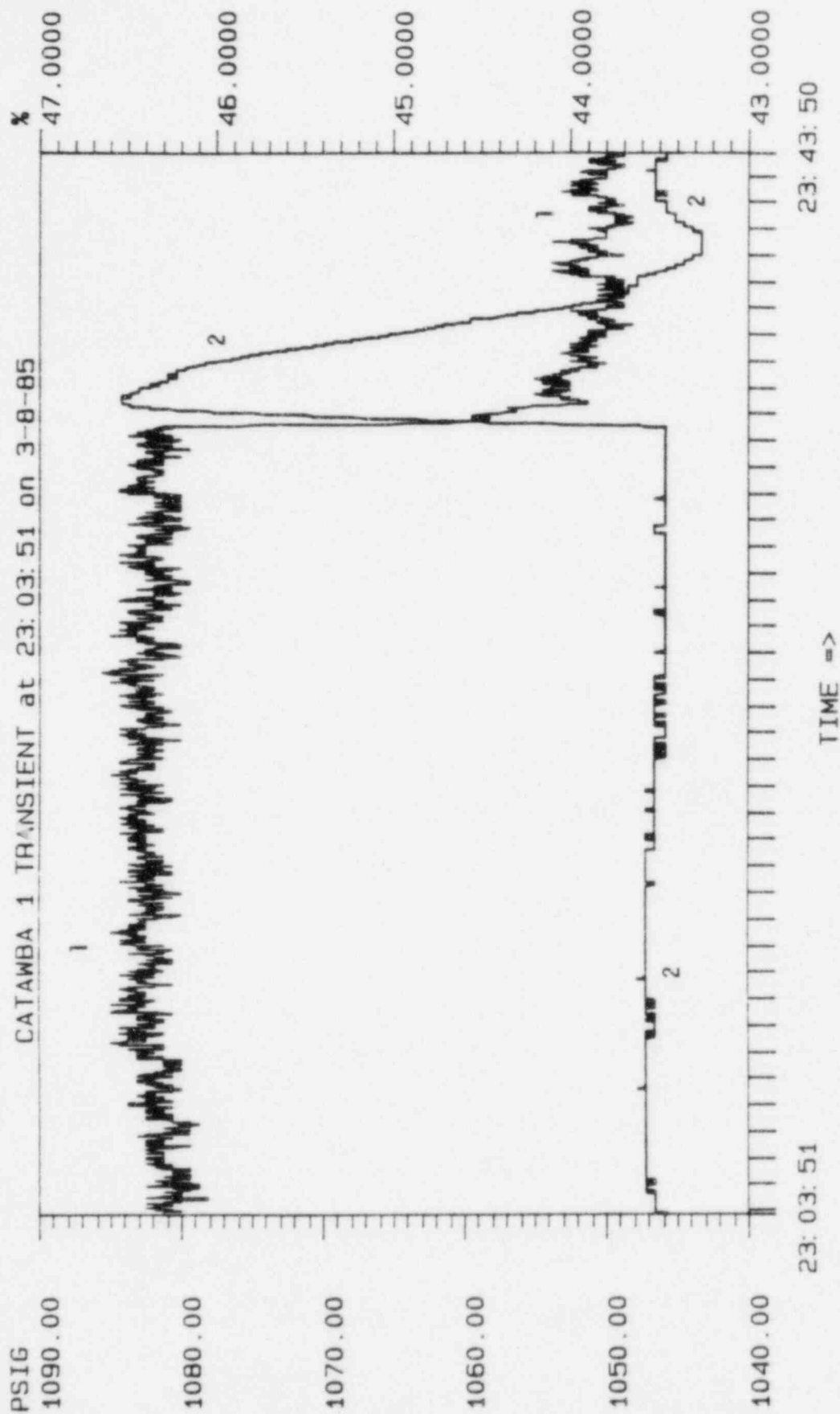


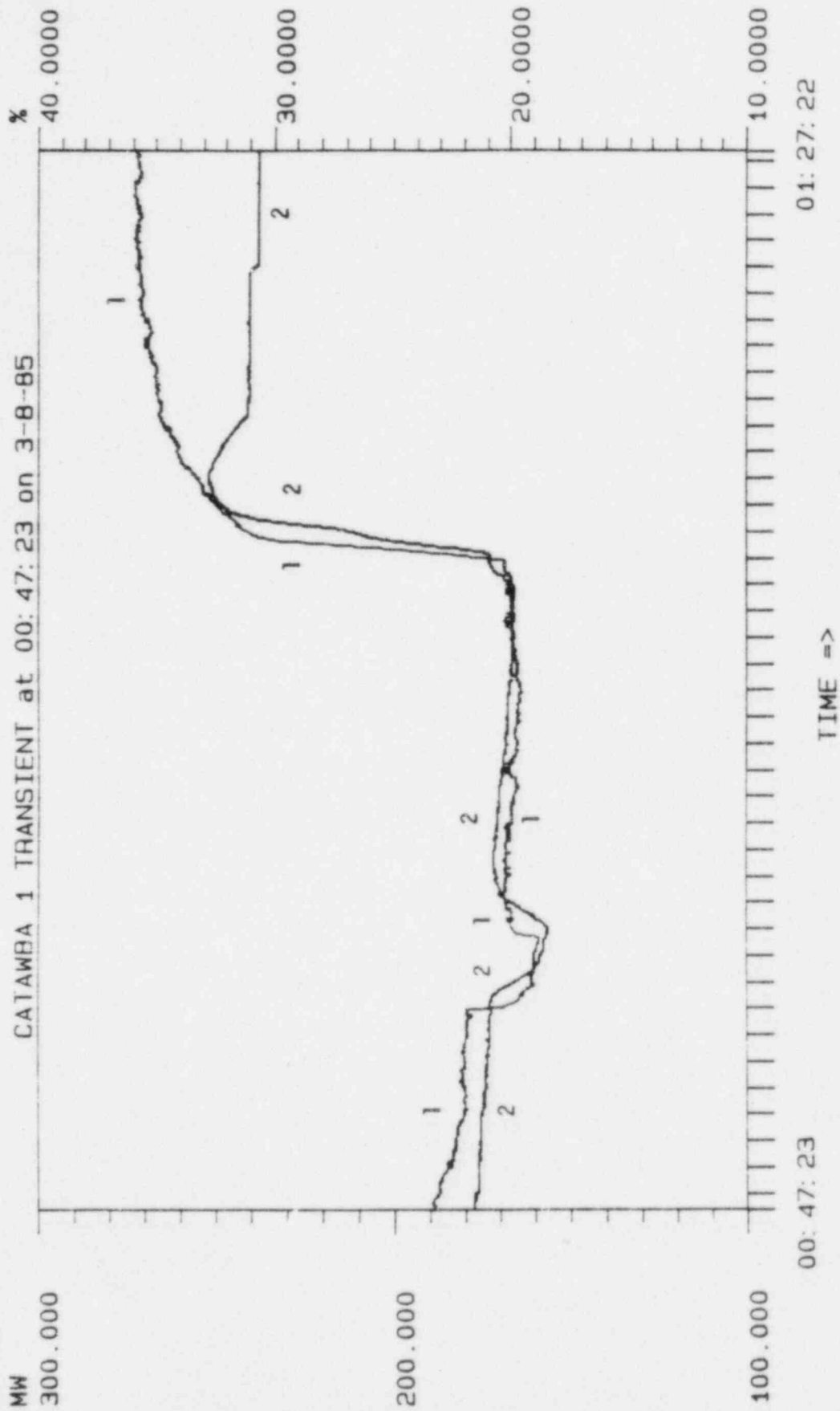
Figure 8.5-5

CATAWBA 1

	MAX	MIN	PID	MESSAGE
> 1	46.6500	43.6500	A0537	S/G B NARROW RANGE LEVEL CH 2
< 2	1084.20	1043.25	A1286	S/G B STEAM PRESS CH #2

< Manual Freeze >

UNIT LOAD TRANSIENT TEST - 30% INCREASE
PRIMARY AND SECONDARY POWER TREND



	MAX	MIN	PID	MESSAGE
< 1	275.223	158.736	A0697	GENERATOR MEGAWATTS
> 2	32.8200	18.4800	A0758	POWER RANGE AVG LEVEL QUADRANT 1 (N-43)

< Manual Freeze >

Figure 8.5-6

UNIT LOAD TRANSIENT TEST - 30% INCREASE
 NC AVERAGE TEMPERATURE TREND

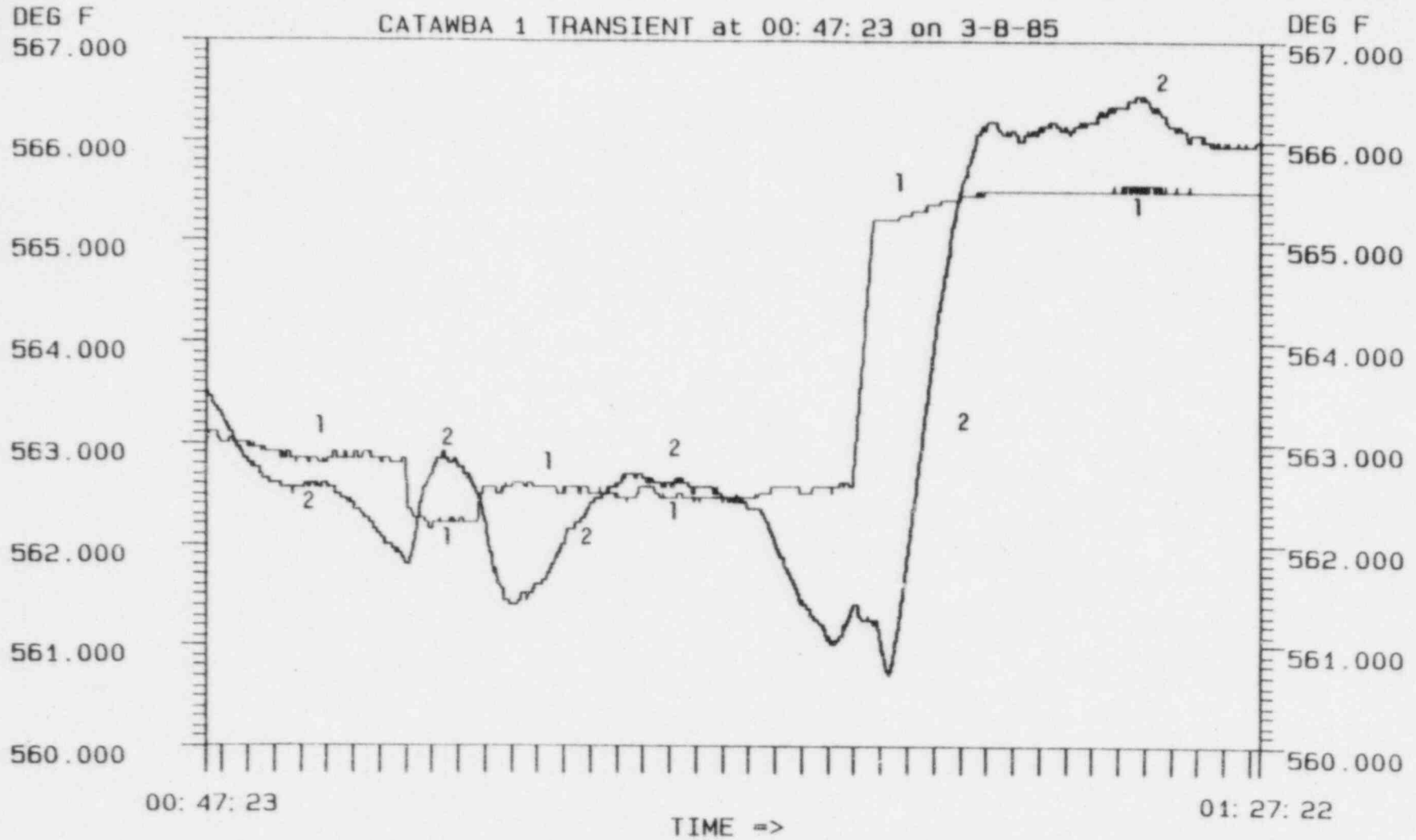


Figure 8.5-7

CATAWBA 1

	MAX	MIN	PID	MESSAGE
1	565.550	562.150	A0725	T-REF
2	566.450	560.700	A0701	NC LOOP HIGHEST T-AVG

< Manual Freeze >

UNIT LOAD TRANSIENT TEST - 30% INCREASE
TYPICAL NC LOOP TEMPERATURE TREND

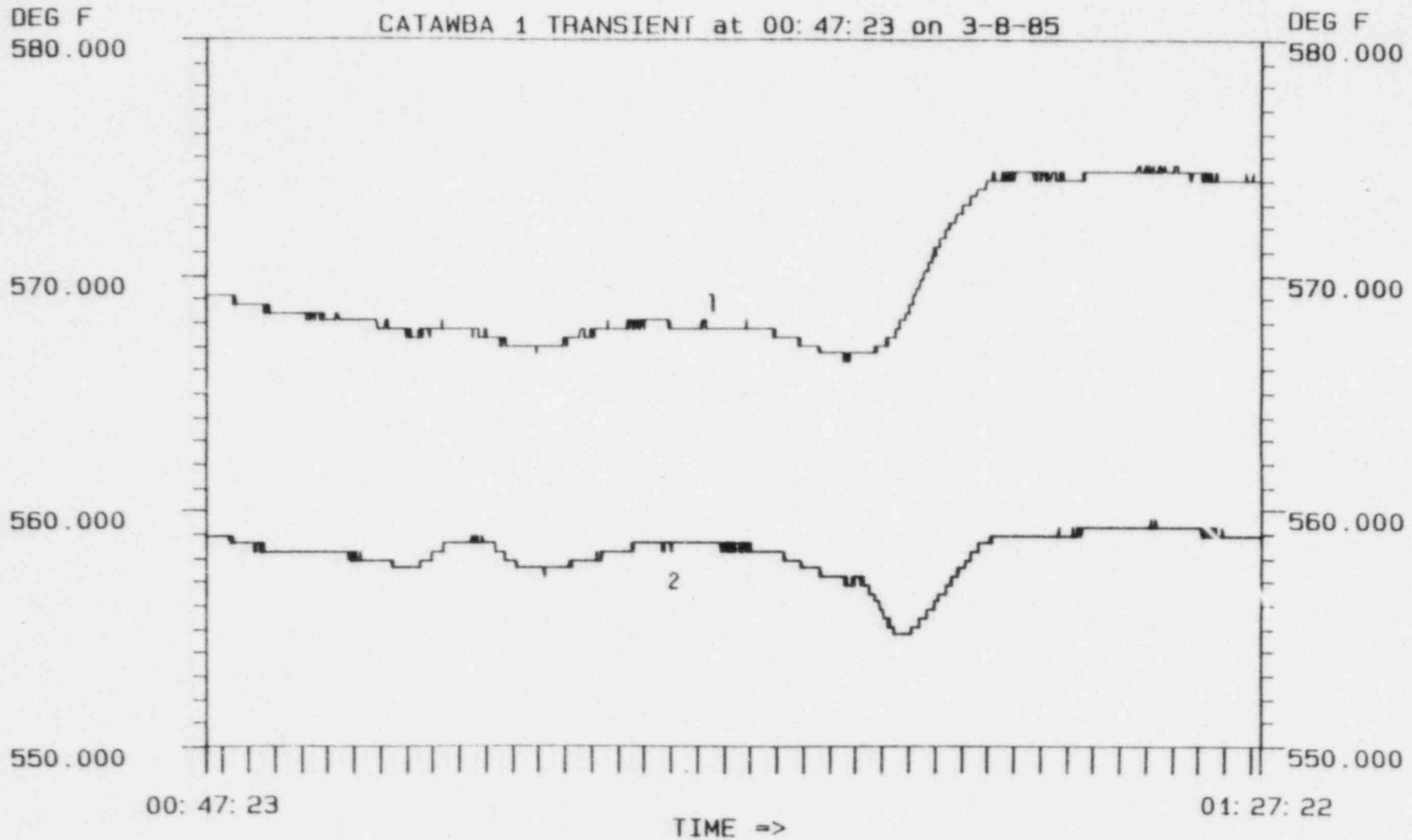


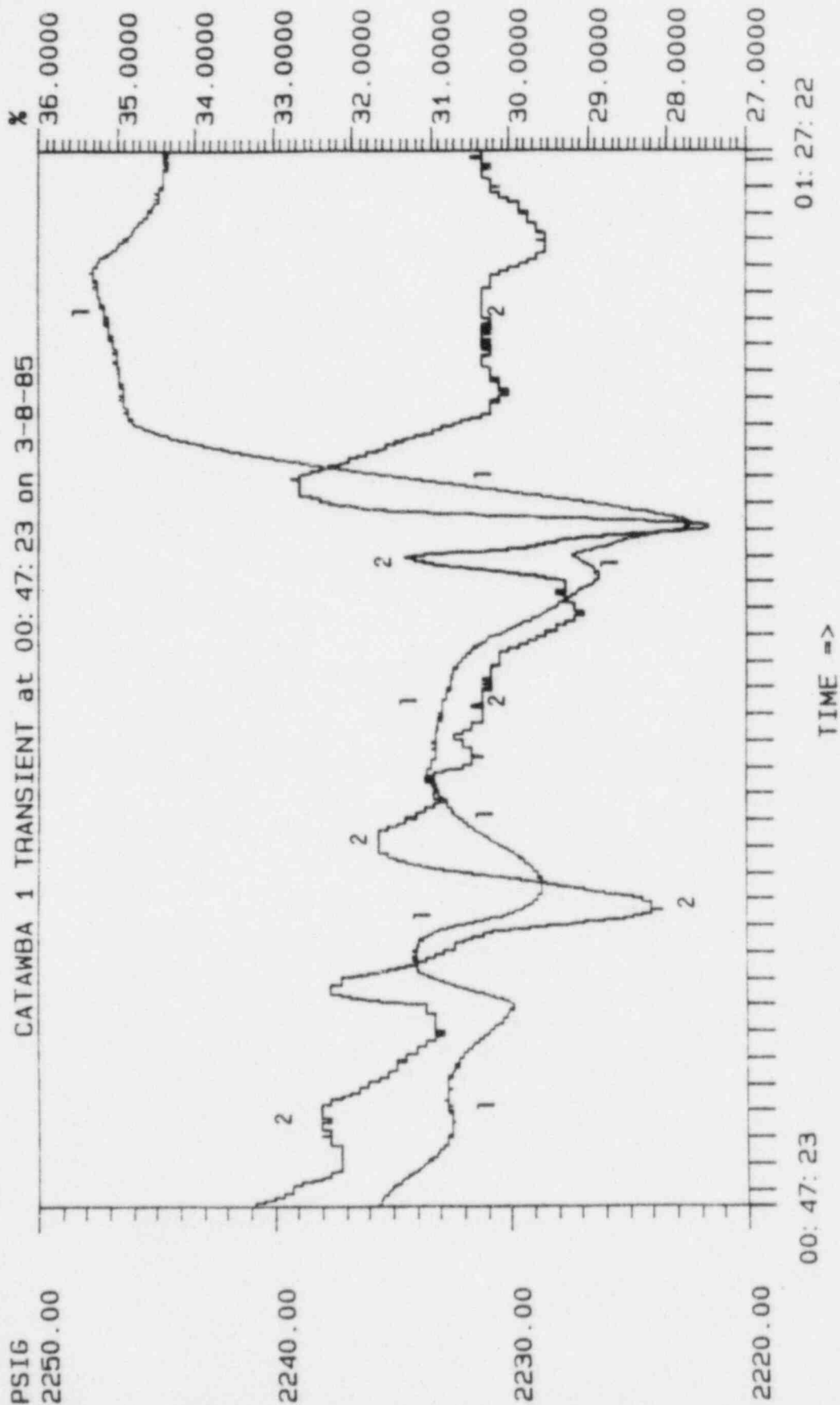
Figure 8.5-8

CATAWBA 1

	MAX	MIN	PID	MESSAGE
1	574.700	566.300	A0668	NC LOOP A WIDE RANGE HOT LEG TEMP
2	559.650	554.750	A0700	NC LOOP A WIDE RANGE COLD LEG TEMP

< Manual Freeze >

UNIT LOAD TRANSIENT TEST - 30% INCREASE
PRESSURIZER PRESSURE AND LEVEL TREND



	MAX	MIN	PID	MESSAGE
> 1	35.3500	27.6500	A0707	PZR LEVEL CH I
< 2	2240.80	2221.60	A0713	PZR PRESS CH 1

< Manual Freeze >

Figure 8.5-9

UNIT LOAD TRANSIENT TEST - 30% INCREASE
TYPICAL S/G PRESSURE AND LEVEL TREND

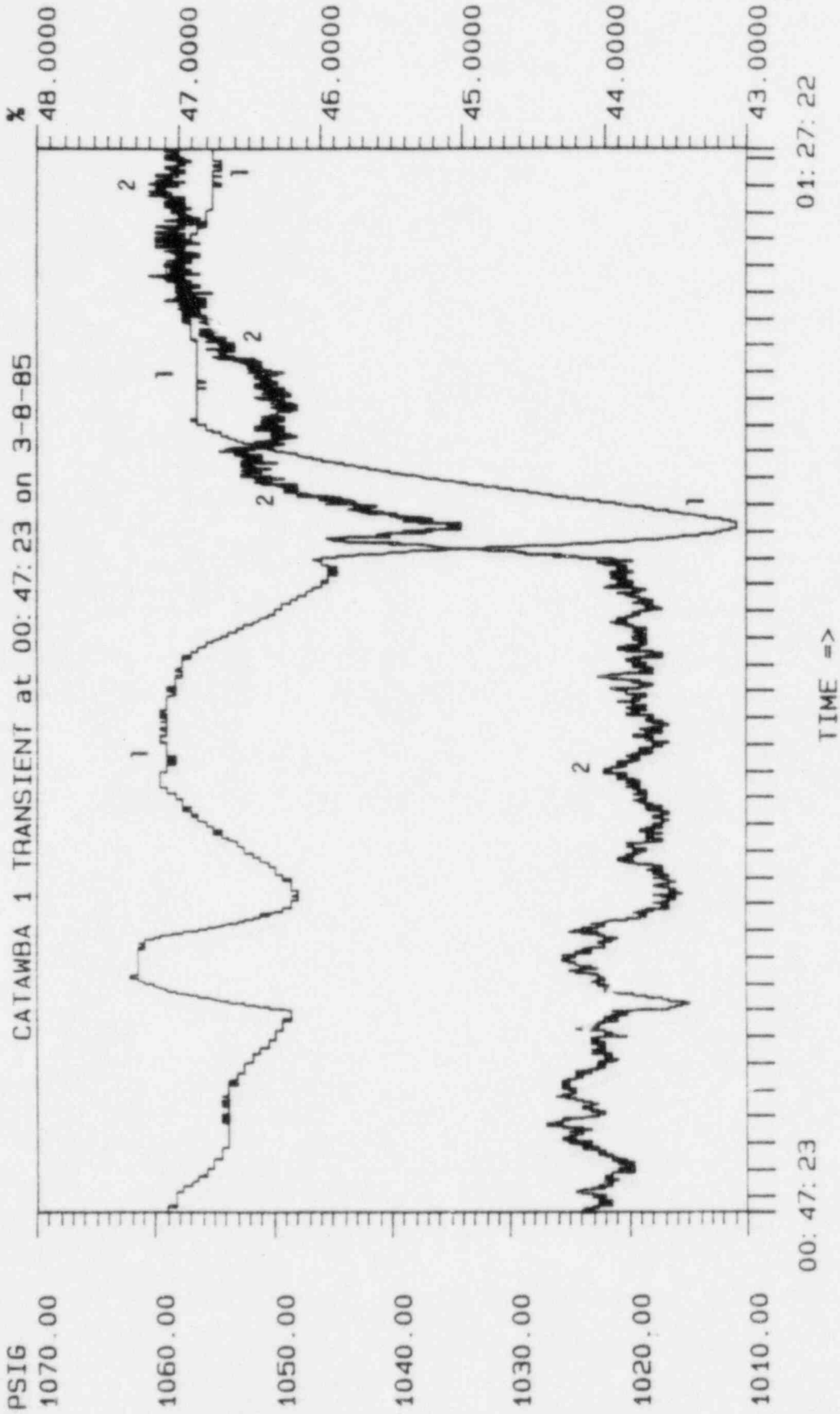


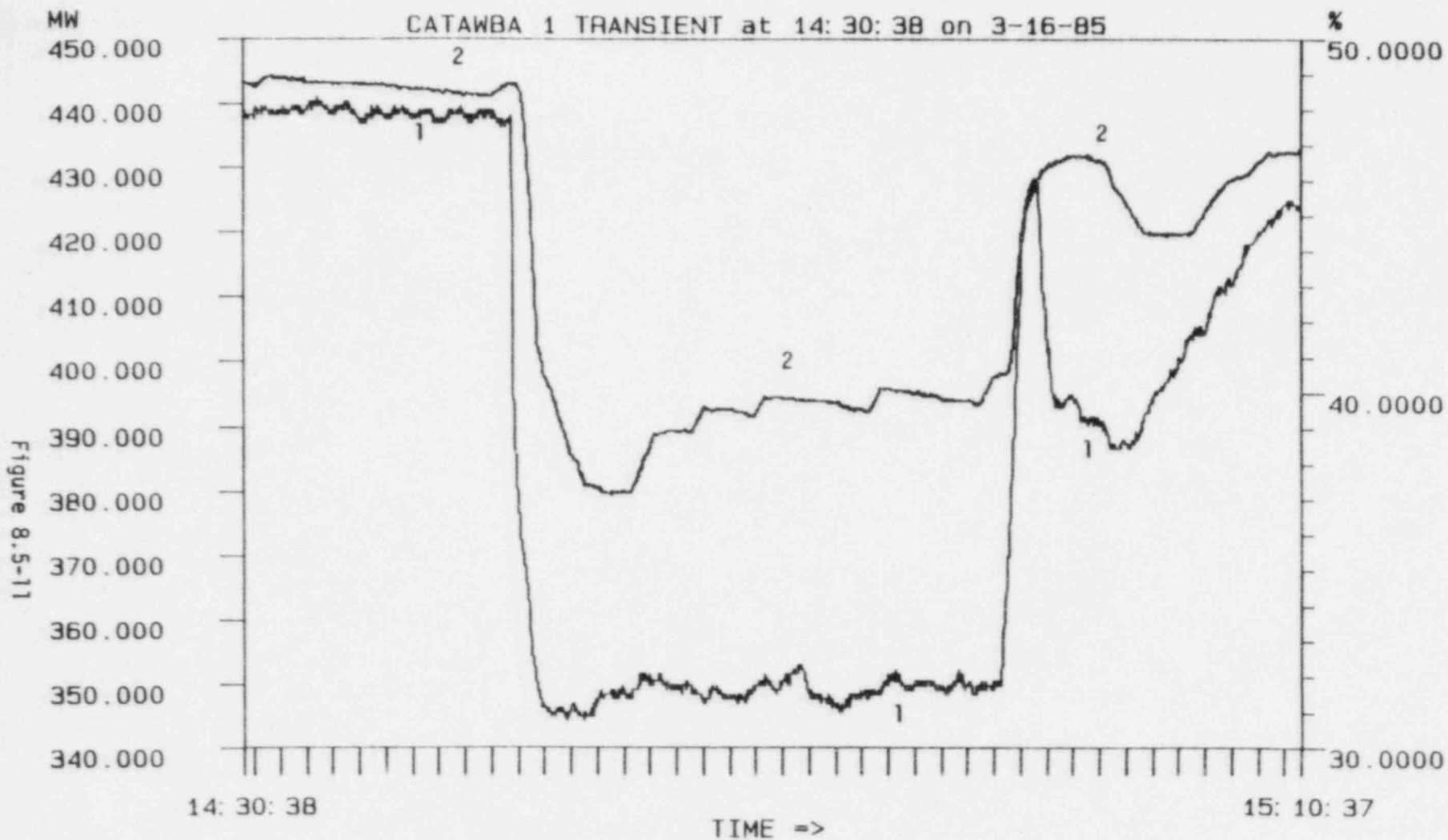
Figure 8.5-10

CATAWBA 1

	MAX	MIN	PID	MESSAGE
< 1	1062.10	1010.75	A1286	S/G B STEAM PRESS CH #2
> 2	47.2000	43.4000	A0537	S/G B NARROW RANGE LEVEL CH 2

< Manual Freeze >

UNIT LOAD TRANSIENT TEST - 50% DECREASE
PRIMARY AND SECONDARY POWER TREND



CATAWBA 1		MAX	MIN	PID	MESSAGE
>	1	441.090	344.162	A0697	GENERATOR MEGAWATTS
>	2	49.0200	37.1400	A0758	POWER RANGE AVG LEVEL QUADRANT 1 (N-43)

< Manual Freeze >

UNIT LOAD TRANSIENT TEST - 50% DECREASE
 NC AVERAGE TEMPERATURE TREND

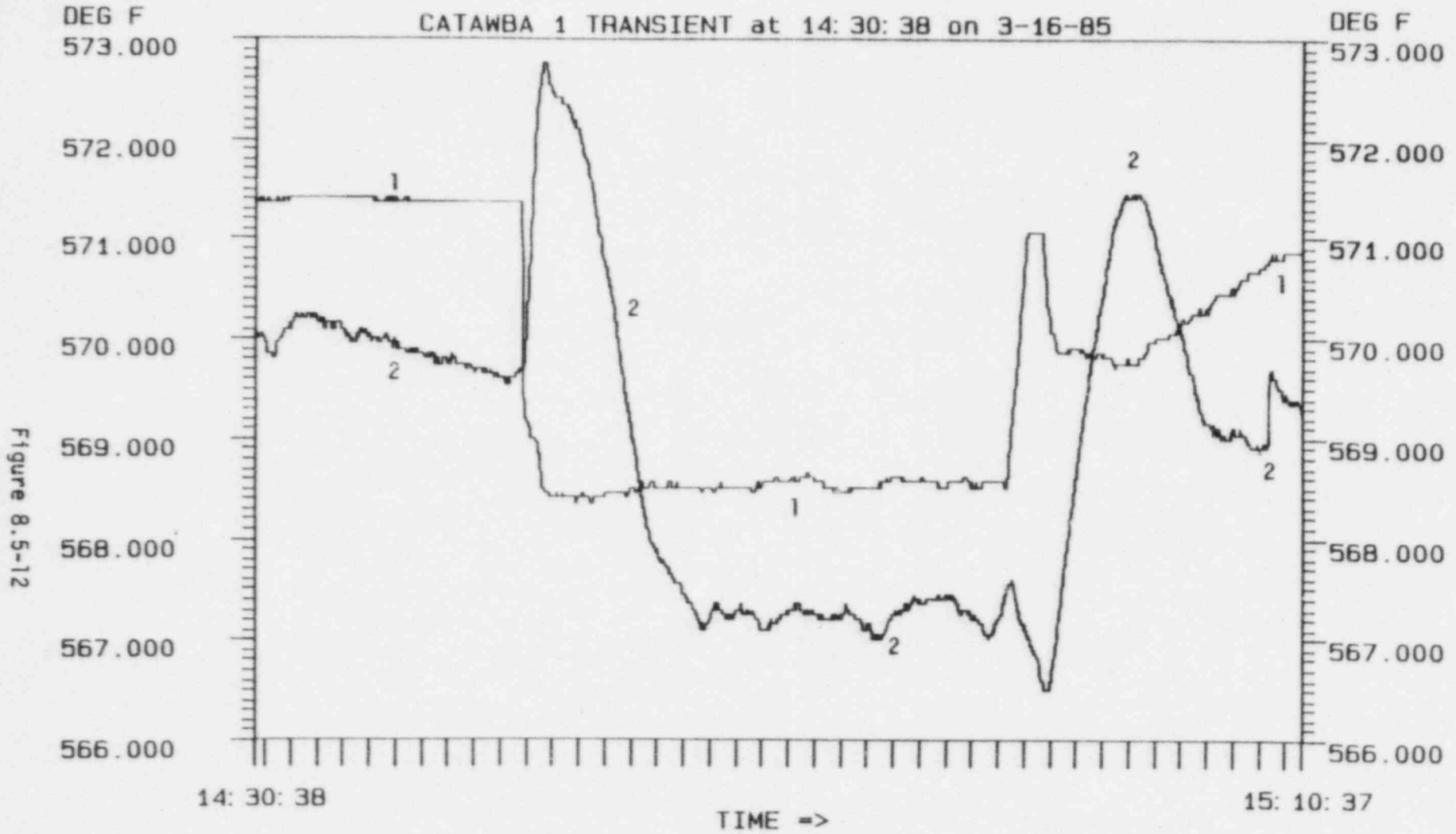


Figure 8.5-12

CATAWBA 1

	MAX	MIN	PID	MESSAGE
1	571.400	568.350	A0725	T-REF
2	572.750	566.500	A.7.1	NC LOOP HIGHEST T-AVG < Manual Freeze >

UNIT LOAD TRANSIENT TEST - 50% DECREASE
TYPICAL NC LOOP TEMPERAUTE TREND

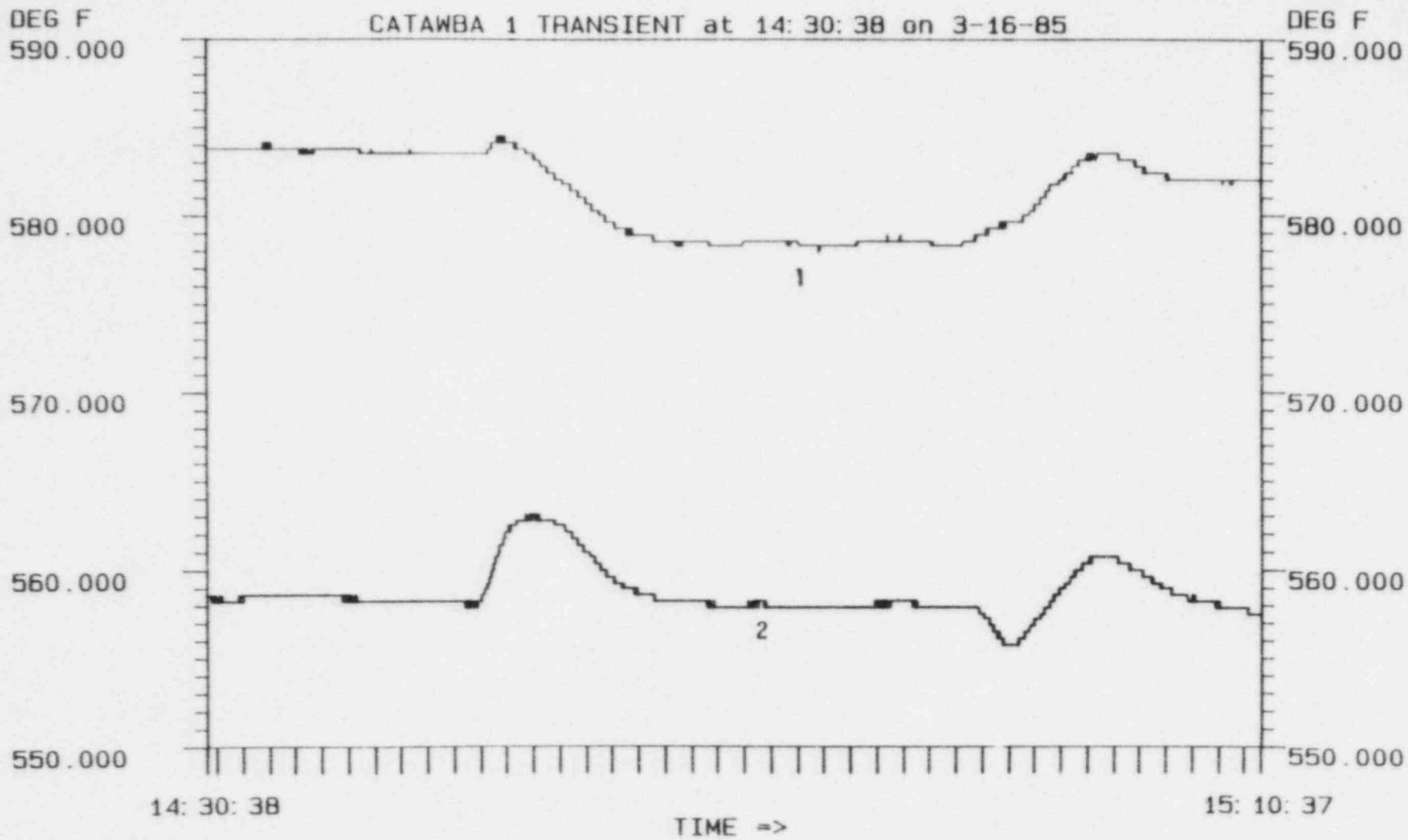


Figure 8.5-13

CATAWBA 1

	MAX	MIN	PID	MESSAGE
1	584.500	577.850	A0668	NC LOOP A WIDE RANGE HOT LEG TEMP
2	563.150	555.800	A0700	NC LOOP A WIDE RANGE COLD LEG TEMP

< Manual Freeze >

UNIT LOAD TRANSIENT TEST - 50% DECREASE
PRESSURIZER PRESSURE AND LEVEL TREND

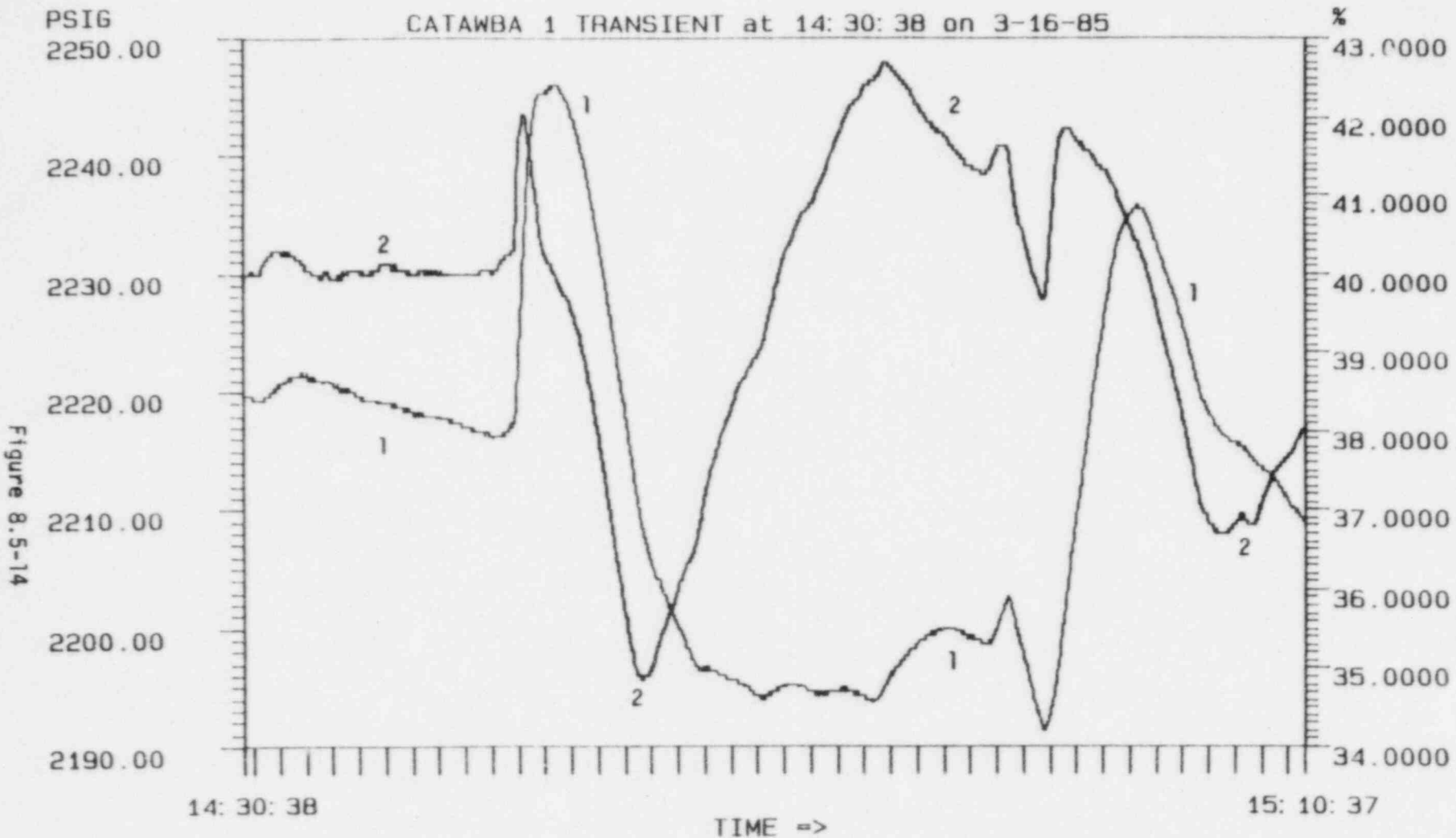


Figure 8.5-14

CATAWBA 1

	MAX	MIN	PID	MESSAGE
> 1	42.4000	34.2000	A0707	PZR LEVEL CH I
> 2	2248.00	2195.60	A0713	PZR PRESS CH 1

< Manual Freeze >

UNIT LOAD TRANSIENT TEST - 50% DECREASE
TYPICAL S/G PRESSURE AND LEVEL TREND

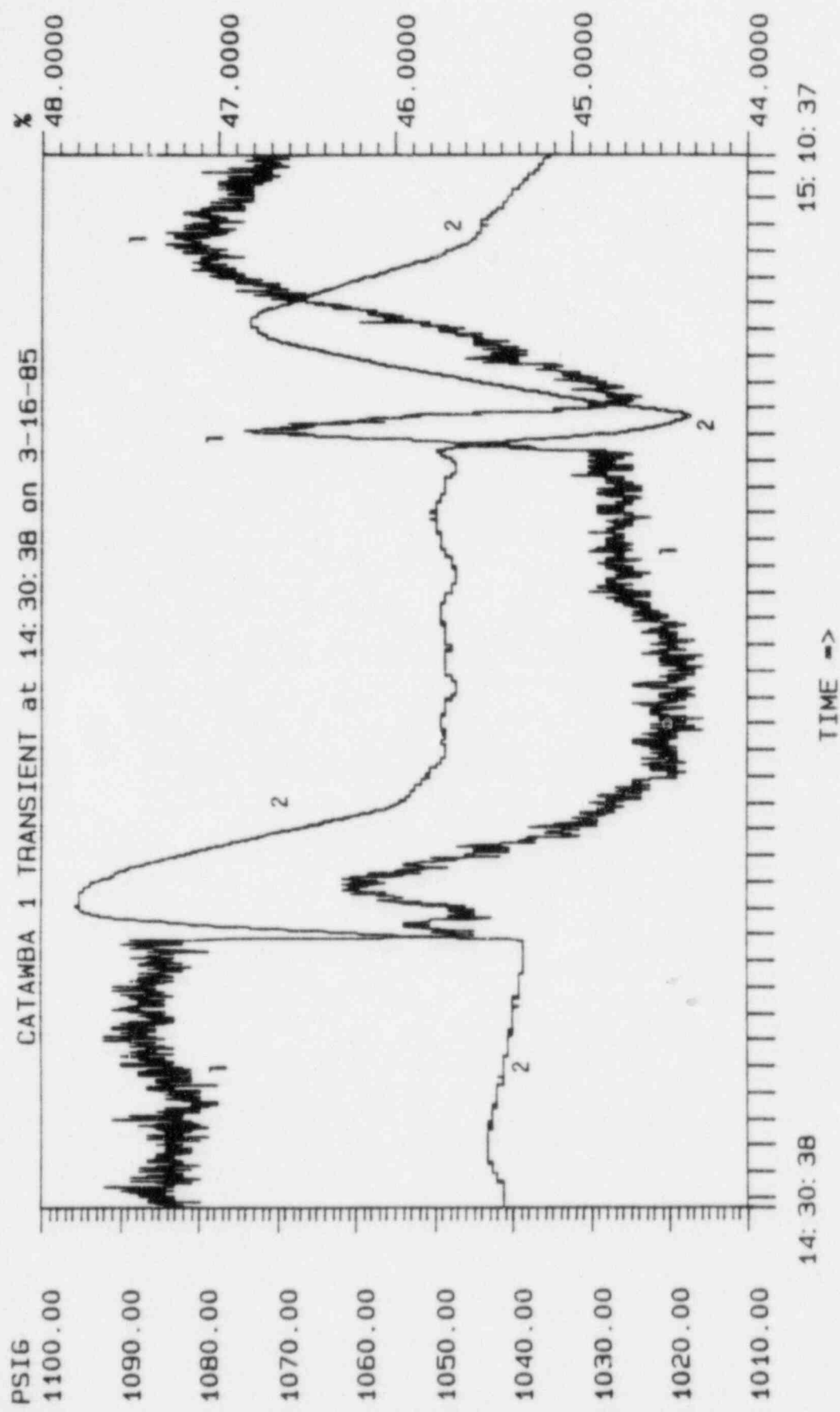


Figure 8.5-15

CATAWBA 1

	MAX	MIN	PID	MESSAGE
> 1	47.6500	44.2500	A0537	S/G B NARROW RANGE LEVEL CH 2
< 2	1095.90	1017.25	A1286	S/G B STEAM PRESS CH #2

< Manual Freeze >

UNIT LOAD TRANSIENT TEST - 50% INCREASE
PRIMARY AND SECONDARY POWER TREND

CATAWBA 1 TRANSTENT at 15:04:06 on 3-16-85

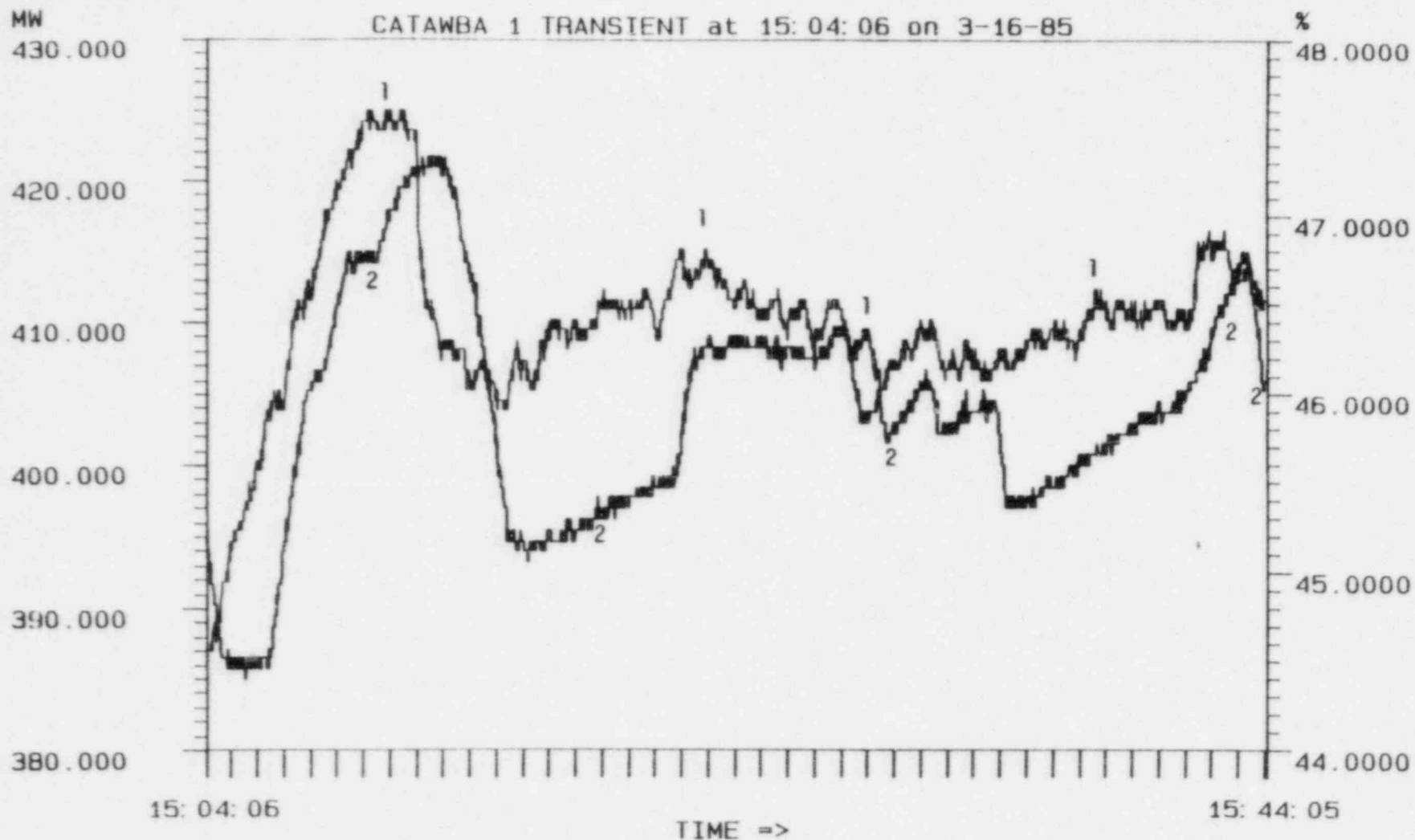


Figure 8.5-16

CATAWBA 1

	MAX	MIN	PID	MESSAGE
>	424.935	386.305	A0697	GENERATOR MEGAWATTS
>	47.3400	44.4000	A0758	POWER RANGE AVG LEVEL QUADRANT 1 (N-43)
	< Manual Freeze >			

UNIT LOAD TRANSIENT TEST - 50% INCREASE
 NC AVERAGE TEMPERATURE TREND

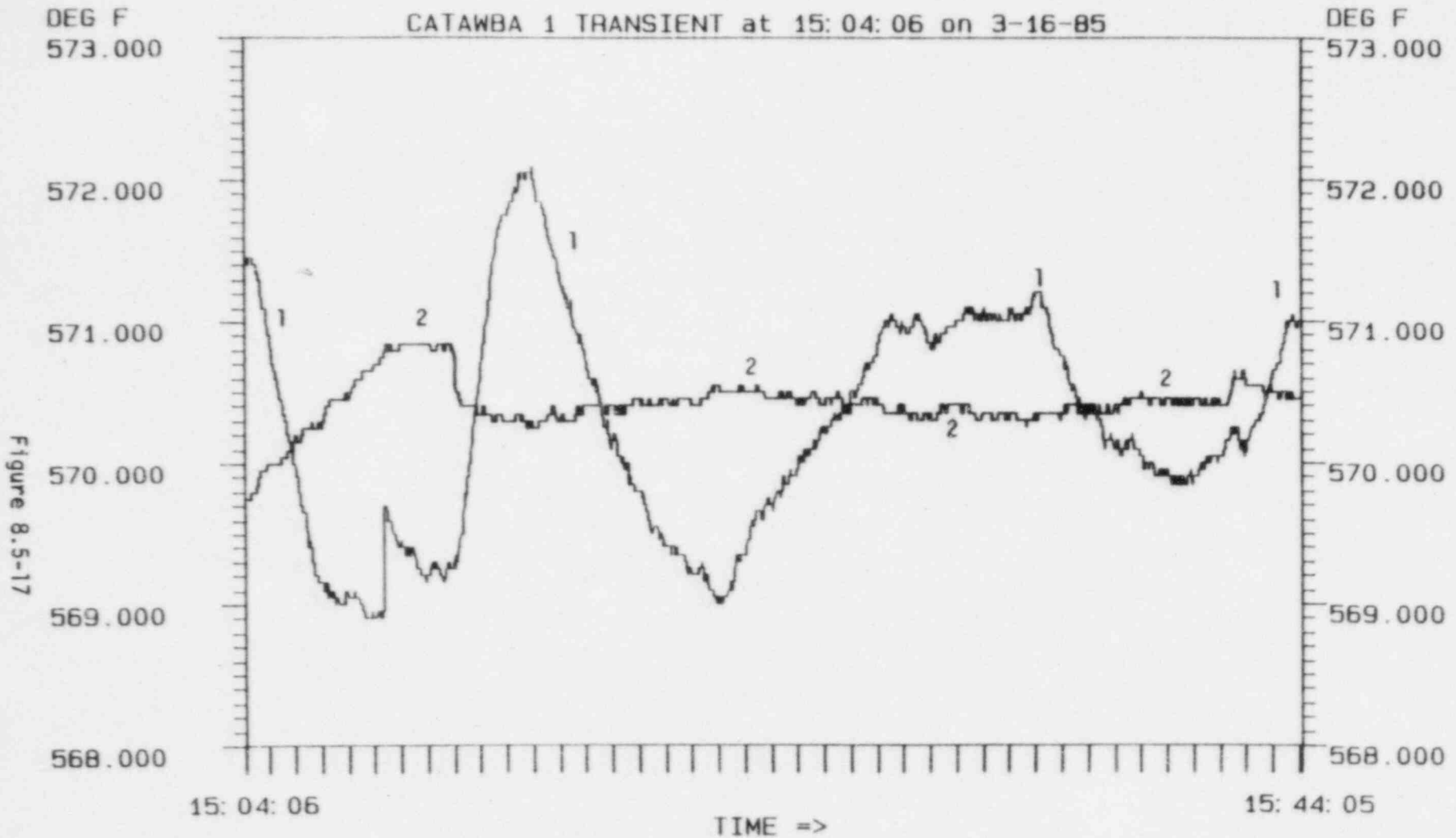


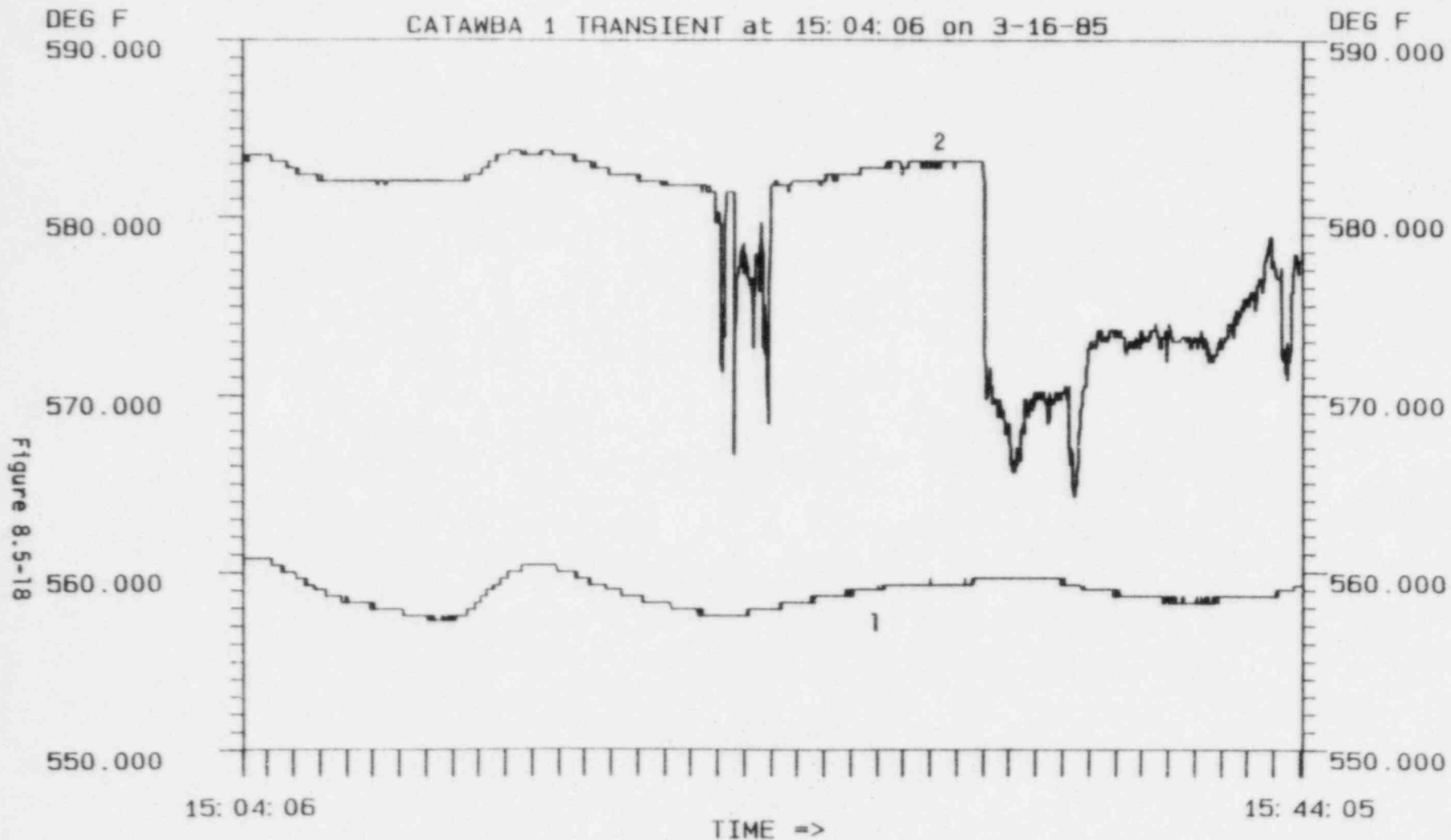
Figure 8.5-17

CATAWBA 1

	MAX	MIN	PID	MESSAGE
1	572.100	568.900	A0701	NC LOOP HIGHEST T-AVG
2	570.850	569.750	A0725	T-REF

< Manual Freeze >

UNIT LOAD TRANSIENT TEST - 50% INCREASE
TYPICAL NC LOOP TEMPERATURE TREND



CATAWBA 1

	MAX	MIN	PID	MESSAGE
1	560.700	557.200	A0700	NC LOOP A WIDE RANGE COLD LEG TEMP
2	583.800	564.200	A0668	NC LOOP A WIDE RANGE HOT LEG TEMP

< Manual Freeze >

UNIT LOAD TRANSIENT TEST - 50% INCREASE
PRESSURIZER PRESSURE AND LEVEL TREND

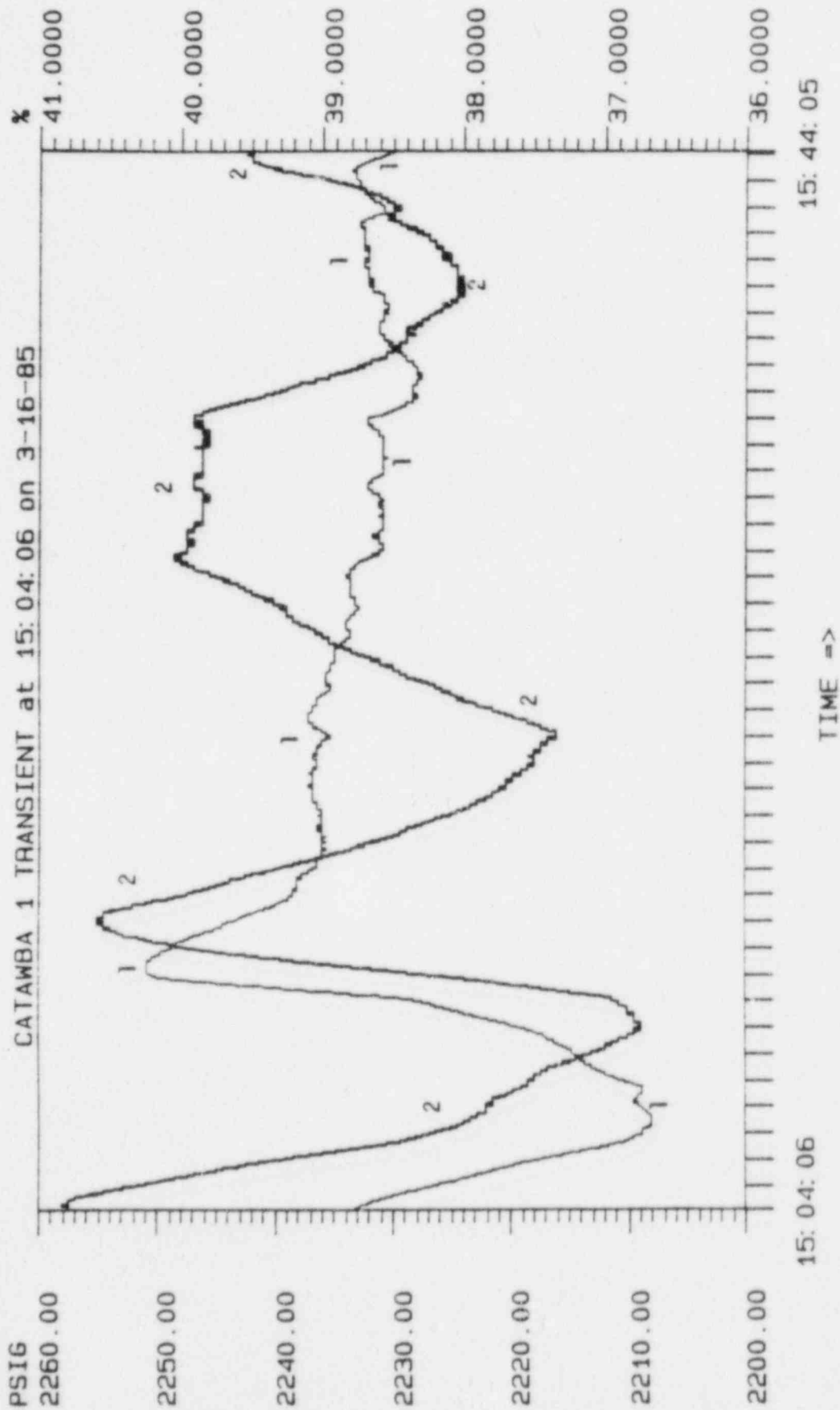


Figure 8.5-19

CATAWBA 1

	MAX	MIN	PID	MESSAGE
< 1	2250.80	2208.00	A0713	PZR PRESS CH 1
> 2	40.8500	36.7500	A0707	PZR LEVEL CH I

< Manual Freeze >

UNIT LOAD TRANSIENT TEST - 50% INCREASE
TYPICAL S/G PRESSURE AND LEVEL TREND

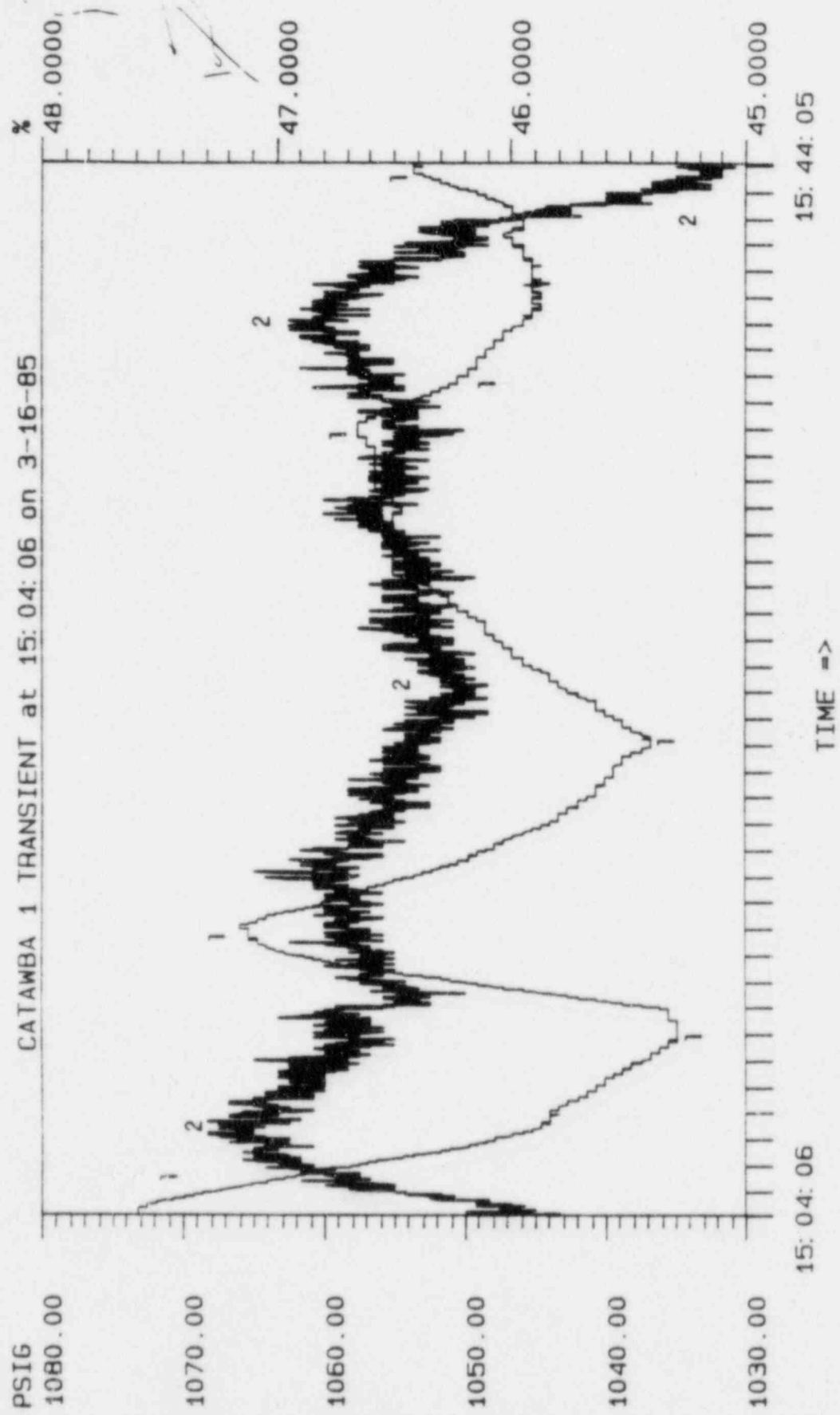


Figure 8.5-20

CATAWBA 1

	MAX	MIN	PID	MESSAGE
< 1	1073.15	1034.80	A1286	S/G B STEAM PRESS CH #2
> 2	47.3000	45.0500	A0537	S/G B NARROW RANGE LEVEL CH 2

< Manual Freeze >

UNIT LOAD TRANSIENT TEST - 75% DECREASE
PRIMARY AND SECONDARY POWER TREND

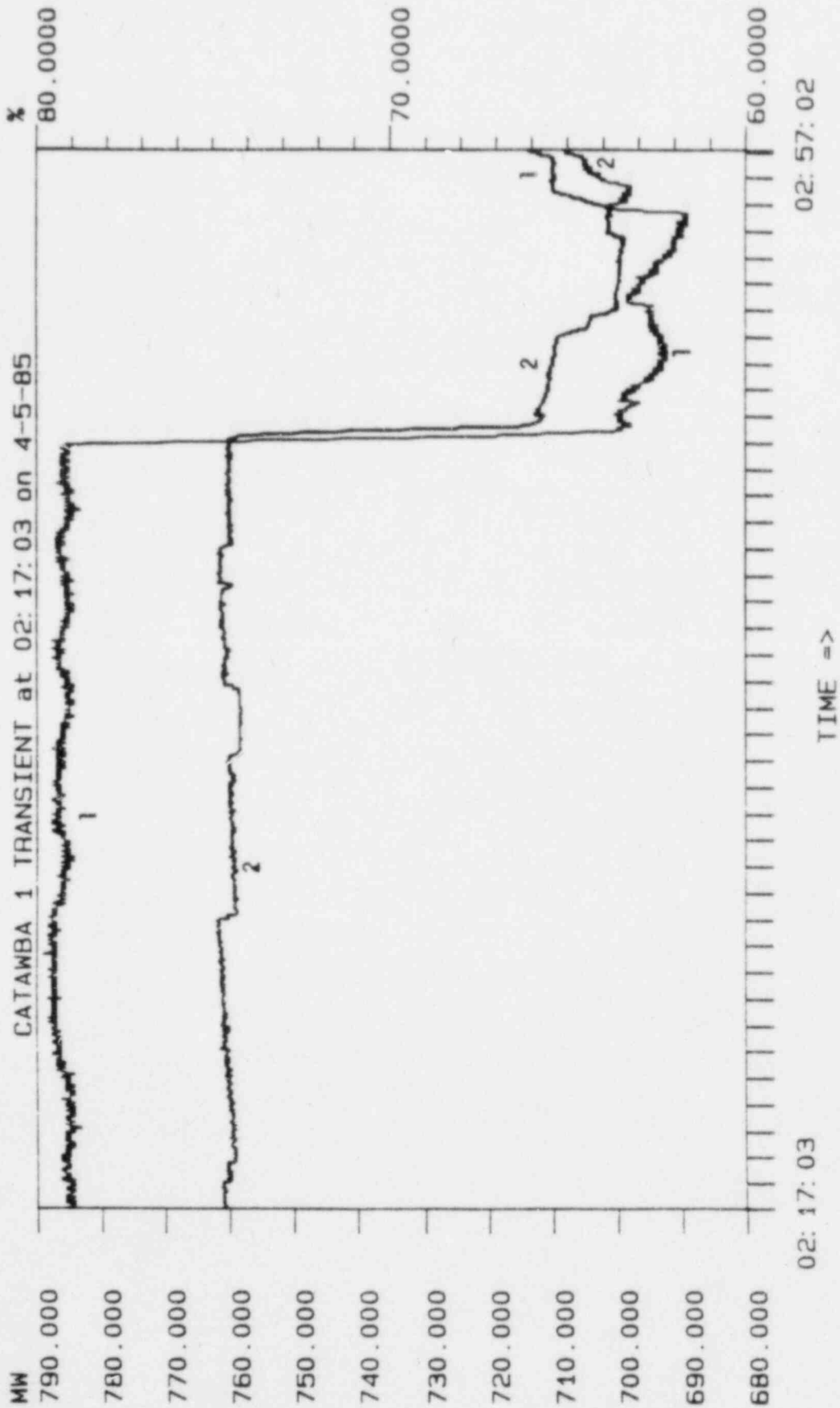


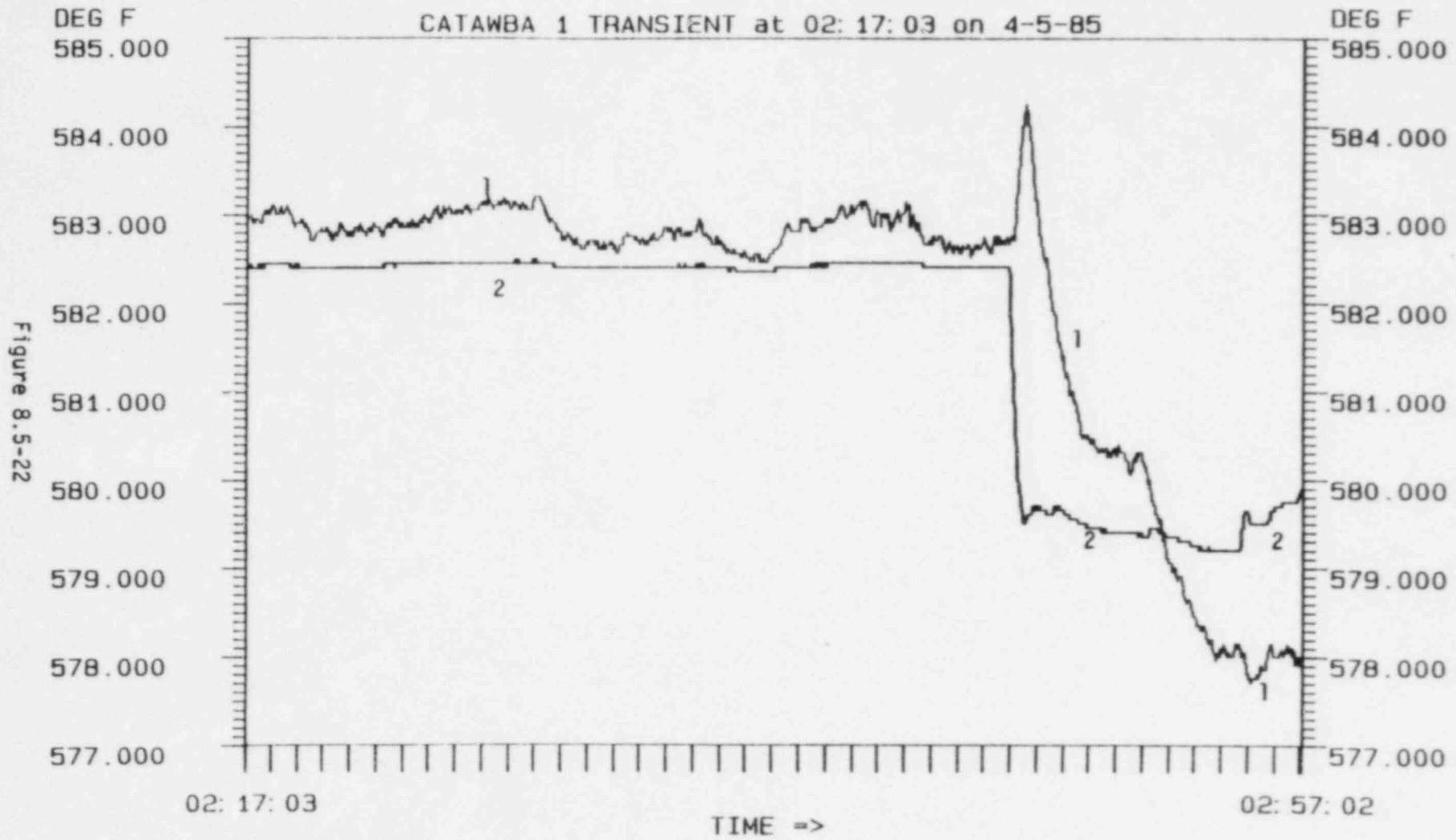
Figure 8.5-21

CATAWBA 1

	MAX	MIN	PID	MESSAGE
< 1	788.873	689.090	A0697	GENERATOR MEGAWATTS
> 2	74.9400	63.3600	A0758	POWER RANGE AVG LEVEL QUADRANT 1 (N-43)

< Manual Freeze >

UNIT LOAD TRANSIENT TEST - 75% DECREASE
NC AVERAGE TEMPERATURE TREND



CATAWBA 1

	MAX	MIN	PID	MESSAGE
1	584.250	577.700	A0701	NC LOOP HIGHEST T-AVG
2	582.500	579.200	A0725	T-REF

< Manual Freeze >

UNIT LOAD TRANSIENT TEST - 75% DECREASE
TYPICAL NC LOOP TEMPERATURE TREND

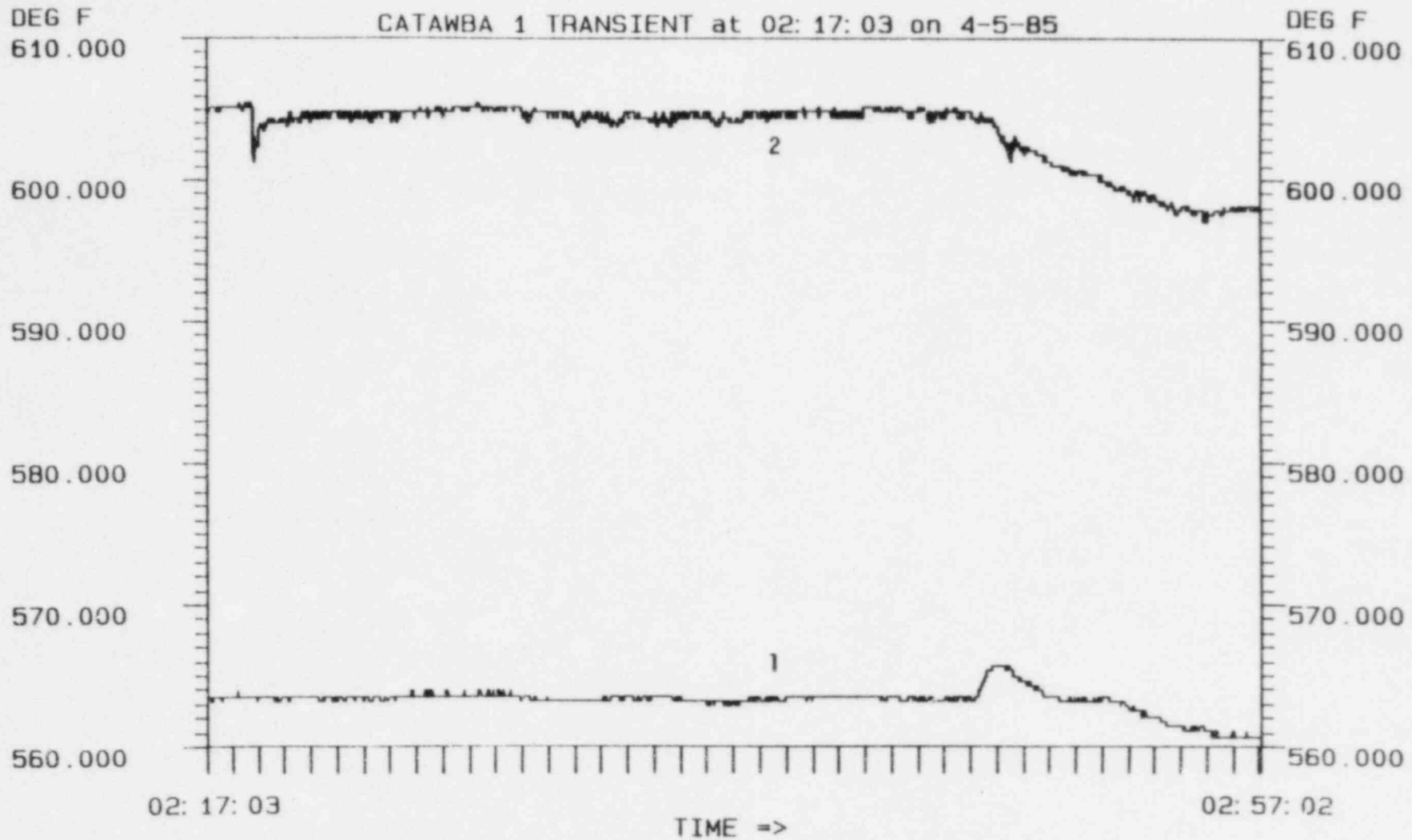


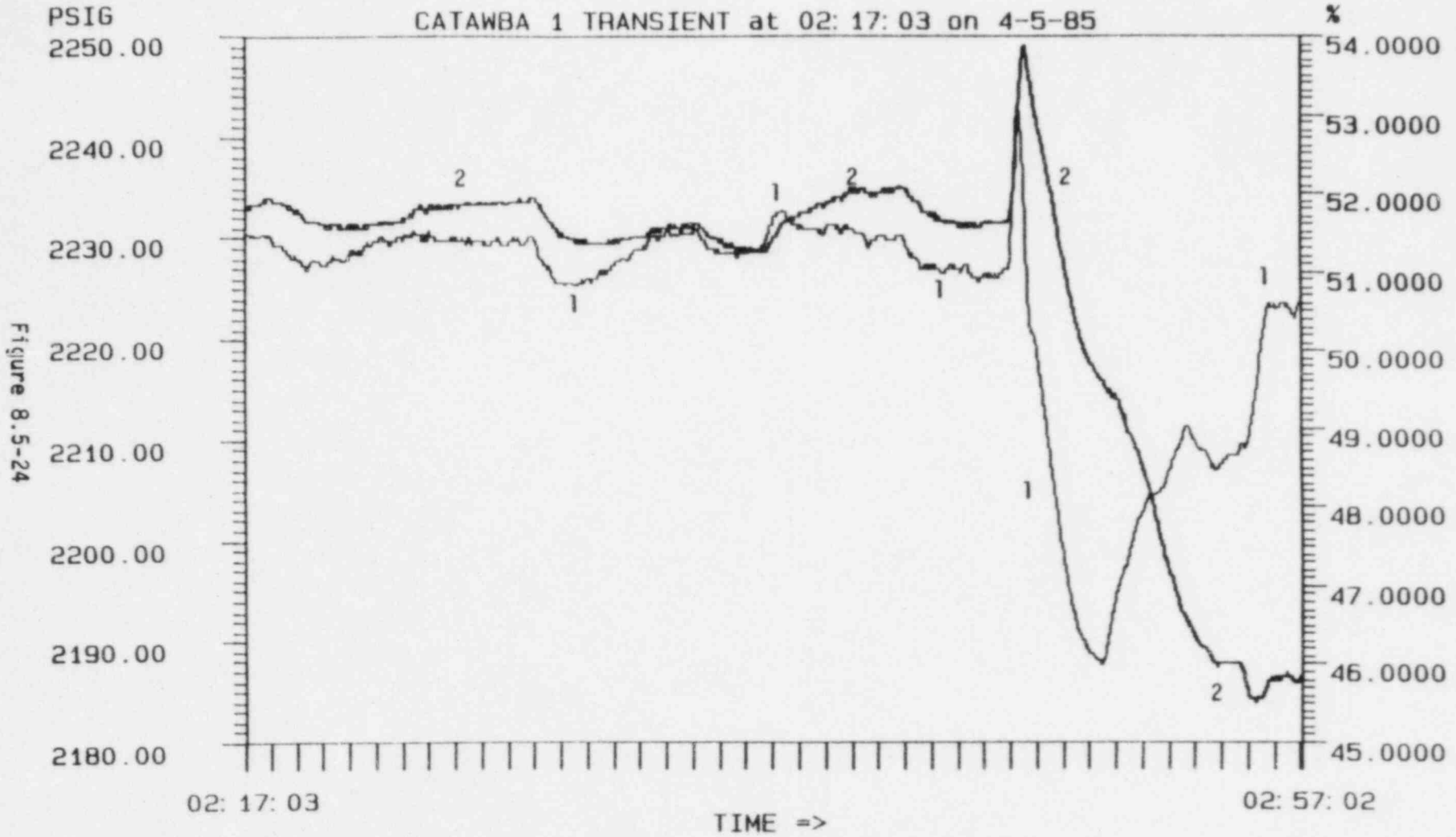
Figure 8.5-23

CATAWBA 1

	MAX	MIN	PID	MESSAGE
1	565.600	560.350	A0700	NC LOOP A WIDE RANGE COLD LEG TEMP
2	605.500	597.100	A0668	NC LOOP A WIDE RANGE HOT LEG TEMP

< Manual Freeze >

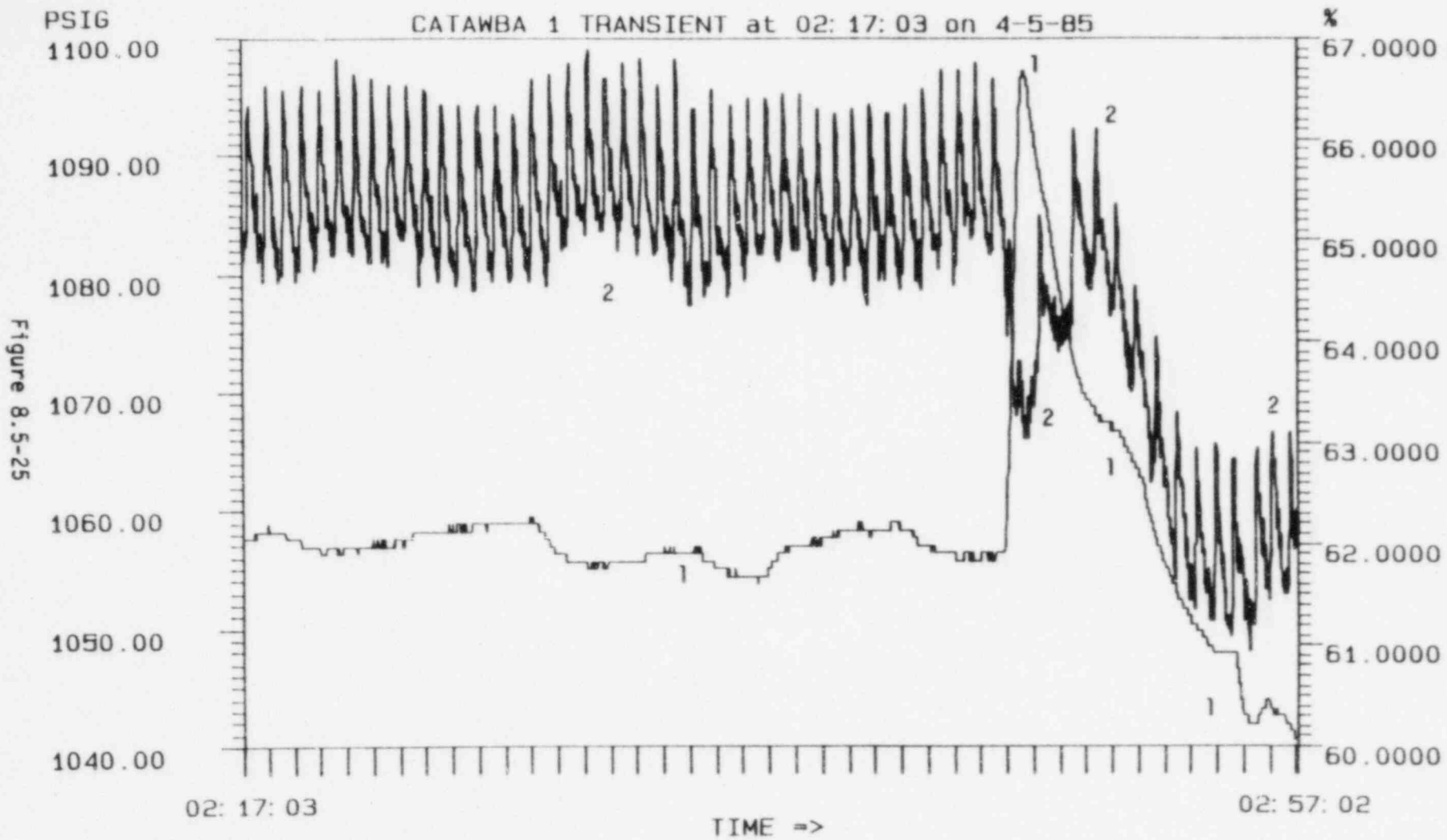
UNIT LOAD TRANSIENT TEST - 75% DECREASE
PRESSURIZER PRESSURE AND LEVEL TREND



CATAWBA 1		MAX	MIN	PID	MESSAGE
<	1	2242.80	2187.60	A0713	PZR PRESS CH 1
>	2	53.9000	45.5000	A0707	PZR LEVEL CH I

< Manual Freeze >

UNIT LOAD TRANSIENT TEST - 75% DECREASE
TYPICAL S/G PRESSURE AND LEVEL TRENDS



CATAWBA 1

	MAX	MIN	PID	MESSAGE
< 1	1097.20	1040.65	A1286	S/G B STEAM PRESS CH #2
> 2	66.9000	60.9500	A0537	S/G B NARROW RANGE LEVEL CH 2

< Manual Freeze >

UNIT LOAD TRANSIENT TEST - 75% INCREASE
PRIMARY AND SECONDARY POWER TREND

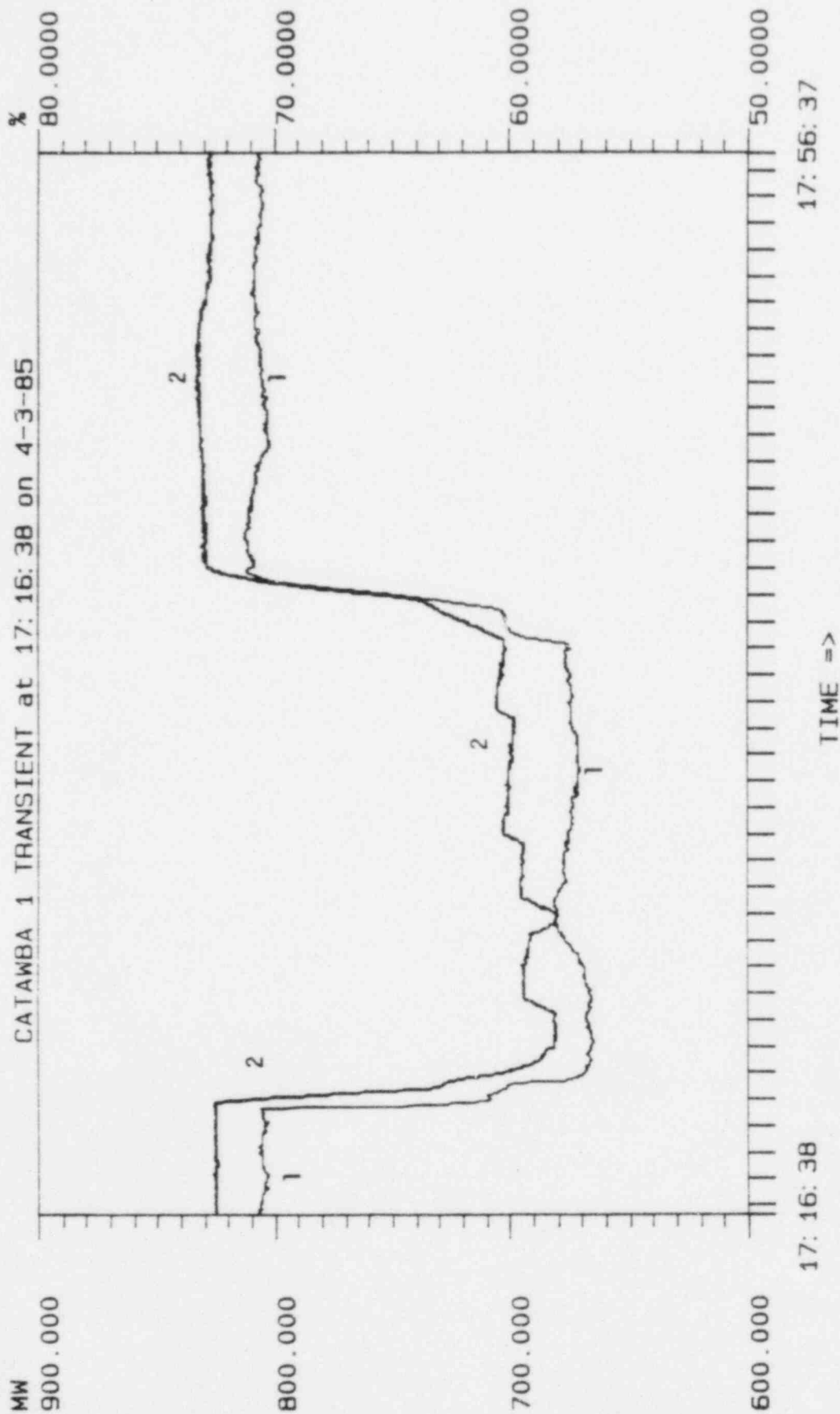


Figure 8.5-26

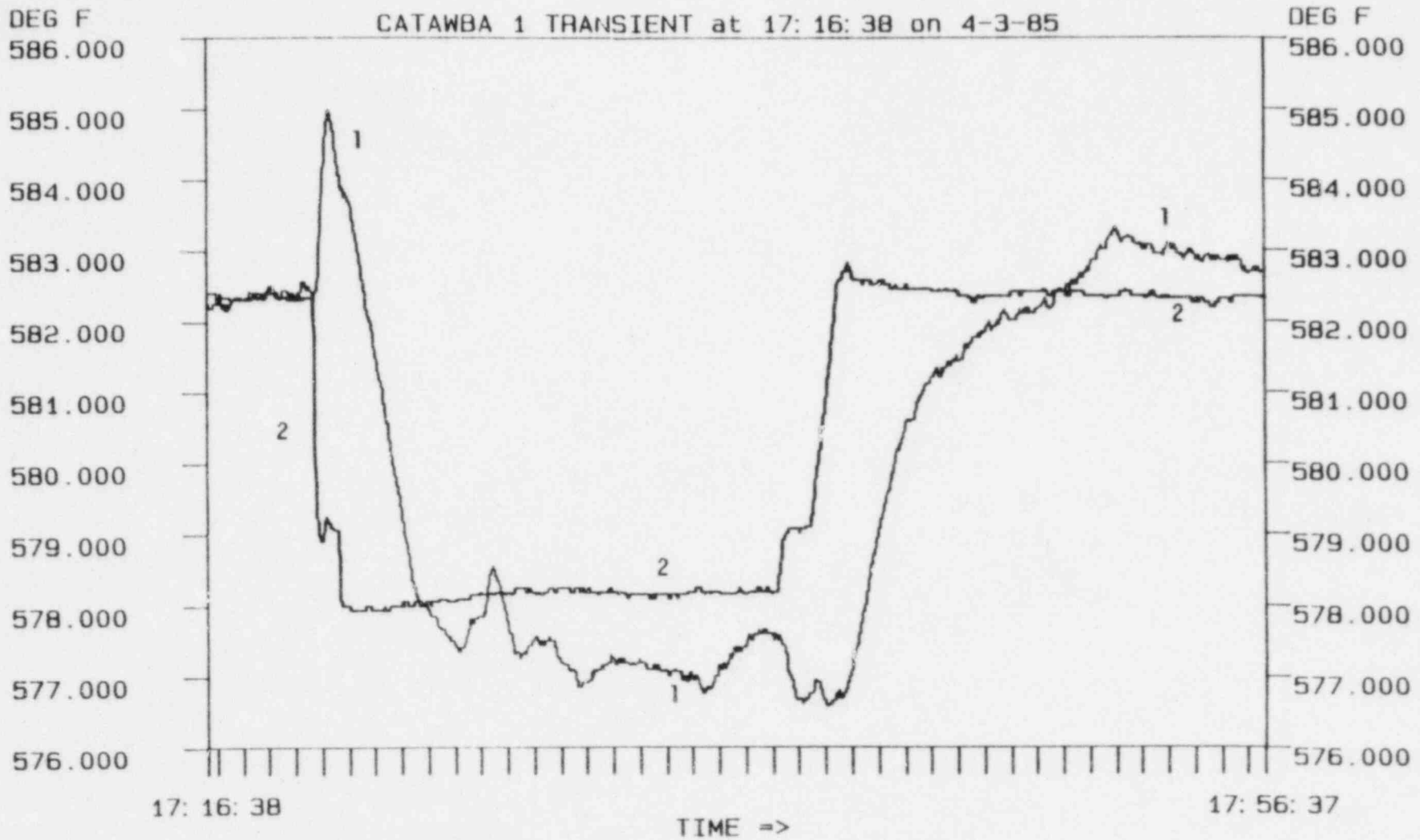
CATAWBA 1

	MAX	MIN	PID	MESSAGE
< 1	812.597	664.668	A0697	GENERATOR MEGAWATTS
> 2	73.3200	57.9600	A0758	POWER RANGE AVG LEVEL QUADRANT 1 (N-43)

< Manual Freeze >

UNIT LOAD TRANSIENT TEST - 75% INCREASE
 NC AVERAGE TEMPERATURE TREND

Figure 8.5-27



CATAWBA 1

	MAX	MIN	PID	MESSAGE
1	585.000	576.550	A0701	NC LOOP HIGHEST T-AVG
2	582.800	577.950	A0725	T-REF

< Manual Freeze >

UNIT LOAD TRANSIENT TEST - 75% INCREASE
TYPICAL NC LOOP TEMPERATURE TREND

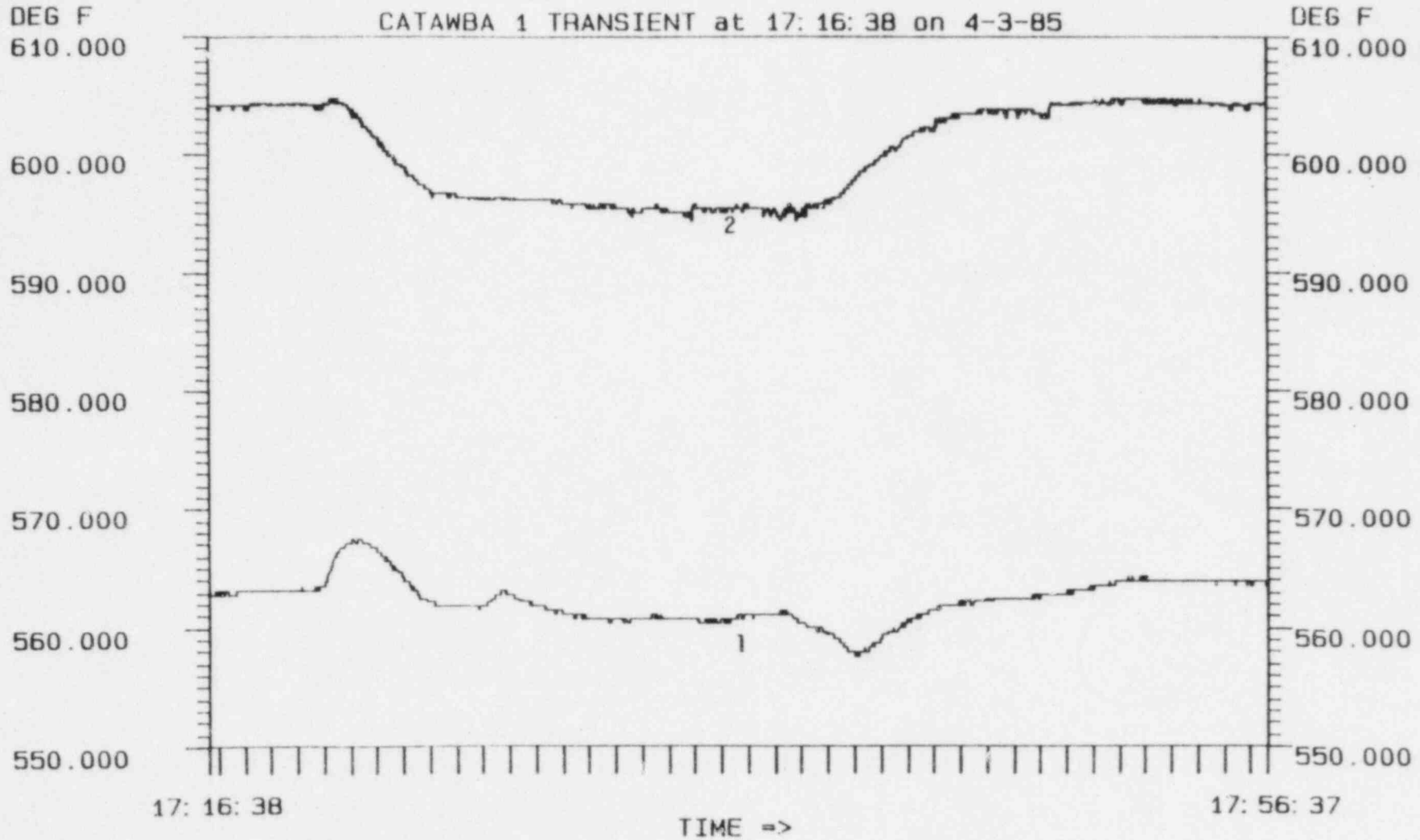


Figure 8.5-28

CATAWBA 1

	MAX	MIN	PID	MESSAGE
1	567.350	557.550	A0700	NC LOOP A WIDE RANGE COLD LEG TEMP
2	604.800	594.300	A0668	NC LOOP A WIDE RANGE HOT LEG TEMP

< Manual Freeze >

UNIT LOAD TRANSIENT TEST - 75% INCREASE
PRESSURIZER PRESSURE AND LEVEL TREND

CATAWBA 1 TRANSIENT at 17:16:38 on 4-3-85

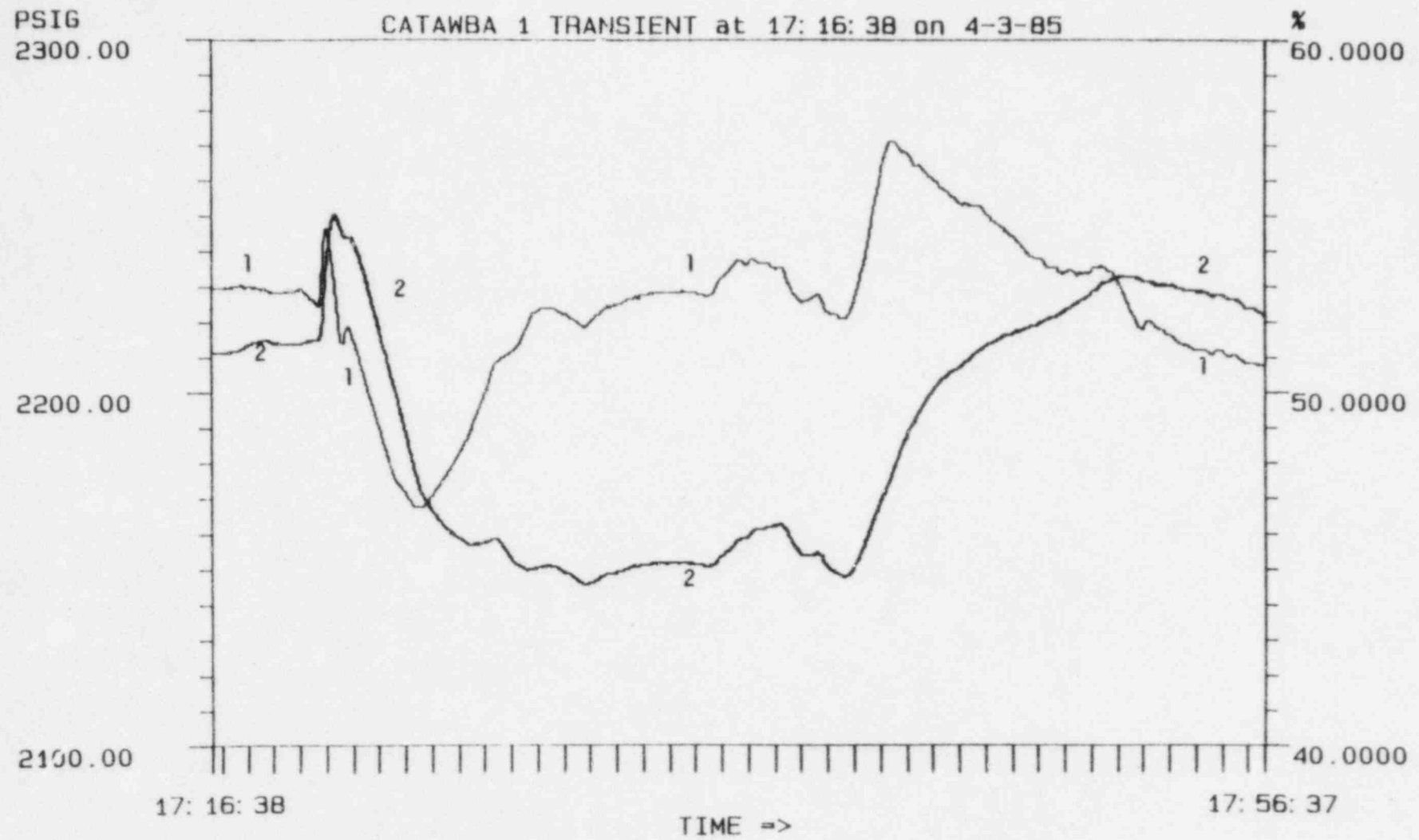


Figure 8.5-29

CATAWBA 1		MAX	MIN	PID	MESSAGE
<	1	2270.80	2167.20	A0713	PZR PRESS CH 1
>	2	55.0500	44.5500	A0707	PZR LEVEL CH I
					< Manual Freeze >

UNIT LOAD TRANSIENT TEST - 75% INCREASE
TYPICAL S/G PRESSURE AND LEVEL TREND

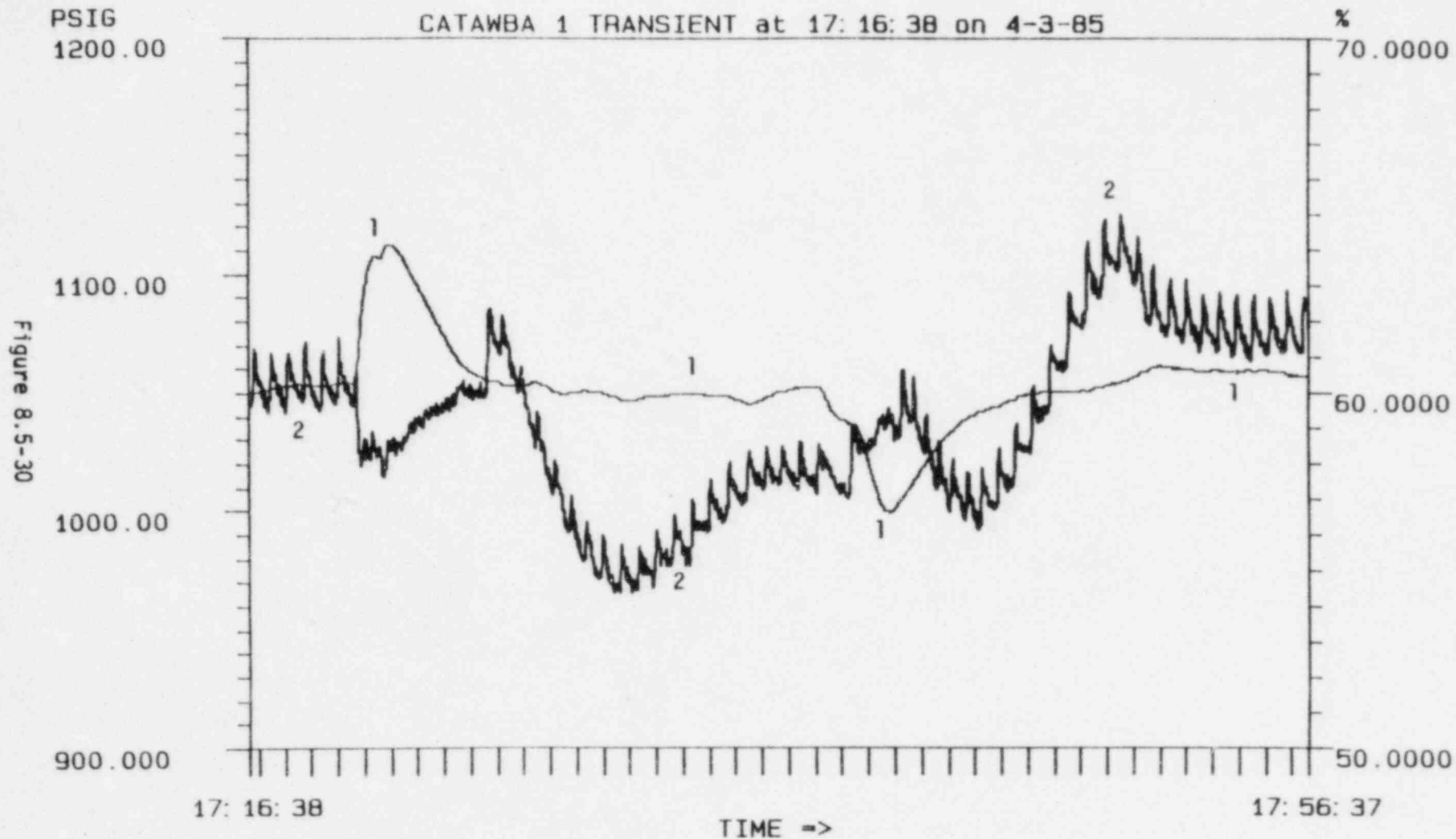


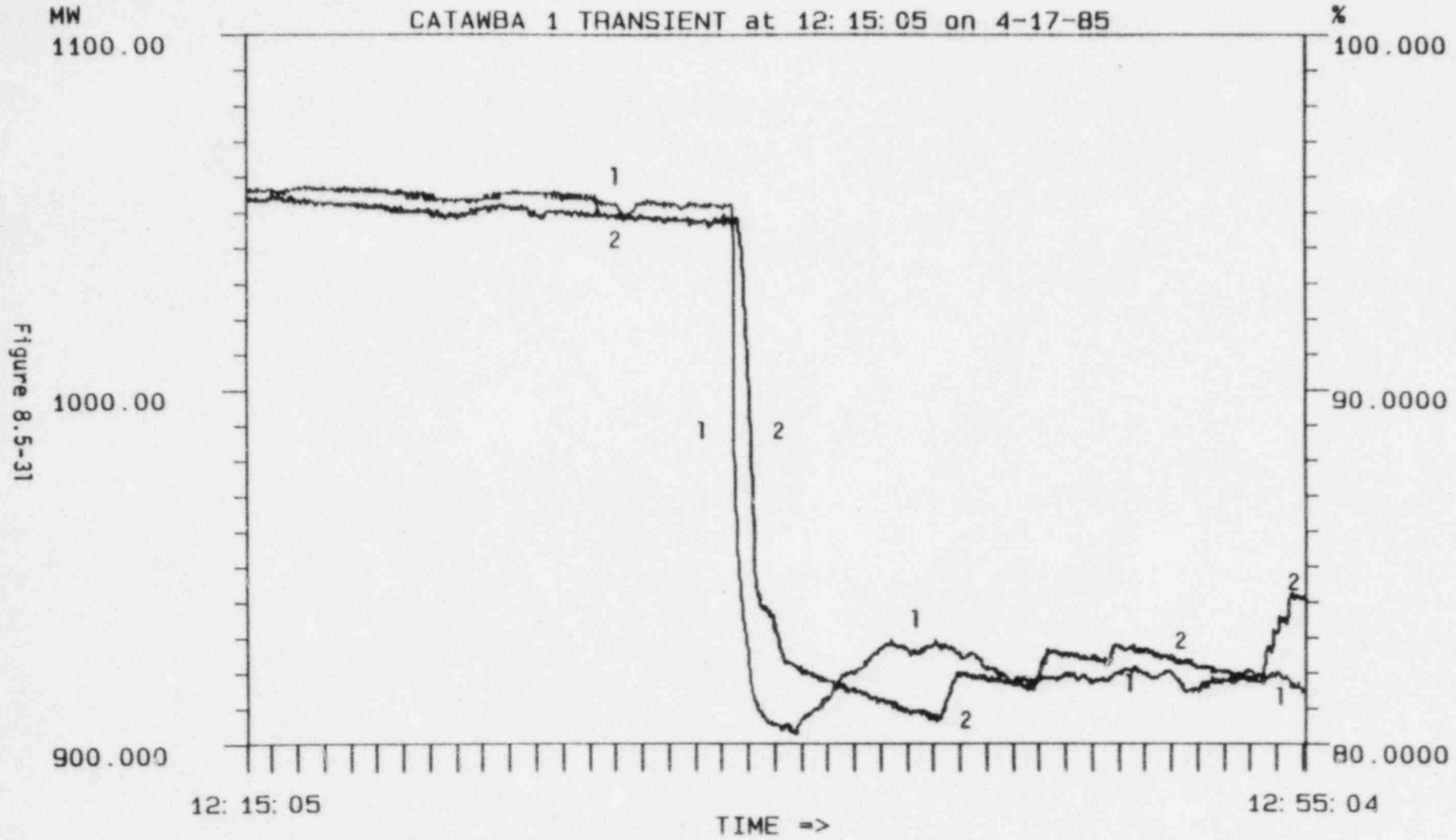
Figure 8.5-30

CATAWBA 1

	MAX	MIN	PID	MESSAGE
< 1	1112.15	998.400	A1286	S/G B STEAM PRESS CH #2
> 2	65.0000	54.4000	A0537	S/G B NARROW RANGE LEVEL CH 2

< Manual Freeze >

UNIT LOAD TRANSIENT TEST - 100% DECREASE
PRIMARY AND SECONDARY POWER TREND



CATAWBA 1		MAX	MIN	PID	MESSAGE
>	1	1057.51	902.610	A0697	GENERATOR MEGAWATTS
>	2	95.5800	80.6400	A0758	POWER RANGE AVG LEVEL QUADRANT 1 (N-43)

< Manual Freeze >

UNIT LOAD TRANSIENT TEST - 100% DECREASE
 NC AVERAGE TEMPERATURE TREND

CATAWBA 1 TRANSIENT at 12:15:05 on 4-17-85

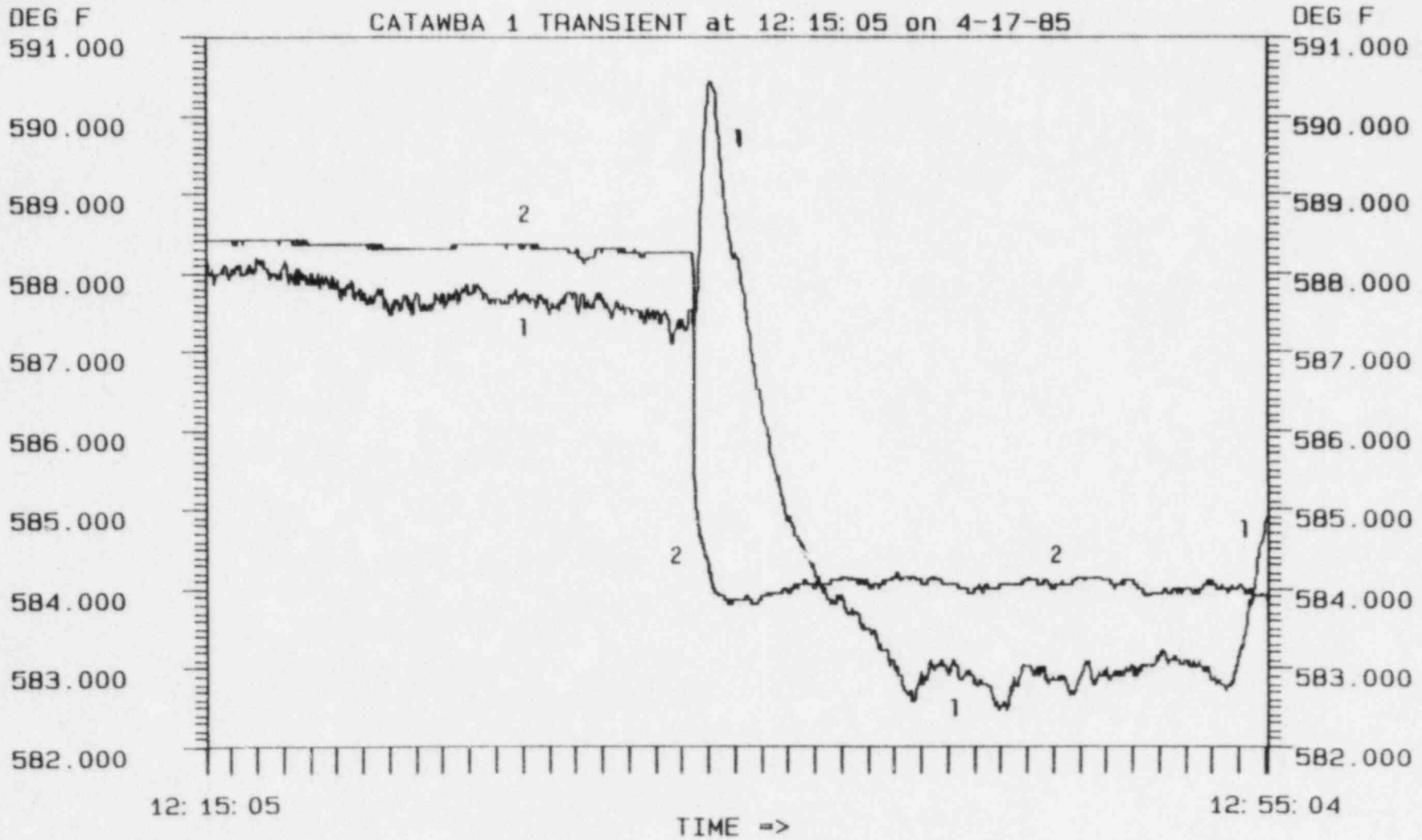


Figure 8.5-32

CATAWBA 1

	MAX	MIN	PID	MESSAGE
1	590.450	582.450	A0701	NC LOOP HIGHEST T-AVG
2	588.400	583.800	A0725	T-REF

< Manual Freeze >

UNIT LOAD TRANSIENT TEST - 100% DECREASE
TYPICAL NC LOOP TEMPERATURE TREND

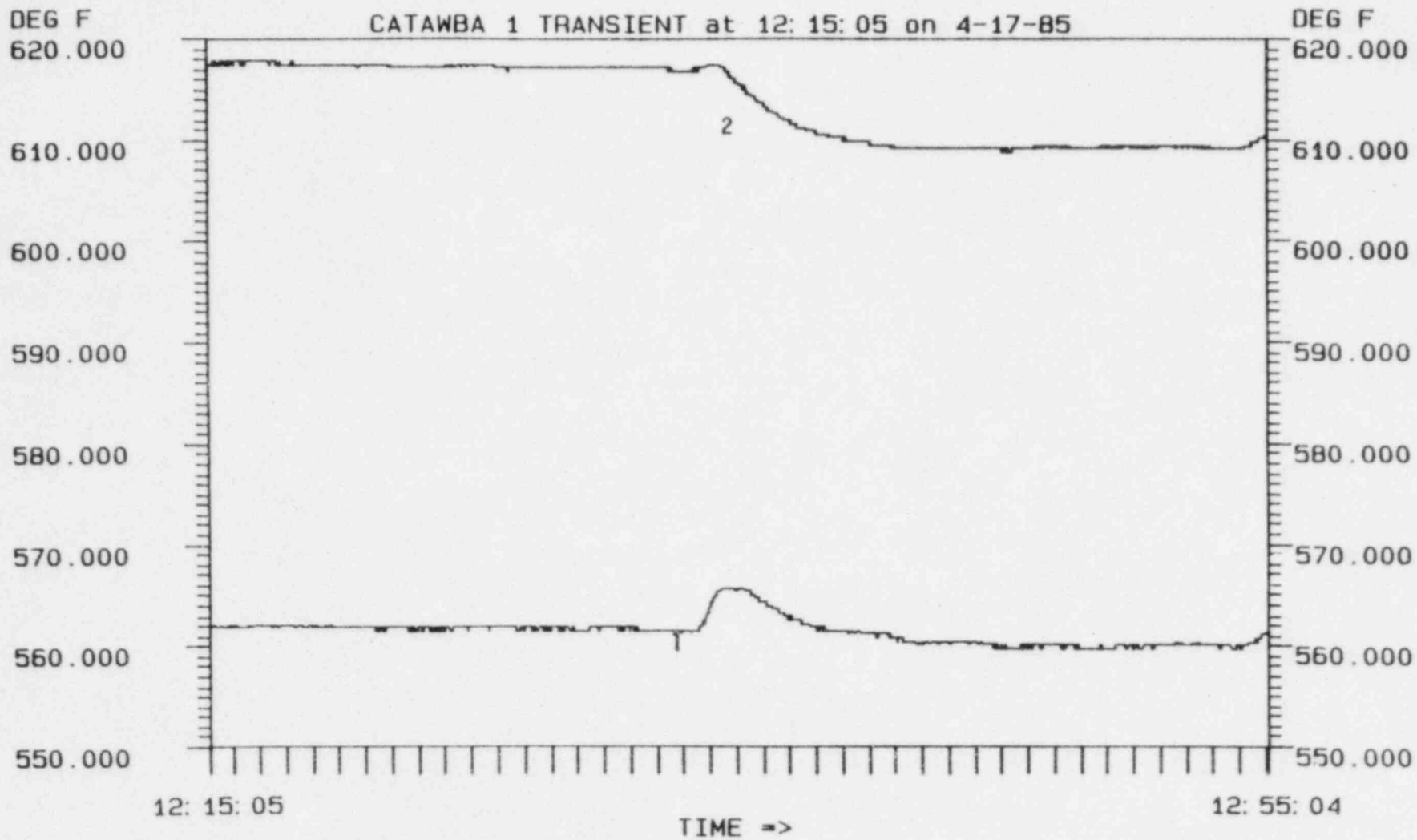


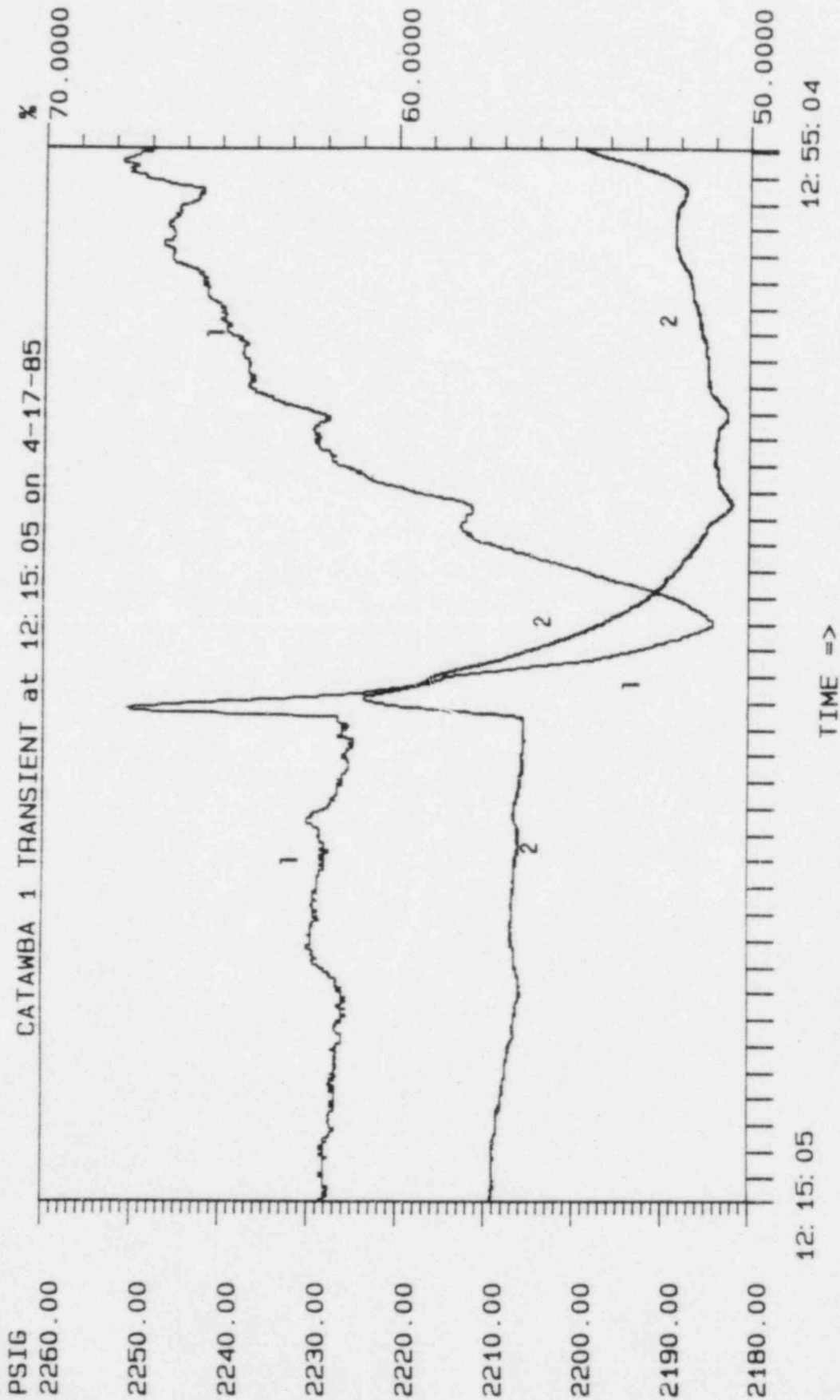
Figure 8.5-33

CATAWBA 1

	MAX	MIN	PID	MESSAGE
1	565.600	559.650	A0700	NC LOOP A WIDE RANGE COLD LEG TEMP
2	617.750	608.650	A0668	NC LOOP A WIDE RANGE HOT LEG TEMP

< Manual Freeze >

UNIT LOAD TRANSIENT TEST - 100% DECREASE
PRESSURIZER PRESSURE AND LEVEL TREND



	MAX	MIN	PID	MESSAGE
< 1	2251.20	2184.00	A0713	PZR PRESS CH 1
> 2	60.9500	50.4500	A0707	PZR LEVEL CH I

< Manual Freeze >

Figure 8.5-34

UNIT LOAD TRANSIENT TEST - 100% DECREASE
TYPICAL S/G PRESSURE AND LEVEL TREND

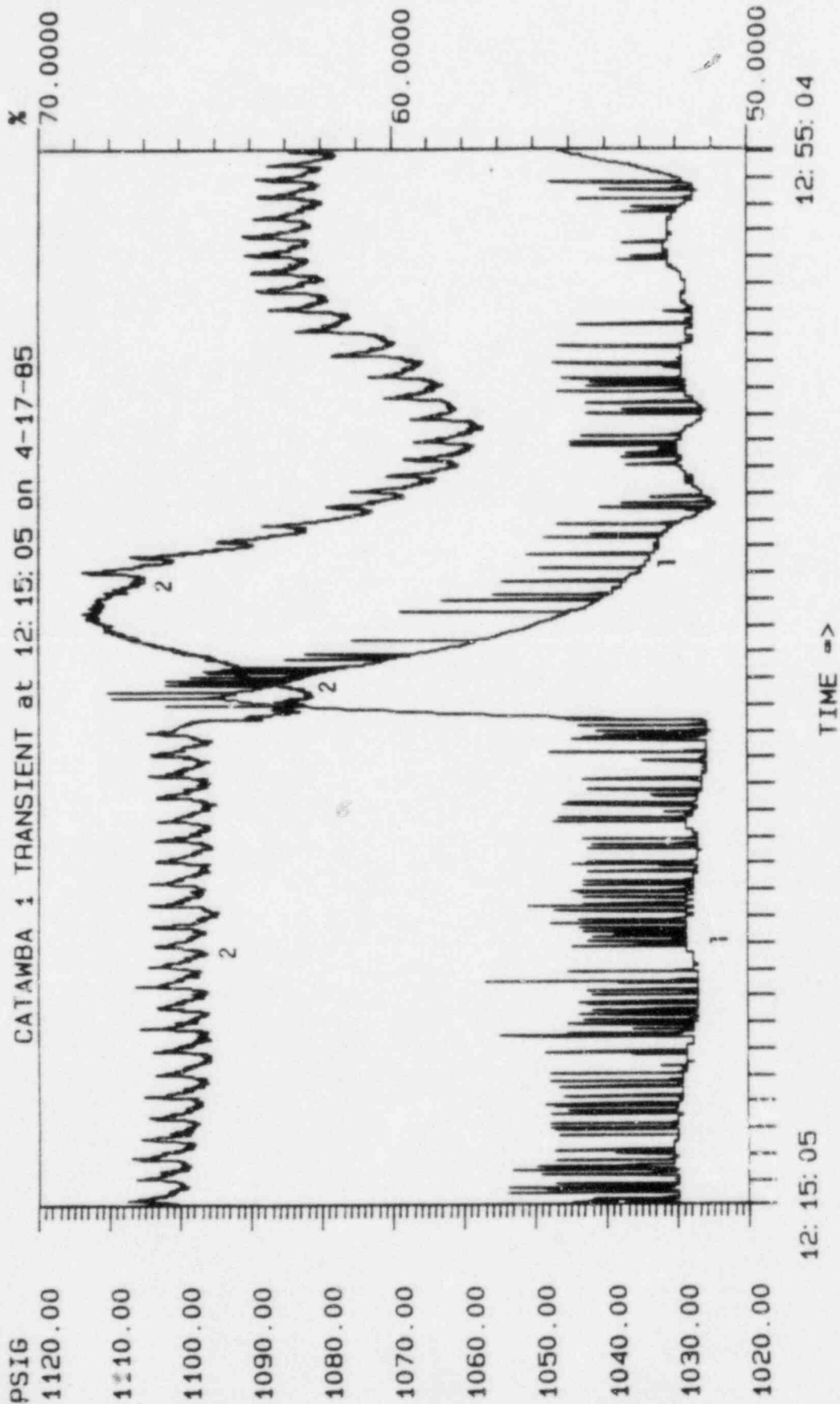


Figure 8.5-35

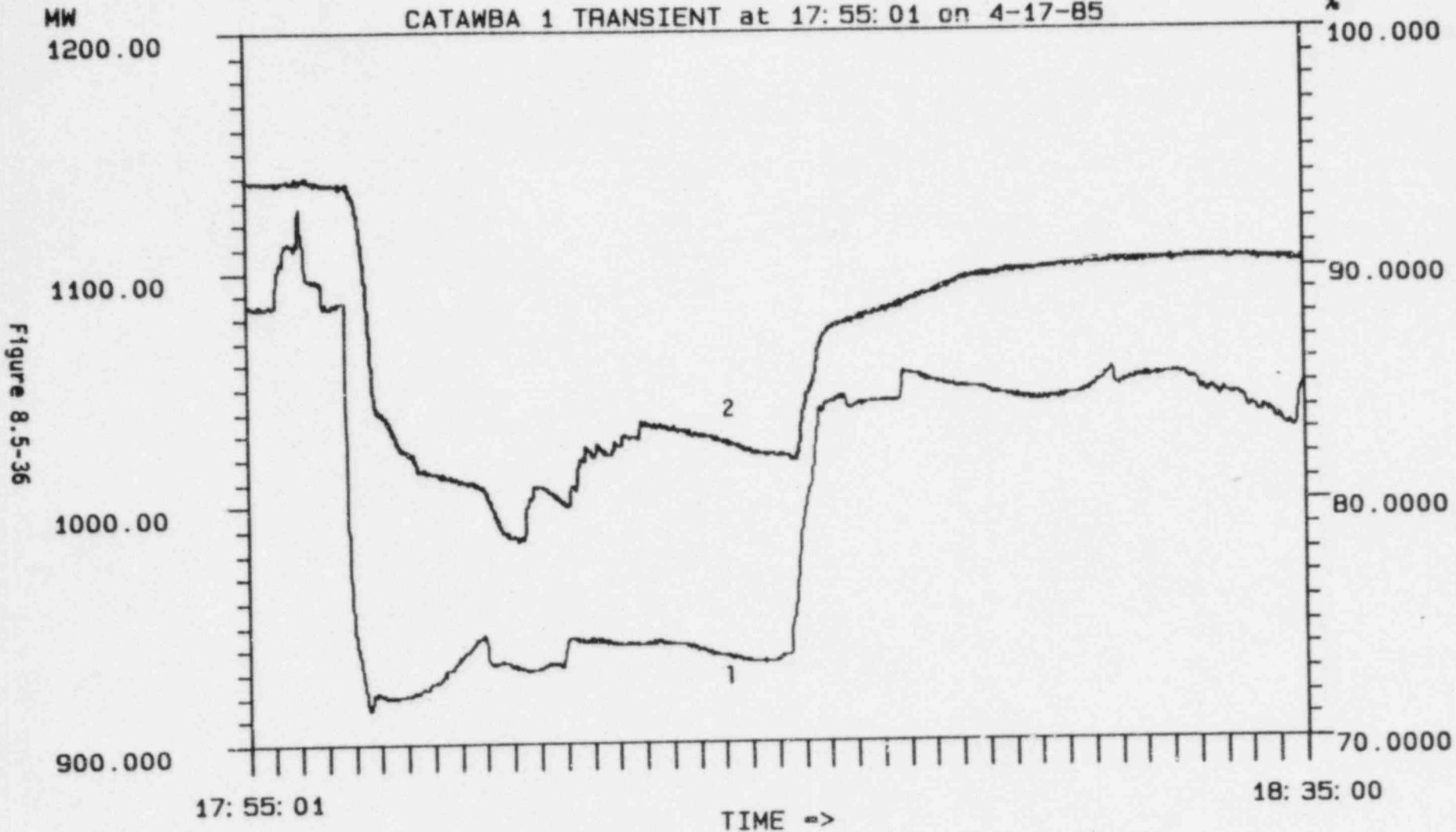
CATAWBA 1

	MAX	MIN	PID	MESSAGE
< 1	1110.20	1024.40	A0723	S/G A STEAM PRESS CH #1
> 2	68.7500	57.4000	A0531	S/G A NARROW RANGE LEVEL CH 1

< Manual Freeze >

UNIT LOAD TRANSIENT TEST - 100% INCREASE
PRIMARY AND SECONDARY POWER TREND

CATAWBA 1 TRANSIENT at 17:55:01 on 4-17-85



CATAWBA 1

	MAX	MIN	PID	MESSAGE
> 1	1126.59	914.473	A0697	GENERATOR MEGAWATTS
> 2	94.0200	78.4800	A0758	POWER RANGE AVG LEVEL QUADRANT 1 (N-43)

< Manual Freeze >

UNIT LOAD TRANSIENT TEST - 100% INCREASE
NC AVERAGE TEMPERATURE TREND

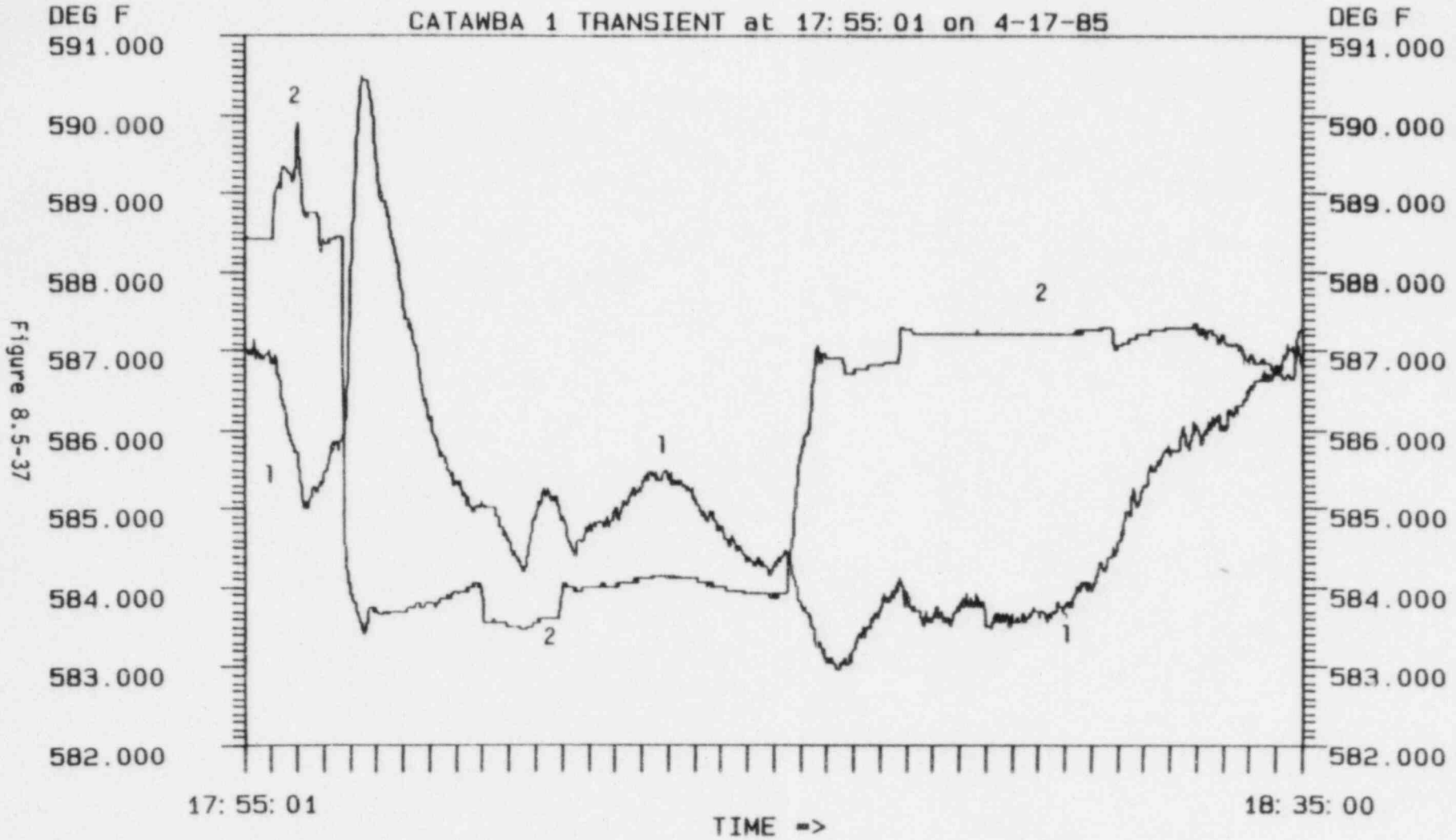


Figure 8.5-37

CATAWBA 1

	MAX	MIN	PID	MESSAGE
1	590.500	582.950	A0701	NC LOOP HIGHEST T-AVG
2	589.900	583.400	A0725	T-REF

< Manual Freeze >

UNIT LOAD TRANSIENT TEST - 100% INCREASE
TYPICAL NC LOOP TEMPERATURE TREND

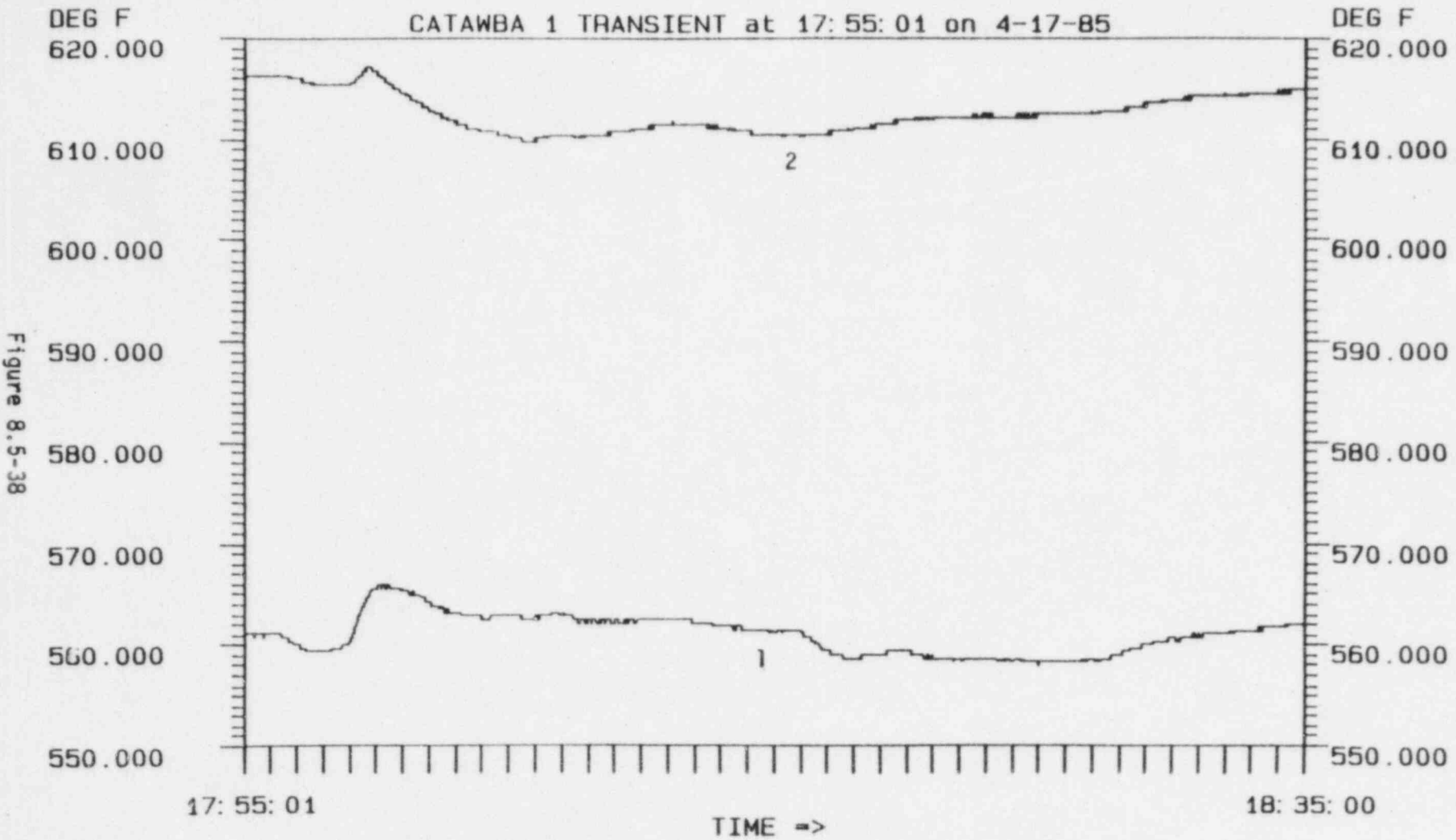


Figure 8.5-38

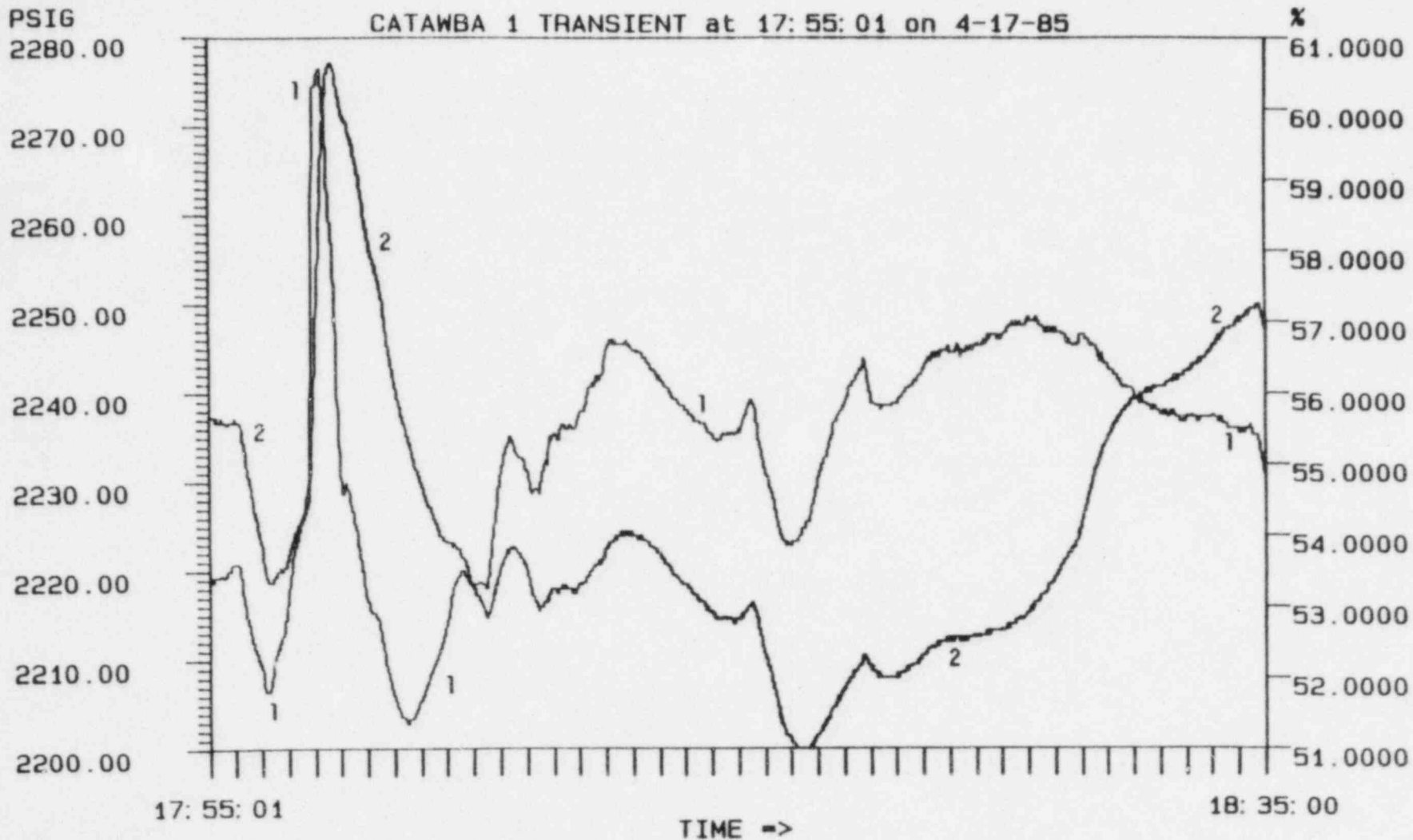
CATAWBA 1

	MAX	MIN	PID	MESSAGE
1	565.950	557.900	A0700	NC LOOP A WIDE RANGE COLD LEG TEMP
2	617.050	609.700	A0668	NC LOOP A WIDE RANGE HOT LEG TEMP

< Manual Freeze >

UNIT LOAD TRANSIENT TEST - 100% INCREASE
 PRESSURIZER PRESSURE AND LEVEL TREND

Figure 8.5-39

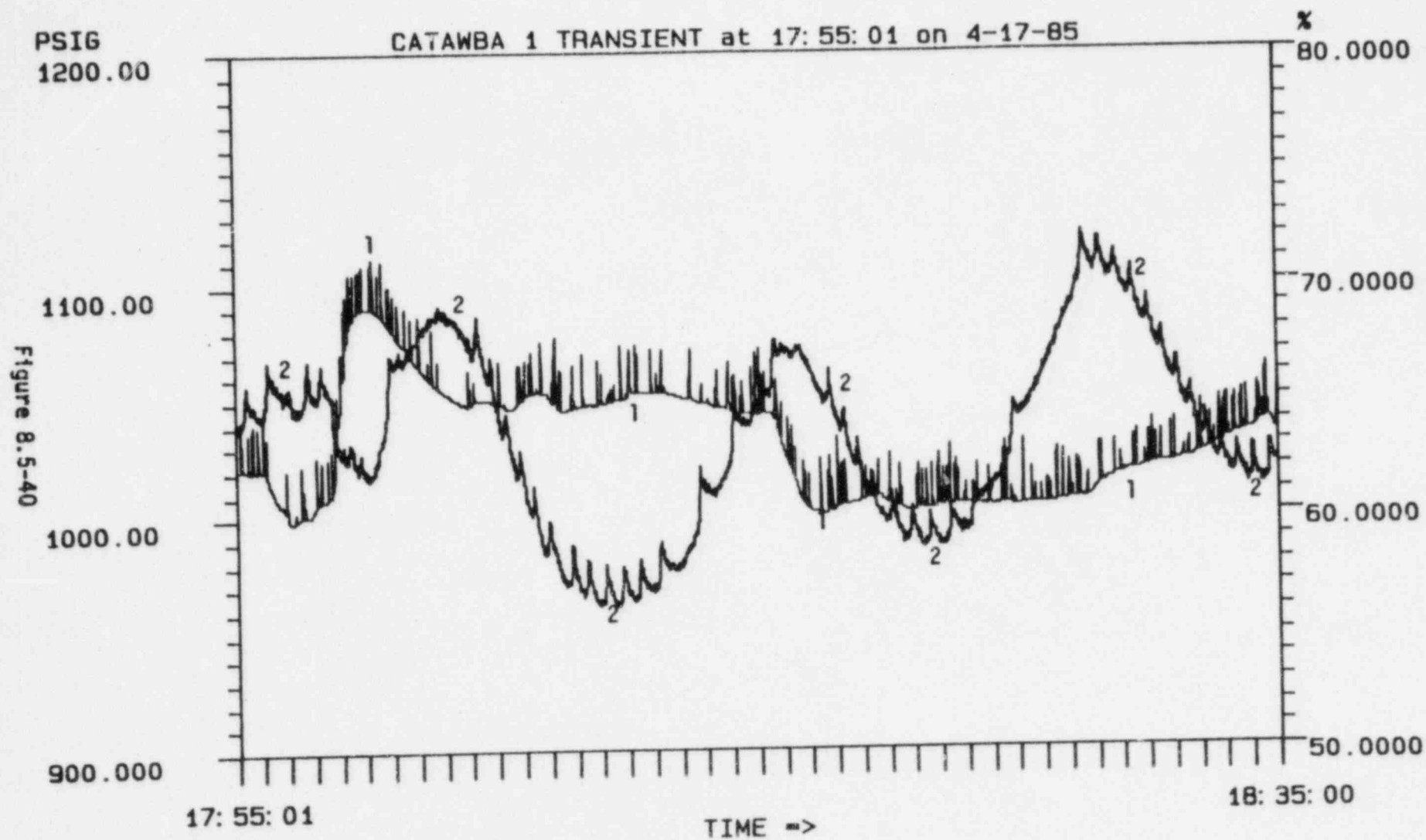


CATAWBA 1

		MAX	MIN	PID	MESSAGE
<	1	2276.40	2202.80	A0713	PZR PRESS CH 1
>	2	60.6500	51.0000	A0707	PZR LEVEL CH I

< Manual Freeze >

UNIT LOAD TRANSIENT TEST - 100% INCREASE
TYPICAL S/G PRESSURE AND LEVEL TRENDS



CATAWBA 1		MAX	MIN	PID	MESSAGE
1	—	1111.50	998.400	A0723	S/G A STEAM PRESS CH #1
2	—	72.1500	56.1500	A0531	S/G A NARROW RANGE LEVEL CH 1

< Manual Freeze >

8.6 UNIT LOSS OF ELECTRICAL LOAD TEST - TP/1/A/2650/06

Date(s) Performed: 4/19/85

I. PURPOSE

The objectives of this test were:

- A. To demonstrate the ability of the unit to sustain a total loss of electrical load without exceeding turbine design overspeed conditions.
- B. To evaluate the response of control systems during the transient to determine if any changes are required to improve response.
- C. To demonstrate proper steam dump control system operation.
- D. To demonstrate the adequacy of AP/1/A/5500/02, Turbine Generator Trip.

II. METHOD

With the Unit initially stable at 100% RTP both Main Generator breakers 1A and 1B were opened. This caused a Reactor trip followed by a turbine trip. Data was gathered during the transient via the OAC Transient Monitor, Events Recorder and Alarm Summary programs.

III. RESULTS

All Acceptance Criteria associated with this test were met. Following the trip stable no-load conditions were re-established using the Turbine Generator Trip AP. During the transient, neither safety injection was initiated nor were any Pressurizer safety relief valves lifted. Figures 8.6-1 through 8.6-5 show the Unit's transient response. Turbine speed never exceeded the 2007 RPM Acceptance Criterion. The maximum turbine speed recorded by the Transient Monitor was 1863 RPM. Reactor power, Pressurizer pressure, and NC T-AVG did not exceed the Tech Spec safety limits. The Steam Dump Control System functioned adequately to control NC T-AVG.

IV. CORRECTIVE ACTION

Three of the steam dump to condenser valves did not open during this test. Minor control system problems were found to be the cause. These have since been resolved. Overall system performance was adequate as evidenced by the post-trip NC T-AVG trend.

UNIT LOSS OF ELECTRICAL LOAD TEST
PRIMARY AND SECONDARY POWER TREND

CATAWBA 1 TRANSIENT at 10:05:49 on 4-19-85

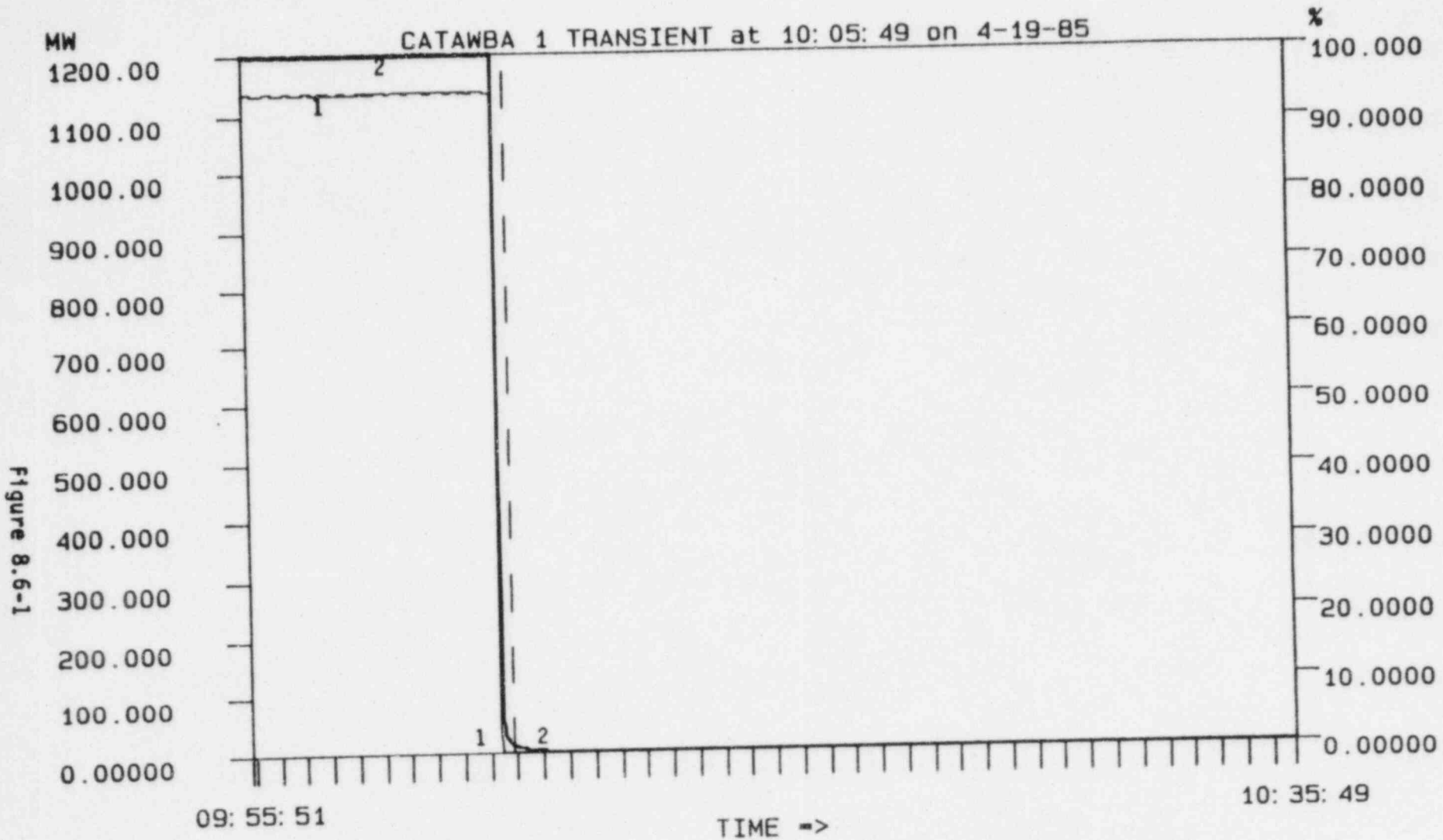


Figure 8.6-1

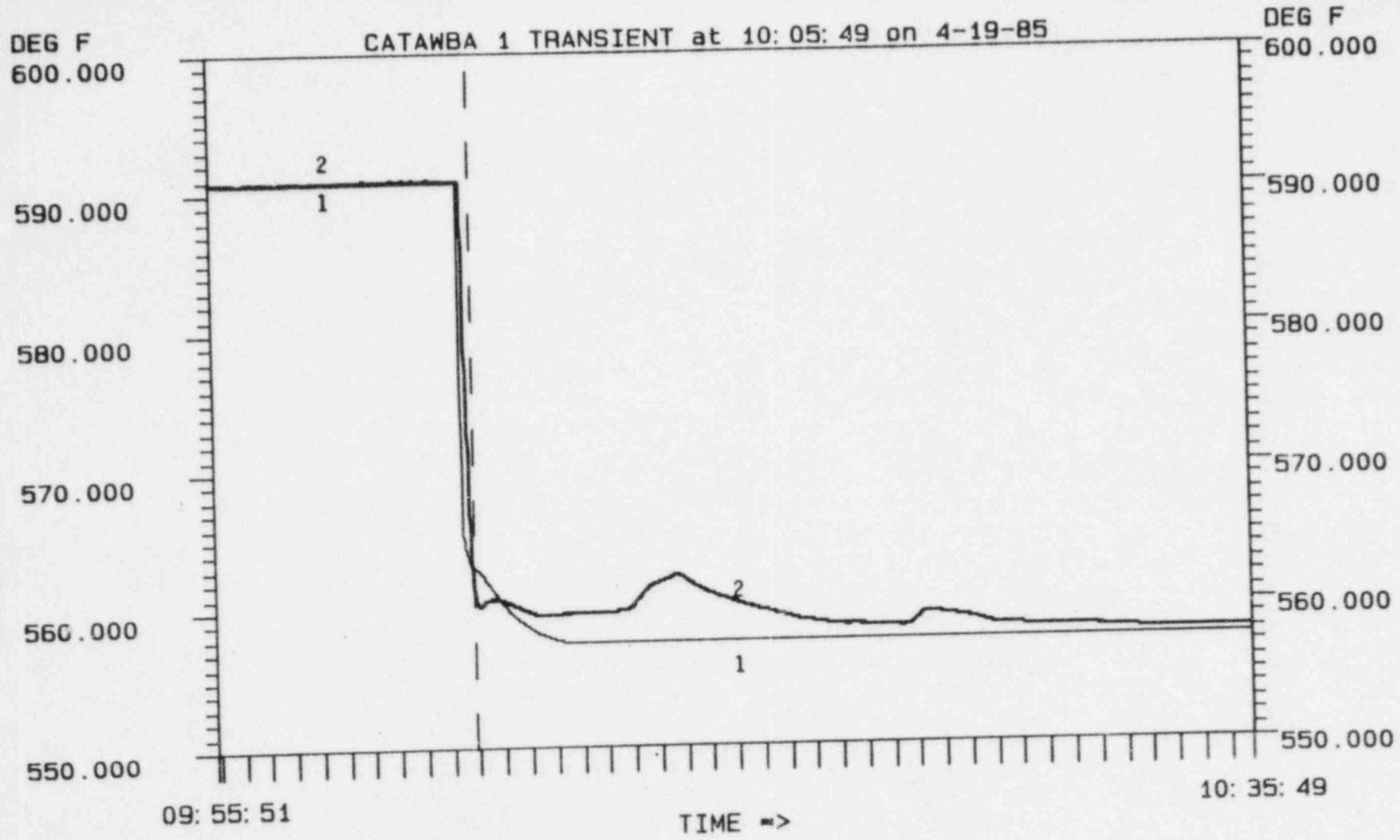
CATAWBA 1

	@TRIP	MAX	MIN	PID	MESSAGE
1	0.00000	1137.06	0.00000	A0697	GENERATOR MEGAWATTS
2	1.08000	99.9600	0.00000	A0758	POWER RANGE AVG LEVEL QUADRANT 1 (N-43)

Initiating point was D2725, CF PUMP TURB B

CATAWBA 1 TRANSIENT at 10:05:49 on 4-19-85

Figure 8.6-2

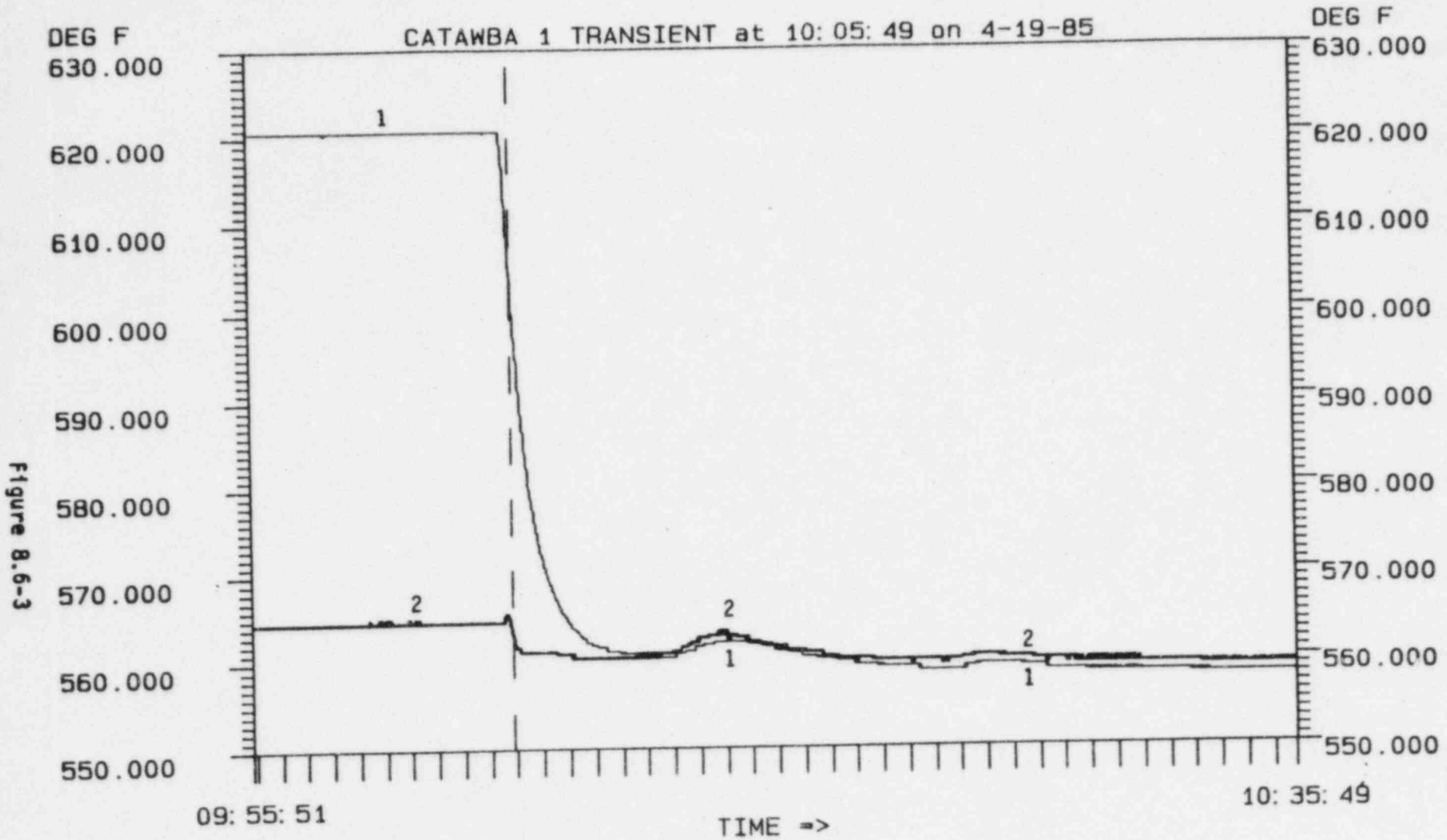


	@TRIP	MAX	MIN	PID	MESSAGE
1	562.950	590.600	557.450	A0725	T-REF
2	561.800	590.950	557.900	A0701	NC LOOP HIGHEST T-AVG

Initiating point was D2725, CF PUMP TURB B

CATAWBA 1

UNIT LOSS OF ELECTRICAL LOAD TEST
TYPICAL NC LOOP TEMPERATURE TREND



CATAWBA 1

	@TRIP	MAX	MIN	PID	MESSAGE
1	606.200	620.550	557.900	A0668 NC	LOOP A WIDE RANGE HOT LEG TEMP
2	563.150	565.600	558.950	A0700 NC	LOOP A WIDE RANGE COLD LEG TEMP

Initiating point was D2725, CF PUMP TURB B

UNIT LOSS OF ELECTRICAL LOAD TEST
PRESSURIZER PRESSURE AND LEVEL TREND

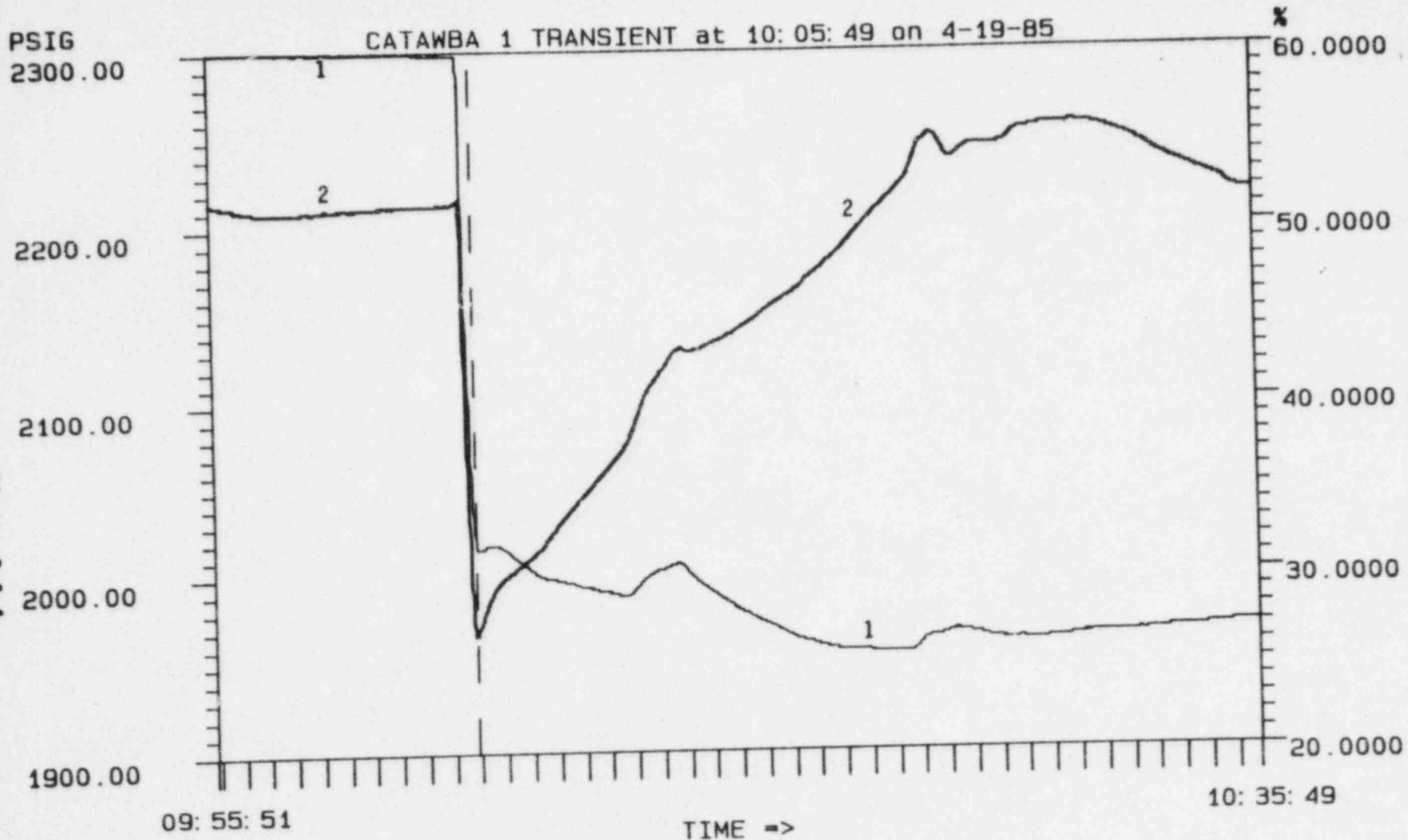


Figure 8.6-4

CATAWBA 1

	@TRIP	MAX	MIN	PID	MESSAGE
1	32.6000	60.0000	25.4500	A0707	PZR LEVEL CH I
2	1966.40	2257.20	1966.40	A0713	PZR PRESS CH 1

Initiating point was D2725, CF PUMP TURB B

UNIT LOSS OF ELECTRICAL LOAD TEST
TYPICAL S/G PRESSURE AND LEVEL TRENDS

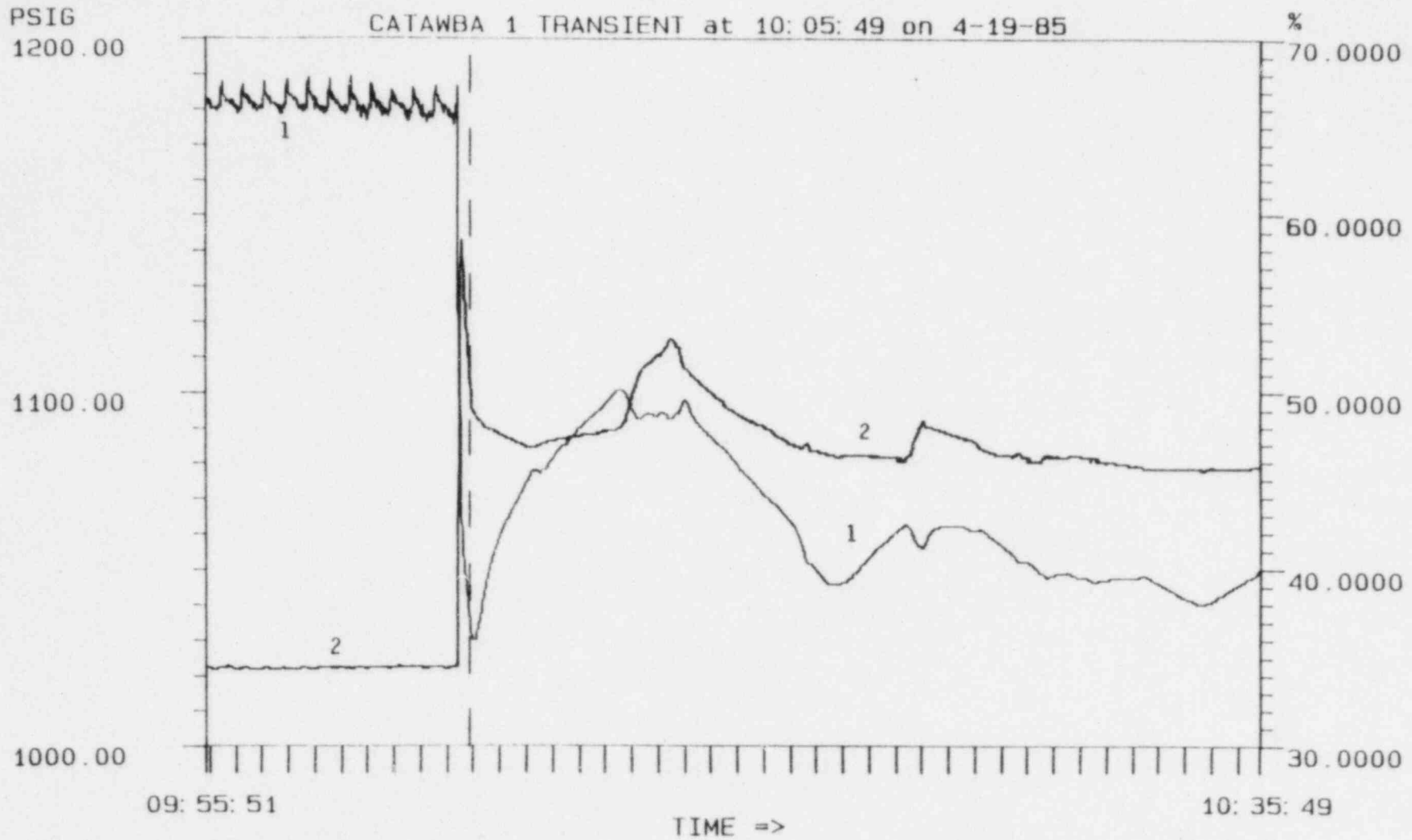


Figure 8.6-5

CATAWBA 1

	@TRIP	MAX	MIN	PID	MESSAGE
1	37.9000	67.9000	36.0000	A0537	S/G B NARROW RANGE LEVEL CH 2
2	1105.00	1143.35	1021.80	A1286	S/G B STEAM PRESS CH #2

Initiating point was D2725, CF PUMP TURB B

8.7 TURBINE TRIP TEST - TP/1/A/2650/07

Date(s) Performed: 3/27/85 - 3/28/85

I. PURPOSE

- A. To measure the unit response during and after a turbine trip from $\leq 68\%$ FP.
- B. To evaluate the interaction between plant control systems and transient response in order to determine whether any adjustments to the control systems are required.
- C. To verify the adequacy of the turbine trip procedure.

II. METHOD

With the Unit initially stable at $\approx 66\%$ F.P. a manual turbine trip was initiated. This is just below the power level at which a turbine trip causes an anticipatory Reactor trip. Steady state conditions were re-established using AP/1/A/5500/02, Turbine Trip. Transient response was evaluated using data gathered by the OAC Transient Monitor.

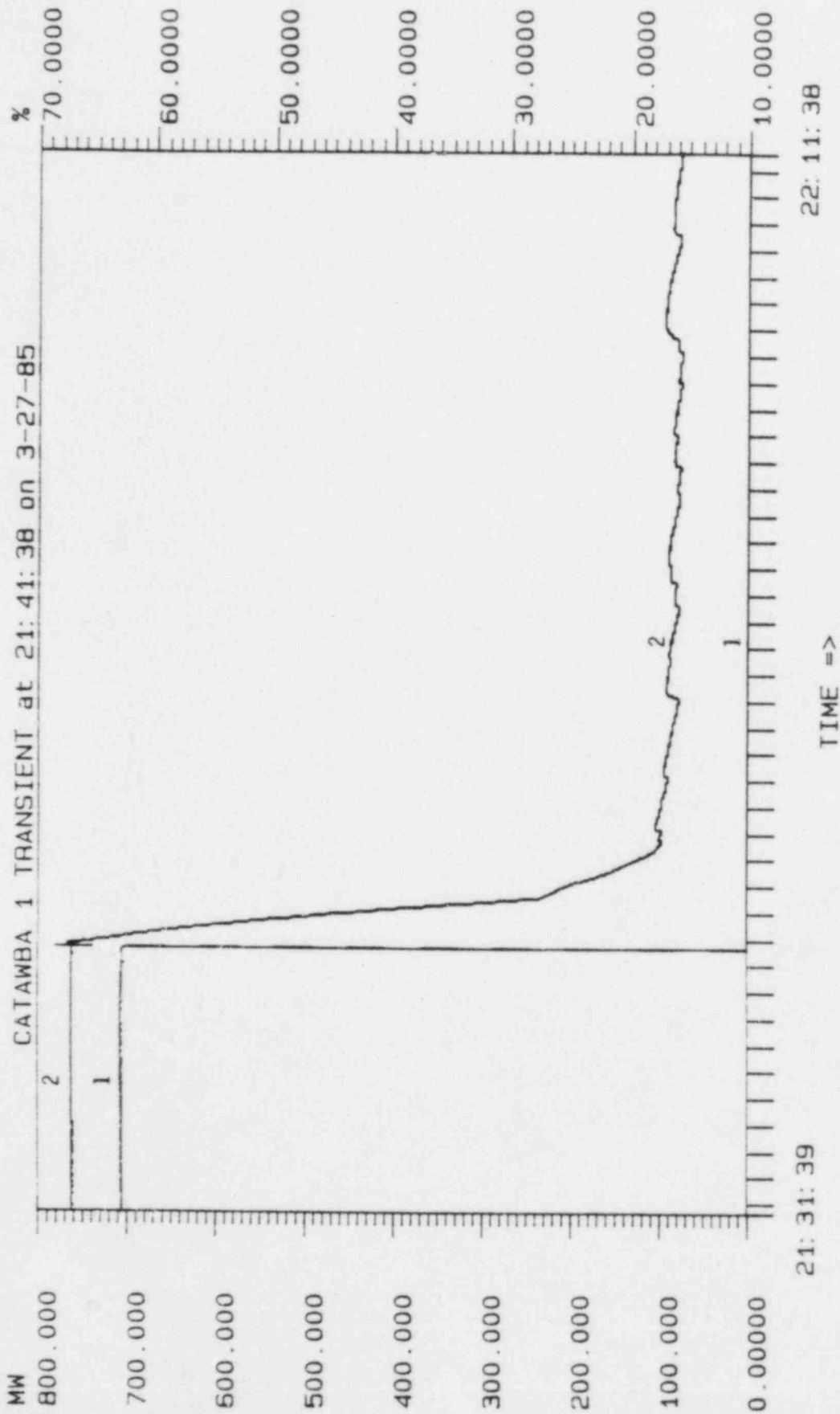
III. RESULTS

All Acceptance Criteria associated with the test were met. Stable conditions were successfully re-established following the turbine trip. Safety injection was not initiated. Neither the Main Steam nor the Pressurizer code safety valves lifted during the transient. The pressurizer PORV's were not challenged. Reactor power, NC pressure, and NC T-AVG did not exceed the Tech Spec safety limits. Figures 8.7-1 through 8.7-5 show the transient response of several important parameters.

IV. CORRECTIVE ACTIONS

None

TURBINE TRIP TEST
PRIMARY AND SECONDARY POWER TREND

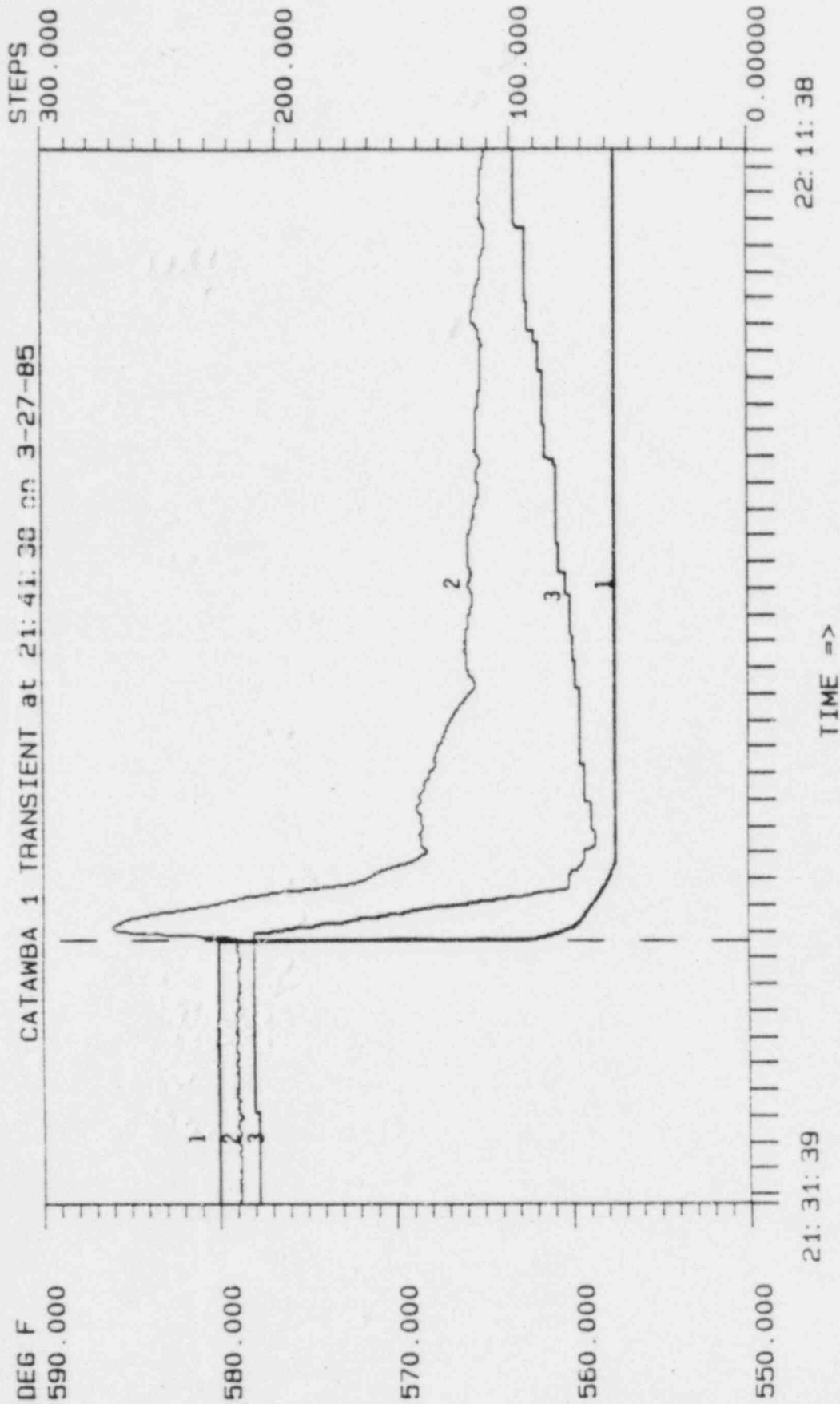


@TRIP	MAX	MIN	PID	MESSAGE
1	349.079	707.233	0.00000	A0697 GENERATOR MEGAWATTS
2	67.1400	67.6200	15.5400	A0758 POWER RANGE AVG LEVEL QUADRANT 1 (N-43)

Initiating point was D2802, TURB STATUS

Figure 8.7-1

TURBINE TRIP TEST
 NC AVERAGE TEMPERATURE AND
 CONTROL BANK D POSITION TREND



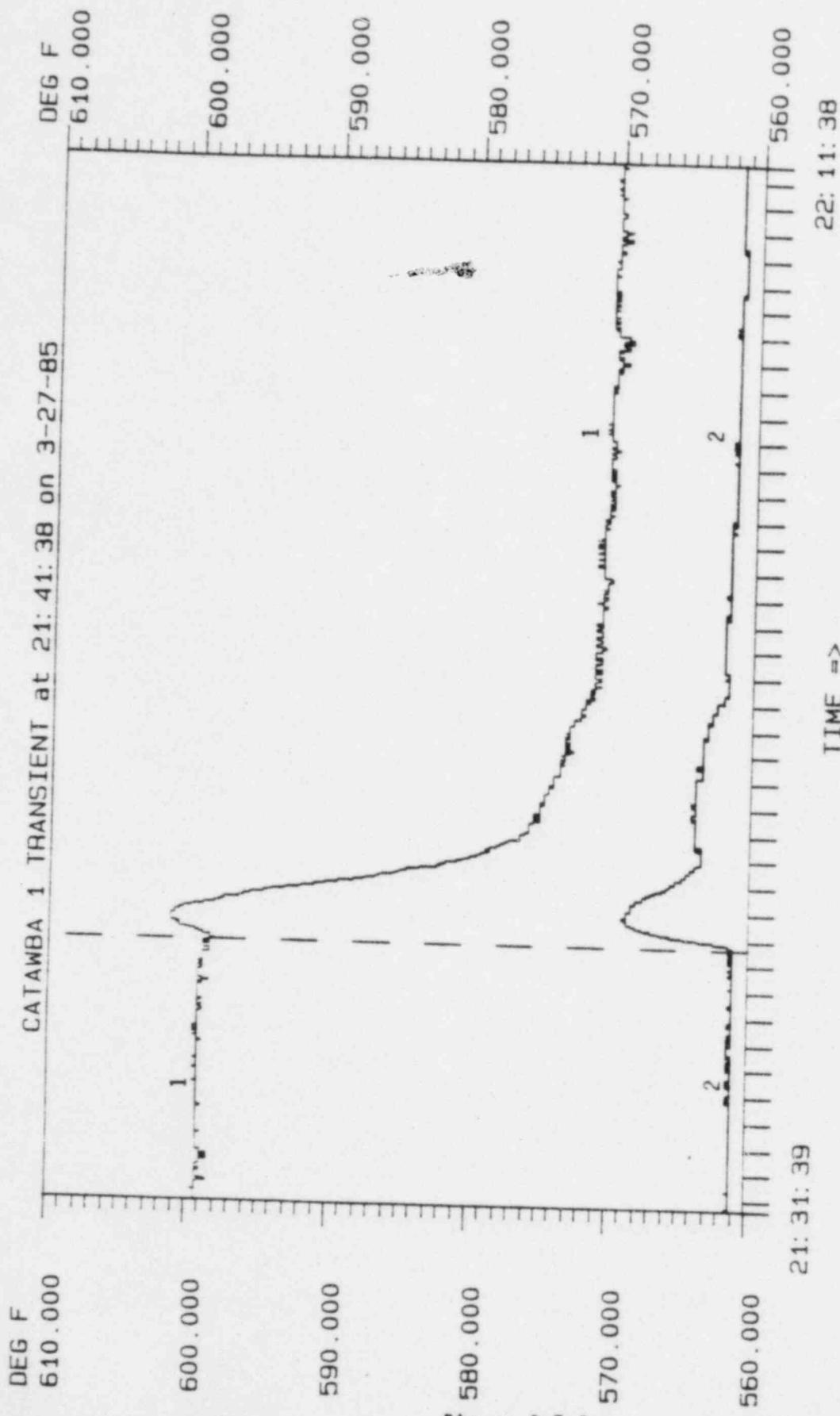
CATAWBA 1

eTRIP	MAX	MIN	PID	MESSAGE
1	577.700	580.050	557.450	A0725 T-REF
2	579.000	585.000	564.750	A0701 NC LOOP HIGHEST T-AVG
3	209.572	209.572	64.5883	A0749 CONTROL ROD BANK D DEMAND POSITION

Initiating point was D2802, TURB STATUS

Figure 8.7-2

TURBINE TRIP TEST
TYPICAL NC LOOP TEMPERATURE TREND

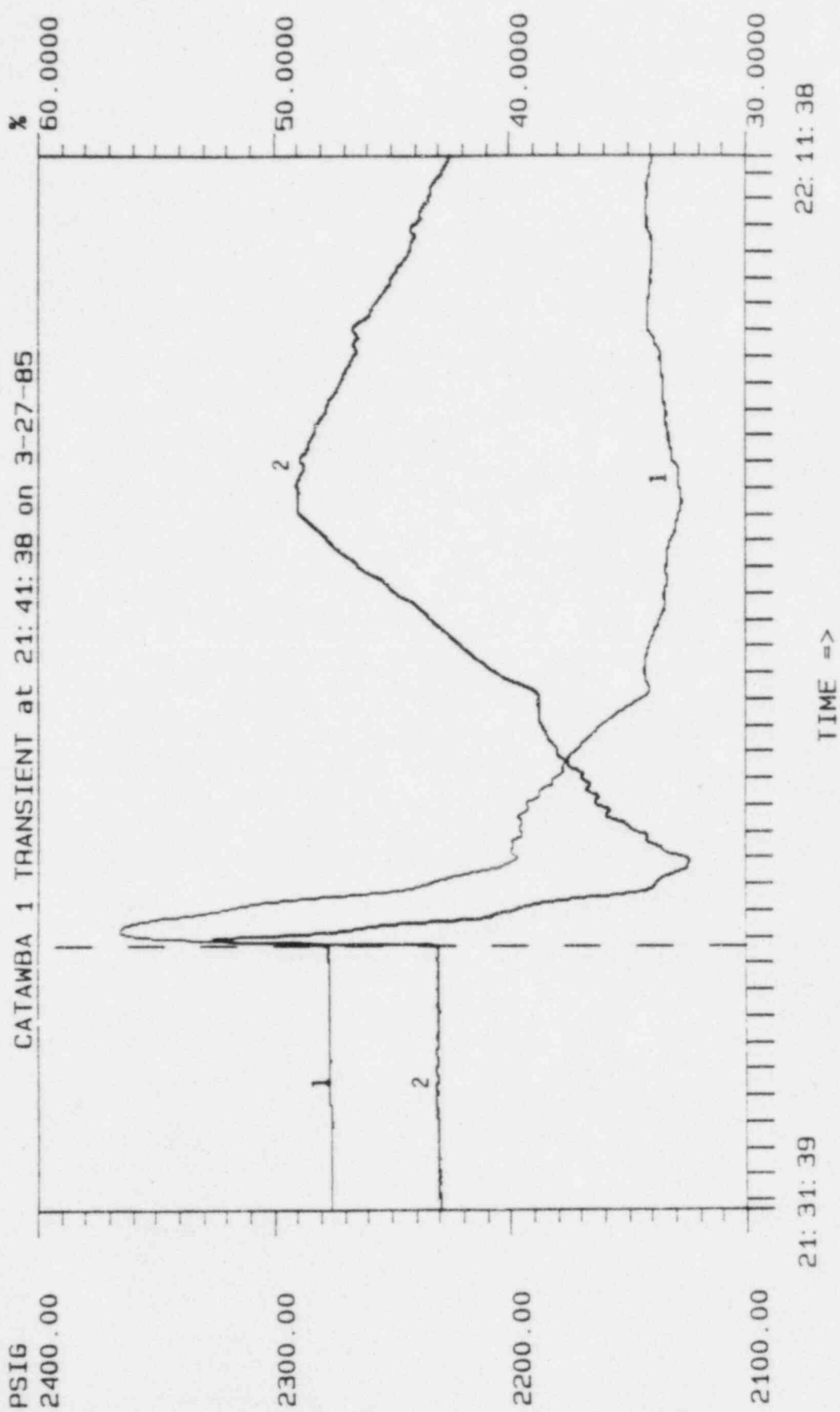


	eTRIP	MAX	MIN	PID	MESSAGE
1	598.500	601.300	569.100	A0668	NC LOOP A WIDE RANGE HOT LEG TEMP
2	561.400	569.100	561.050	A0700	NC LOOP A WIDE RANGE COLD LEG TEMP

Initiating point was D2802, TURB STATUS

Figure 8.7-3

TURBINE TRIP TEST
PRESSURIZER PRESSURE AND LEVEL TREND



CATAWBA 1 TRANSIENT at 21:41:38 on 3-27-85

	TRIP	MAX	MIN	PID	MESSAGE
1	47.7000	56.5500	32.6500	A0707	PZR LEVEL CH I
2	2230.80	2326.40	2123.60	A0713	PZR PRESS CH 1

Initiating point was D2802, TURB STATUS

Figure 8.7-4

TURBINE TRIP TEST
TYPICAL S/G PRESSURE AND LEVEL TREND

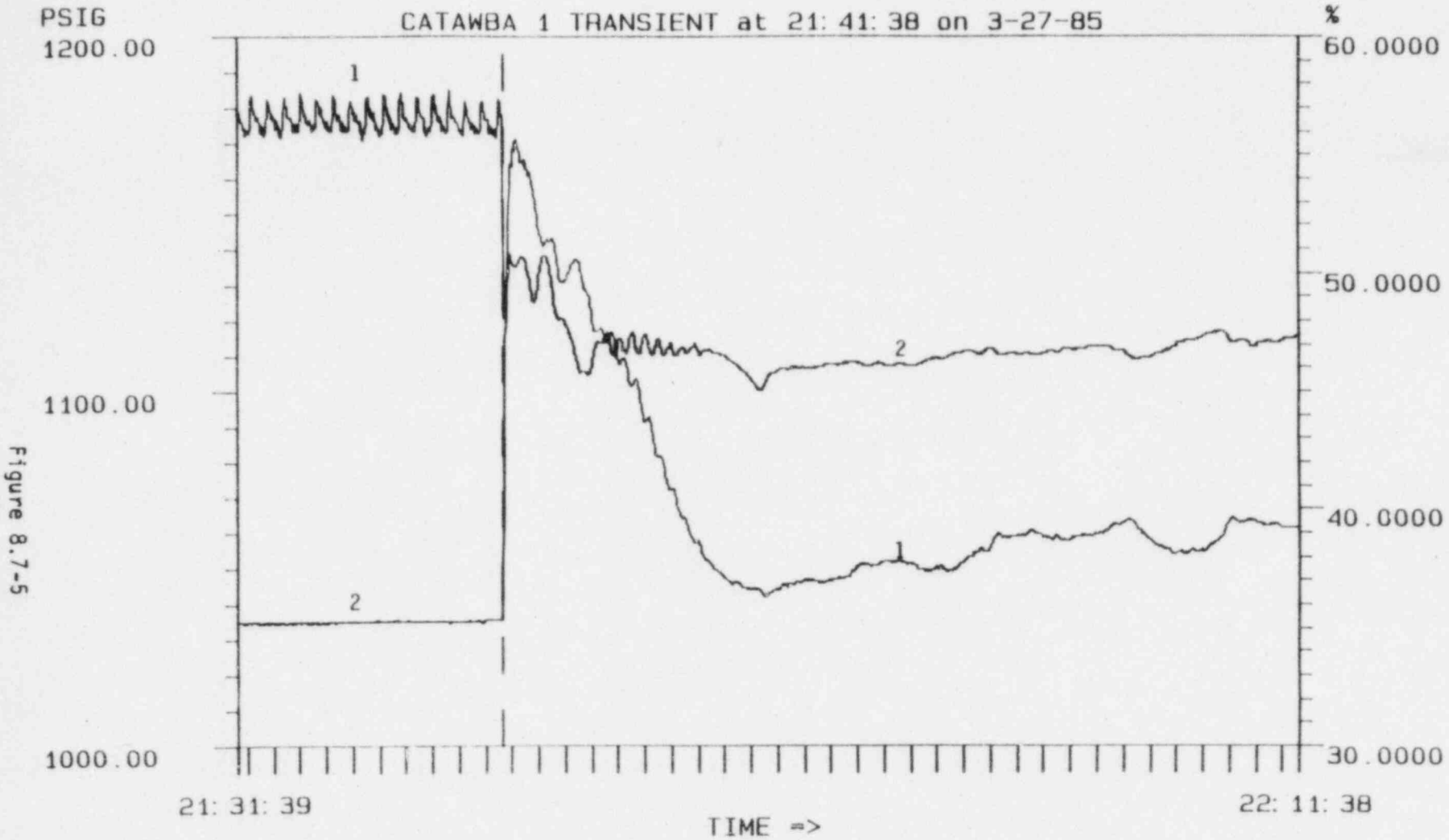


Figure 8.7-5

CATAWBA 1

	@TRIP	MAX	MIN	PID	MESSAGE
1	56.2000	57.7500	36.2500	A0537	S/G B NARROW RANGE LEVEL CH 2
2	1045.85	1138.80	1034.15	A1286	S/G B STEAM PRESS CH #2

Initiating point was D2802, TURB STATUS

8.8 STATION BLACKOUT TEST - TP/1/A/2650/12

Date(s) Performed: 1/29/85 - 1/30/85

I. PURPOSE

The objectives of this test were:

- A. To demonstrate the ability of the Unit to sustain a Turbine-Generator trip concurrent with an isolation from the offsite power distribution system.
- B. To evaluate proper control systems response to the transient including auto start and loading of the Diesel Generators.
- C. To verify that the unit can be brought to and maintained in a safe Hot Shutdown condition and Natural Circulation established for a minimum of 30 minutes using appropriate emergency procedures.

II. METHOD

Beginning from a stable condition at approximately 20% RTP with the generator loaded at 134 MWe (11.6%) both Main Generator breakers 1A and 1B and the 4 main switchyard PCB's (PCB's 14, 15, 17, and 18) were all simultaneously opened. This created a station blackout condition. Loss of power to the Rod Control System caused a Reactor trip. Also, all NC pumps tripped. The Reactor trip caused a Turbine trip. The Diesel Generators automatically started and sequenced on the designated B/O loads. Plant response was monitored via the OAC Transient Monitor, Events Recorder and Alarm Summary programs. Operators brought the plant to a Hot Standby condition and established Natural Circulation. This was maintained for 30 minutes after which plant recovery was per the appropriate OPS procedures.

III. RESULTS

All Acceptance Criteria for this test were met as follows:

- A. A Reactor trip occurred as a result of loss of power to the Rod Control system. Turbine trip subsequently occurred. This was evidenced by Visicorder traces taken from the power cabinet of the Rod Control system and from the Events Recorder.
- B. All Auxiliary Feedwater pumps started automatically as a result of the transient.
- C. Natural Circulation was established and maintained for 31 minutes while the unit was in Hot Standby.

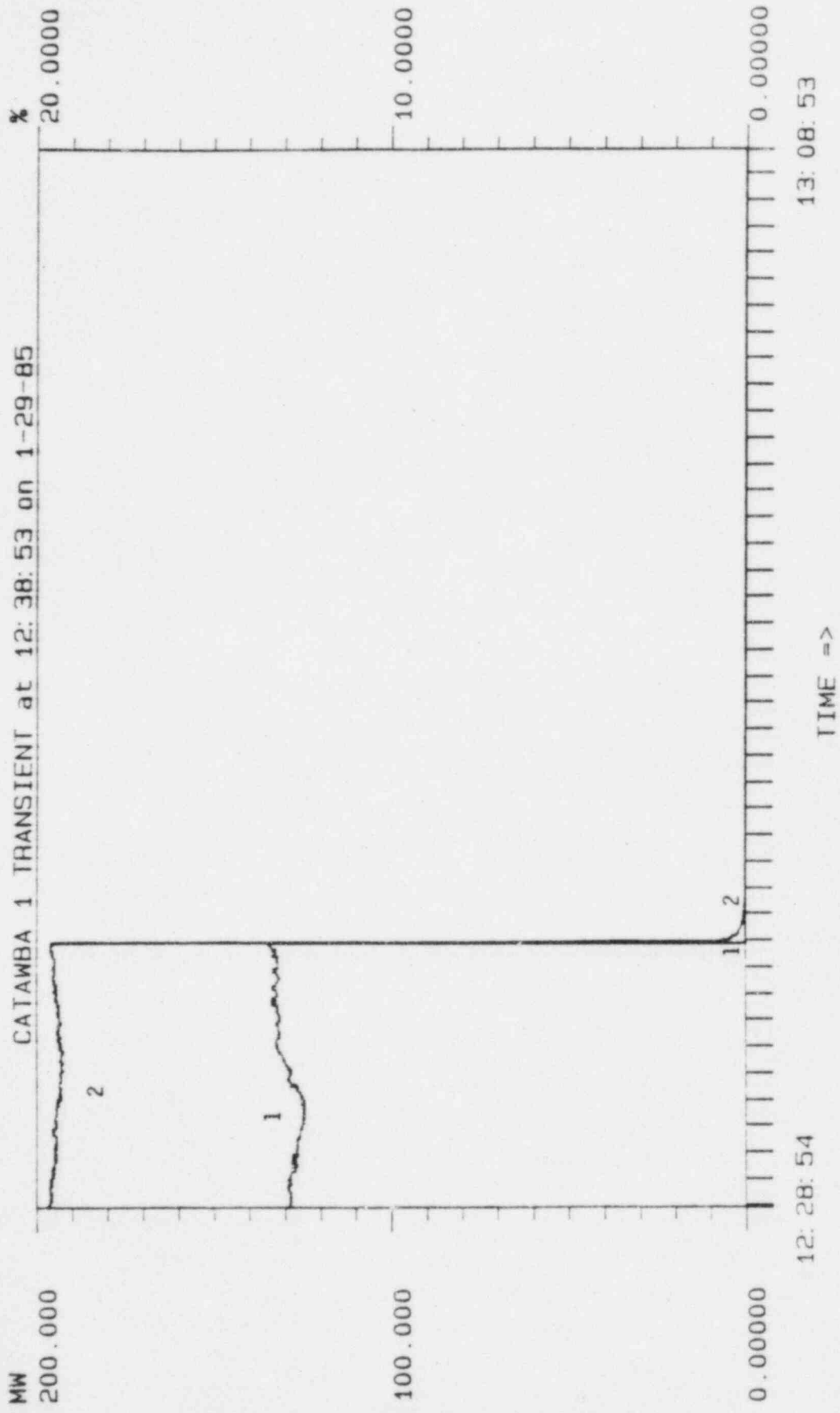
- D. Both Diesel Generators automatically started and sequenced on all required Essential Load groups.
- E. Control of Steam Generator (S/G) levels was maintained such that no level ever decreased below 10% N/R.
- F. Control of S/G pressure was maintained and no S/G PORV's were challenged.
- G. Control over Pressurizer (PZR) pressure and level was maintained. Level never went above the 70% maximum nor decreased below the 17% minimum. No PZR PORV's were challenged during the transient.

Figures 8.8-1 through 8.8-5 show data obtained during the transient by the OAC Transient Monitor.

IV. CORRECTIVE ACTION

The "B" Pressurizer heater group failed to come on during or after the B/O. It is powered from a B/O protected source. It was found that the breaker at the MCC had spuriously tripped. This problem has since been corrected.

STATION BLACKOUT TEST
PRIMARY AND SECONDARY POWER TREND

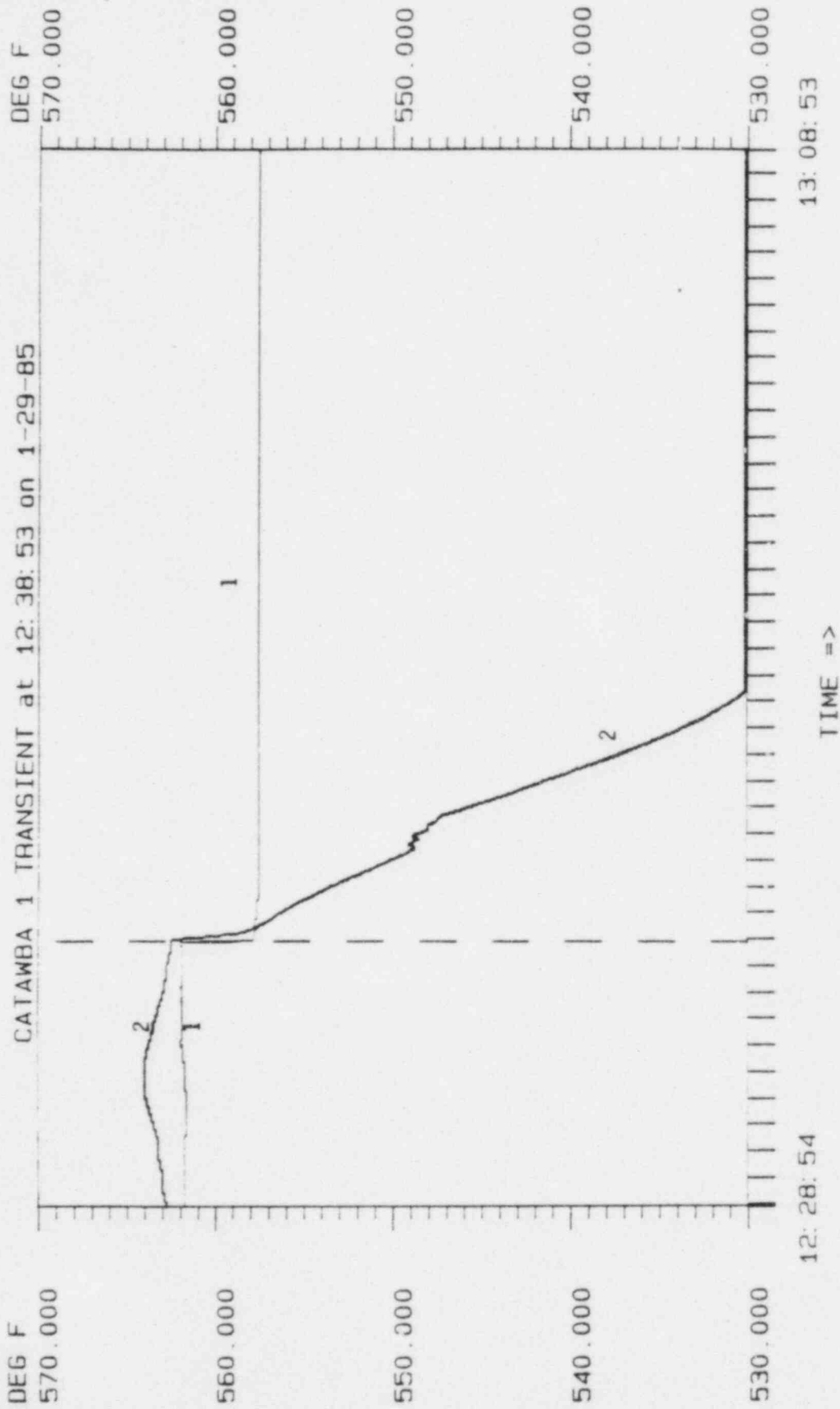


	TRIP	MAX	MIN	PID	MESSAGE
1	1.40474	134.855	0.00000	A0697	GENERATOR MEGAWATTS
2	9.06000	19.6800	0.00000	A0758	POWER RANGE AVG LEVEL QUADRANT 1 (N-43)

Initiating point was D2720, CF PUMP TURB A

Figure 8.8-1

STATION BLACKOUT TEST
NC AVERAGE TEMPERATURE TREND

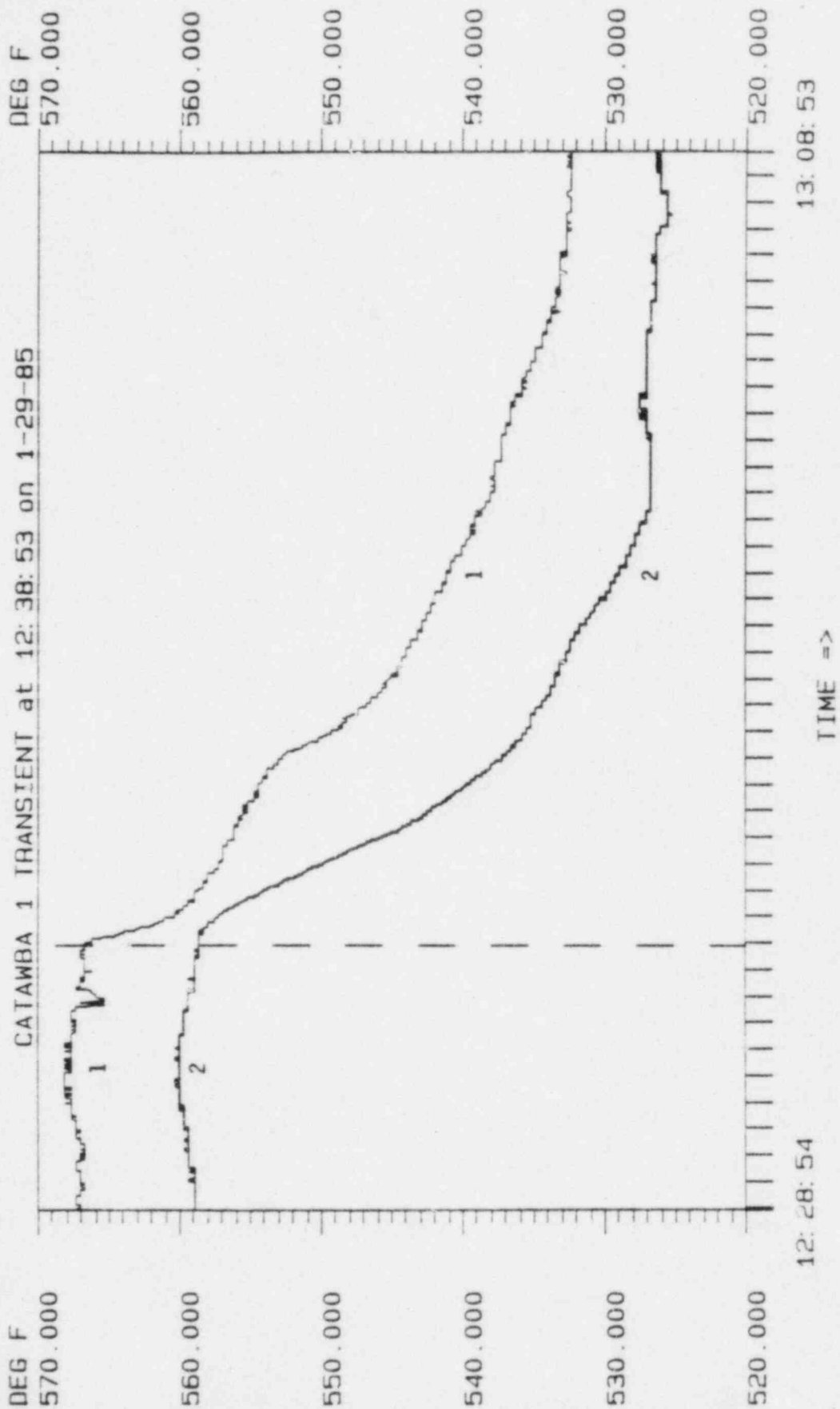


	@TRIP	MAX	MIN	PID	MESSAGE
1	559.550	561.950	557.550	A0725	T-REF
2	562.450	564.000	530.150	A0701	NC LOOP HIGHEST T-AVG

Initiating point was D2720, CF PUMP TURB A

Figure 8.8+2

TYPICAL NC LOOP TEMPERATURE TREND



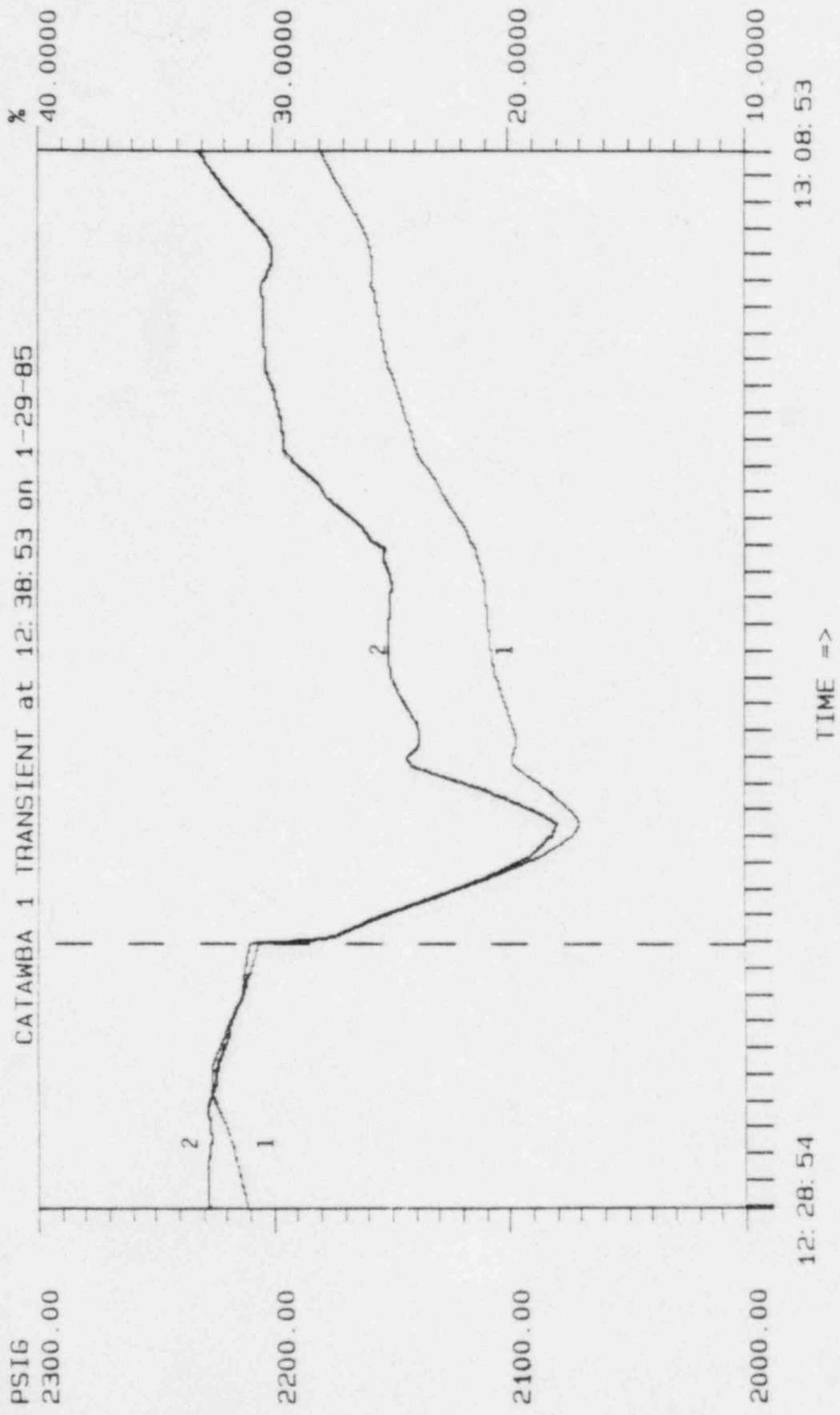
	TRIP	MAX	MIN	PID	MESSAGE
1	566.650	568.050	532.350	A0668	NC LOOP A WIDE RANGE HOT LEG TEMP
2	558.600	560.350	525.350	A0700	NC LOOP A WIDE RANGE COLD LEG TEMP

Initiating point was D2720, CF PUMP TURB A

Figure 8.8-3

CATAWBA 1

STATION BLACKOUT TEST
PRESSURIZER PRESSURE AND LEVEL TREND

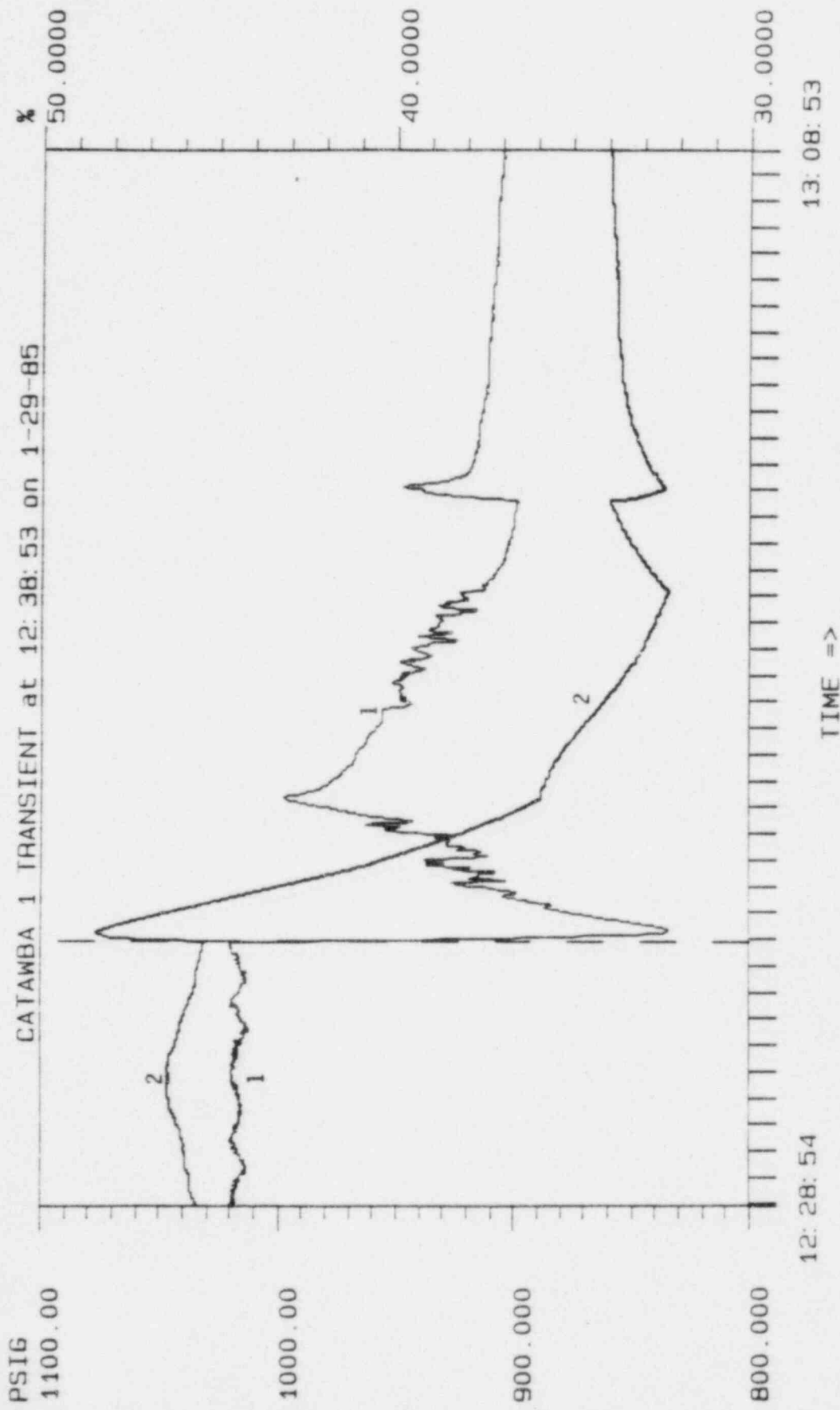


	@TRIP	MAX	MIN	PID	MESSAGE
1	30.5000	32.6000	17.0000	A0707	PZR LEVEL CH I
2	2209.60	2231.20	2080.00	A0713	PZR PRESS CH 1

Initiating point was D2720, CF PUMP TURB A

Figure 8.8-4

TYPICAL S/G LEVEL AND PRESSURE TREND



13:08:53

12:28:54

MESSAGE

PID

MIN

MAX

@TRIP

	@TRIP	MAX	MIN	PID	MESSAGE
1	42.9500	44.7000	32.2500	A0531	S/G A NARROW RANGE LEVEL CH 1
2	1036.75	1077.05	833.950	A0723	S/G A STEAM PRESS CH #1

Initiating point was D2720, CF PUMP TURB A

PSIG
1100.00

1000.00

900.000

800.000

%
50.0000

40.0000

30.0000

TIME =>

Figure 8.8-5

CATAWBA 1

8.9 PRELIMINARY INCORE AND NUCLEAR INSTRUMENTATION SYSTEMS CORRELATION -
PT/1/A/4600/05C

Date(s) Performed: 3/13/85

I. PURPOSE

The purpose of this test was to determine, in a preliminary fashion, the relationship between axial offset, as measured by the movable incore detector system, and the excore detector upper and lower currents; and then to calculate new factors for the excore power distribution monitor and the $f(\Delta I)$ function in the overtemperature ΔT reactor trip setpoints.

II. METHOD

Excore detector currents were recorded during flux maps taken at various power levels and axial offsets. This data was used to determine the relationship between axial offset and excore detector currents and to setup the proper $f(\Delta I)$ function for the overtemperature ΔT setpoints.

III. RESULTS

All criteria for this test were met. The excore detectors were recalibrated and the constants used by the OAC Excore Power Distribution Monitor program adjusted using the results of the test. The data for the $f(\Delta I)$ functions was forwarded to IAE for incorporation into the overtemperature ΔT trip setpoints.

Table 8.9-1 contains a summary of the data utilized in this test. Table 8.9-2 shows the results which were used to calibrate the excore detectors.

IV. CORRECTIVE ACTIONS

None required.

PRELIMINARY INCORE AND NUCLEAR INSTRUMENTATION SYSTEMS CORRELATION
 INCORE AND NIS TEST DATA

Excore Detector Currents, 10-6 Amps

Map No.	% Power	Incore Axial Offset, %	N41		N42		N43		N44	
			Top	Bottom	Top	Bottom	Top	Bottom	Top	Bottom
6	32.0	-11.749	151.7	199.8	157.1	209.6	145.5	196.3	130.8	193.2
9	47.2	-8.747	233.5	288.5	239.5	303.5	223.5	285.0	200.0	280.0
10	48.7	4.340	251.5	261.0	256.2	275.9	238.6	258.5	217.9	253.8

NOTE: All maps used were full core maps.

TABLE 8.9-1

CATAWBA I

PRELIMINARY INCORE AND NUCLEAR INSTRUMENTATION SYSTEMS CORRELATION
INCORE AND NIS TEST RESULTS

Full Power Excore Detector Currents, 10⁻⁶ Amps

Incore Axial Offset, %	N41		N42		N43		N44	
	Top	Bottom	Top	Bottom	Top	Bottom	Top	Bottom
30	578.5	394.0	589.1	424.0	540.9	396.9	506.0	387.8
20	555.0	449.5	566.6	479.8	521.6	449.4	483.6	440.0
10	531.4	505.1	544.1	535.6	502.4	501.9	461.1	492.3
0	507.8	560.6	521.6	591.4	483.1	554.4	438.7	544.5
-10	484.2	616.1	499.1	647.3	463.8	606.9	416.3	596.8
-20	460.6	671.7	476.7	703.1	444.5	659.4	393.9	649.1
-30	437.0	727.2	454.2	758.9	425.3	711.9	371.4	701.3

AFD Incore/Excore Ratios (M_i factors) for Quadrants 1-4

Quad 4	Quad 2	Quad 1	Quad 3
N41	N42	N43	N44
M4 = 1.374	M2 = 1.455	M1 = 1.486	M3 = 1.360

TABLE 8.9-2

CATAMBA 1

8.10 INCORE AND NUCLEAR INSTRUMENTATION SYSTEMS DETECTOR CORRELATION -
PT/1/A/4600/05A

Date(s) Performed: 4/1/85 - 4/4/85

I. PURPOSE

The purpose of this test was to determine the relationship between axial offset, as measured by the movable incore detector system, and the excore detector upper and lower currents; and then to calculate new factors for the OAC Excore Power Distribution Monitor program and the $f(\Delta I)$ function in the overtemperature ΔT reactor trip setpoints. Another purpose was to demonstrate the ability of the Rod Control System to control induced axial Xenon transients.

II. METHOD

A full core power distribution map was obtained at stable Axial Flux Difference (AFD), all-rods-out conditions. Control Bank D was diluted in to obtain a negative AFD transient. Three quarter-core maps were taken at approximately equal intervals while AFD was swinging negative. A full-core map was obtained with AFD held near its negative limit. Control Bank D was borated back out to its original position, starting a positive AFD swing. A quarter core map was obtained immediately after returning Bank D to its original position. Three more quarter-core maps were obtained at approximately equal intervals during the positive AFD swing.

Excore detector currents were measured during each of the core power distribution maps. This data was used to determine the relationship between incore axial offset and excore detector currents and to set up the proper $f(\Delta I)$ functions for the overtemperature ΔT setpoints.

After data was obtained, the Xenon-induced AFD oscillation was quenched by reinserting Control Bank D.

III. RESULTS

All criteria for this test were met. The excore detectors were recalibrated to extrapolated 100% currents with zero offset. The OAC Excore Power Distribution Monitor program was adjusted using the results of the test. The $f(\Delta I)$ function for overtemperature ΔT trip setpoints was set per Technical Specifications. The maximum indicated excore quadrant tilt ratio was < 1.02 (1.0138 in Quadrant 2, lower chamber).

Table 8.10-1 contains a summary of the data obtained during this test. Table 8.10-2 shows the results which were used to calibrate the excore detectors. Figure 8.10-1 is a plot of AFD versus time.

IV. CORRECTIVE ACTIONS

None required.

INCORE AND NUCLEAR INSTRUMENTATION SYSTEMS DETECTOR CORRELATION
INCORE AND NIS TEST DATA

Excore Detector Currents, 10⁻⁶ Amps

Map No.	% Power	Incore Axial Offset, %	N41		N42		N43		N44	
			Top	Bottom	Top	Bottom	Top	Bottom	Top	Bottom
15*	74.0	-10.729	354.6	443.4	363.8	469.9	338.0	439.4	306.0	433.0
16	74.8	-12.594	353.0	457.0	362.0	482.0	337.0	452.0	304.0	445.0
17	74.3	-14.398	343.0	459.0	349.0	485.0	326.0	456.0	294.0	449.0
18	74.9	-18.008	329.0	468.0	339.0	495.0	316.0	464.0	283.0	458.0
19*	73.2	-21.994	311.7	462.5	321.3	489.9	298.5	458.0	268.1	452.7
20	74.1	-10.363	354.7	440.2	364.7	466.6	338.0	453.3	306.7	430.3
21	74.1	+0.500	383.0	417.4	394.4	443.0	363.8	412.7	332.3	406.8
22	73.5	+8.886	399.8	390.5	412.9	415.2	380.1	387.2	347.9	380.6
23	73.6	+12.206	408.8	380.0	423.0	404.9	388.5	377.3	356.6	360.5

*indicates full-core maps. All other maps were quarter-core maps.

Table 8.10-1

CATWBA 1

INCORE AND NUCLEAR INSTRUMENTATION SYSTEMS DETECTOR CORRELATION
INCORE AND NIS TEST RESULTS

Full Power Excure Detector Currents, 10⁻⁶ Amps

Incore Axial Offset, %	N41		N42		N43		N44	
	Top	Bottom	Top	Bottom	Top	Bottom	Top	Bottom
30	625.1	458.3	647.4	490.4	591.8	454.6	548.0	434.8
20	587.8	492.8	607.8	525.6	557.3	488.7	513.7	471.5
10	550.5	527.3	568.3	560.9	522.8	522.7	479.4	508.2
0	513.1	561.8	528.8	596.2	488.3	556.7	445.0	544.8
-10	475.8	596.3	489.2	631.5	453.8	590.7	410.7	581.5
-20	438.4	630.8	449.7	666.8	419.2	624.8	376.4	618.2
-30	401.1	665.2	410.1	702.1	384.7	658.8	342.1	654.9

AFD Incore/Excure Ratios (Mj factors) for Quadrants 1-4

Quad 4	Quad 2	Quad 1	Quad 3
N41	N42	N43	N44
M4 = 1.491	M2 = 1.493	M1 = 1.517	M3 = 1.385

Table 8.10-2

CATAMBA 1

INCORE AND NUCLEAR INSTRUMENTATION SYSTEMS CORRELATION

AFD Versus Time

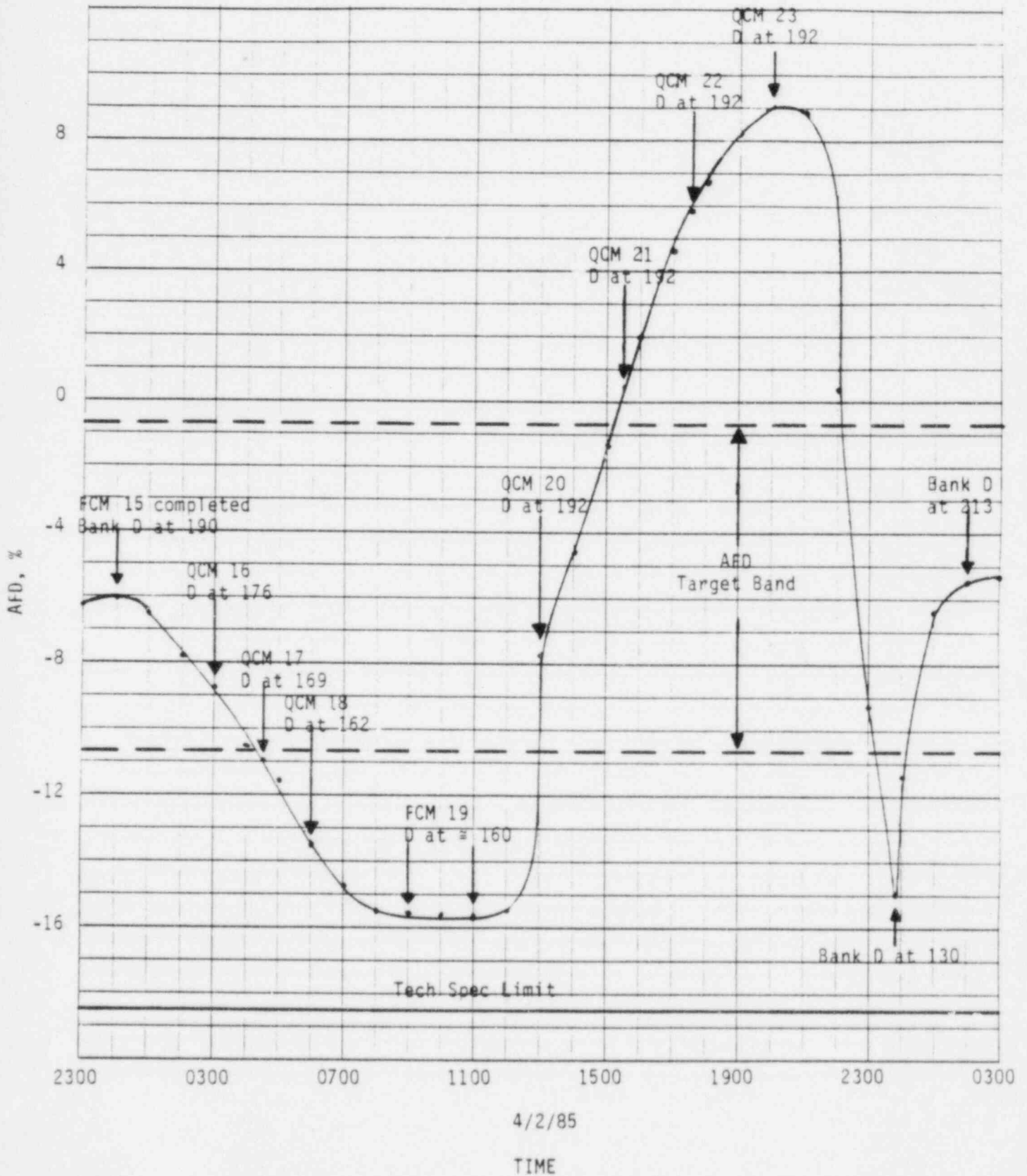


Figure 8.10-1

8.11 BELOW BANK ROD TEST - TP/1/A/2150/05

Date(s) Performed: 3/17/85 - 3/23/85

I. PURPOSE

The objectives of the Below Bank Rod Test were the following:

- A. To demonstrate response of the incore and excore instrumentation systems to a power maldistribution caused by an RCCA (rod D-12) moving below its bank (Control Bank D).
- B. To demonstrate that core power limits (F_Q and $F_{\Delta H}$) are not exceeded with the rod partially and fully misaligned.
- C. To obtain the integral and differential worth of rod D-12.
- D. To benchmark the Trace Pair Analysis technique per OP/0/A/6150/12, Control Rod Position Verification.
- E. To verify that Operator Aid Computer and Control Board Annunciator alarms associated with Rod Insertion Limits actuate and clear at established setpoints.

II. METHOD

Stable conditions were established with the unit at \approx 49% F. P. and Control Bank D at \approx 200 steps withdrawn. A baseline full core flux map was then obtained in this configuration (FCM/1/01/011).

RCCA D-12 was slowly diluted into the core as Trace Pair and Excore Detector data was taken each 25 steps.

A full core flux map was obtained with the RCCA fully inserted (FCM/1/01/012). The RCCA was then borated to a position \approx 25 steps below Control Bank D. The reactivity computer was used to obtain rod worth data during this withdrawal. A full core flux map was taken at this configuration, (FCM/1/01/013), before the rod was fully recovered.

During the insertion and withdrawal of rod D-12, the OAC Alarm and Control Board Annunciators dealing with rod insertion limit were monitored to ensure proper actuation.

III. RESULTS

All Acceptance Criteria for the stated test objectives were met. Discussion of these specific objectives follows.

- A. Response of Incore and Excore NIS to rod misalignment as follows:

1. Good response by the Excore Nuclear Instrumentation System to the rod misalignment was noted. These results are illustrated on Table 8.11-1.
2. The measured assembly powers obtained from the Incore Nuclear Instrumentation System (utilizing the SNC Core Program on the VAX Computer) clearly show the effects of the fully misaligned rod (Figure 8.11-1) and the partially misaligned rod (Figure 8.11-2).
3. The effects of the rod misalignment on Incore Axial Offset and Quadrant Power Tilt Ratio were also noted. These parameters are displayed as follows:

Base Case Map Results - Figure 8.11-3

Fully Misaligned Map Results - Figure 8.11-4

Partially Misaligned Map Results - Figure 8.11-5

4. The effects of the rod misalignment on Excore Quadrant Power Tilt Ratio are displayed on Figures 8.11-6 through 8.11-9.
- B. A summary of the measured core peaking factors, F_Q and $F_{\Delta H}$, for the Base Case, Full Misalignment, and Partial Misalignment Flux Maps is provided on Tables 8.11-2 and 8.11-3.
- C. The integral worth of D-12 was obtained by analyzing the reactivity trace shown on Figure 8.11-10, obtained during the withdrawal of the rod. The predicted worth of this rod was 150 pcm and the measured worth was 15.2 pcm. A differential worth of 0.0745 pcm/step could only be inferred due to the large number of steps involved with each rod withdrawal necessary to obtain a discernable change in reactivity. The huge discrepancy between predicted and measured integral worth can be explained by the tremendous uncertainty involved with at power reactivity measurements using a reactimeter. This is due mainly to the amplified spatial effects at power. There were no quantitative acceptance criteria associated with this measurement, merely a requirement to perform it.
- D. The Trace Pair Analysis technique was applied every 25 steps during insertion of Rod D-12. The selected traces were E-11, a location diagonal to D-12, and L-05, a mirrored location in the core quadrant diagonal to the one in which D-12 is located. The base case trace pair results (D-12 fully aligned with Bank D) are shown on Figure 8.11-11. The Trace pair with D-12 inserted 54 steps below Bank D is shown on Figure 8.11-12. This trace pair is representative of the others obtained at 25 step intervals. It clearly demonstrates the ability of this technique to detect misaligned control rods, which requires the axial peaks of the two traces to differ by greater than one axial data point. The axial peaks are circled on the associated figures.

- E. All Operator Aid Computer alarms and Control Board Annunciators associated with control rod insertion limits actuated and cleared at the appropriate setpoints. A summary of their operation is displayed on Table 8.11-4.

IV. CORRECTIVE ACTIONS

All plant systems and components pertinent to this test performed well with two exceptions. The first was the malfunction of Incore Detectors B, C, and D during the baseline flux map and then during the first attempt at the fully misaligned flux map. This faulty operation resulted in an extension of the time during which D-12 was misaligned, thereby, contributing to an undesirable power maldistribution.

This delay was further aggravated by the failure of the OAC to store flux map pass data during the fully misaligned map. The computer's scanner which gathers ENA data went down during the map without alarming. This prohibited any type of malfunction diagnosis and led to the abortion of the flux map and the retrieval of the RCCA to prevent a violation of Tech Specs.

The RCCA was unfortunately withdrawn during a period of peak Xenon in the vicinity of the rod (\approx 9 hours after misalignment). This caused a substantial oscillating Quadrant Power Tilt (in excess of 60% at peak) which forced Operations to decrease power. Daily surveillance procedures required a channel check of NI's and ΔT 's. With a power tilt, channel deviation of $\leq \pm 2\%$ could not be met. The oscillation eventually converged and allowances for tilts in the surveillance procedure were made allowing testing to be resumed. This occurred four days later.

Indicated Reactor Coolant Flow was unreliable during this test due to "preliminary" elbow tap correction factors in the OAC. The final corrections were eventually obtained during 75% F. P. testing. This precluded analysis of the calculated R_1 factors but this requirement was waived due to acceptable $F_{\Delta H}$ values. The subsequent calorimetric reactor coolant flow measurement verified that adequate flow was indeed available to satisfy the R_1 criterion.

BELOW BANK ROD TEST
EXCORE DETECTOR RESPONSE TO ROD INSERTION

Initial Thermal Power: 48.9% F.P.

Shutdown Bank Positions: A 228 B 228 C 228 D 228 E 228

Control Bank Positions: A 228 B 228 C 228 D 204

Date	TIME	UNIT RCCA POSITION	N-41 (Volts)			N-42 (Volts)			N-43 (Volts)			N-44 (Volts)		
			TOP	BOT.	Q	TOP	BOT.	Q	TOP	BOT.	Q	TOP	BOT.	Q
3/18/85	0036	204	238.8	271.7	47.7	246.2	287.7	47.9	227.7	269.3	47.8	207.4	265.4	45.8
3/18/85	0130	192	240.8	273.8	48.1	239.8	288.0	47.3	228.8	270.7	48.1	208.5	267.0	46.1
3/18/85	0144	179	242.6	275.5	48.5	229.8	287.0	46.3	229.8	273.0	48.4	209.5	269.0	46.4
3/18/85	0212	150	245.5	279.8	49.1	299.2	277.0	42.5	229.3	274.7	48.5	209.0	270.1	46.4
3/18/85	0243	125	252.2	288.6	50.6	172.8	262.0	38.7	231.8	280.0	49.2	211.5	275.8	47.2
3/18/85	0318	100	261.5	301.0	52.6	149.8	236.5	34.3	235.8	285.5	50.1	214.3	281.0	48.0
3/18/85	0341	75	267.4	308.1	53.8	132.9	201.0	29.7	237.0	286.4	50.4	214.8	281.8	48.1
3/18/85	0414	50	278.3	320.9	56.0	122.6	166.8	25.8	241.4	292.0	51.3	219.3	287.0	49.0
3/18/85	0431	25	282.3	327.6	57.0	117.7	144.6	23.5	243.3	292.9	51.6	220.8	287.8	49.3
3/23/85	0253	0	290.2	353.0	60.1	109.5	130.7	21.5	243.5	306.0	52.8	219.8	299.2	50.2

Notes:

- 1) Excore currents taken on 3/18/85 were during initial insertion of the RCCA.
- 2) Excore currents taken on 3/23/85 were after the RCCA was reinserted to complete the test after delay due to Xenon oscillation.
- 3) Unit RCCA was rod D-12 in Control Bank D.

Table 8.11-1

CATAMBA 1

BELOW BANK ROD TEST
CORE PEAKING FACTOR SUMMARY
(HEAT FLUX HOT CHANNEL FACTOR)

FLUX MAP I.D.	ROD D-12 POSITION (steps wd)	MAXIMUM F _{xy} UNEXCLUDED	CORE LOC.	TECH SPEC LIMIT	MAXIMUM TOTAL F _Q	CORE LOC.	TECH SPEC LIMIT
			AXIAL LOC.			AXIAL LOC.	
FCM/1/01/011	204	1.5612	C-13	1.8112	2.1269	D-12	4.64
			13			27	
FCM/1/01/012	0	1.9199*	M-04	1.8227*	2.6043	M-04	4.64
			13			25	
FCM/1/01/013	179	1.7382	L-02	1.8228	2.3604	L-02	4.64
			13			25	

*Note: Effects of F_{xy} violation on F_Q x P(REL) not required to be evaluated due to abnormal Control Rod configuration.

Table 8.11-2

CATAMBA 1

BELOW BANK ROD TEST
CORE PEAKING FACTOR SUMMARY
(ENTHALPY RISE HOT CHANNEL FACTOR)

FLUX MAP I.D.	ROD D-12 POSITION (steps wd)	MAXIMUM MEASURED F _{ΔH}	CORE LOCATION	TEST LIMIT	CALCULATED R ₁ VALUE	TECH SPEC LIMIT
FCM/1/01/011	204	1.3988	D-12	1.84**	0.8137*	1.0
FCM/1/01/012	0	1.7043	M-04	1.84**	0.9852*	1.0
FCM/1/01/013	179	1.5629	L-02	1.84**	0.9034*	1.0

*Note: R₁ versus Reactor Coolant Flow was not evaluated due to RCS flow uncertainty.

**Note: F_{ΔH} limit established by Westinghouse Safety Review Criterion documented in DCP/DDP-SU-3.1.1

Table 8.11-3

CATANBA 1

BELOW BANK ROD TEST
SUMMARY OF CONTROL ROD INSERTION LIMIT ALARMS

<u>ALARM</u>	<u>SOURCE</u>	<u>ALARMED</u> (D-12 Insertion)		<u>CLEARED</u> (D-12 Withdrawal)	
		<u>SET PT.</u>	<u>ACTUAL</u>	<u>SET PT.</u>	<u>ACTUAL</u>
"Control Rod Bank Lo Limit"	Control Board Annunciator	88 ± 3	89	87 ± 3	89
"Control Rod Bank Lo Lo Limit"	Control Board Annunciator	78 ± 3	80	91 ± 3	92
"Lo Bank Insertion Limit"	Operator Aid Computer	71 ± 3	68	>71	91
"Control Bank Tech Spec Insertion Limit Reached"	Operator Aid Computer	63 ± 3	60	82 ± 3	81

Table 8.11-4

CATWBA 1

BELOW BANK ROD TEST

COMPARISON OF MEASURED $F_{\Delta H}$ VALUES - BASE CASE TO PARTIAL MISALIGNMENT

Map	01	02	03	04	05	06	07	08	09	10	11	12	13	14	15
Base Case Map (D-12 @ 204)	0.698	0.903	0.972	0.982	0.953	0.909	0.718								
FCM/T/01/011	0.760	1.059	1.079	0.908	1.133	0.944	1.125	0.913	1.109	1.102	0.797				
	0.760	1.098	1.076	0.978	1.027	0.944	1.000	0.941	1.043	0.990	1.098	1.138	0.797		
	1.054	1.069	1.027	1.106	0.975	1.119	0.986	1.127	0.993	1.129	1.342	1.088	1.118		
	0.657	1.028	0.944	1.092	1.006	1.137	0.997	1.143	1.011	1.170	1.041	1.129	1.005	1.130	0.732
	0.848	0.865	0.990	0.951	1.139	0.993	1.113	0.976	1.141	1.024	1.172	0.991	1.061	0.950	0.946
	0.928	1.105	0.922	1.099	1.005	1.131	0.943	1.011	0.964	1.149	1.008	1.126	0.955	1.151	0.964
	0.962	0.938	0.991	0.973	1.143	0.983	1.007	0.893	1.008	0.979	1.141	0.977	1.025	0.921	0.956
	0.924	1.111	0.929	1.106	1.007	1.138	0.952	0.981	0.935	1.128	1.002	1.123	0.946	1.117	0.940
	0.876	0.897	1.015	0.969	1.150	1.015	1.126	0.956	1.111	0.997	1.148	0.984	1.031	0.911	0.907
	0.687	1.066	0.968	1.102	1.006	1.145	0.984	1.111	0.980	1.130	1.006	1.117	0.982	1.084	0.701
	1.049	1.065	1.322	1.089	0.950	1.093	0.946	1.089	0.961	1.093	1.299	1.060	1.060		
	0.755	1.075	1.013	0.931	0.983	0.911	0.986	0.893	0.956	0.926	1.021	1.049	0.774		
	0.730	0.998	1.013	0.854	1.068	0.878	1.054	0.861	1.062	1.028	0.742				
				0.646	0.832	0.892	0.910	0.896	0.870	0.686					
	01	02	03	04	05	06	07	08	09	10	11	12	13	14	15
					0.635	0.796	0.865	0.809	0.836	0.709	0.538				
			0.756	1.047	1.022	0.886	1.006	0.858	0.979	0.789	0.849	0.812	0.573		
		0.746	1.040	1.062	1.001	1.073	0.922	0.950	0.869	0.818	0.768	0.809	0.796	0.586	
		1.040	1.073	1.300	1.119	1.022	1.082	0.962	1.014	0.815	0.871	0.994	0.830	0.837	
	0.698	1.075	1.021	1.130	1.067	1.177	1.000	1.103	0.957	1.008	0.842	0.871	0.809	0.862	0.546
	0.875	0.949	1.070	1.051	1.219	1.058	1.110	0.995	1.088	0.968	1.033	0.878	0.878	0.757	0.680
	1.008	1.173	1.049	1.224	1.106	1.213	0.991	0.991	0.929	1.077	0.953	0.999	0.855	0.933	0.814
	1.006	1.083	1.109	1.134	1.265	1.102	1.093	0.937	1.002	0.971	1.081	0.927	0.955	0.863	0.812
	1.024	1.215	1.082	1.256	1.154	1.247	1.039	1.066	0.995	1.118	1.001	1.092	0.918	1.013	0.849
	0.924	1.043	1.208	1.108	1.294	1.173	1.235	1.068	1.180	1.057	1.177	0.996	1.023	0.883	0.800
	0.747	1.180	1.135	1.184	1.145	1.289	1.130	1.223	1.083	1.180	1.067	1.100	0.999	1.034	0.652
	1.153	1.180	1.416	1.213	1.103	1.226	1.085	1.181	1.013	1.093	1.289	1.063	1.033		
	0.816	1.129	1.169	1.108	1.131	1.041	1.104	1.033	1.096	1.038	1.074	1.025	0.756		
FCM/T/01/013	0.813	1.156	1.160	0.984	1.165	1.001	1.127	0.962	1.125	1.062	0.742				
Map With D-12 @ 179 Steps	0.727	0.891	1.010	0.985	0.987	0.879	0.713								

Figure 8.11-2

CATAWBA 1

BELOW BANK ROD TEST
CORE PEAKING FACTOR SUMMARY
(ENTHALPY RISE HOT CHANNEL FACTOR)

FLUX MAP I.D.	ROD D-12 POSITION (steps wd)	MAXIMUM MEASURED $F_{\Delta H}$	CORE LOCATION	TEST LIMIT	CALCULATED R_1 VALUE	TECH SPEC LIMIT
FCM/1/01/011	204	1.3988	D-12	1.84**	0.8137*	1.0
FCM/1/01/012	0	1.7043	M-04	1.84**	0.9852*	1.0
FCM/1/01/013	179	1.5629	L-02	1.84**	0.9034*	1.0

*Note: R_1 versus Reactor Coolant Flow was not evaluated due to RCS flow uncertainty.

**Note: $F_{\Delta H}$ limit established by Westinghouse Safety Review Criterion documented in DCP/DDP-SU-3.1.1

Table 8.11-3

CATAMBA 1

BELOW BANK ROD TEST
 AXIAL OFFSETS AND INCORE TILTS - ROD D-12 AT 0 STEPS WITHDRAWN

AXIAL OFFSET RATIOS FROM ASSEMBLAGE POWERS

TOTAL CORE AXIAL OFFSET = -0.07479

AXIAL OFFSET FOR QUADRANTS	1 2 4 3	AND FOR QUADRANTS	5 6 8 7		
-0.07678	-0.06205			-0.08360	-0.06602
-0.07744	-0.07807				-0.07189
					-0.07438

AXIAL OFFSET BY OCTANTS	2 3 1 4 8 5 7 6				
-0.08152	-0.07139	-0.05816		-0.06592	
-0.08556	-0.06918	-0.07990		-0.07602	

TILTING FACTORS FROM ASSEMBLAGE POWERS FOR ENTIRE CORE HEIGHT

TILTING FACTORS FOR QUADRANTS	1 2 4 3	AND FOR QUADRANTS	5 6 8 7	FOR ENTIRE CORE HEIGHT	
1.06747	0.68508			0.84148	
1.19329	1.05416		1.16979	1.14786	0.84087

TILTING FACTORS FOR OCTANTS	2 3 1 4 8 5 7 6	FOR ENTIRE CORE HEIGHT			
1.13507	0.99986	0.68309		0.68706	
1.20451	1.18207	1.11364		0.99468	

TILTING FACTORS FOR QUADRANTS	1 2 4 3	AND FOR QUADRANTS	5 6 8 7	FOR TOP HALF, NORMALIZED	
1.06518	0.69452			0.84946	
1.18987	1.05043		1.15866	1.14837	0.84351

TILTING FACTORS FOR QUADRANTS	1 2 4 3	AND FOR QUADRANTS	5 6 8 7	FOR BOT HALF, NORMALIZED	
1.06943	0.67696			0.83461	
1.19623	1.05738		1.17938	1.14741	0.83860

Figure 8.11-4

CATAWBA 1

BELOW BANK ROD TEST
 AXIAL OFFSETS AND INCORE TILTS - ROD D-12 AT 179 STEPS WITHDRAWN

AXIAL OFFSET RATIOS FROM ASSEMBLAGE POWERS

TOTAL CORE AXIAL OFFSET = -0.06544

AXIAL OFFSET FOR QUADRANTS	1 2 4 3	AND FOR QUADRANTS	5 6 8 7	
-0.05578	-0.08648			
-0.06079	-0.06229	-0.06038	-0.06899	-0.07530
			-0.05901	

AXIAL OFFSET BY OCTANTS	1 2 3 4 5 6 7 8		
-0.05568	-0.05589	-0.08390	-0.08904
-0.06482	-0.05667	-0.06147	-0.06316

TILTING FACTORS FROM ASSEMBLAGE POWERS FOR ENTIRE CORE HEIGHT

TILTING FACTORS FOR QUADRANTS	1 2 4 3	AND FOR QUADRANTS	5 6 8 7	FOR ENTIRE CORE HEIGHT
1.01916	0.86448			
1.10460	1.01177	1.08553	0.92287	0.92316
			1.06844	

TILTING FACTORS FOR OCTANTS	1 2 3 4 5 6 7 8	FOR ENTIRE CORE HEIGHT	
1.05548	0.98283	0.86291	0.86605
1.11557	1.09362	1.04326	0.98028

TILTING FACTORS FOR QUADRANTS	1 2 4 3	AND FOR QUADRANTS	5 6 8 7	FOR TOP HALF, NORMALIZED
1.02969	0.84502			
1.11010	1.01518	1.09141	0.91937	0.91342
			1.07579	

TILTING FACTORS FOR QUADRANTS	1 2 4 3	AND FOR QUADRANTS	5 6 8 7	FOR BOT HALF, NORMALIZED
1.00992	0.88154			
1.09977	1.00877	1.08037	0.92594	0.93171
			1.06199	

Figure 8.11-5

BELOW BANK ROD TEST
 EXCORE TILT VS ROD POSITION - QUAD. 1 (N-43)

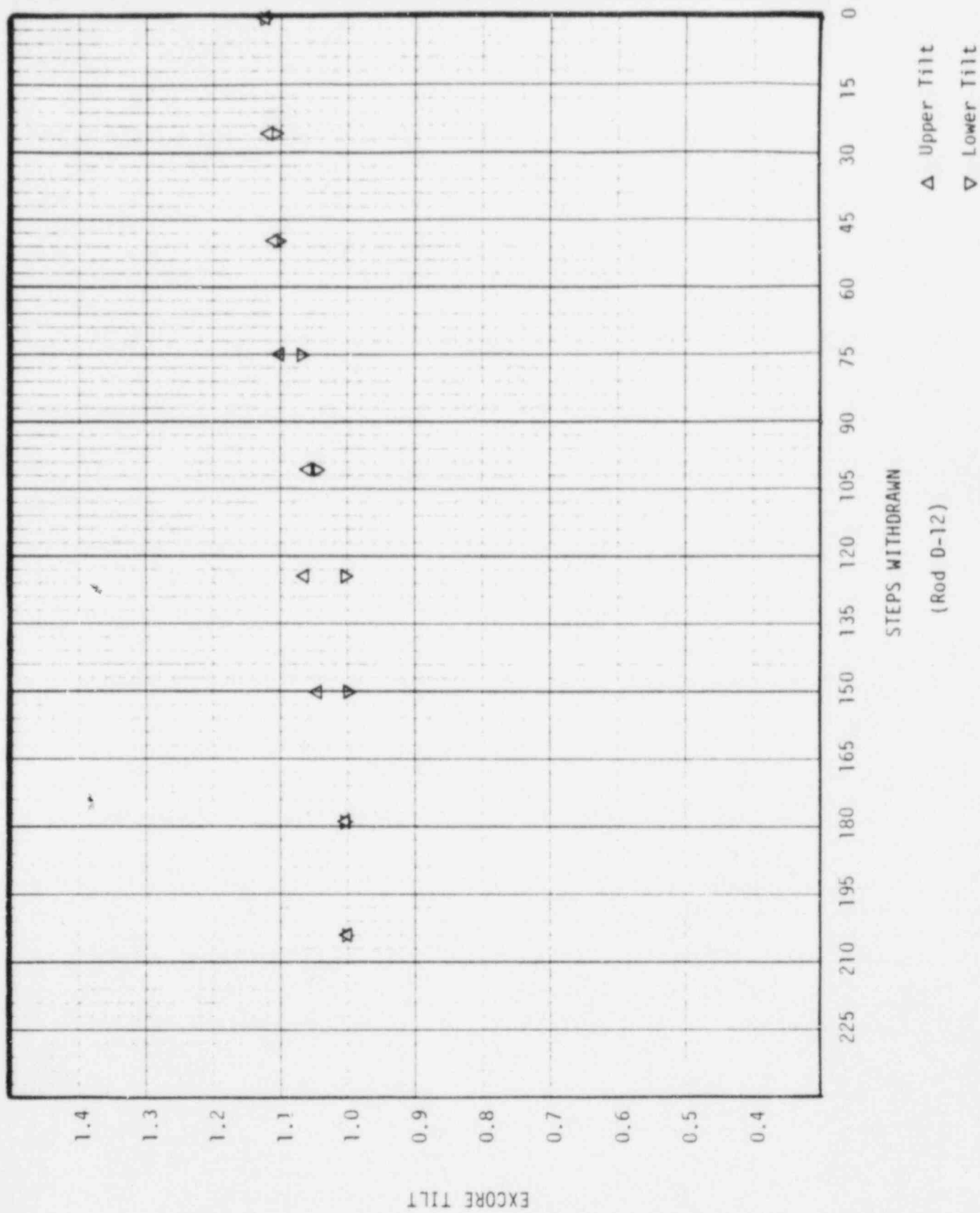


Figure 8.11-6

BELOW BANK ROD TEST
EXCORE TILT vs ROD POSITION - QUAD. 2 (N-42)

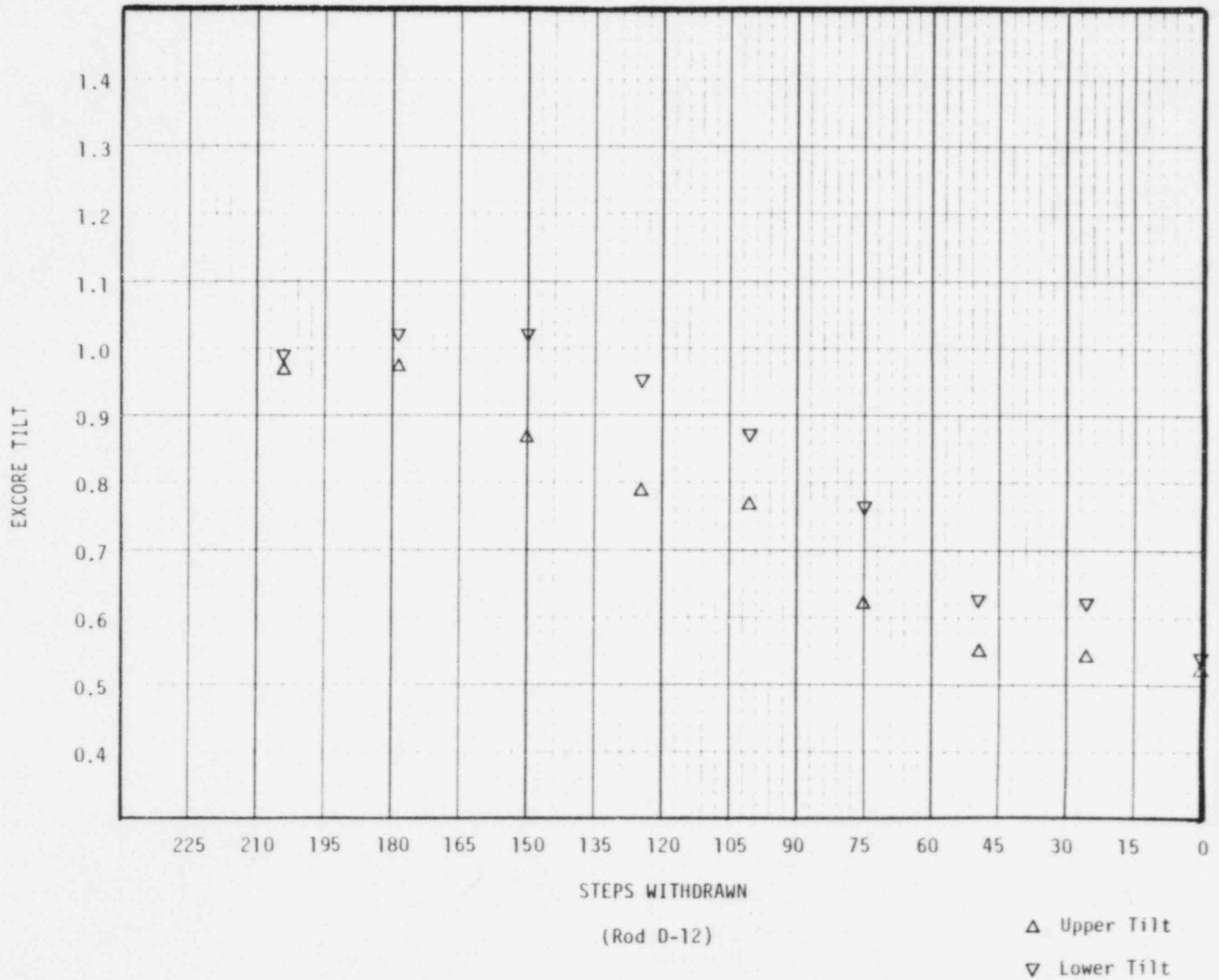


Figure 8.11-7

CATAMBA 1

BELLOW BANK ROD TEST
EXCORE TILT vs ROD POSITION -QUAD. 3 (N-44)

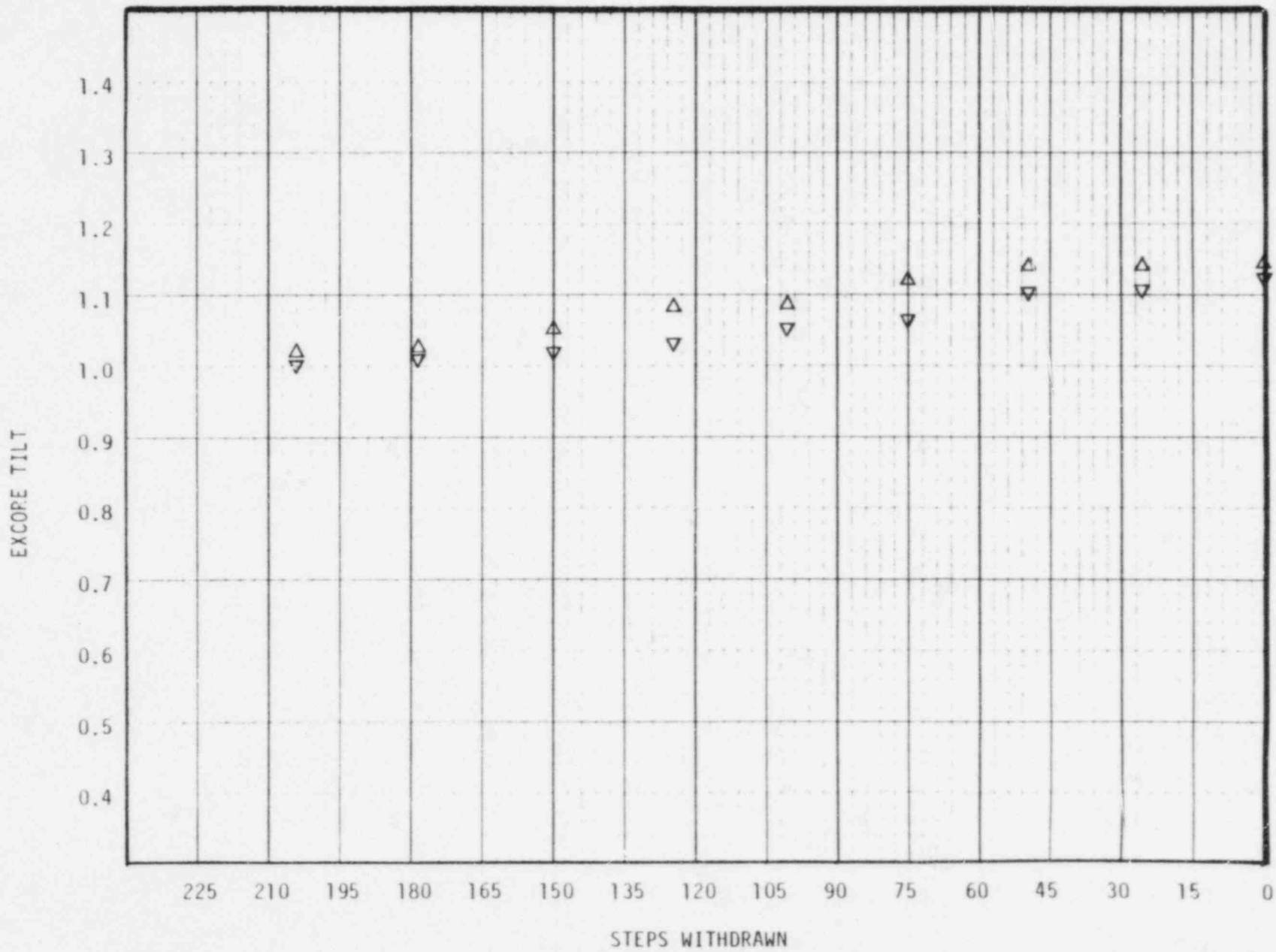


Figure 8.11-8

CATAWBA 1

(Rod D-12)

△ Upper Tilt

▽ Lower Tilt

BELOW BANK ROD TEST
EXCORE TILT vs ROD POSITION - QUAD. 4 (N-41)

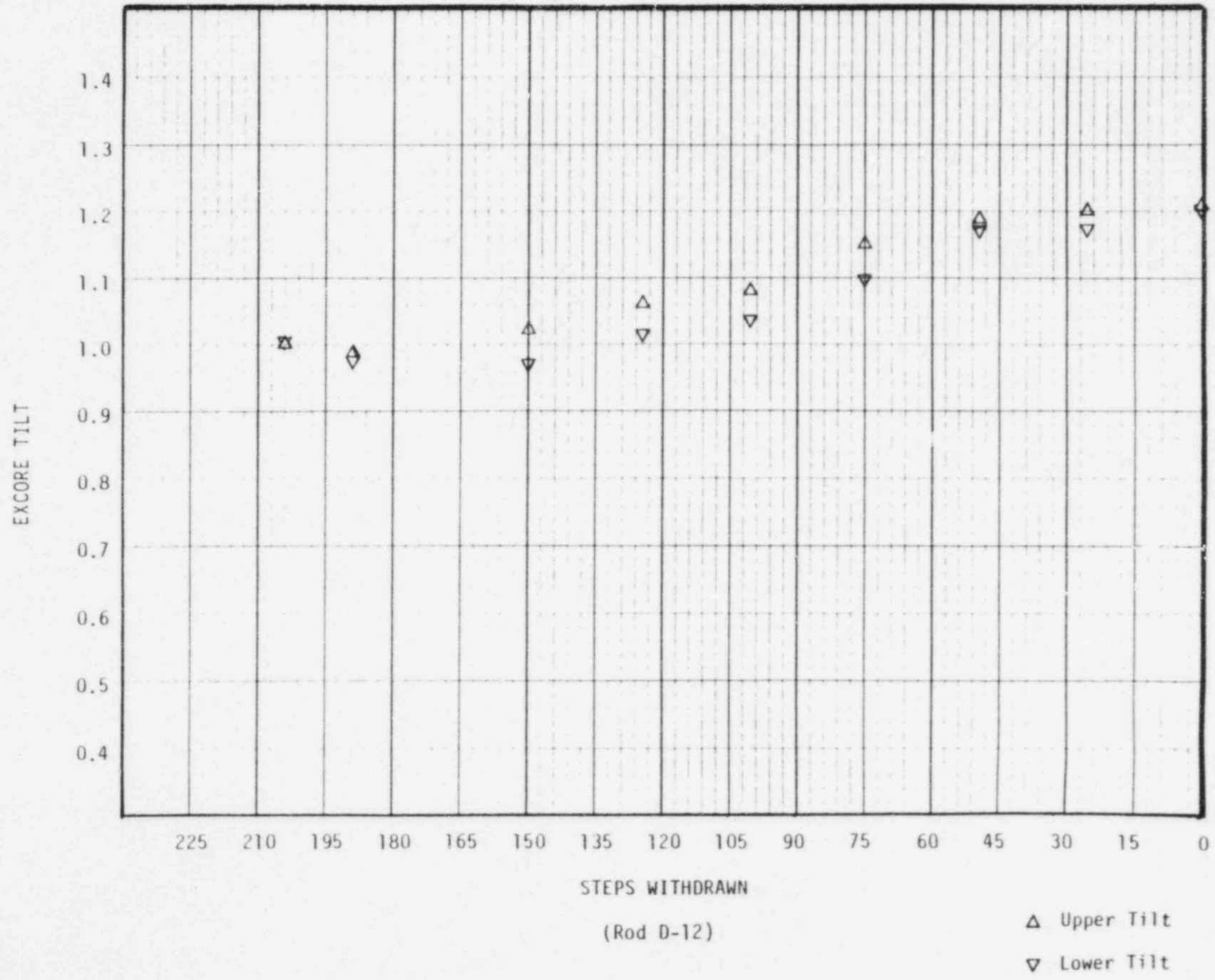


Figure 8.11-9

CATAMBA 1

BELOW BANK ROD TEST

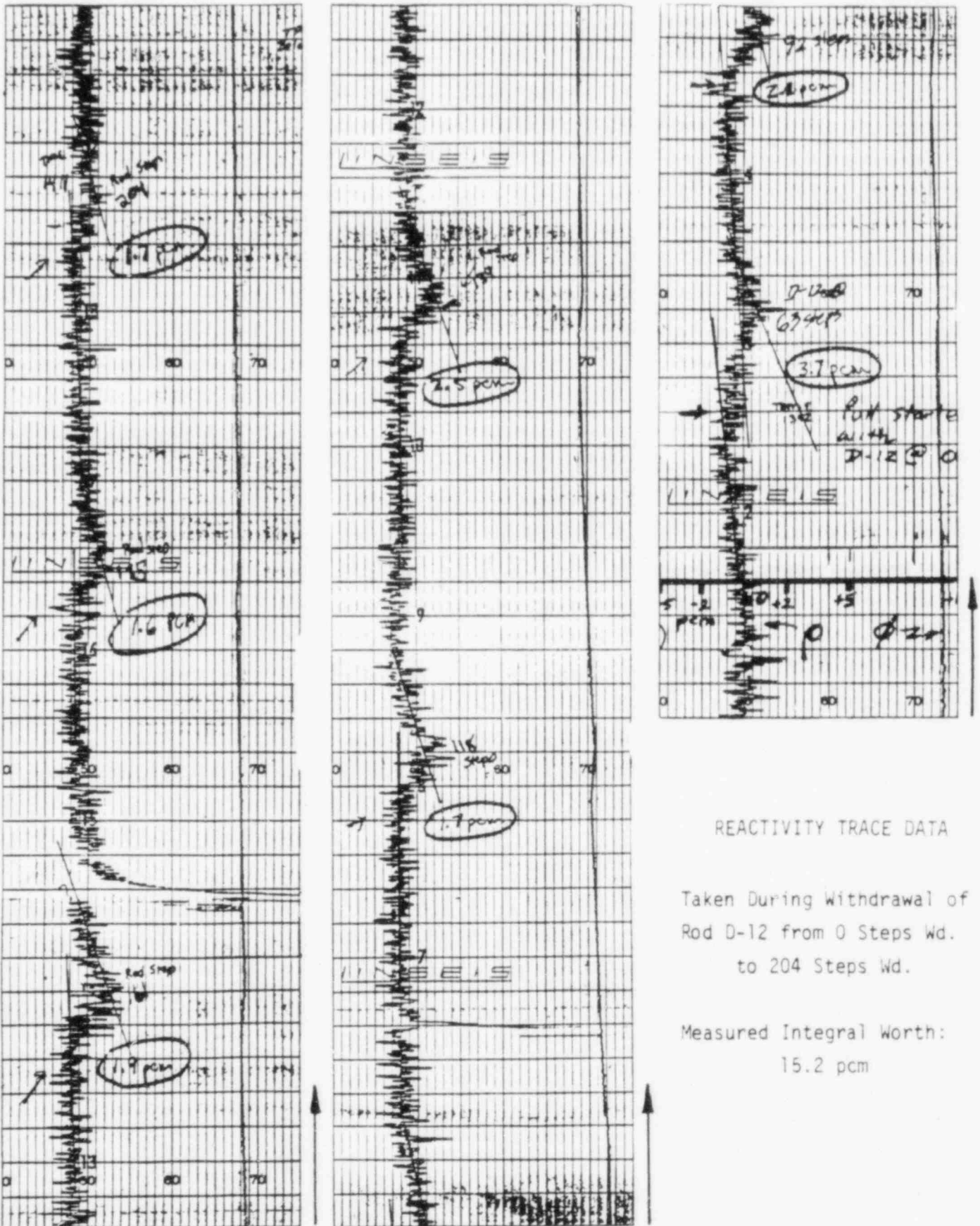


Figure 8.11-10

BELOW BANK ROD TEST
TRACE PAIR ANALYSIS WITH ROD D-12 ALIGNED WITH BANK D

DATE 03/18/85 TIME 01:02:33 MOVEABLE DEFLECTOR TRACE PAIR ANALYSIS

ROD BANK POSITION (SIRP) CH 228 CH 228 CC 228 CD 204

MW THERMAL POWER 1654.0 RPM BURON CONCENTRATION 698

	TRACE L05	TRACE F11
30	1.5506	1.8347
29	2.4541	2.9796
28	3.6997	4.1747
27	4.7649	5.2734
26	6.0326	6.7047
25	6.9328	7.4971
24	7.7691	8.2959
23	8.3280	8.7546
22	9.0255	9.5371
21	9.6444	10.096
20	10.447	10.478
19	10.274	10.546
18	10.395	10.977
17	10.845	11.311
16	11.022	11.408
15	11.077	11.231
14	10.751	11.157
13	10.935	11.322
12	10.826	11.120
11	10.850	10.874
10	10.125	10.325
9	9.9214	10.171
8	9.5661	9.7358
7	9.0593	9.1245
6	5.2604	8.1495
5	7.4354	7.4109
4	6.6102	6.9434
3	5.4850	5.2347
2	4.3221	3.7744
1	2.8746	2.4309

Figure 8.11-11

BELOW BANK ROD TEST
TRACE PAIR ANALYSIS WITH ROD D-12 54 STEPS BELOW BANK

DATE 03/18/85 TIME 02:21A MOVABLE DETECTOR TRACE PAIR ANALYSIS

ROD BANK POSITION (STEPS) CA 228 CB 228 CC 228 CD 150

MW THERMAL POWER 1665.5 PPM BOREN CONCENTRATION 692

	TRACE E11	TRACE L05
30	1.1546	1.6356
29	1.2574	2.5929
28	2.7368	3.2341
27	3.4960	4.7441
26	4.4077	6.1403
25	5.0105	7.3216
24	5.6907	7.9104
22	7.2512	9.1666
21	7.7089	9.8454
20	8.4342	10.329
19	9.1668	10.543
18	9.5654	10.652
17	10.342	11.112
16	10.723	11.289
15	10.922	11.523
14	10.772	10.455
13	11.100	11.207
12	11.159	11.147
11	11.050	10.948
10	10.540	10.382
9	10.377	10.217
8	10.073	9.8417
7	9.5052	9.3406
6	8.7573	8.5553
5	7.8403	7.6216
4	7.0249	6.8742
3	5.8749	5.7541
2	4.5445	4.8828
1	2.9599	2.9479

Figure 8.11-12

8.12 CALORIMETRIC REACTOR COOLANT FLOW MEASUREMENT - PT/1/A/4150/13B

Date(s) Performed: 3/7/85, 3/31/85

I. PURPOSE

The Calorimetric Reactor Coolant Flow Measurement was performed during power ascension testing with the following objectives:

- A. To determine the reactor coolant system flow rate by precision heat balances about the steam generators with a minimum of measurement uncertainty.
- B. To determine Reactor Thermal Power from the precision heat balance data.
- C. To calculate correction factors for the reactor coolant flow elbow taps to correlate their indications of flow with the precision measurement results.
- D. To ensure that adequate reactor coolant system flow is present as required by Tech Specs.

II. METHOD

In order to minimize the uncertainty associated with the determination of reactor coolant flow, plant parameters were measured manually with the use of precision test instruments.

A comparison of the uncertainties involved with plant parameters as measured by process instrumentation (and subsequently input to the OAC) and the precision test instrumentation is as follows:

<u>PARAMETER</u>	<u>UNCERTAINTY</u>	
	<u>OAC</u>	<u>PRECISION CALOR.</u>
Feedwater Flow	± 1.75%	± 0.5%
Feedwater Pressure	± 0.0%	± 0.003%
Feedwater Temperature	± 0.6%	± 0.07%
S/G Steam Pressure	± 0.26%	± 0.006%
NC Hot Leg Temperature	± 5.89%	± 2.9%
NC Cold Leg Temperature	± 4.27%	± 0.59%

These uncertainties account for a total of 3.76% uncertainty assigned to the OAC Reactor Coolant Flow Calculation. The significantly smaller uncertainties involved with the use of precision instrumentation to measure these parameters reduces the total uncertainty on the precision calorimetric measurement to 1.51%.

The other plant parameters essential to the precision calorimetric were measured with the process instrumentation available. These parameters are:

- Pressurizer pressure
- Reactor Coolant Pump voltage
- Reactor Coolant Pump current
- S/G Blowdown flow
- CVCS Letdown flow, temperature, and pressure
- CVCS Charging flow, temperature, and pressure
- S/G Tempering flow

The necessary parameters were trended for a one hour period. The data obtained was input to the NCFLOW 2 Program which is established on a micro computer (the IBM 9000). The measurements were repeated twice to ensure reproduceability.

III. RESULTS

This test was performed at power levels of 68% F.P., 74% F.P., and 100% F.P. One of the three test runs at 68% F.P. was invalid so another one hour run was performed at 74% F.P. The single one hour run at 100% F.P. was an "information only" measurement and was not used to calculate the Reactor Coolant System elbow tap flow correction factors as the previous three runs were.

A typical output of the NCFLOW 2 Program is shown on Figure 8.12-1 (test run #1 at 68% F.P.). The results of the four valid test runs are summarized on Table 8.12-1.

The test Acceptance Criteria were met as follows:

- A. Reactor Coolant flowrates in excess of the Tech Spec Limit of 396100 gpm were measured by the four valid test runs. These results are shown on Table 8.12-1.
- B. Reactor Coolant Elbow Tap flow correction factors were calculated from 3 of the 4 test runs (the 100% F.P. results were not used). These calculations are detailed on Table 8.12-2.

IV. CORRECTIVE ACTIONS

Due to the fact that the second one hour test run at 68% F.P. was invalid an additional run had to be performed at 74% F.P. This was the only corrective action necessary.

CALORIMETRIC REACTOR COOLANT FLOW MEASUREMENT
SUMMARY OF TEST RESULTS

Run #	CALC. THERMAL POWER MW Thermal	PERCENT OF Full Power	CALC. REACTOR COOLANT FLOW Mib/hr	FLOW GPM	PERCENT OF TECH SPEC FLOW
1	2301.9	67.49	148.042	399180	100.78
2*	2291.2	67.17	145.751	393210	99.27
3	2313.1	67.81	147.816	398860	100.70
4	2527.4	74.10	147.220	397950	100.47
5*	3430.6	100.58	147.822	399960	100.97

*NOTE: These runs were not utilized to calculate Reactor Coolant Flow Elbow Tap Correction Factors

Table 8.12-1

CATAMBA 1

CALORIMETRIC REACTOR COOLANT FLOW MEASUREMENT
 REACTOR COOLANT FLOW ELBOW TAP CORRECTION FACTOR CALCULATION

	ELBOW TAP ΔP (IN H ₂ O)			CALC FLOW, M̄ (MLB/HR)			CORRECTION FACTOR, K _{ETI}			AVG K _{ETI}
	RUN #1	RUN #3	RUN #4	RUN #1	RUN #3	RUN #4	RUN #1	RUN #3	RUN #4	
Loop A, Ch I	332.770	332.456	331.366	38.314	38.252	38.006	0.30901	0.30877	0.30757	0.3085
Loop A, Ch II	346.726	346.518	345.262	38.314	38.252	38.006	0.30273	0.30244	0.30132	0.3022
Loop A, Ch III	343.646	343.465	342.115	38.314	38.252	38.006	0.30408	0.30379	0.30261	0.3035
Loop B, Ch I	308.247	307.813	306.651	35.394	35.374	35.319	0.29591	0.29606	0.29640	0.2961
Loop B, Ch II	305.127	304.873	304.040	35.394	35.374	35.319	0.29742	0.29749	0.29767	0.2975
Loop B, Ch III	308.173	308.055	307.173	35.394	35.374	35.319	0.29595	0.29595	0.29615	0.2960
Loop C, Ch I	354.426	354.154	352.830	37.924	37.818	37.626	0.29635	0.29572	0.29504	0.2957
Loop C, Ch II	347.426	347.077	345.666	37.924	37.818	37.626	0.24929	0.29872	0.29809	0.2987
Loop C, Ch III	352.487	352.054	350.655	37.924	37.818	37.626	0.29714	0.29660	0.29596	0.2966
Loop D, Ch I	317.761	317.466	316.128	36.410	36.371	36.269	0.30039	0.30031	0.30037	0.3004
Loop D, Ch II	322.170	321.933	320.794	36.410	36.371	36.269	0.29833	0.29882	0.29817	0.2982
Loop D, Ch III	301.194	300.961	299.845	36.410	36.371	36.269	0.30854	0.30844	0.30841	0.3085

$$K_{ETI} = \frac{\text{Calc. Reactor Coolant Flow } (\dot{M})}{[(\text{Reactor Coolant Density, } \rho) \times (\text{Elbow Tap } \Delta P)]^{\frac{1}{2}}}$$

Table 8.12-2

CALORIMETRIC REACTOR COOLANT FLOW MEASUREMENT
EXAMPLE OF NCFLOW2 OUTPUT

CATAWBA NUCLEAR STATION SECONDARY HEAT BALANCE
AND NC FLOW CALCULATIONS (UNIT 1)

	FEEDWATER FLOW	NC LOOP ENTHALPIES, BTU/LBM				NC PUMP POWER	THERMAL POWER	NC LOOP FLOW	
		PRIMARY	SECONDARY						
	10 ⁶ LBM/HR	HOT	COLD	SH	CT	10 ⁶ BTU/HR	10 ⁶ BTU/HR	10 ⁶ LBM/HR	10 ⁴ GPM
A	2.5317	611.77	559.79	1189.47	375.98	15.537	1991.87	38.314	10.346
B	2.3874	609.26	555.68	1189.43	376.28	14.578	1896.46	35.394	9.513
C	2.4937	612.11	559.46	1189.58	376.51	15.877	1996.48	37.924	10.236
D	2.4754	613.14	559.85	1189.49	375.48	14.642	1969.69	36.418	9.822
TOTAL							7854.49	148.842	39.918

POWER LEVEL FROM SECONDARY HEAT BALANCE IS 1301.9 MWt = 67.49 %

TOTAL NC FLOW IS 399179. GPM @ 358.97 DEG F

TECH SPEC PRIMARY FLOW = 396100. GPM

PERCENT TECH SPEC FLOW = 100.78 %

8.13 LARGE LOAD REDUCTION - TI/1/A/2650/10

Date(s) Performed: 4/12/85

I. PURPOSE

The objectives of this test were to:

- A. Demonstrate the ability of the plant with plant controls in automatic to withstand a large ($\approx 44\%$) step load decrease.
- B. Evaluate the response and interactions of control systems.
- C. Verify adequacy of AP/1/A/5500/03, Load Rejection, in establishing stable conditions.

II. METHOD

One of the two Main Generator Breakers (Breaker B) was opened to initiate the load reduction. Transient monitor data was taken via the plant computer and was analyzed to verify that the acceptance criteria were met.

III. RESULTS

The Reactor and Turbine ran back successfully (i.e. without incurring a trip). Safety injection was not initiated. Pressurizer and Steam Generator Safety relief valves did not open.

No manual intervention was required during the runback except that the feedwater control valve for Steam Generator C was momentarily placed in manual to stop a feedwater "swing". Plant conditions were stable approximately 12 minutes after the runback. Figures 8.13-1 through 8.13-6 show plant response to the transient. Following the runback, reactor power was further reduced to work on one of the feedwater pumps.

IV. CORRECTIVE ACTIONS

Feedwater swings, which occurred when feedwater control valves would oscillate closed and opened, occurred frequently during power escalation. Manual intervention was required to terminate the swing. Work requests were written and repair work was performed during the outage following the Unit Loss of Electrical Load Test. The test criteria was considered to have been met since feedwater flow was controlled automatically except for the momentary manual control of the Steam Generator C Feedwater Control Valve.

LARGE LOAD REDUCTION
REACTOR POWER RESPONSE TO LOAD REDUCTION

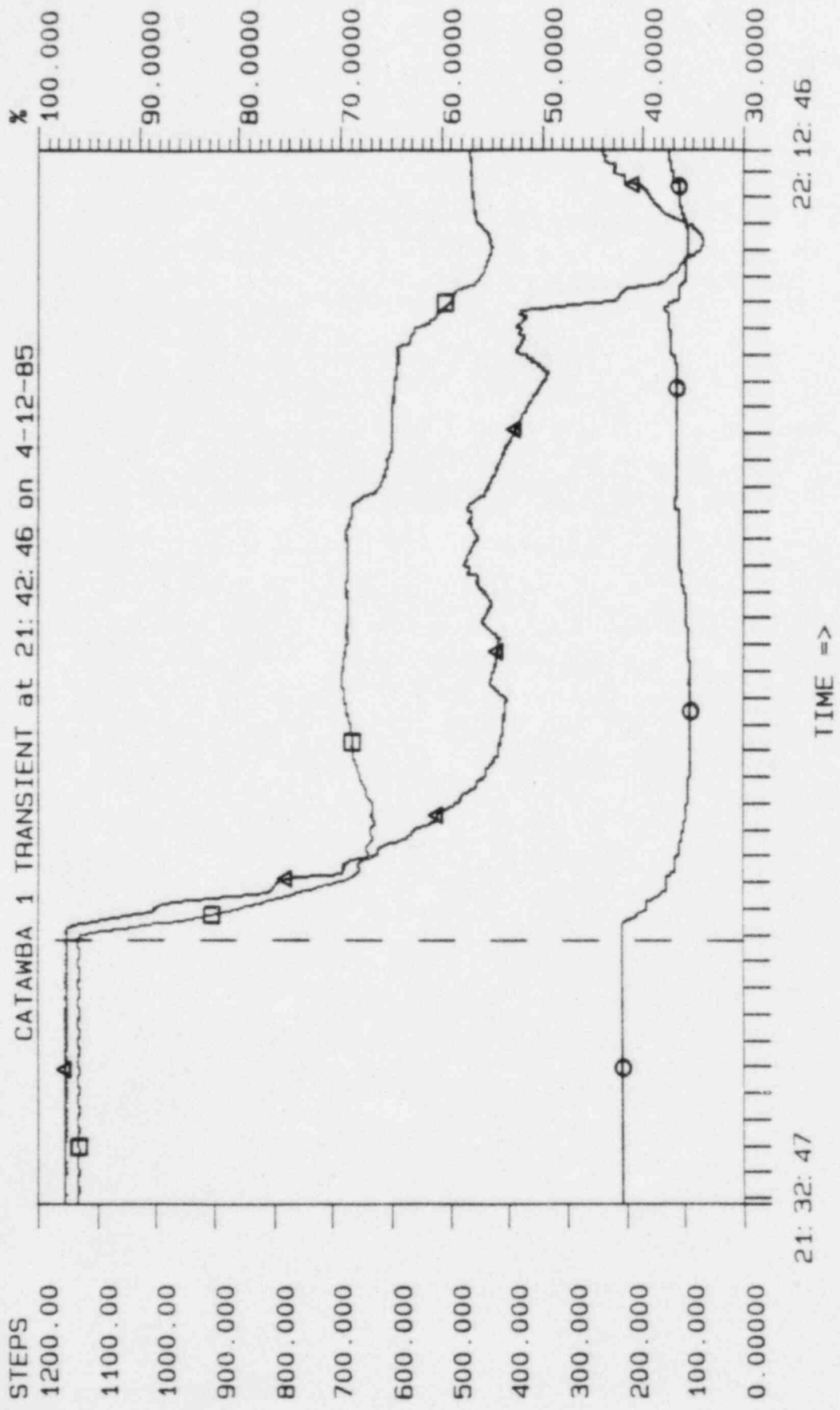


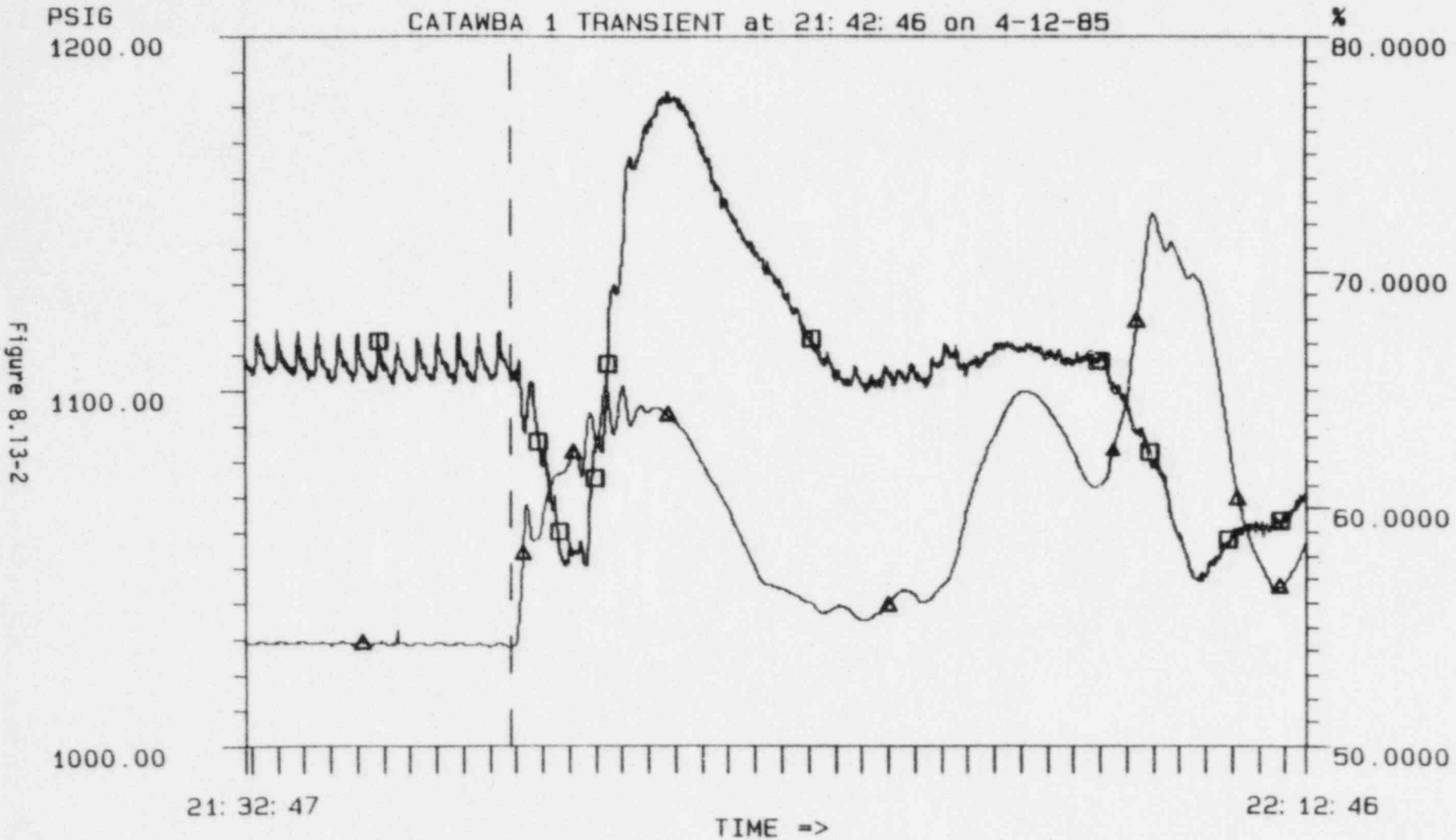
Figure 8.13-1

CATAWBA 1

>	<	<	@TRIP	MAX	MIN	PID	MESSAGE
	△	—	97.2000	97.4400	33.9000	A0758	POWER RANGE AVG LEVEL QUADRANT 1 (N-43)
	<	□	1145.43	1145.43	424.935	A0697	GENERATOR MEGAWATTS
	<	○	207.494	207.753	89.0122	A0749	CONTROL ROD BANK D DEMAND POSITION

Initiating point was D1427, GEN BKR B

LARGE LOAD REDUCTION
 STEAM GENERATOR PRESSURE AND LEVEL RESPONSE TO LOAD REDUCTION

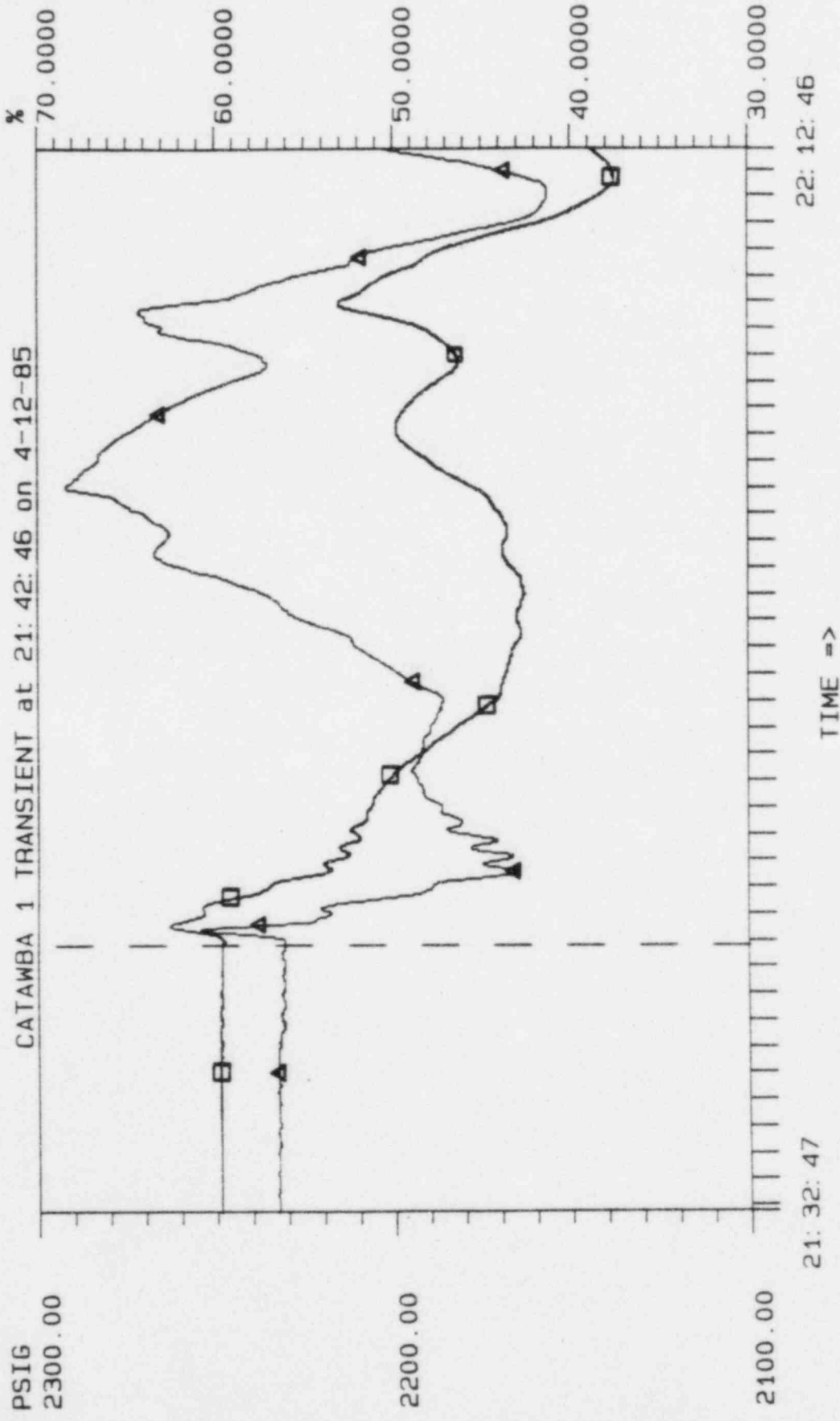


CATAWBA 1

	@TRIP	MAX	MIN	PID	MESSAGE
> Δ —	1027.65	1149.85	1027.65	A0723	S/G A STEAM PRESS CH #1
> \square —	65.7500	77.7000	56.9500	A0531	S/G A NARROW RANGE LEVEL CH 1

Initiating point was D1427, GEN BKR B

LARGE LOAD REDUCTION
PRESSURIZER PRESSURE AND LEVEL RESPONSE TO LOAD REDUCTION



<	@TRIP	MAX	MIN	PID	MESSAGE
>	2230.80	2291.60	2156.00	A0713	PZR PRESS CH 1
>	59.6000	62.6500	37.5000	A0707	PZR LEVEL CH I

Initiating point was D1427, GEN BKR B

Figure 8.13-3

LARGE LOAD REDUCTION
 REACTOR COOLANT TEMPERATURE RESPONSE TO LARGE LOAD REDUCTION

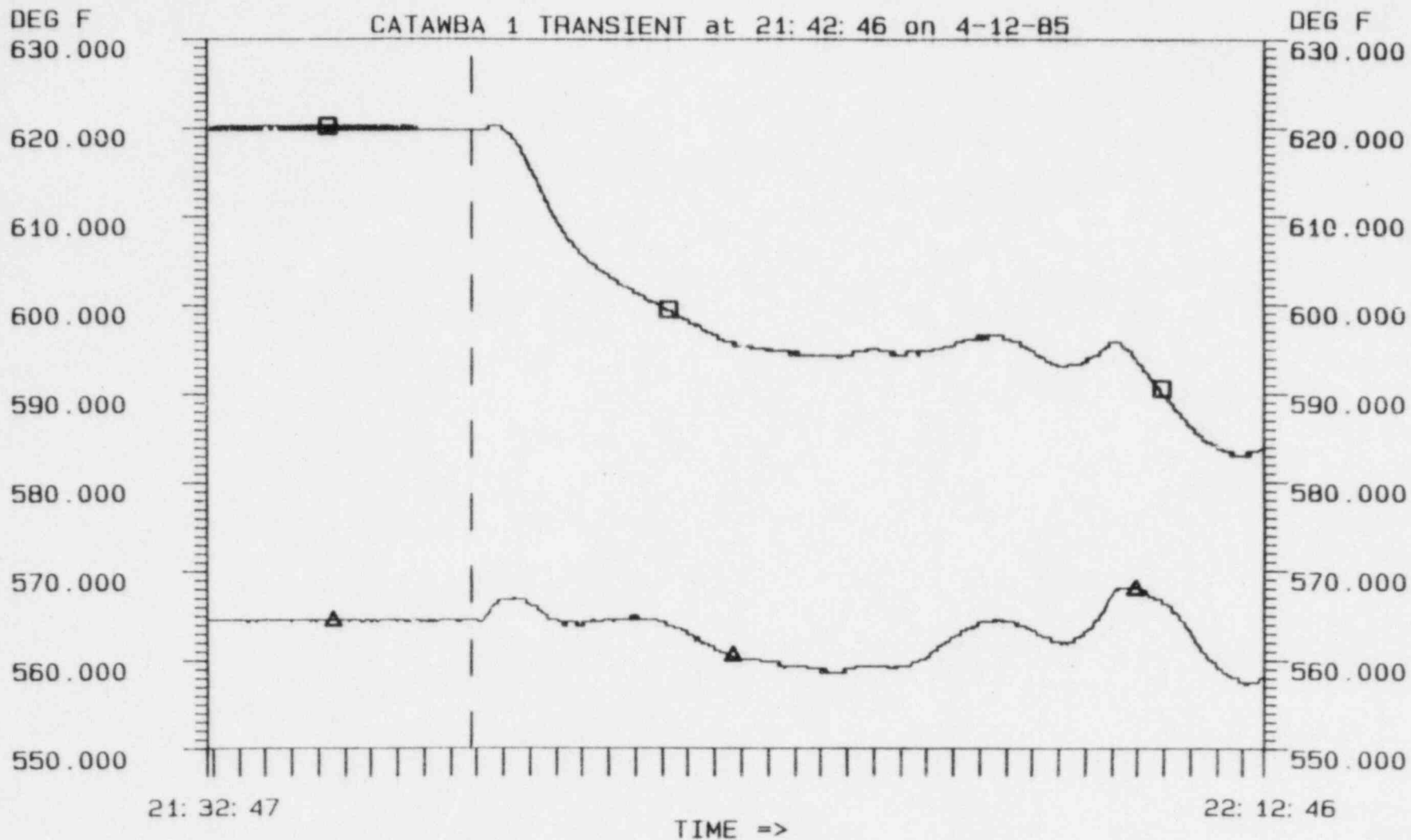


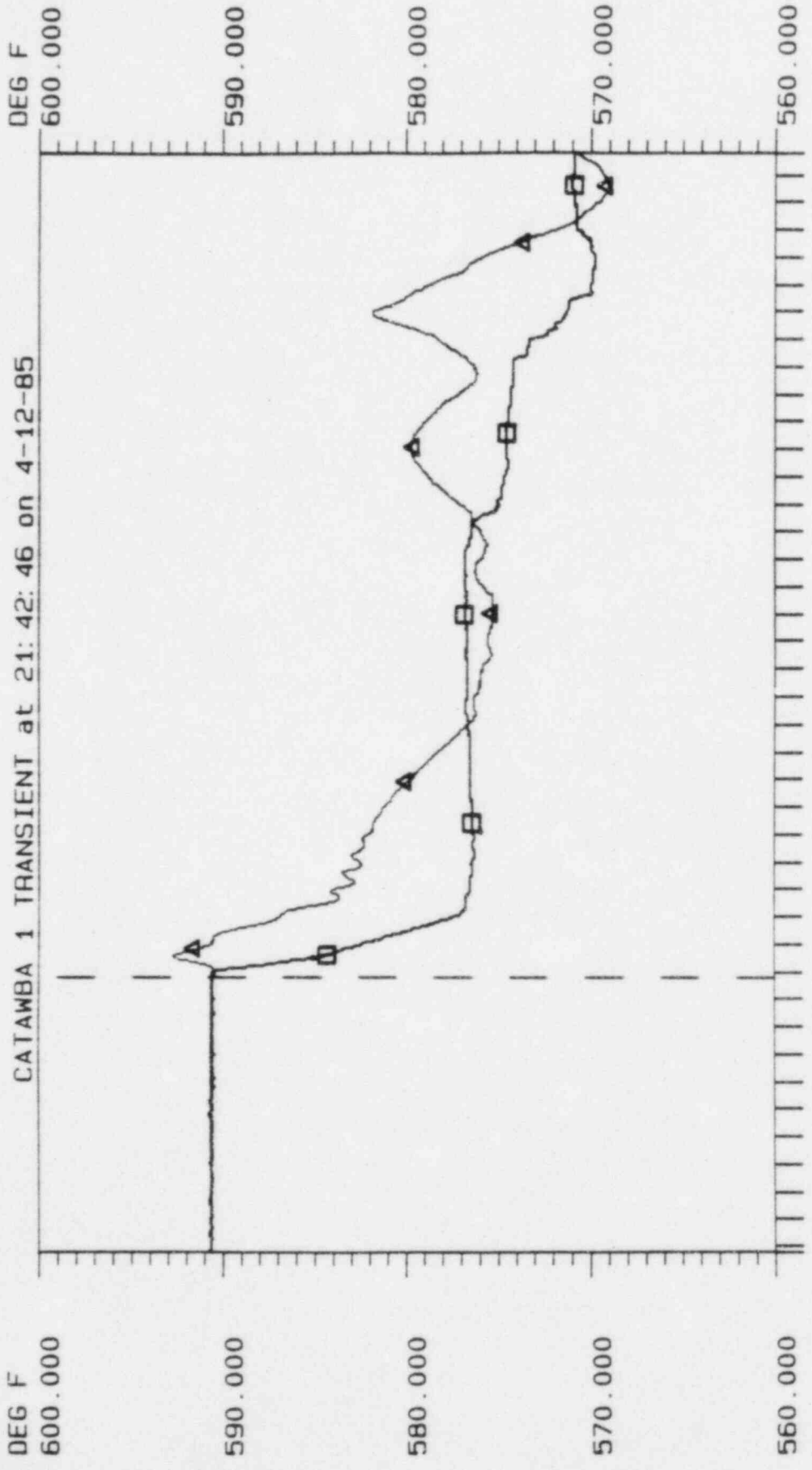
Figure 8.13-4

CATAWBA 1

	@TRIP	MAX	MIN	PID	MESSAGE
△	564.550	568.050	557.200	A0700 NC	LOOP A WIDE RANGE COLD LEG TEMP
□	619.850	620.200	583.100	A0668 NC	LOOP A WIDE RANGE HOT LEG TEMP

Initiating point was D1427, GEN BKR B

LARGE LOAD REDUCTION
 PROGRAM T-REF AND REACTOR COOLANT AVERAGE TEMPERATURE RESPONSE TO LOAD REDUCTION



21: 32: 47

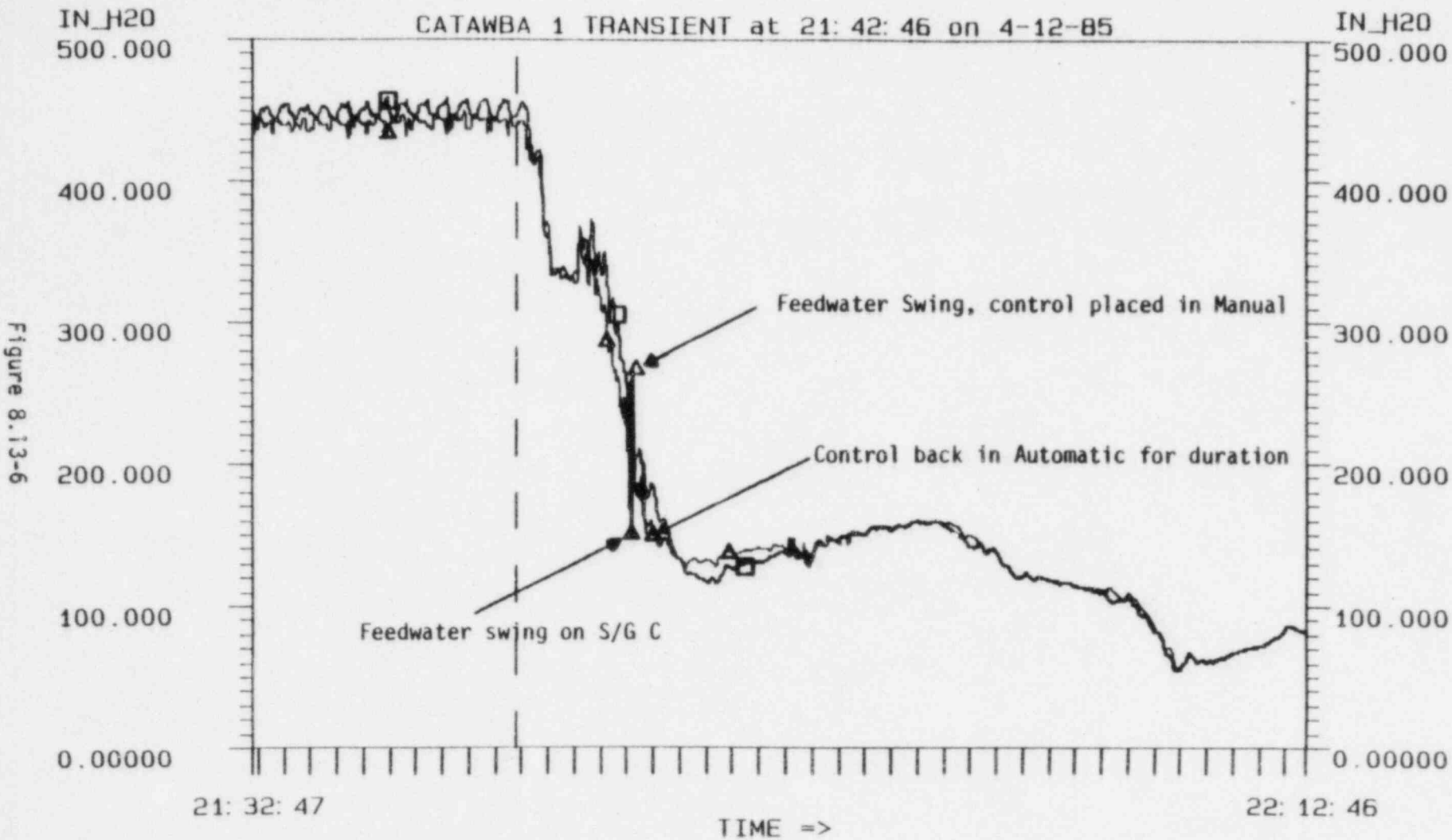
22: 12: 46

△	□	TRIP	MAX	MIN	PID	MESSAGE
△	□	590.450	592.700	568.950	A0701	NC LOOP HIGHEST T-AVG
△	□	590.600	590.600	569.750	A0725	T-REF

Initiating point was D1427, GEN BKR B

Figure 8.13-5

LARGE LOAD REDUCTION
 FEEDWATER FLOW RESPONSE TO LOAD REDUCTION



CATAWBA 1

	@TRIP	MAX	MIN	PID	MESSAGE
△	440.037	449.640	55.5619	A0635	S/G C FEEDWATER FLOW CH 1
□	445.450	459.118	54.0945	A0640	S/G A FEEDWATER FLOW CH 2

Initiating point was D1427, GEN BKR B

9.0 POWER ESCALATION TESTING - INSTRUMENTATION AND CONTROLS

The Instrumentation and Control section of power escalation testing concerns the initial setup and calibration of certain plant systems to ensure that they function properly in all modes. This testing included the following:

- Control Rod System at Power Test
- Nuclear Instrumentation Initial Calibration
- Pressurizer Pressure and Level Control System Test
- Tuning of the Steam Dump Control System

These tests are described on the following pages.

9.1 CONTROL ROD SYSTEM AT POWER TEST - TP/1/A/2600/10

Date(s) Performed: 3/2/85 - 3/3/85

I. PURPOSE

The Control Rod System at Power Test was performed to meet three objectives:

- A. To demonstrate the ability of the Rod Control System to automatically compensate for Reactor Coolant system temperature deviations.
- B. To verify that all setpoints associated with the Rod Control System are properly adjusted.
- C. To verify that no anomalous oscillations in reactor control occur as a result of automatic Rod Control System actuation.

II. METHOD

- A. To verify the capability of the Rod Control System to automatically control Reactor Coolant temperature at steady state conditions the reactor was first adjusted manually as necessary to bring T-AVE within $\pm 1^{\circ}\text{F}$ of T-REF. The Rod Control System was then placed in AUTO. Steady state data was trended to verify proper control.
- B. To verify the capability of the Rod Control System to automatically compensate for Reactor Coolant temperature deviations from programmed reference temperature T-AVE was first manually increased to a value $\approx 6^{\circ}\text{F}$ above T-REF. The Rod Control System was then placed in AUTO and its ability to bring T-AVE back to within $\pm 1.5\%$ of T-AVE and stabilize it there was monitored via the Transient Monitor Program on the Operator Aid Computer. T-AVE was then manually decreased to a value $\approx 6^{\circ}\text{F}$ below T-REF. Following placement of Rod Control in AUTO the Transient Monitor Program was again used to verify the capability of the system to raise T-AVE to a value within $\pm 1.5^{\circ}\text{F}$ of T-REF and stabilize it there.

III. RESULTS

All specified Acceptance Criteria for this test were met. Discussion of these individually follows:

- A. The Rod Control System was proven capable of controlling Reactor Coolant T-AVE to within $\pm 1.5^{\circ}\text{F}$ of Programmed T-REF while the reactor was at steady state conditions. This was verified by observing T-AVE for a period of 10 minutes with the Rod Control System in AUTO. No diverging oscillation were propagated as T-AVE was maintained with $\pm 0.5^{\circ}\text{F}$ of T-REF during this interval.

- B. The capability of the Rod Control System to correct a temperature error of + 6°F was verified per the test method previously detailed. The results of this test are shown on Figure 9.1-1.
- C. The capability of the Rod Control System to correct a temperature error of - 6°F was verified per the test method previously detailed. The results of this test are shown on Figure 9.1-2.

IV. CORRECTIVE ACTIONS

The Rod Control System was successfully tested without the need to adjust any of the setpoints associated with it.

CONTROL ROD SYSTEM AT POWER TEST
CORRECTION OF +6°F TEMPERATURE ERROR IN AUTO

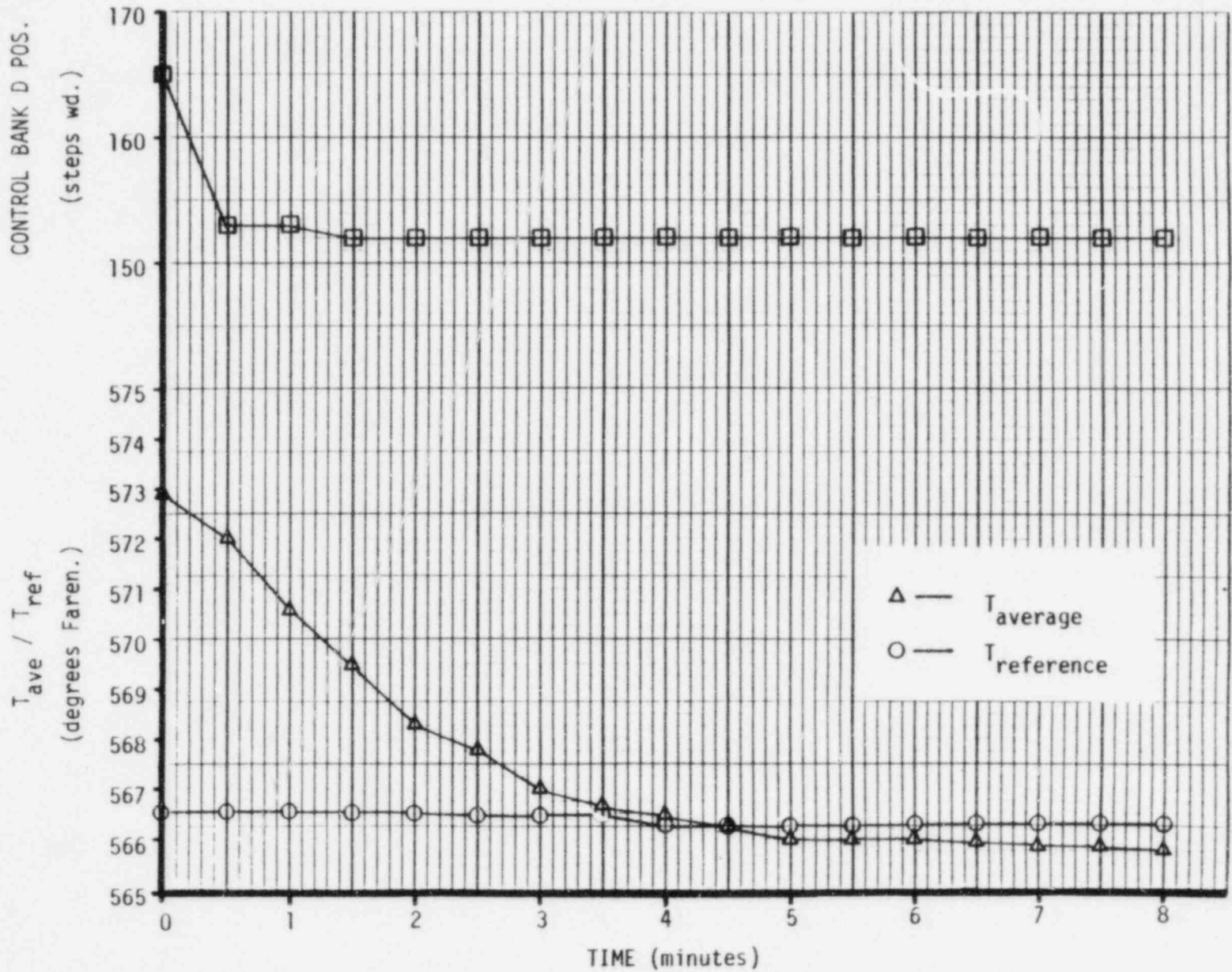


Figure 9.1-1

CATAMBA 1

CONTROL ROD SYSTEM AT POWER TEST
CORRECTION OF -6°F TEMPERATURE ERROR IN AUTO

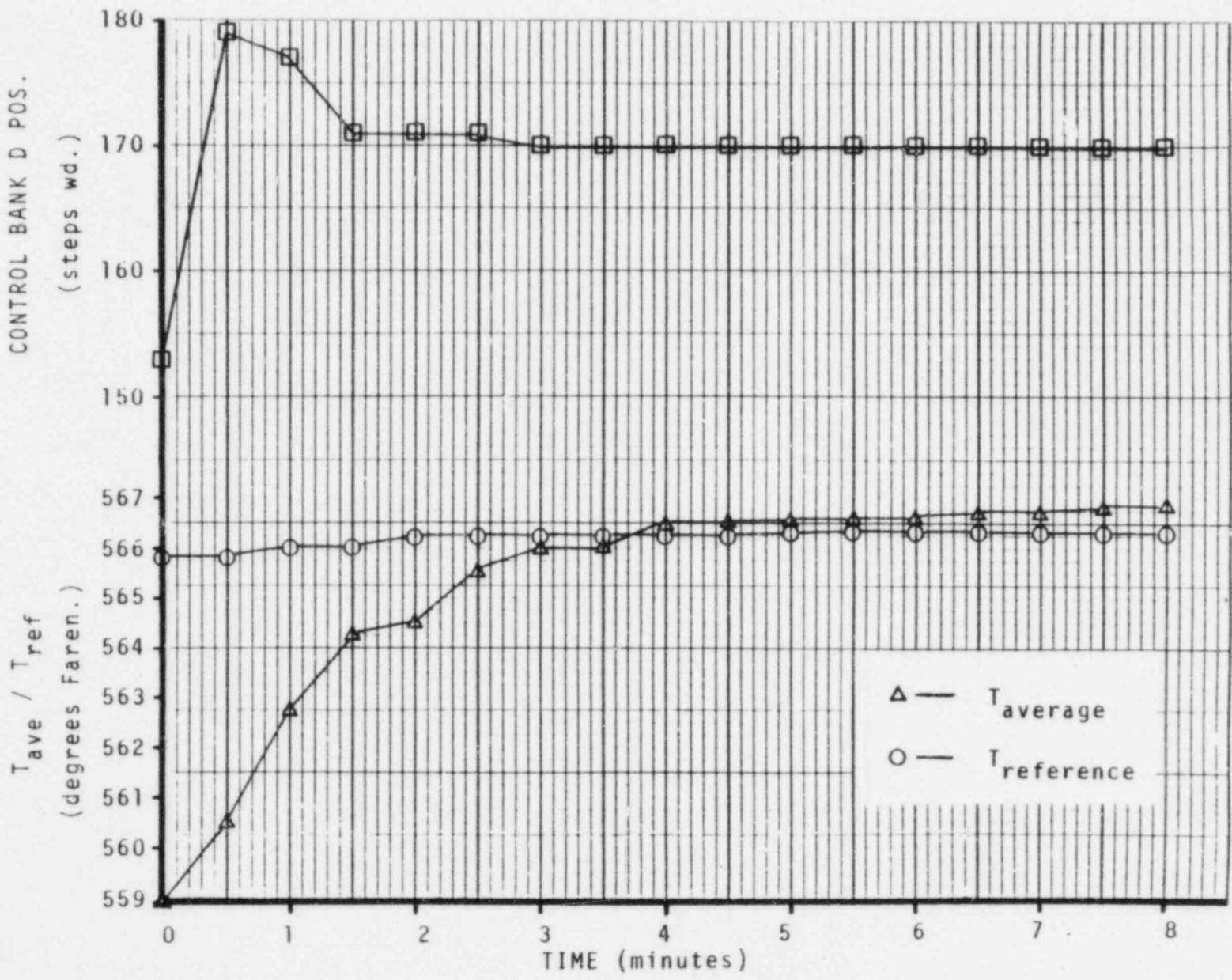


Figure 9.1-2

9.2 NUCLEAR INSTRUMENTATION INITIAL CALIBRATION - TP/1/A/2600/14

Date(s) Performed: 1/20/85, 1/21/85, 1/26/85, 2/5/85, 3/14/85,
3/22/85, 3/31/85, 4/8/85 - 4/9/85, 4/12/85

I. PURPOSE

The objectives of this test were to determine the linearity and uniformity of power range detector output; to calibrate the power range channels to match actual power levels; and to obtain overlap data between the intermediate and the power range channels.

II. METHOD

At various power levels, as specified in the controlling procedure, intermediate and power range detector currents were measured. Data was plotted and extrapolated to determine trip setpoints and full power currents. If necessary, trip setpoints were readjusted and power range detectors recalibrated to agree with actual power level. A least squares fit was performed on detector currents versus power to evaluate the linearity of the detectors.

III. RESULTS

All criteria for this test were met. The Power Range channels were adjusted to be within $\pm 2\%$ of actual power levels. Overlap between the intermediate and power range channels was observed to be more than one decade. Power Range channels gave uniform power indications, within $\pm 4\%$. Power Range detectors displayed linear output over the range of power operation with two minor exceptions. At 46.7% RTP, data for power range channels N42 and N44 fell slightly outside the $\pm 2\sigma$ band drawn around the straight line fit. Reactor power had just started to be calculated from a secondary heat balance, instead of the primary heat balance, which changed the value of Reactor Thermal Power, Best Estimate by approximately 1.5%.

Table 9.2-1 shows the Excore Detector data gathered during the test. Figure 9.2-1 shows the overlap that exists between the Intermediate Ranges and the Power Ranges. Figures 9.2-2 through 9.2-5 demonstrate the linearity of the Power Range channels.

IV. CORRECTIVE ACTIONS

Data was reviewed to determine that the data points for N42 and N44 would have been within the error band if the basis for reactor thermal power had not been changed. The effect of deleting these points from the data used to extrapolate to obtain Full power currents and trip setpoints was evaluated and determined to be negligible (less than 0.3% change).

NUCLEAR INSTRUMENTATION INITIAL CALIBRATION

EXCORE DETECTOR CURRENTS VERSUS POWER LEVEL

Best Est. Power, %	Intermediate Ranges Currents, 10 ⁻⁶ Amps		Power Ranges Currents, 10 ⁻⁶ Amps			
	N35	N36	N41	N42	N43	N44
0.55	1.308	1.697	4.039	3.987	4.024	4.040
2.80	5.191	7.585	14.32	16.84	15.17	14.85
5.86	21.70	26.60	51.26	57.88	51.43	48.37
8.71	34.00	41.50	82.87	90.77	83.42	78.75
19.55	76.20	87.90	207.7	221.3	206.9	195.5
25.42*	92.47	108.2	-	-	-	-
29.57	109.2	129.0	320.0	336.0	313.0	296.0
48.95	183.9	210.5	524.0	540.3	508.3	481.6
46.72	179.8	210.5	505.2	549.2	498.1	476.7
73.70	290.3	327.7	796.3	827.8	774.3	734.4
89.22	344.9	389.3	948.7	988.0	924.7	877.7
98.15	387.5	433.1	1041.9	1079.3	1013.1	961.0

*Data was taken at 25.42% for the intermediate range channels to ensure an accurate estimation of the intermediate range High Flux Setpoint (set to correspond to 25% power). This was not a testing plateau, so no Power Range currents were obtained.

Table 9.2-1

CATAMBA 1

NUCLEAR INSTRUMENTATION INITIAL CALIBRATION
OVERLAP BETWEEN THE INTERMEDIATE AND POWER RANGES

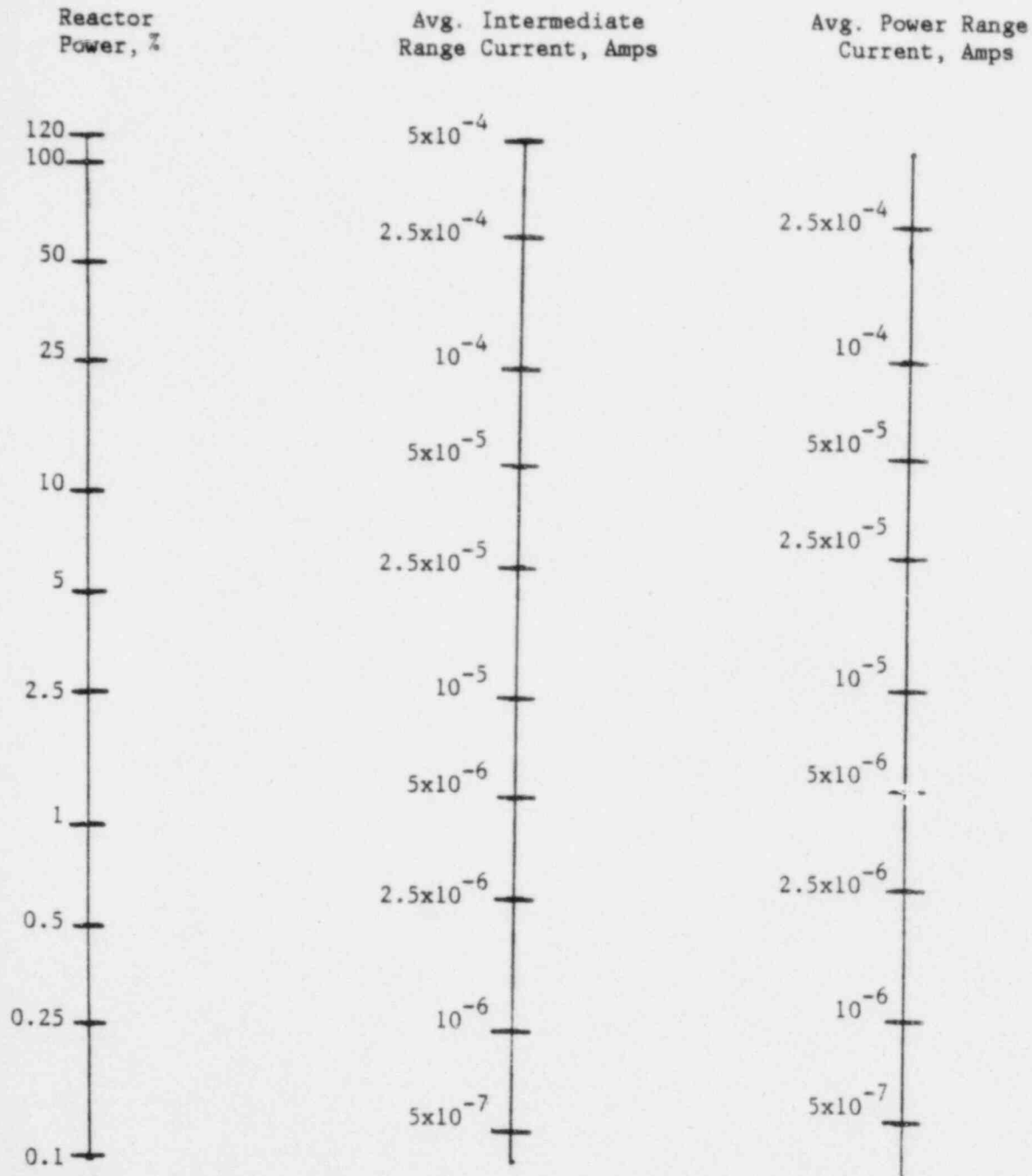


FIGURE 9.2-1

CATAWBA 1

NUCLEAR INSTRUMENTATION INITIAL CALIBRATION

POWER RANGE CURRENT VERSUS POWER LEVEL - N41

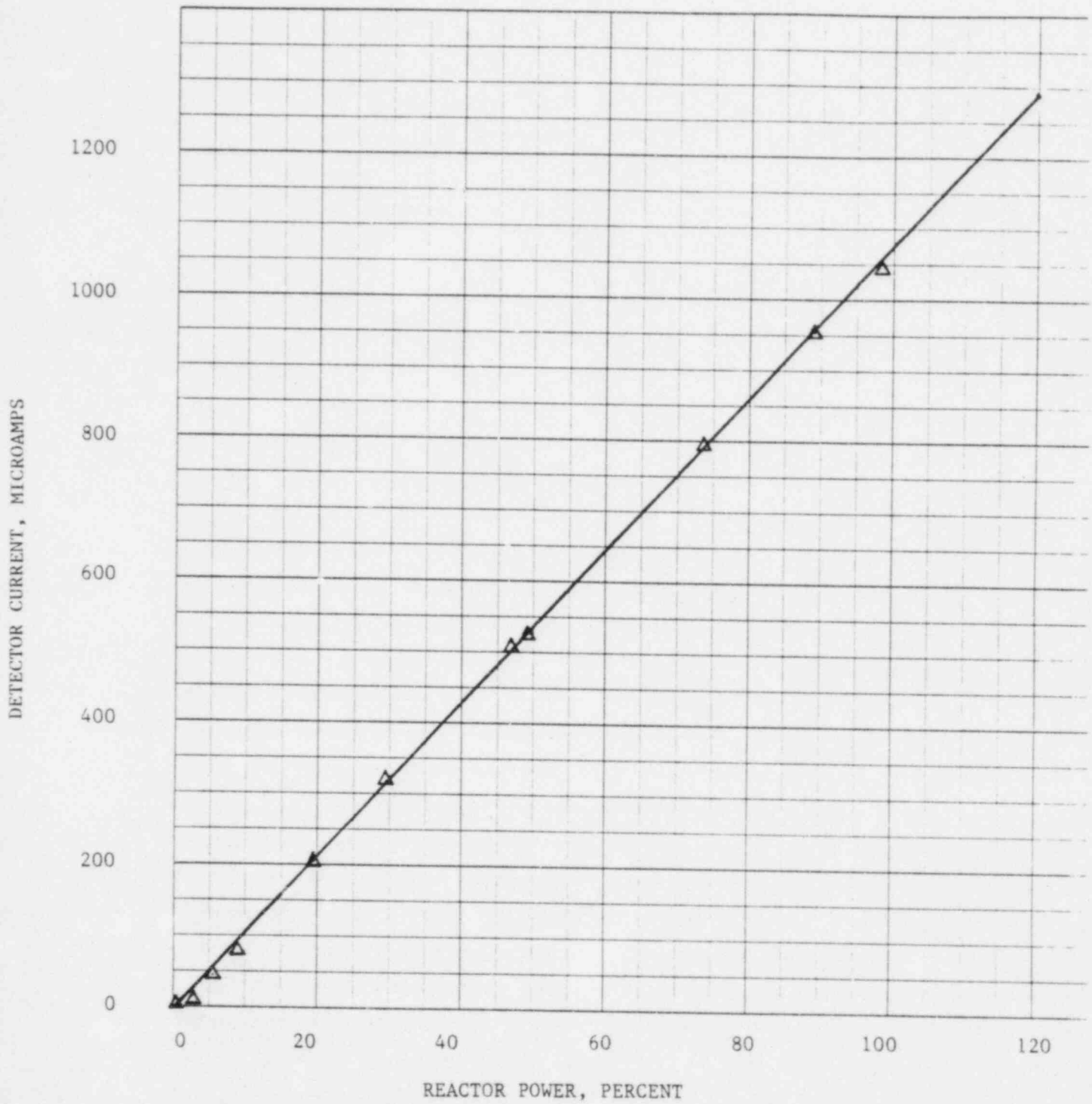


FIGURE 9.2-2

CATAWBA 1

NUCLEAR INSTRUMENTATION INITIAL CALIBRATION
POWER RANGE CURRENT VERSUS POWER LEVEL - N42

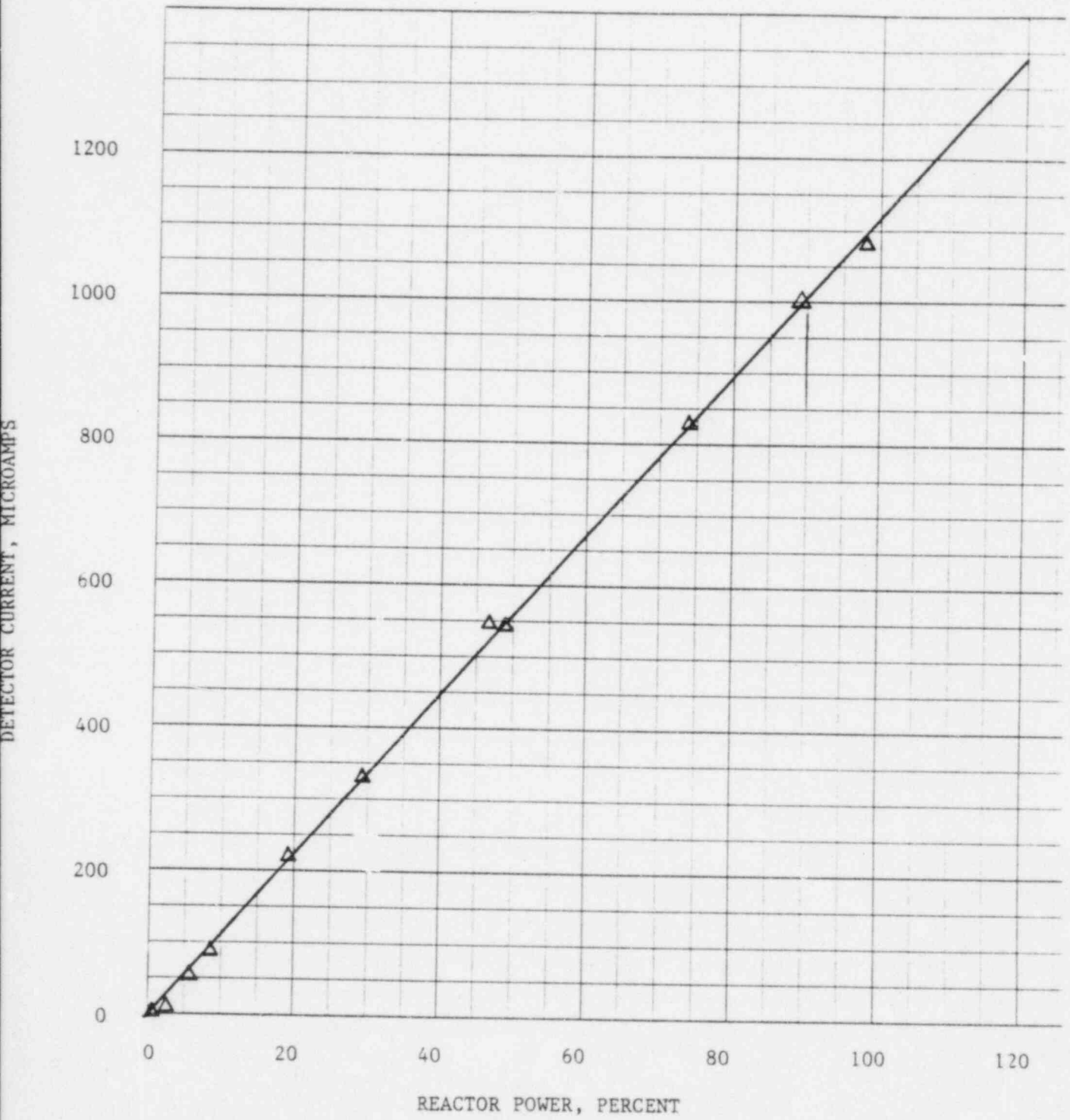


FIGURE 9.2-3

NUCLEAR INSTRUMENTATION INITIAL CALIBRATION
POWER RANGE CURRENT VERSUS POWER LEVEL - N43

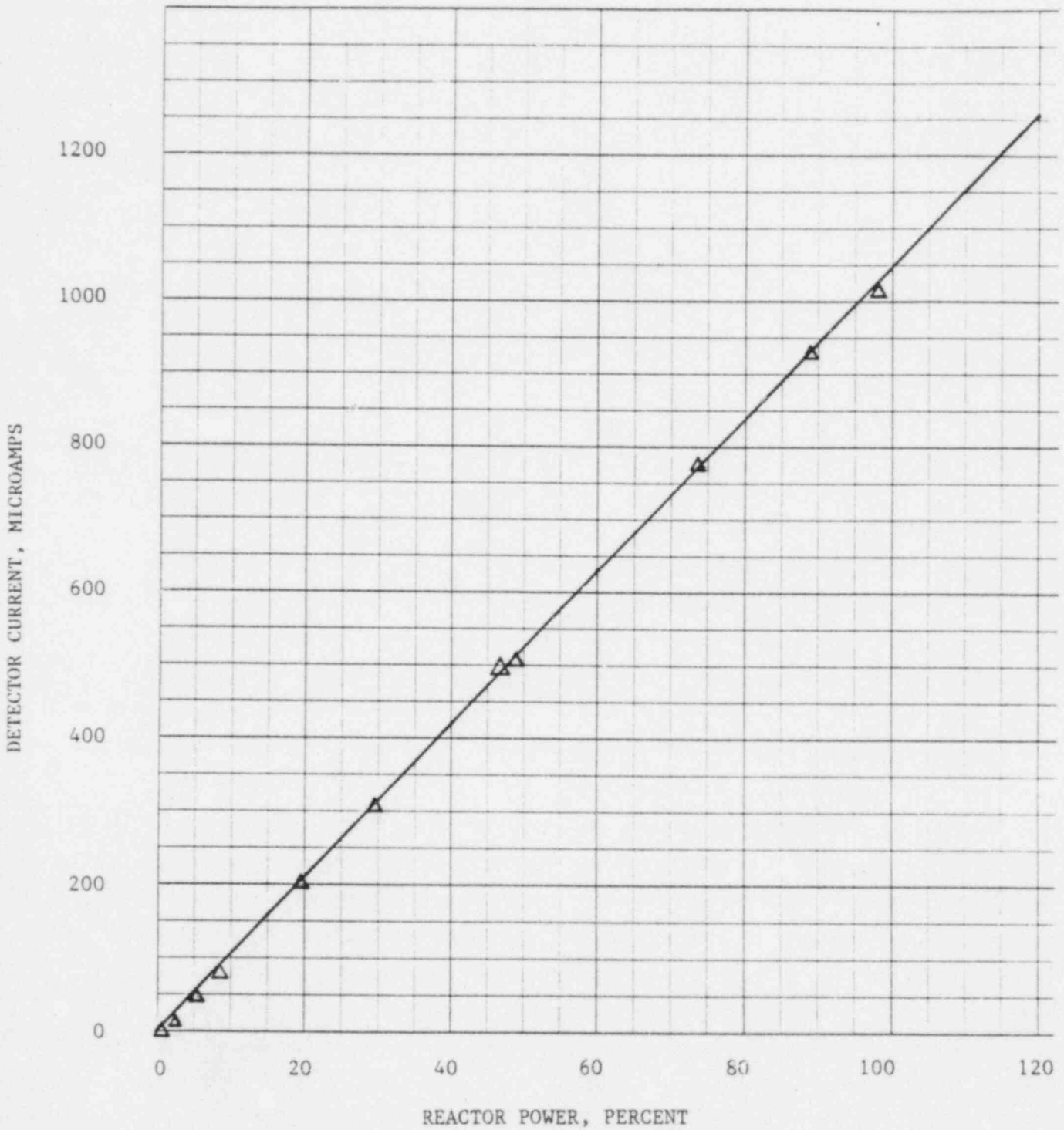


FIGURE 9.2-4

NUCLEAR INSTRUMENTATION INITIAL CALIBRATION
POWER RANGE CURRENT VERSUS POWER LEVEL - N44

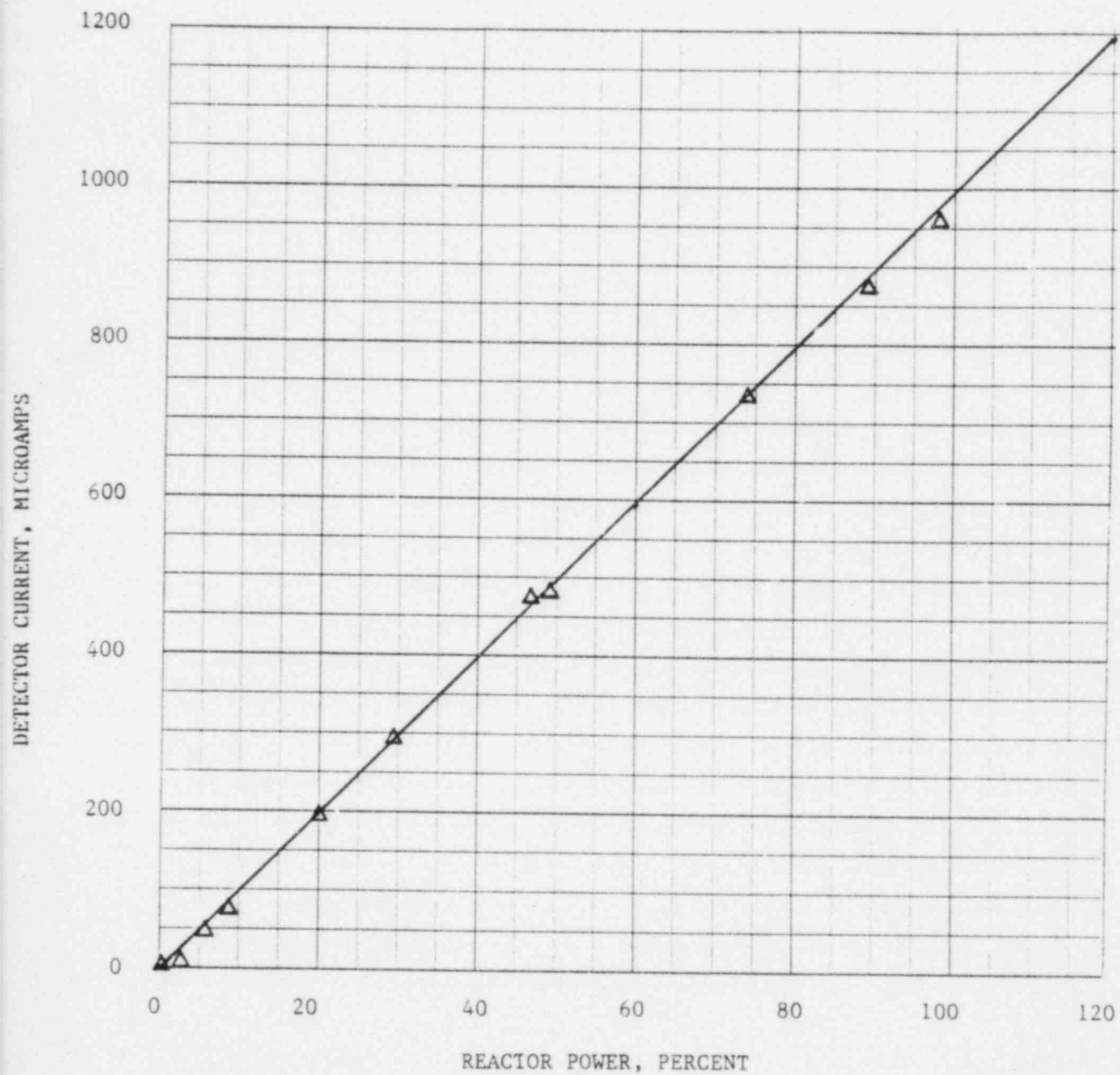


FIGURE 9.2-5

9.3 PRESSURIZER PRESSURE AND LEVEL CONTROL SYSTEM TEST - TP/1/A/2600/11

Date(s) Performed: 2/4/85, 3/01/85, 3/02/85

I. PURPOSE

The purpose of this test was to verify that Pressurizer Pressure and Level Controllers could maintain pressurizer conditions while in automatic and to verify that pressurizer heaters and spray actuate within the appropriate limits under the following conditions:

- A. Stable Conditions.
- B. Pressurizer Level perturbations with charging flow provided by the Positive Displacement Pump.
- C. Pressurizer Level perturbations with charging flow provided by a Centrifugal Charging Pump.
- D. Pressurizer Pressure perturbations.

II. METHOD

- A. With Pressurizer controls in automatic, Pressurizer parameters were monitored by the Operator Aid Computer for 30 minutes to verify the ability of Pressurizer Controllers to maintain stable conditions.
- B. With Pressurizer level controller in automatic and charging flow provided by the Positive Displacement Pump, perturbations in Pressurizer level were induced through manual control of pump speed and/or letdown flow.
- C. With Pressurizer level controller in automatic and charging flow provided by a Centrifugal Charging Pump, perturbations in Pressurizer level were induced through manual control of charging flow and/or letdown flow.
- D.
 - 1. With Pressurizer spray valves in automatic, perturbations in Pressurizer pressure were induced through manual control of the Pressurizer heaters.
 - 2. With Pressurizer heater controls in automatic, perturbations in Pressurizer pressure were induced through manual control of the Pressurizer spray valves.

III. RESULTS

All Acceptance Criteria were met by the test. The specific results were as follows:

- A. With all controls in automatic:
 - 1. Pressure remained within 18.1 psi of average (Acceptance Criteria \pm 30 psi of average).
 - 2. Level remained within 2.7% of average (Acceptance Criteria \pm 5% of average).
- B. The Pressurizer level controller automatically adjusted charging flow to maintain level within 5% of programmed level with charging flow supplied by the Positive Displacement Pump.
- C. The Pressurizer level controller automatically adjusted charging flow to maintain level within 5% of programmed level with charging flow supplied by a Centrifugal Charging Pump.
- D. The Pressurizer heaters and spray valves operated to maintain or return pressure to within 30 psi of the setpoint.

IV. CORRECTIVE ACTION

None Required.

9.4 TUNING OF THE STEAM DUMP CONTROL SYSTEM - IP/O/B/3222/85

Date(s) Performed - 1/19/85 - 1/20/85

I. PURPOSE

The Tuning of the Steam Dump Control System was performed to meet three objectives:

- A. To verify closed loop operation of the Plant Trip and Load Rejection modes of the Steam Dump Control System.
- B. To demonstrate the adequacy of the controller setpoints and to adjust them if necessary.
- C. To obtain settings for the Steam Header Pressure controller.

II. METHOD

- A. To verify the closed loop operation of the Plant Trip Controller, a reactor trip was simulated to the Steam Dump Control System and T-AVG was increased by withdrawing control rods. Data was taken to verify that the steam dump valves opened to control T-AVG to approximately 5°F above the no-load T-AVG.
- B. To verify the closed loop operation of the Load Rejection Controller, reactor power was first brought to approximately 3% with the Steam Header Pressure Controller in Manual. Steam Dump Control was switched to the T-AVG Mode and data was taken to verify that the steam dump valves opened to control T-AVG to approximately 5°F above the no-load T-AVG.
- C. To verify the closed loop operation of the Steam Header Pressure Controller, the Steam Dump Control System was placed in the STEAM PRESSURE MODE, and reactor power was increased to approximately 5%. Data was taken to verify that Main Steam Pressure was maintained at 1092 PSIG.

III. RESULTS

All acceptance criteria for the test were met.

- A. The Plant Trip controller controlled T-AVG to approximately 562°F satisfactorily.
- B. The Load Rejection controller controlled T-AVG to approximately 562.5°F satisfactorily.
- C. The Steam Header Pressure Controller controlled steam pressure to 1092 PSIG as power was increased to approximately 5%.

IV. CORRECTIVE ACTIONS

No corrective actions or Steam Dump controller adjustments were deemed necessary based on the results of the test.

10.0 POWER ESCALATION TESTING - MISCELLANEOUS

The miscellaneous section of power escalation testing contains startup tests that are not related to core physics or instrumentation and control of systems. This testing included the following:

- Loss of Control Room Test
- Biological Shield Survey
- Process and Effluent Radiation Monitor Test
- Support System Verification Test
- Feedwater Temperature Variation Test
- Steam Generator Water Hammer Test

These tests are described on the following pages.

10.1 LOSS OF CONTROL ROOM FUNCTIONAL TEST - TP/1/A/2650/03

Date(s) Performed: 1/21/85 - 2/01/85

I. PURPOSE

The Loss of Control Room Functional Test was performed for the following purposes:

- A. To demonstrate that the plant can be brought to hot standby conditions from a moderate power level (10-25%) using Auxiliary Shutdown Panel controls and following Operations procedure AP/1/A/5500/17 (Loss of Control Room).
- B. To demonstrate that the plant can be maintained at this condition for 30 minutes from the Auxiliary Shutdown Panels.
- C. To demonstrate that the plant can be brought to hot standby and maintained in that condition with the minimum shift requirements of Technical Specifications.
- D. To demonstrate that the Reactor Coolant System can be cooled down at least 50°F from a steady state hot standby condition while being operated from the Auxiliary Shutdown Panels.

II. METHOD

During Power Escalation Testing with the reactor at a power level of \approx 20% F.P. and the turbine/generator on line the test was initiated by the manual tripping of the reactor trip breakers from the trip breaker switchgear cabinets. Control of the unit was then immediately transferred from the Control Room to the Auxiliary Shutdown Panels (ASP's) per abnormal procedure AP/1/A/5500/17. The Operations personnel at the ASP's then brought the unit to a stable Hot Standby condition and maintained it for 30 minutes. The Reactor Coolant System was subsequently cooled down 50°F from the ASP's. Control of the unit was then returned to the Control Room.

III. RESULTS

All Acceptance Criteria stated in the purpose section were satisfied.

IV. COLLECTIVE ACTIONS

All plant systems and components functioned properly with the following exceptions:

- A. After control was transferred to local, it was discovered that both ICA40 and ICA44 controllers on IASPB were inoperable. This resulted in uncontrolled CA flow to steam generators C and D. Steam generator isolation valves had to be located and manually throttled.

- B. Following operation of Auxiliary Pressurizer Spray Valve INV37A, the valve stuck in the intermediate position. This resulted in limited control of pressurizer pressure. The Auxiliary Building NEO had to contact Health Physics, dress out and enter containment to manually isolate Pressurizer Spray.
- C. While proceeding to cooldown from the panels, it was discovered that the steam generator PORV's would not open from the AFWPTCP. The Service Building NEO was dispatched to the Dog House to manually open the PORV's as directed by the SRO.
- D. Approximately three quarters of the way through the cooldown, a Safety Injection Signal was received in the Control Room resulting from Pressurizer low pressure. All valves not blocked by transfer to the Shutdown Panels aligned to the required position. No pumps started due to the Safety Injection signal to the Load Sequencer being blocked with control transferred. This appears to have occurred due to two factors. First, the stuck Pressurizer Spray Valve was hindering the Operators in increasing Pressurizer pressure. Secondly, the Pressurizer pressure instrumentation on the Panels never indicated pressure below the Safety Injection setpoint of 1845 psig. The lowest pressure recorded from the panel was 1880 psig. With instrument uncertainty, this pressure could have dropped below the setpoint for Safety Injection without the knowledge of the operators.

None of these deficiencies had any effect on the satisfactory completion of the test.

10.2 BIOLOGICAL SHIELD SURVEY - TP/1/B/2200/01

Date(s) Performed: 6/22/84, 1/9/85, 2/4/85, 3/11/85,
3/30/85, 4/12/85

I. PURPOSE

The Biological Shield Survey was performed in order to determine the radiological integrity of the shielding constructed in the Containment building by measuring radiation levels in and around the Containment building at various power levels.

II. METHOD

All walls adjacent to the Reactor building as well as the various levels in Containment were surveyed to determine the amount of gamma and neutron radiation passing through them at Reactor power levels ranging from 0% to 100% F.P.

The major survey areas were:

- A. Reactor Building Pipe Chase
- B. Reactor Building Annulus
- C. Upper and Lower Ice Condenser
- D. Upper Containment Operating Deck
- E. Auxiliary Building Walls adjacent to Reactor Building 522' -638' Elevations
- F. Radiation Zones as described in FSAR

Each survey location consisted of three survey points of 2 feet, 4 feet, and 6 feet from the floor. Each survey location was \approx fifteen feet apart from the next one. Rooms adjacent to the Reactor Building, but separated by a wall had a survey location on each side of the separating wall. A contact gamma and neutron reading was taken at each survey point.

III. RESULTS

All Final Acceptance Criteria for this test were met. Radiation levels were found to be within the limits specified in FSAR Section 12.3. Specific results at the various power levels are detailed in the following sections.

- A. The 0% power survey was performed after initial criticality. Survey results showed only background radiation levels, except for a small increase in gamma readings in the lower ice condenser.

- B. The 30% power survey resulted in gamma readings increasing slightly in the Auxiliary building, upper ice condenser, upper containment operating deck, and the upper half of the annulus. 35% of the lower ice condenser had contact gamma readings greater than 100 mR/hr, with a high contact reading of 480 mR/hr. Highest neutron contact reading in the lower ice condenser was 2.25 mR/hr. The Reactor building pipe chase had a high general area gamma readings of 300 mR/hr between 176°-224°, due to a letdown line. One survey point in the lower annulus had a contact gamma reading of 200 mR/hr due to a letdown line being near the wall in the pipe chase. Neutrons were detectable in the lower annulus with contact readings as high as 2.25 mRem/hr.
- C. At 50% power readings remained unchanged in the upper ice condenser and the upper containment operating deck. Gamma readings increased slightly in the Auxiliary building and the upper half of the annulus. 43% of the lower ice condenser had contact gamma readings greater than 100 mR/hr, with a high contact reading of 900 mR/hr and a high general area reading of 500 mR/hr. The Reactor building pipe chase had increases in contact gamma readings between 128°-224°, due to a letdown line. It had a contact reading of 3.25 R/hr. The general area reading was between 300-700 mR/hr. Five survey points in the lower annulus had contact gamma readings of greater than 100 mR/hr, with a high contact reading of 300 mR/hr and a high general area reading of 100 mR/hr. The high annulus readings were due to a letdown line on the other side of the wall. Neutron levels in the lower annulus remained consistent with the 30% power survey.
- D. The 75% power readings remained unchanged in the upper ice condenser, the upper containment operating deck, and the Auxiliary building. Gamma readings increased slightly in the upper half of the annulus. Surveying of the lower ice condenser was discontinued when the general area exceeded 500 mR/hr. Nine survey points in the lower annulus had contact gamma reading of greater than 100 mR/hr, with a high contact gamma reading of 300 mR/hr and a high general area reading of 100 mR/hr. The highest neutron level found in the lower annulus was 3 mRem/hr at contact. Gamma readings in the Reactor building pipe chase continued to increase between 128°-224°, due to a letdown line having a 6R/hr contact reading.
- E. The 100% power survey resulted in readings remaining unchanged in the upper ice condenser, the upper containment operating deck, and the Auxiliary building and the upper annulus. Gamma readings in the Reactor Building pipe chase continued to increase between 128°-224°, due to a letdown line having an 8 R/hr contact gamma reading. Twenty-four survey points in the lower annulus had contact gamma readings of greater than 100 mR/hr, with a high contact reading 500 mR/hr and a high general area reading of 175 mR/hr. The highest neutron level found in the lower annulus was 3 mRem/hr. Neutrons detected were between 75°-105°, due to the in-core instrumentation tunnel, and also near penetrations in the shield wall.

Survey results did not meet the initial Acceptance Criteria for platforms 1 to 5 in the annulus. This necessitated a change in the FSAR to redesignate these areas from radiation zone level VI to radiation zone level VII. This allowed the Acceptance Criteria to be met.

IV. CORRECTIVE ACTIONS

Surveys near various electrical and mechanical penetrations showed undesirably high radiation levels on the outside of the Containment building wall. Although all Acceptance Criteria were met, Design action has been taken to reduce these problems. Also, administrative changes have been made to provide extra radiological controls in areas adjacent to the Incore Instrumentation system during flux mapping.

10.3 PROCESS AND EFFLUENT RADIATION MONITOR TEST - TP/1/B/2600/15

Date(s) Performed: 1/9/85 - 1/17/85, 3/12/85 - 3/16/85,
4/12/85 - 4/18/85

I. PURPOSE

The Process and Effluent Radiation Monitor Test was performed for the purpose of verifying proper operation of the station's process and effluent radiation monitors under actual operating/discharge conditions. The test was performed at power levels of 0% F.P., 50% F.P., and 100% F.P. as required by the Final Safety Analysis Report.

II. METHOD

The test method involved the following actions, performed at each of the three test plateaus:

- A. Obtaining background count rates, instrument time constants, and calculating minimum monitor sensitivities.
- B. Simultaneously sampling process or effluent streams and scaling corresponding count rates.
- C. Performing radioisotopic analyses on samples and converting the results to monitor count rate equivalent by application of appropriate correlation constant (obtained from initial vendor calibration documents).
- D. Verifying that monitor vs sample analysis percent differential meets the $\pm 20\%$ acceptance criterion.

III. RESULTS

A. Zero Percent Test Plateau

The zero percent (0%) test was completed with one monitor (1EMF48-Reactor Coolant) meeting acceptance criteria out of 26 monitors tested. Twenty-five tested monitors had insufficient process system activity. Four monitors were not tested. Of those not tested, three were inoperable (2EMF31, 2EMF52, 2EMF55B), and the fourth (EMF50) did not have prerequisite activity.

B. Fifty Percent Test Plateau

The fifty percent (50%) test was completed with the following four monitors meeting acceptance criteria out of 25 monitors tested:

1EMF39	-	Containment Noble Gas
1EMF46A	-	Component Cooling/Train A
1EMF46B	-	Component Cooling/Train B
1EMF49	-	Liquid Waste Discharge (Effluent)

One monitor (1EMF48) failed to meet acceptance criteria due to a high sample chamber contamination. A flush procedure and work request was written for test implementation at 100% reactor power.

Twenty tested monitors had insufficient process system activity. Four monitors were not tested. Of those not tested, two were inoperable (1EMF34 and 2EMF31), and two were impractical to test (EMF50 and 1EMF52).

EMF50 (Waste Gas) was impractical to test since an unnecessary gas release would have been required that would have resulted in dose to the public. 1EMF52 (Floor Drain) was impractical to test due to a very low water source (sump level depleted).

C. One Hundred Percent Test Plateau

The one hundred percent (100%) test was completed with the following six monitors meeting acceptance criteria out of 27 monitors tested:

1EMF39	-	Containment Noble Gas
1EMF46A	-	Component Cooling/Train A
1EMF46B	-	Component Cooling/Train B
EMF47	-	Boron Recycle Evaporator
1EMF48	-	Reactor Coolant
EMF49	-	Liquid Waste Discharge (Effluent)

Twenty-one tested monitors had insufficient process system activity. One monitor (EMF50 - Waste Gas) was not tested since an unnecessary gas release would have been required that would have resulted in dose to the public.

2EMF31 (Unit 2 Turbine Bldg Sump) was deleted from test as not required for Unit One Start-Up/Operation.

The Unit One process and effluent radiation monitors were verified to be performing in accordance with their design objectives and should be considered acceptable for continued operation. All monitors that had prerequisite system activities met acceptance criteria when backgrounds were established and responded predictably when considering vendor calibration documentation.

Correlation studies on 1EMF38 (Containment Particulate) and 1EMF40 (Containment Iodine) should be conducted with regard to sample line losses due to particulate and iodine plateout and deposition. After baseline information is obtained the representative correlation values should be changed to reflect line losses if needed.

The correlation factors for all other monitors will be based on the respective vendor Primary Calibration Reports.

All monitors which had insufficient process system activity will be tested annually per HP/O/B/1000/03, EMF correlation. At such time that sufficient activity is present for these monitors they will be tested. It is doubtful that this situation will ever exist since these monitors are on secondary plant systems.

IV. CORRECTIVE ACTIONS

Testing of the Process and Effluent Monitoring System was performed over the course of the power escalation testing program. The testing of several monitors was deferred to higher power levels when adequate activities were present.

One monitor 1EMF48 had contamination in its sample chamber when it was tested at 50% F.P. This necessitated a work request to perform a flush procedure. The monitor was retested successfully at 100% F.P.

The requirements to test EMF50 has been transferred to TP/2/B/2600/15, to be done during Unit 2 power escalation when adequate activity will be present to test it. A change to the FSAR was made accordingly.

10.4 FEEDWATER TEMPERATURE VARIATION TEST - TP/1/A/2650/11

Date(s) Performed: 4/10/85 - 4/11/85

I. PURPOSE

The purpose of this test was to demonstrate the unit's ability to sustain a reduction in feedwater temperature due to inadvertent opening of the bypass valve around the high pressure feedwater heaters. Also, system response during the transient was evaluated to determine if any control system changes were required.

II. METHOD

With the unit at 90% Rated Thermal Power and all major control systems in automatic the High Pressure Heaters A & B Bypass Valve (1CF-22) was slowly opened. Plant parameters were monitored via the OAC during the ensuing transient. The data obtained was analyzed to verify that the Acceptance Criteria had been met. This sequence was then repeated with the bypass valve being opened as rapidly as possible.

III. RESULTS

All Acceptance Criteria were met during both test runs as follows:

- A. No turbine or reactor trip occurred as a result of NC system transients.
- B. Safety injection did not occur.
- C. Pressurizer or main steam line safety valves did not lift.
- D. Following the transient stable plant conditions were established as follows:

<u>Parameter</u>	<u>Required Stability Criteria*</u>	<u>Measured Stability Deviation Absolute Value</u>	
		<u>Run #1</u>	<u>Run #2</u>
		1. Reactor Power	± 1%
2. NC T _{ave}	± 2°F	0.190°F	.082°F
3. CF Temperature	± 2°F	0.687°F	.908°F
4. CF Flow	± 4%	2.683%	1.836% (Loop A)
		2.491%	2.079% (Loop B)
		3.265%	1.605% (Loop C)
		2.151%	2.143% (Loop D)
5. S/G Level	± 2%	1.232%	.964% (Loop A)
		1.501%	1.166% (Loop B)
		1.454%	.513% (Loop C)
		1.518%	.874% (Loop D)

<u>Parameter</u>	<u>Required Stability Criteria*</u>	<u>Measured</u>	
		<u>Stability Deviation</u>	
		<u>Absolute Value</u>	
		<u>Run #1</u>	<u>Run #2</u>
6. Pzr Pressure	2235 ± 25 psig	2239 psig	2246 psig (Minimum)
		2242 psig	2252 psig (Maximum)
7. Pzr Level	± 2%	.506%	.157%

*Maximum Permissible Deviation From Average Value Over a 5-minute Period.

- E. Feedwater temperature did not decrease more than 48°F during the transient compared to the value recorded just prior to the transient. The actual change recorded was 47.22°F in run #1 and 44.6°F in run #2.

During run #1, the bypass valve was opened over a period of 25 minutes whereas the opening time for Run #2 was 2½ minutes. The resulting decrease in feedwater temperature enhanced heat transfer across the Steam Generators such that primary side power automatically increased. This feedback caused nuclear power to increase 4½% during each of the test runs. These results are all conservative compared to the Westinghouse safety analysis for this event. Figures 10.4-1 and 10.4-2 show the Reactor power and feedwater temperature trend for each test run.

IV. CORRECTIVE ACTION

No corrective action was required.

FEEDWATER TEMPERATURE VARIATION TEST
 POWER AND CF TEMPERATURE TREND

SLOW VALVE OPENING

CATAWBA 1 TRANSIENT at 16:10:04 on 4-10-85

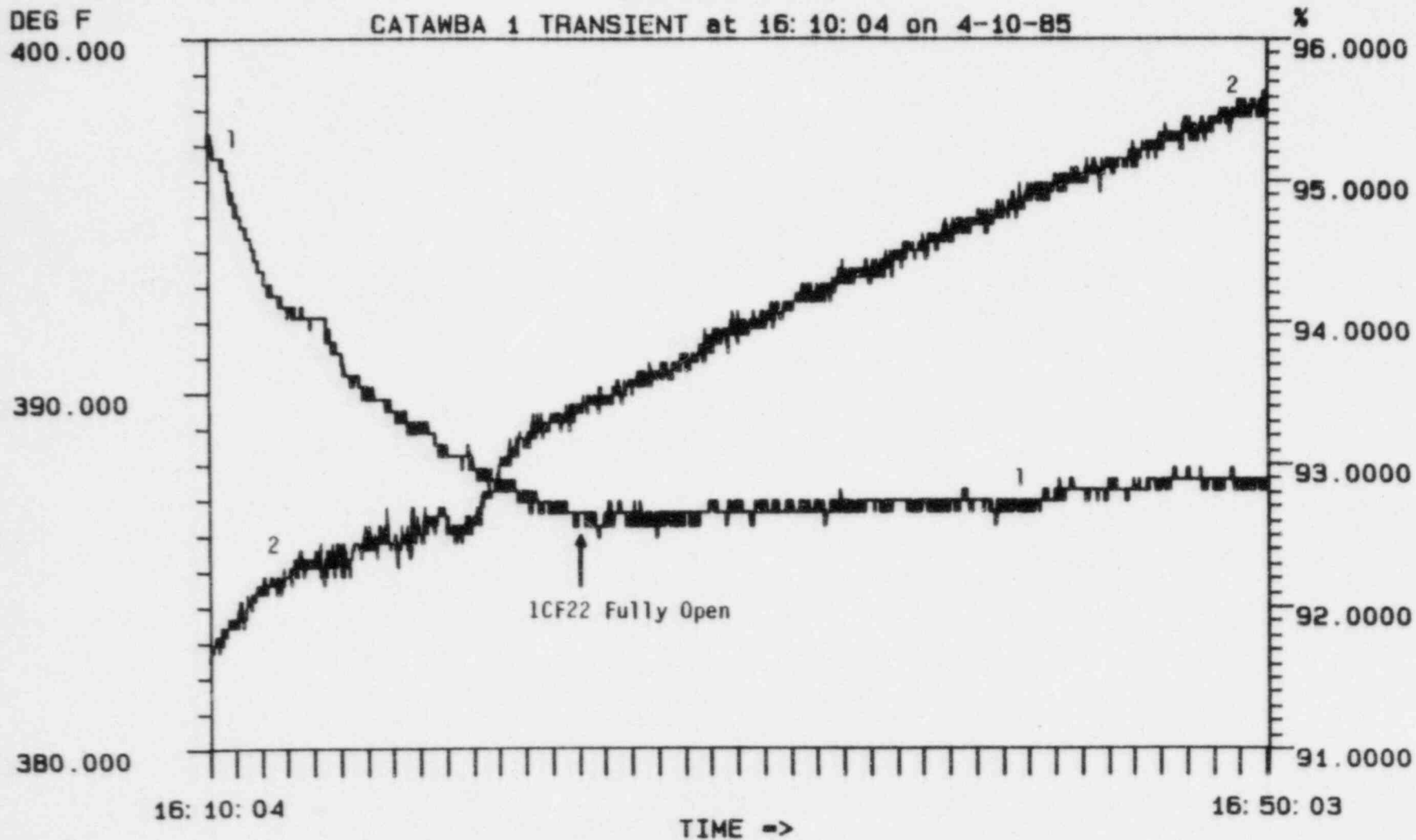


FIGURE 10.4-1

CATAWBA 1

		MAX	MIN	PID	MESSAGE
>	1	397.293	385.954	A0147	S/G A CF INLET TEMP
>	2	95.6400	91.6200	A0758	POWER RANGE AVG LEVEL QUADRANT 1 (N-43)

< Manual Freeze >

FEEDWATER TEMPERATURE VARIATION TEST
POWER AND CF TEMPERATURE TREND

FAST VALVE OPENING

CATAWBA 1 TRANSIENT at 04:15:53 on 4-11-85

ICF22 Fully Open

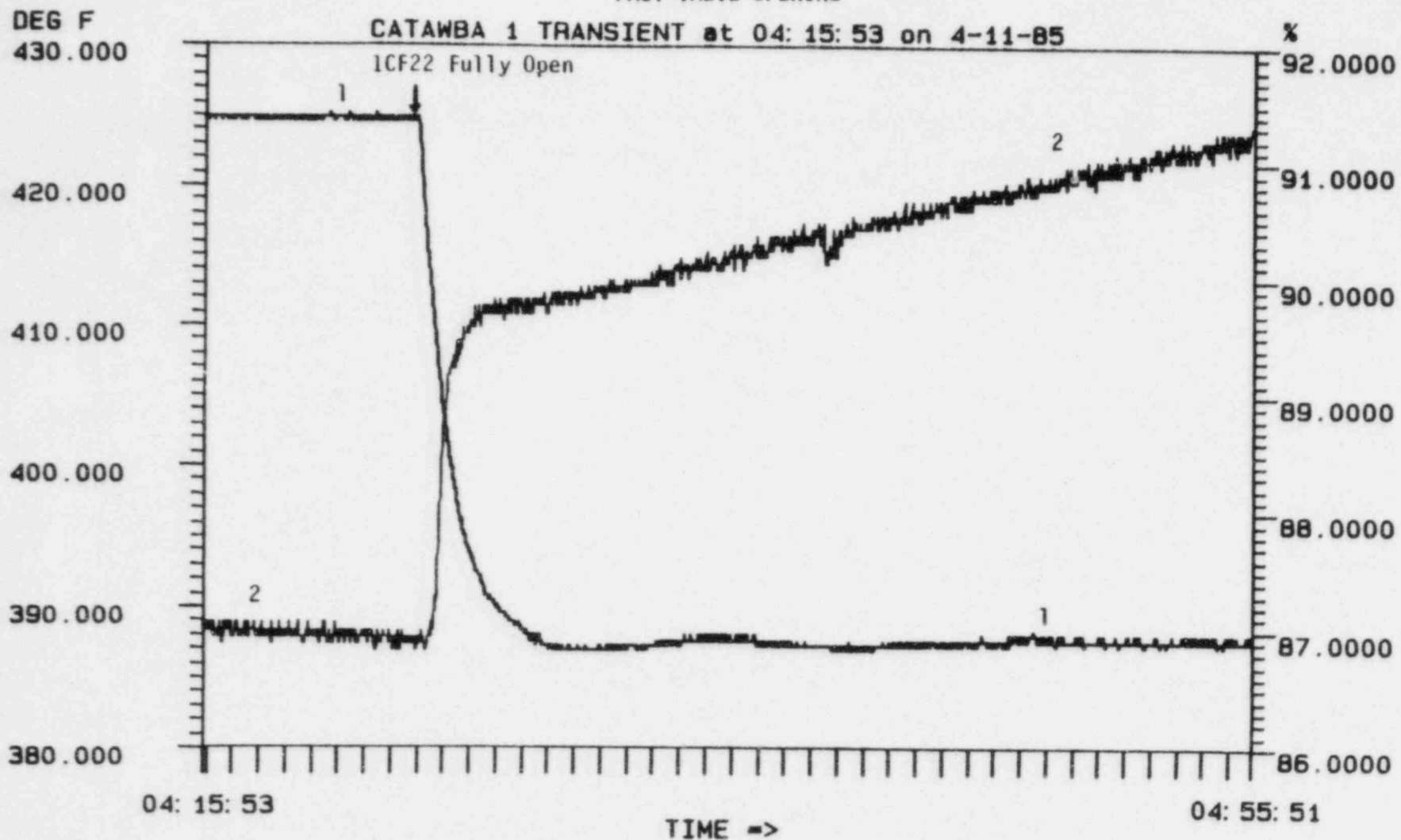


FIGURE 10.4-2

CATAWBA 1

		MAX	MIN	PID	MESSAGE
<	1	425.142	386.926	A0147	S/G A CF INLET TEMP
>	2	91.3200	86.8200	A0758	POWER RANGE AVG LEVEL QUADRANT 1 (N-43)

< Manual Freeze >

10.5 SUPPORT SYSTEMS VERIFICATION TEST - TP/1/A/2650/14

Date(s) Performed: 03/10/85, 04/17/85

I. PURPOSE

The Support Systems Verification Test was performed on the following days:

<u>DATE</u>	<u>ACTIVITY</u>
03/10/85	Support Systems Verification Test at 50% Full Power
04/17/85	Support Systems Verification Test at 100% Full Power

The objective of the Support Systems Verification Test was to verify that air temperature within rooms in the Auxiliary Building which contain Engineered Safety Features (ESF) pumps and motors are maintained within design limits during power operation by normal operation of the respective cooling systems serving those areas.

II. METHOD

Temperature measurements were taken from a combination of Operator Aid Computer points and hand held digital thermometers for the following rooms that contain ESF pumps and motors:

Room Numbers

- 102 - Containment Spray (NS) Pump Motor A
- 103 - Containment Spray (NS) Pump Motor B
- 104 - Residual Heat Removal (ND) Pump Motor A
- 105 - Residual Heat Removal (ND) Pump Motor B
- 230 - Chemical and Volume Control System (NV)
Centrifugal Charging Pump A
- 231 - Chemical and Volume Control System (NV)
Centrifugal Charging Pump B
- 234 - Safety Injection (NI) Pump B
- 235 - Safety Injection (NI) Pump A
- 255 - Auxiliary Feedwater (CA) Pump B
- 256 - Auxiliary Feedwater (CA) Pump A
- 300 - Component Cooling (KC) Pumps A1 and A2
- 300 - Component Cooling (KC) Pumps B1 and B2

These temperature measurements were taken at the following two power levels:

- A. 50% of Full Power
- B. 100% of Full Power

These temperature measurements were then compared to the Temperature Design Limits for the rooms.

III. RESULTS

- A. The temperature measurements at 50% of Full Power were as follows for each room containing ESF pumps and motors:

<u>Room #</u>	<u>Measured Value (°F)</u>	<u>Design Limits (°F)</u>
102	77.3	61 < T < 104
103	78.0	61 < T < 104
104	82.2	61 < T < 104
105	85.6	61 < T < 104
230	83.7	60 < T < 105
231	86.9	60 < T < 105
234	86.7	60 < T < 105
235	86.7	60 < T < 105
255	94.3	60 < T < 105
256	92.1	60 < T < 105
300 (Between KC Pumps 1B1 and 1B2)	80.7	60 < T < 105
300 (Between KC Pumps 1A1 and 1A2)	87.3	60 < T < 105

All acceptance criteria for the Support Systems Verification Test for 50% of Full Power were satisfied.

- B. The temperature measurements at 100% of Full Power were as follows for each room containing ESF pumps and motors:

<u>Room #</u>	<u>Measured Value (°F)</u>	<u>Design Limits (°F)</u>
102	82.9	61 < T < 104
103	83.8	61 < T < 104
104	84.7	61 < T < 104
105	83.7	61 < T < 104
230	93.2	60 < T < 105
231	90.1	60 < T < 105
234	89.1	60 < T < 105
235	88.9	60 < T < 105
255	93.8	60 < T < 105
256	93.8	60 < T < 105
300 (Between KC Pumps 1B1 and 1B2)	84.5	60 < T < 105
300 (Between KC Pumps 1A1 and 1A2)	84.0	60 < T < 105

All acceptance criteria for the Support Systems Verification Test for 100% of Full Power were satisfied.

All temperatures in rooms containing ESF pumps and motors were within the design limits. All acceptance criteria for the test were met.

IV. CORRECTIVE ACTION

No corrective action was needed.

10.6 STEAM GENERATOR WATER HAMMER TEST - TF/1/A/2650/15

Date(s) Performed: 2/2/85 through 2/3/85

I. PURPOSE

The purpose of this test was to verify that no damaging bubble-collapse pressure pulses occur in the steam generator feedline and preheater sections during feedwater flow switchover from auxiliary (CA) nozzle to the main (CF) feedwater nozzle.

II. METHOD

Steady-state operation at 20% Reactor power was maintained while bringing feedwater inlet temperature to within 5°F of 255°F upstream of the steam generator feedwater nozzle to the main feedwater nozzle on Steam Generator A.

Data was obtained from the Loose Parts Monitoring System, the Transient Monitor on the plant computer, a selected data-trending table on the computer, and special test equipment used to record feedwater nozzle pressure. Mechanical Maintenance performed a post-test walk-through of the Main Feedwater system to visually inspect piping and restraints associated with Steam Generator A.

The data that was gathered was reviewed to verify that the Acceptance Criteria was met. The test was then repeated on Steam Generator B to confirm the results obtained using Steam Generator A.

III. RESULTS

A review of data obtained during feedwater nozzle switchover and the visual inspection following switchover showed no indication of damage to the steam generators or associated feedwater piping and restraints.

No pressure pulse in excess of 50 psi (peak to peak) in the main feedwater lines occurred following the switch to the main feedwater nozzle.

IV. CORRECTIVE ACTION

The validity of feedwater nozzle pressure data as recorded by the special test equipment on Steam Generator A was questionable. The test was repeated on Steam Generator B to ensure that reliable data was taken.

11.0 POWER ESCALATION TESTING/PREOPERATIONAL TESTING
TO BE PERFORMED

This testing included:

Boron Thermal Regeneration System Functional Test
Secondary Systems Functional Test

These tests are described on the following page.

11.1 BORON THERMAL REGENERATION SYSTEM FUNCTIONAL TEST

There is not a specific FSAR Chapter 14 commitment to perform this test. However, this system will be tested prior to use in accordance with the recommendations of Regulatory Guide 1.68, Rev. 2, Appendix A, Section 1.b.2. Major system design changes have been identified to install a recirculation loop to speed up the temperature stabilization times between boration/dilution modes. To minimize personnel exposures for implementation of this change, use of this system will be delayed. Administrative actions have been taken to prohibit the use of this system. This system is non-safety related and is not required for unit operation. When the system changes have been performed, the system will be tested as stated in FSAR Section 3.9.2.1.1 and 9.3.6.6 and Table 14.2.12-1 (Page 37).

11.2 SECONDARY SYSTEMS FUNCTIONAL TEST - TP/1/B/2650/09

This test was written to complete outstanding items not completed in the Reactor Coolant System Hot Functional Testing program (FSAR 14.2.12-1 Page 3). Specifically, vacuum in the Main Condenser C is less than expected (higher absolute pressure). Efforts are underway to identify/correct the problem. The effect of the condenser vacuum being lower than expected is to reduce the turbine cycle efficiency.

12.0 PREOPERATIONAL TESTING COMPLETED AS PART OF THE
STARTUP TESTING PHASE

Various preoperational tests required to be performed were completed after initial fuel loading. This testing included the following:

- Steam Generator Blowdown Functional Test
- Secondary Systems Functional Test
- Containment Air Release and Addition Pressure Control Functional Test
- Ice Condenser Region Functional Test
- Diesel Generator 1A(1B) Preoperational Test
- CO2 Fire Protection System Test - D/G Building
- Thermal Expansion Testing on ASME Code Piping
- Steady State Piping Systems Operational Vibration Measurement
- Post Transient Piping Survey
- Annulus Ventilation Filter Train Functional Test
- Spent Fuel Pool Exhaust Filter Train Functional Test
- Electrical Heat Tracing
- Preservice Inspection of PSA Mechanical Snubbers
- Auxiliary Building Filtered Exhaust Filter Train Functional Test
- Process Radiation Monitoring System Functional Test

These tests are described on the following pages.

12.1 STEAM GENERATOR BLOWDOWN FUNCTIONAL TEST - TP/1/B/2650/04

Date(s) Performed: 10/24/84 - 12/6/84

I. PURPOSE

The objectives of the Steam Generator Blowdown (BB) Functional Test were:

- A. To verify proper operation of the BB pump mini-flow valve
- B. To verify proper operation of the BB tank level interlocks
- C. To verify proper operation of the BB high radiation interlock
- D. To verify proper operation of the BB tank high pressure interlock
- E. To verify proper operation of the CA pump auto start interlock
- F. To verify proper operation of the BB recovery heat exchanger exit high temperature interlock
- G. To balance the blowdown flow rates from the two nozzles on each steam generator
- H. To verify proper BB system operation at Hot No-Load conditions

II. METHOD

Starting with a blowdown tank level of 50% a BB pump was started. The "mini-flow" rate was recorded when the level control valve closed and the mini-flow control valve opened to maintain a minimum flow.

The BB tank level was raised until all high and high-high level alarms and interlocks were verified. The level was then lowered until the low level setpoint and associated alarm were verified. The BB pumps were started to further reduce tank level until all interlocks associated with the low-low setpoint were verified.

A high radiation alarm on the steam generator blowdown was simulated and proper response verified. Override mode was then initiated and proper response verified.

BB tank high pressure was simulated by applying pressure to the instrument sensing line. Proper response was verified after the high pressure setpoint was reached.

CA pump autostart was simulated and proper system response verified.

A high BB recovery heat exchanger exit temperature was simulated and proper response at the high and high-high setpoints verified.

The D Heater bleed isolation valves were cycled and the proper response of the BB tank vent to this action was verified.

With the BB System in operation the low resistance flow path was isolated on each steam generator and the resulting flow rates were measured. The high resistance flow paths were isolated and the low resistance flow paths throttled on each steam generator to obtain the same flows as those obtained through the high resistance paths alone at the same conditions of pressure and temperature in the steam generators. Finally the high resistance flow paths were restored to complete the flow balance.

With the steam generators at normal No-Load temperature and pressure the BB System was operated in normal alignment. Trend data was recorded to verify stable operation.

III. RESULTS

All acceptance criteria were met. All alarms and interlocks functioned as designed. Flow rates were balanced at each nozzle of the four steam generators. The BB System functioned properly at Hot No-Load Conditions.

IV. CORRECTIVE ACTIONS

Problems encountered in this test were due to controls not operating properly. These problems are summarized below:

- A. Mini-flow controls would not maintain flow above minimum allowed flow. Control dead band was too great to keep flow at the required setpoint. The controller was recalibrated with a new setpoint higher than the previous setpoint.
- B. During the Hot No-Load operation test the system could not maintain a stable level in the blowdown tank. This was determined to be due to the following reasons:
 - 1. The blowdown flow being used was found to be significantly higher than design because the orifice plate calculations used by design were incorrect. They used 6% moisture rather than 6% quality. New calculations were performed to determine the proper flowrate.
 - 2. The level control setpoint was setup in the lower 25% of the controller band. The controller was not designed to operate in this manner. The level controls were recalibrated to operate as the controller was designed.

12.2 SECONDARY SYSTEMS FUNCTIONAL TEST - TP/1/B/2650/09

This test has not been completed. See Section 11.2.

12.3 CONTAINMENT AIR RELEASE AND ADDITION PRESSURE CONTROL FUNCTIONAL TEST -
TP/1/B/2100/03

Date(s) Performed: 9/27/84 - 10/24/84, 1/10/85, 4/19/85,
6/14/85 - 6/27/85

I. PURPOSE

This test was performed for the purpose of demonstrating the capability of the Containment Air Release and Addition (VQ) System to maintain containment pressure within Tech Spec limits. This completed a requirement of the VQ System Functional Test (performed during Hot Functional Testing) which could not be successfully performed previously.

II. METHOD

Containment pressure was trended during normal heatup of the Reactor Coolant System (in preparation for Initial Criticality). Comparison of the increasing pressure with VQ System actuation was made to verify containment pressure control capability (Air Release Mode).

Once steady state operation at temperature was achieved, the flow path for Air Addition Mode was verified by opening the Air Addition Valve.

III. RESULTS

During unit heatup the VQ System was operated per normal operating procedure. This involved manual actuation of the system in Air Release Mode following receipt of a containment pressure alarm (setpoint 0.15 PSIG). The Air Release Valve automatically closes when containment pressure decreases to 0.0 PSIG and the VQ fans are subsequently manually secured. By review of the system operating log and the containment pressure trend data it was verified that the VQ System adequately maintained containment pressure below the Tech Spec limit of 0.3 PSIG.

Circumstances necessitating containment air addition did not arise during the testing period. Therefore, in order to test this facet of system capability the Air Addition Mode was established by manually actuating the VQ Fans and opening the air addition valves. Air flow into containment was physically verified at the discharge of the containment isolation valve.

IV. CORRECTIVE ACTIONS

None Required.

12.4 ICE CONDENSER REGION SUBSYSTEM FUNCTIONAL TEST - TP/1/A/1100/02

Date(s) Performed: 7/8/83 - 8/24/84

I. PURPOSE

The Ice Condenser Region Subsystem Functional Test was performed for the purpose of:

- A. Verifying that the total ice inventory present is in excess of that required by Technical Specifications.
- B. Verifying that a minimum weight of 1250.8 lbs. of ice is present in each of the 1944 baskets.
- C. Verifying that a minimum concentration of 1800 ppm of boron is present in each basket.
- D. Ensuring unrestricted operation of the Lower Inlet Doors and Intermediate Deck Doors.
- E. Verifying that the Ice Bed and Wear Slab temperatures are less than or equal to 20°F.
- F. Verifying the absence of obstructions in the Ice Condenser Floor Drains and Ice Bed Lattice Structure Flow Paths.

II. METHOD

The methodologies employed to meet the objectives of this test are summarized as follows:

- A. The ICECSAS Computer Program was utilized with basket weights obtained by TP/1/A/1200/23, Initial Ice Basket Weight Determination to verify with a 95% degree of confidence that total ice inventory is in excess of 2,431,418 lbs.
- B. The weight results of each basket (obtained by TP/1/A/1200/23) were surveyed to ensure that none were lighter than 1250.8 lbs.
- C. The results of CP/0/A/8100/21, Chemistry Procedure for Determination of Borax, were checked to ensure that the Tech Spec limit of ≥ 1800 ppm boron was present.
- D. Satisfactory completion of MP/0/A/7150/06 and MP/0/A/7150/07 which verify proper operation of the Lower Inlet Doors and Intermediate Deck Doors, respectively, was documented in this procedure.
- E. A survey of the Ice Bed Temperature monitoring system and Ice Condenser Wear Slab temperature monitoring system was conducted to verify that temperatures did not exceed 20°F.

F. Satisfactory completion of MP/O/A/7150/08 and MP/O/A/7150/10 which verify clearance of the floor drains and lattice structure flow paths, respectively, was documented in this procedure.

III. RESULTS

All Acceptance Criteria were met. A summary of the quantitative results of this test is as follows:

<u>Parameter</u>	<u>Tech Spec Limit</u>	<u>Acceptance Criterion</u>	<u>Measured Value</u>
Avg. Ice Bed Temp.	$\leq 27^{\circ}\text{F}$	$< 20^{\circ}\text{F}$	12.8 $^{\circ}\text{F}$
Avg. Wear Slab Temp.	N/A	$< 20^{\circ}\text{F}$	18.5 $^{\circ}\text{F}$
Total Ice Bed Inventory	$\geq 2,368,652 \text{ lbs.}$	$> 2,431,424 \text{ lbs.}$	2,962,297 lbs.

All other Acceptance Criteria were verified via the respective Chemistry and Maintenance procedures.

IV. CORRECTIVE ACTIONS

Some maintenance was required on the Lower Inlet Doors and Intermediate Deck Doors to obtain passable operation per the Maintenance procedures. Most doors met their Acceptance Criteria without this requirement, however.

The Floor Wear Slab temperatures in a few locations had to be obtained with a contact pyrometer due to malfunctions of some of the temperature elements. These elements were subsequently repaired or replaced.

12.5 DIESEL GENERATOR 1A(1B) PREOPERATIONAL TEST - TP/1/A/1100/02A(B)

Date(s) Performed: 7/6/83 - 9/11/84 (1A)
6/27/83 - 9/28/84 (1B)

I. PURPOSE

The objectives of these tests were:

- A. To demonstrate that the Diesel Generator Auxiliary Systems [Cooling Water (KD), Lube Oil (LD), Fuel Oil (FD), Starting Air (VG), and Building Ventilation (VD)] perform in accordance with design.
- B. To demonstrate that all logic, trip devices, initiating devices and permissive and prohibit interlocks function properly.
- C. To demonstrate that each Diesel Generator can be started and loaded to 7000 KW in ≤ 60 seconds, and operated with this load for ≥ 60 minutes.
- D. To demonstrate that each Diesel Generator: 1) starts from ambient conditions and accelerates to at least 427 RPM in < 11 seconds and 2) attains a generator voltage and frequency of 4160 ± 420 volts and 60 ± 1.2 Hz within 11 seconds after the start signal on an ESF actuation test signal.
- E. To demonstrate on a simulated loss of offsite power the de-energization of the emergency buses and load shedding from the emergency buses.
- F. To demonstrate that each Diesel Generator and its load group can function without any dependence upon any other load group or portion thereof.
- G. To demonstrate that each Diesel Generator is capable of: 1) starting and accelerating to rated speed, 2) loading in the required sequence, all of the Engineered Safeguard Feature Loads, 3) maintaining the frequency and voltage during the loading sequence at no less than 95% of nominal and 75% of nominal, respectively, and 4) restoring the frequency to within 2% of nominal, and the voltage to within 10% of nominal within 60% of each load-sequence time interval.
- H. To demonstrate, by simulating loss of AC voltage, that each Diesel Generator can start automatically and attain the required voltage and frequency within acceptable limits.
- I. To demonstrate the proper operation of each Diesel Generator during the design-accident-loading sequence and to verify that the voltage and frequency are maintained within required limits.

- J. To demonstrate the full-load-carrying capability of each Diesel Generator for an interval of ≥ 24 hours, of which at least 22 hours are 100% load and at least 2 hours at 110% load.
- K. To demonstrate the Fuel Oil Filter and Strainer can be changed with the Diesel Generator operating, using the diverter valve, without a drop or loss of fuel oil pressure.
- L. To demonstrate the functional capability of each Diesel Generator at full-load temperature. By actual loss of all AC voltage, verify that the Diesel Generator can start, in the required time, and sequence loads automatically while maintaining voltage and frequency within required limits.
- M. To demonstrate the capability of each Diesel Generator to start and accept $\geq 50\%$ rated load for ≥ 1 hour, 35 consecutive times.
- N. To demonstrate the transient following a complete loss of load should not cause either Diesel Generator to reach 500 rpm.
- O. To demonstrate that the capability of each Diesel Generator to supply emergency power within the required time is not impaired during testing.
- P. To demonstrate that each Diesel Generator is capable of:
1) starting and accelerating to rated speed, 2) loading, in the required sequence, all of the Black-Out loads, 3) maintaining the frequency and voltage during the loading sequence at no less than 95% of nominal and 75% of nominal, respectively, and 4) restoring the frequency to within 2% of nominal, and the voltage to within 10% of nominal within 60% of each load-sequence time interval.
- Q. To demonstrate the capability of each Diesel Generator to reject a load of ≈ 825 KW and maintain voltage at 4160 ± 416 volts and frequency at 60 ± 1.2 Hz.
- R. To demonstrate the ability to: 1) synchronize each Diesel Generator while connected to the emergency load, with offsite power, 2) transfer the emergency load to the offsite power, and 3) isolate the Diesel Generator to standby status.
- S. To verify that auto-connected loads to each Diesel Generator do not exceed 7700 KW.
- T. To verify that load sequencing times are within acceptable tolerances.
- U. To demonstrate a simultaneous start of both Diesel Generators.
- V. To demonstrate the ability of either Main Fuel Oil Storage Tank to supply fuel oil to each Fuel Oil Day Tank.

II. METHOD

- A. Each of the Diesel Generator Auxiliary Systems, (KD, LD, VG, and VD) was checked during its standby mode of operation. The ability of those systems to maintain the Diesel Generators was verified during tests on these systems. All significant interlocks were checked, along with those alarms that require operator action.
- B. The Diesel Generator Load Sequencers were tested to verify proper operation in all various situations. During these tests all interlocks, resets, and logic were verified. Also, the ability of the Sequencers to come out of Test and return to Standby upon receipt of an emergency signal was verified.
- C. Diesel Generator starting capability was tested by decreasing the pressure in the Starting Air Tanks to the minimum start pressure of the Starting Air Compressor. Then a normal start of each Diesel Generator was demonstrated. After returning the VG System to Standby a fast start and load test, consisting of starting and loading each Diesel Generator to 7000 KW in ≤ 60 seconds, was performed. Next, a simultaneous start of both Diesel Generator 1A and Diesel Generator 1B was demonstrated.
- D. Each Diesel Generator's engine trip devices were tested to verify proper function. While operating in a normal situation each trip device was actuated individually and shown to trip the Diesel Generator. Then each Diesel Generator was operated in a simulated emergency situation and each trip device individually actuated. This was to demonstrate that only the Overspeed, Low Low Lube Oil Pressure, and Generator Fault trips would trip the Diesel Generator during an emergency.
- E. Local and remote control of each Diesel Generator was tested. Each Diesel Generator was shown to be able to be operated locally in a normal situation. The controls in the Control Room were then operated and shown to have no affect on Diesel Generator speed or voltage. Each Diesel Generator was then operated from the Control Room in a normal situation. The controls on the Engine and Generator Control Panels were operated and shown to have no affect on Diesel Generator speed or voltage. Next, each Diesel Generator was operated locally, by using the Control Room Override, in a normal situation. The controls in the Control Room were operated and shown to have no affect on Diesel Generator speed or voltage.

- F. The ability of the Diesel Generator Auxiliary Systems to sustain Diesel Generator operation was tested. The Lube Oil System and the Cooling Water System were aligned and operated for normal standby service. System parameters were recorded for 8 hours to verify that the systems could maintain the Diesels at warm Standby. While data was being recorded, the Maintenance Mode and Barring Device interlocks were verified. Then with each Diesel Generator running the ability to supply fuel oil from either Main Fuel oil Storage Tank to each Fuel Oil Day Tank was demonstrated.
- G. Each Diesel Generator was then tested to verify total system capability and emergency response. Each Diesel Generator was started from ambient conditions by a simulated LOCA signal. Response times were verified to be within limits. Then each Diesel Generator was loaded to 7000 KW until equilibrium engine temperature was obtained. During this warm-up period, Fuel Oil Filter and Fuel Oil Strainer swap-over capability was demonstrated. After engine warm-up the load was increased to 7700 KW for a period of at least two hours. During this time, data was taken to verify the capabilities of the KD, LD, and FD systems to maintain the Diesel Generators. Next the load was decreased to 7000 KW for a period of at least 22 hours. During this time data was again taken on the KD, LD, and FD systems. Next, a simulated blackout + LOCA was initiated to demonstrate the functional capability of each Diesel Generator at full-load temperature.
- H. The ability of the Diesel Generators to accept sequenced loads and long-term system reliability were tested. Each Diesel Generator was started by a blackout signal. The proper load sequence was then observed while data was recorded to verify voltage and frequency requirements. Next, a complete loss of load test was performed to verify each diesel's capability to withstand such a transient. Then a partial loss of load test was performed. Train separation was also verified during this test. Each Diesel Generator was then demonstrated to successfully start, be loaded to > 3500 KW and maintain this load for > 60 minutes, 35 consecutive times.

III. RESULTS

The following sections describe the results of the corresponding tests described in Part II. All Acceptance Criteria associated with this test were met by both Diesel Generators.

- A. The Diesel Generator Auxiliary Systems [Cooling Water (KD), Lube Oil (LD), Fuel oil (FD), Starting Air (VG), and Building Ventilation (VD)] performed in accordance with design.

- B. All the logic and initiating devices associated with the Load Sequencers operated properly. The sequencers returned to Standby from test receipt of an emergency signal.
- C. Diesel Generator 1A(1B) was started and loaded to 7000 KW in 51.24(30.16) seconds and then maintained that load for ≥ 60 minutes. This met the Acceptance Criterion of ≤ 60 seconds for loading. Both Diesel Generators were successfully started simultaneously and achieved rated speed within the Acceptance Criterion of ≤ 11 seconds.
- D. The tested engine trip devices and associated event recorder points operated properly on both Diesel Generators during both Emergency and Normal Modes of Operations.
- E. The redundant emergency start circuits were demonstrated to operate properly. Local and Control Room controls were verified to function properly in each control mode.
- F. The Diesel Generator Auxiliary Systems performed in accordance with design. All interlocks functioned properly. The capability of either Main Fuel Oil Storage Tank to Supply fuel to the Fuel Oil Day Tanks was successfully demonstrated with the Diesel Generators running.
- G. The following Acceptance Criteria were successfully met by both Diesel Generators:
1. Each Diesel Generator and its load group functions without any dependance upon any other load group or portion thereof.
 2. Each Diesel Generator is capable of: 1) starting and accelerating to rated speed, 2) loading in the required sequence, all of the Engineered Safeguard Features loads, 3) maintaining, during the loading sequence, the frequency and voltage within 95% of nominal and 75% of nominal, 4) restoring the frequency to within 2% of nominal and the voltage to within 10% of nominal within 60% of each load-sequence time interval.
 3. Each Diesel Generator starts from ambient conditions and reaches 427 RPM within 11 seconds. The Generator achieves 4160 ± 420 volts and 60 ± 1.2 Hz within 11 seconds after the start signal on an ESF actuation test signal.
 4. Each Diesel Generator can start automatically, by simulating loss of all AC voltage, and attain the required voltage and frequency within acceptable limits and time.

5. Each Diesel Generator operates properly during the design-accident-loading sequence and maintains the voltage and frequency within the required limits.
 6. Each Diesel Generator is capable of carrying full load for an interval ≥ 24 hours, of which at least 22 hours shall be at 100% and at least 2 hours shall be at 110% load. During testing Diesel Generator 1A(1B) was capable of carrying full load for an interval ≥ 24 hours, of which 26(30) hours were at 7000 KW and 2(2) hours were at 7700 KW.
 7. The Fuel Oil Filter can be changed while each Diesel Generator is operating, using the diverter valve, without a drop or loss of fuel oil pressure.
 8. The Fuel Oil Strainer can be changed while each Diesel Generator is operating, using the diverter valve, without a drop or loss of fuel oil pressure.
 9. Upon the simulated loss of all AC voltage concurrent with a LOCA signal each Diesel Generator, at full load temperature, is capable of starting automatically, in the required time, and automatically sequencing the loads while maintaining voltage and frequency within required time limits.
- H. The following Acceptance Criteria were successfully met by both Diesel Generators:
1. The capability of each Diesel Generator to supply emergency power within the required time is not impaired during testing.
 2. Each Diesel Generator is capable of: 1) starting and accelerating to rated speed, 2) loading in the required sequence, all of the Black-Out loads 3) maintaining, during the loading sequence, the frequency and voltage within 95% of nominal and 75% of nominal, respectively, and 4) restoring the frequency to within 2% of nominal, and the voltage to within 10% of nominal within 60% of each load-sequence time interval.
 3. Each Diesel Generator and its load group functions without any dependance upon any other load group or portion thereof.
 4. The transient following the complete loss of load does not cause either Diesel Generator to reach 500 RPM.
 5. Each Diesel Generator is capable of rejecting a load of ≈ 825 KW while maintaining voltage at 4160 ± 416 volts and frequency at 60 ± 1.2 Hz.

6. The ability exists to: 1) synchronize each Diesel Generator while connected to the emergency load, with offsite power, 2) transfer the emergency load to the offsite power, 3) isolate the Diesel Generator and 4) restore the Diesel Generator to standby status.
7. Each Diesel Generator is successfully started and loaded, to \geq 3500 KW for a period of \geq 60 minutes, 35 consecutive times.

IV. CORRECTIVE ACTION

None

12.6 CO2 FIRE PROTECTION SYSTEM TEST-D/G BUILDING - TP/1/A/1400/05B

Date(s) Performed: 5/12/84 - 6/3/84, 8/4/84, 8/20/84 - 8/26/84,
9/18/84, 10/6/84 - 10/17/84

I. PURPOSE

The objectives of this test were:

- A. To verify proper operation of the CO2 Storage Tank refrigeration system, the high/low pressure local alarms and Control Room annunciator, and the low level signal switch.
- B. To verify proper operation of the Diesel Generator Room thermostats.
- C. To verify that the supervisory functions of the system operated properly and provided the correct alarm responses.
- D. To verify that system alarms, internal interlocks, ventilation interlocks, and valves operated as designed.
- E. To verify that the system would reinitiate and complete a trip sequence on receipt of a manual trip signal after a auto trip had occurred without requiring a reset.
- F. To measure the pre-discharge and discharge periods.
- G. To verify that the master and selector valves could be manually opened to discharge CO2 during a complete loss of all power.
- H. To verify that the system battery was capable of providing the minimum required power if normal power is lost.
- I. To verify that the continued operation of the system was possible, including a system trip, using battery power after a loss of normal power.
- J. To verify that the CO2 System discharged at least the minimum concentration of CO2 and maintained at least the minimum concentration for a specified time during an actual system full discharge in each D/G Room without apparent damage to other equipment.
- K. To verify that the presence of CO2 would not degrade D/G starting and operation.

II. METHOD

An overlapping test method was utilized to ensure a complete but efficient test of all circuits and alarms. The following circuits/functions were tested:

- A. CO2 Storage Tank instruments and alarms.
- B. D/G Room thermostats.
- C. Each auto trip circuit including supervision (open circuit, ground, and associated status lights, annunciators and buzzer) and trip or alarm action (master valve circuit, ventilation interlock circuits, selector valve circuits, associated status lights, annunciators, buzzer, and the disable switches).
- D. The ability of the panels to complete a manually initiated discharge sequence after a simulated auto trip without first being reset.
- E. Predischarge and discharge electronic timers.
- F. Each manual trip circuit including supervision (open circuit, ground, and associated status lights, annunciators and buzzer) and trip or alarm action (timers, manual pushbutton switches, status lights, annunciators and buzzer).
- G. Solenoid circuit and alarm circuit supervision features (open circuit, open solenoid/horn, shorted solenoid/horn, ground, associated status lights, annunciators and buzzer).
- H. Manual operation of the master and selector valves.

Overall system performance was also tested. The discharge tests were begun by depressing the manual pushbutton. The predischarge and discharge periods and CO2 concentration were measured during the full discharge tests. After a soak period of 20 minutes, the associated D/G was started and run at its maximum qualified load for 10 minutes. The system's capability to trip and complete a simulated discharge sequence without normal AC power was tested. This included verification of all status lights, annunciators, horns, buzzer, valve action, and ventilation interlocks.

III. RESULTS

All Acceptance Criteria for this test were met.

- A. All supervisory functions, thermostats, alarms, interlocks, and valves operated as required with and without normal AC power.
- B. The battery maintained 24.0 ± 1 VDC for 20 ± 5 seconds.

- C. The predischage time for D/G Room 1A was 60.52 seconds and 1B was 59.45 seconds. This met the Acceptance Criteria of 60 ± 3.5 seconds.
- D. The discharge time for D/G Room 1A was 58.65 seconds, and 1B was 59.63 seconds. This met the Acceptance Criteria of 60 ± 3.5 seconds.
- E. The system discharged CO2 into the D/G 1A Room for 59.22 seconds with a peak average concentration of 34%. After a soak period of 10 minutes, the average concentration was 32%. The system discharged CO2 into the D/G Room 1B for 59.64 seconds with a peak average CO2 concentration of 42%. After 10 minutes the average CO2 concentration was 38%. The Acceptance Criteria of $\geq 34\%$ concentration within a discharge period of one minute and a concentration of $\geq 28\%$ after a 10 minute soak period was met.
- F. Each Diesel-Generator started and loaded with CO2 in the associated room.

IV. CORRECTIVE ACTION

Initially it was found that the system could not discharge an adequate amount of CO2 into the 1A D/G Room. This was due to excessive pressure drops. Hydraulic redesign and modification was performed to decrease these pressure drops and increase system response time. The discharge piping was rerouted using bent pipe and shorter runs. The selector valve solenoid was relocated from the D/G room access hall to a few feet from the valve and the actuation line size was increased to $\frac{1}{2}$ ". The master valve pilot lines were increased to $\frac{3}{4}$ " and $\frac{1}{2}$ ". Following these modifications the system was tested successfully.

12.7 THERMAL EXPANSION TESTING ON ASME CODE PIPING - TP/1/A/1150/08A

Date(s) Performed: 8/13/84, 9/28/84-10/1/84, 10/3/84-10/4/84,
11/17/84, 11/28/84-12/5/84

I. PURPOSE

The purpose of the Thermal Expansion Test was to verify that the ASME Code piping greater than 1 inch in diameter and with normal operating temperatures above 200°F is not restricted from expanding.

II. METHOD

Visual inspections and walk-throughs were performed at various reactor coolant temperature plateaus during precritical testing for the ASME Code portions of the following systems:

- BB - Steam Generator Blowdown System
- CA - Auxiliary Feedwater System
- CF - Main Feedwater System
- NC - Reactor Coolant System
- ND - Residual Heat Removal System
- NI - Safety Injection System
- NM - Nuclear Sampling System
- NR - Boron Thermal Regeneration System
- NV - Chemical and Volume Control System
- SA - Main Steam Supply to Auxiliary Equipment
- SM - Main Steam System

The inspections were performed at ambient conditions and at the NC temperature plateaus of 250°F, 350°F, 450°F and 557°F. Piping interferences within 2 inches of the pipe or pipe insulation were identified. Spring support and snubber indications were recorded and verified to be within design range. Discrepancies (i.e. piping interferences or snubber indications out of design range) were evaluated and resolved by Design Engineering.

III. RESULTS

Snubbers were verified not to be within $\frac{1}{2}$ inch of either piston stop through the duration of the test. Spring support settings were within $\pm 10\%$ of the values given on the as-built drawings at ambient conditions and at the reactor coolant temperature plateau of 557°F. Spring support indicators remained on scale throughout the test. All potential interferences within 2 inches of piping were evaluated by Design Engineering. Design Engineering either deemed the potential interference acceptable (generally if no further expansion towards interference was expected) or suggested a method to clear the interference (e.g. notching the insulation). All problems which could restrict piping and components from expanding were resolved. Piping and components did not contact any interferences which may restrict expansion.

IV. CORRECTIVE ACTION

Insulation was notched to allow expansion in 6 places on Steam Generator Blowdown System piping. Reactor Coolant system piping required notches in the insulation in two locations. A bracket was removed in one location to allow for expansion of Reactor Coolant System piping. Insulation was notched in one location on Residual Heat Removal System piping and in 2 places on Safety Injection System piping.

12.8 STEADY STATE PIPING SYSTEMS OPERATIONAL VIBRATION MEASUREMENT -
TP/1/A/1200/21

Date(s) Performed: 11/7/83 - 1/3/84, 7/24/84 - 12/6/84,
1/19/85 - 7/31/85

I. PURPOSE

The objective of this test was to verify that flow induced vibration is sufficiently small to cause no fatigue or stress failures in the piping systems identified in FSAR Table 3.9.2-1. Table 12.8-1 lists the systems which were tested.

II. METHOD

Qualified personnel visually scanned and took readings on each applicable system while in normal operating mode to determine points of maximum vibration. At points of maximum vibration, measurements were taken and compared with the acceptance criteria. If the acceptance criteria were exceeded, the data was sent to Design Engineering for evaluation.

III. RESULTS

All piping systems tested met the original Acceptance Criteria or were reviewed by Design Engineering and found to be acceptable.

The Boron Thermal Regeneration System was not tested as required by FSAR Table 3.9.2-1. This system is not as yet operational.

IV. CORRECTIVE ACTION

The test was initially performed during Hot Functional Testing. It was repeated in its entirety in response to NRC violation 50-413/84-92-01, which found that the prerequisite which required all final piping hangers to be installed and all temporary hangers removed prior to vibration testing had not been met for many of the systems tested.

Data for the 34 locations which failed to meet the initial Acceptance Criteria were evaluated on a case-by-case basis by Design Engineering. The initial criteria were determined using a generic Stress Intensification Factor (SIF) of six. For locations which failed the criteria, new criteria were calculated based upon the highest SIF found in the specific region of piping in question. The highest actual SIF used was 4.034, with most SIF values of approximately 2.5. Thirty of the thirtyfour locations were cleared by re-evaluating the criteria. Two locations were located on vendor supplied piping associated with Diesel Generator 1A. Data was forwarded to the vendor and found by the vendor to be acceptable. The remaining two locations required the addition of restraints. After the restraints were installed, a retest was performed and the Acceptance Criteria were met.

Vibration testing will be performed on the Boron Thermal Regeneration System as part of the Functional Test on the system. The Functional Test will be performed after pending modifications on the system have been completed and before the system is used during normal operation of the plant.

STEADY STATE PIPING SYSTEMS OPERATIONAL VIBRATION MEASUREMENT

PIPING SYSTEMS INCLUDED IN VIBRATION TEST PROGRAM

System Designation	System
NC	Reactor Coolant System
NI	Safety Injection System
ND	Residual Heat Removal System
NS	Containment Spray System
NV	Chemical and Volume Control System
NB	Boron Recycle System
KC	Component Cooling System
WL	Liquid Radwaste System
KF	Fuel Pool Cooling And Cleanup System
FD	Diesel Generator Fuel Oil System
KD	Diesel Generator Cooling Water System
LD	Diesel Generator Lub oil System
RN	Nuclear Service Water System
FW	Refueling Water System
SM	Main Steam System
CF	Feedwater System
CA	Auxiliary Feedwater System
SA, SB	Steam Dump System
YC	Control Area Chilled Water System
BB	Steam Generator Blowdown Recycle System
RC	Recirculated Cooling Water System

12.9 POST TRANSIENT PIPING SURVEY - TP/1/A/1200/26

Date(s) Performed: 11/7/83 - 1/3/84, 7/24/85 - 7/30/85

I. PURPOSE

The objective of the Post Transient Piping Survey was to verify that piping layout and support/restraints are adequate to withstand normal transients without damage to piping systems identified in Table 12.9-1.

II. METHOD

After each transient event, qualified personnel inspected the associated piping and supports to verify the absence of any excessive piping motion. For the NC System exclusively, vibration measurements were taken at Reactor Coolant Pump suction and discharge during pump starts and trips.

III. RESULTS

All acceptance criteria were satisfied. No permanent deformation or damage in any system, structure, or component important to nuclear safety was observed.

All suppressors and restraints responded within their allowable ranges, between stops or with indicators on scale.

The measured piping vibration for the Reactor Coolant System during reactor coolant pump starts and trips did not exceed the value specified by Design Engineering of 7.144 in/sec. peak-to-peak unfiltered.

IV. CORRECTIVE ACTIONS

The Post Transient Piping Survey (TP/1/A/1200/26) had as a prerequisite that all final piping system hangers be installed and all temporary hangers be removed prior to conducting the vibration tests. Contrary to this requirement, the Post Transient Piping Survey was conducted before all permanent hangers were installed and before all temporary hangers were removed. This resulted in NRC Violation 50-143/84-92-01 being received on October 9, 1984. Part of Duke Power's response to this violation was to repeat the test completely.

POST TRANSIENT PIPING SURVEY

PIPING SYSTEMS INCLUDED IN TRANSIENT VIBRATION TEST PROGRAM

System	Transient Type	Type Measurement	Simultaneous Test
NC	NC Pump Start	Vibration measurement at selected points	HFT
	NC Pump Trip	Vibration measurement at selected points	HFT
	NC PORV Cycling	Post Transient inspection	HFT
BB	Initiation of S/G Blowdown	Post Transient inspection	S/G BD Test
	Isolation of S/G Blowdown	Post Transient inspection	S/G BD Test
CA	Motor Driven Pump Start	Post Transient inspection	Aux FDW F.T.
	Motor Driven Pump trip	Post Transient inspection	Aux FDW F.T.
	AFWPT Cold Start	Post Transient inspection	Aux FDW F.T.
	AFWPT Trip	Post Transient inspection	Aux FDW F.T.
CF	Isolation Valve Closure	Post Transient inspection	HFT
NI	NI Pump Start	Post Transient inspection of pump discharge piping	ESF
	CCP Pump Start	Post Transient inspection of pump discharge piping	ESF
NV	Letdown Isolation	Post Transient inspection	HFT
SM	Main Steam Isolation (individually)	Post Transient inspection	SM Isolation HFT
	Main Steam PORV Discharge	Post Transient inspection	HFT
CF, SM	Loss of Electrical Load 100% FP	Post Transient inspection	Power Escalation Testing
CF, SM	Turbine Trip = 70% FP	Post Transient inspection	Power Escalation Testing

TABLE 12.9-1

Date(s) Performed: 7/31/84 - 8/18/84

I. PURPOSE

The objectives of this test were:

- A. To ensure proper system air flow capacity.
- B. To verify acceptable pressure drop across the combined Prefilter/Moisture Separator, Preheaters, HEPA Filters and Carbon Absorber Bed.
- C. To verify proper air flow distribution to the Carbon Absorber and HEPA Filter section.
- D. To verify that tracer injection ports provide proper mixing of the tracer in the air approaching the component stage to be tested.
- E. To perform an In-Place DOP Penetration Test for each HEPA Filter Bank.
- F. To perform an In-Place Refrigerant Bypass and Penetration Test on each Carbon Absorber Bed.

II. METHOD

A series of tests were performed for each filter unit. The first measured air flow rate and filter unit ΔP . Individual train flow was obtained by using the associated air flow monitor device. Individual fan flow was obtained by pitot tube traverse. Flows were adjusted as necessary using the MVD's (Manual Volume Dampers) on the suction of the filter units. Adjustments were made using a combination of methods including: center line static pressure, "Center" point or velocity adjustment for traverse, velocity pressure adjustment for the AFMD (Air Flow Monitoring Device), and filter unit ΔP 's. Following adjustments, fan motor running currents were taken to ensure that nameplate amps were not exceeded. Total filter unit ΔP was measured by attaching a U-Tube across the entire filter unit. Air velocity data was obtained on the downstream side of each HEPA filter and downstream of each air flow channel in the carbon bed. A velometer was used for this measurement.

An Air/Aerosol mixing test was performed on each upstream HEPA bank. A Sparger Tube set-up was used to inject DOP to obtain good mixing. DOP concentration data was obtained by sampling in the middle of each HEPA filter, one foot upstream. After correct air velocity and Air/DOP mixing was proven the HEPA banks were tested for DOP penetration by injecting DOP at the same location and by the same method used to obtain the data for Air/DOP mixing. DOP concentration was obtained both upstream and downstream of the filter bank penetration calculated.

The Air/DOP mixing test for the upstream HEPA bank proved that good mixing existed for both the upstream HEPA bank and the carbon bed, since HEPA filters only straighten the air flow and make it more uniform. The carbon bed was then challenged with R-11. By plotting 5 consecutive penetrations obtained one minute apart and extrapolating back to time zero, the instantaneous percent penetration was determined. After the R-11 test, the filter unit was operated in filtered mode with pre-heaters on to drive off excess refrigerant. Laboratory test data was obtained to prove actual radioiodine efficiencies.

III. RESULTS

All Acceptance Criteria for this test were met as described below (see also Table 12.10-1):

- A. The total combined pressure drop across each filter unit was < 7.96 inwc.
- B. The air flow for each fan with filters installed was in the range of 9000 CFM $\pm 9.0\%$.
- C. The maximum and minimum velocity readings were all within $\pm 20\%$ of the average taken across the upstream HEPA banks carbon absorber beds and downstream HEPA banks.
- D. The maximum and minimum DOP concentrations across each upstream and downstream HEPA filter bank face were within $\pm 20\%$ of the average for that bank.
- E. All HEPA filter bank DOP penetrations were $\leq 1.0\%$.
- F. All carbon bed R-11 penetrations were $\leq 1.0\%$ at rated flow.
- G. The carbon methyl-iodine efficiency for each carbon bed was $\geq 97\%$ when tested as new carbon.

IV. CORRECTIVE ACTIONS

The carbon initially loaded into the VE units did not pass the subsequent methyl-iodide penetration test. This carbon had to be unloaded while other carbon was impregnated with TEDA (Tri-ethylene Di-amine). The end product was a highly efficient co-impregnated carbon (KI₃ and TEDA) which passed the methyl-iodide penetration test.

ANNULUS VENTILATION
FILTER TRAIN FUNCTIONAL TEST

TEST RESULTS SUMMARY

TEST	Acceptance Criteria	Filter Unit	
		1A	1B
Total ΔP	< 7.96 inwc	4.1 inwc	4.3 inwc
Fan Capacity	8190 CFM 9810 CFM	9015 CFM	9203 CFM
HEPA DOP Penetration	Upstream: ≤ 1.0%	0.003%	0.0002%
	Downstream: ≤ 1.0%	0.004%	0.004%
Carbon bed R-11 Penetration	≤ 1.0% of total flow	0.1%	0.25%
Carbon Methyl- Iodide Efficiency	≥ 97%	99.7%	99.7%

TABLE 12.10-1

Date(s) Performed: 5/14/84, 7/10/84 - 7/25/84, 7/30/84

I. PURPOSE

The objectives of this test were:

- A. To ensure proper system air flow capacity.
- B. To verify acceptable pressure drop across the combined Prefilter/Moisture Separator, Preheaters, HEPA Filters and Carbon Absorber Bed.
- C. To verify proper air flow distribution to the Carbon Absorber and HEPA Filter sections.
- D. To verify that tracer injection ports provide proper mixing of the tracer in the air approaching the component stage to be tested.
- E. To perform an In-Place DOP Penetration Test for each HEPA Filter Bank.
- F. To perform an In-Place Refrigerant Bypass and Penetration Test on each Carbon Absorber Bed.

II. METHOD

A series of tests were performed for each filter unit. The first measured air flow rate and filter unit ΔP . Individual train flow was obtained by using the associated air flow monitor device. Individual fan flow was obtained by pitot tube traverse. Flows were adjusted as necessary using the MVD's (Manual Volume Dampers) on the suction of the filter units. Adjustments were made using a combination of methods including: center line static pressure, "Center" point or velocity adjustment for traverse, velocity pressure adjustment for the AFMD (Air Flow Monitoring Device), and filter unit ΔP 's. Following adjustments, fan motor running currents were taken to ensure that nameplate amps were not exceeded. Total filter unit ΔP was measured by attaching a U-Tube across the entire filter unit. Air velocity data was obtained on the downstream side of each HEPA filter and downstream of each air flow channel in the carbon bed. A velometer was used for this measurement.

An Air/Aerosol mixing test was performed on each upstream HEPA bank. A Sparger Tube set-up was used to inject DOP to obtain good mixing. DOP concentration data was obtained by sampling in the middle of each HEPA filter, one foot upstream.

After correct air velocity and Air/DOP mixing was proven the HEPA banks were tested for DOP penetration by injecting DOP at the same location and by the same method used to obtain the data for Air/DOP mixing. DOP concentration was obtained both upstream and downstream of the filter bank and penetration calculated.

The Air/DOP mixing test for the upstream HEPA bank proved that good mixing existed for both the upstream HEPA bank and the carbon bed, since HEPA filters only straighten the air flow and make it more uniform. The carbon bed was then challenged with R-11. By plotting 5 consecutive penetrations obtained one minute apart and extrapolating back to time zero, the instantaneous percent penetration was determined. After the R-11 test the filter unit was operated in filtered mode with pre-heaters on to drive off excess refrigerant. Laboratory test data was obtained to prove actual radioiodine efficiencies.

III. RESULTS

All Acceptance Criteria for this test were met as described below (see also Table 12.11-1):

- A. The total combined pressure drop across each filter unit was < 7.90 in wc.
- B. The air flow for each fan with filters installed was in the range of 16565 CFM \pm 9.0%.
- C. The maximum and minimum velocity readings were all within \pm 20% of the average taken across the face of the upstream HEPA banks, carbon absorber beds, and downstream HEPA banks.
- D. The maximum and minimum DOP concentrations across each upstream and downstream HEPA filter bank face were within \pm 20% of the average for that bank.
- E. All HEPA filter bank DOP penetrations were \leq 1.0%.
- F. All carbon bed R-11 penetrations were \leq 1.0% at rated flow.
- G. The carbon methyl-iodide efficiency for each carbon bed was \geq 97% when tested as new carbon.

IV. CORRECTIVE ACTIONS

The carbon initially loaded into the VF units did not pass the subsequent methyl-iodide penetration test. This carbon had to be unloaded while other carbon was impregnated with TEDA (Tri-ethylene Di-amine). The end product was a highly efficient co-impregnated carbon (KI, and TEDA) which passed the methyl-iodide penetration test.

SPENT FUEL POOL EXHAUST
FILTER TRAIN FUNCTIONAL TEST

TEST RESULTS SUMMARY

TEST	Acceptance Criteria	Filter Unit			
		1A1	1A2	1B1	1B2
Total ΔP	< 7.90 inwc	3.55 inwc	3.60 inwc	3.65 inwc	3.80 inwc
Fan Capacity	15074 CFM- 18056 CFM	17724 CFM	16755 CFM	17433 CFM	16672 CFM
HEPA DOP Penetration	Upstream: ≤ 1.0%	0.004%	0.005%	0.008%	0.016%
	Downstream: ≤ 1.0%	0.003%	0.002%	0.004%	0.007%
Carbon bed R-11 Penetration	≤ 1.0% of rated flow	0.2%	< 0.001%	0.09%	0.1%
Carbon Methyl- Iodide Efficiency	≥ 97%	99.5%	99.5%	99.7%	99.7%

Date(s) Performed: 8/12/84 - 9/17/84

I. PURPOSE

The objective of the EHT System Functional Test was to verify that heat traced pipe in the NV, NB, WL and WS Systems can be maintained at design temperature with primary and backup heat trace systems. Also, proper operation of the temperature controllers, alarms and temperature scanner was verified. Test dates were 8/12, 8/15, 8/16, 8/17, 8/21, 8/29, 8/30, 8/31, 9/6, 9/15, 9/17/84.

II. METHOD

Both the primary and backup heat traced systems were tested by using the scanner to call up the temperature of the system's thermocouples and verifying that the proper temperature range was obtained.

The temperature controllers were tested by recording the temperature at which they energized and deenergized and verifying that they operated within $\pm 5^\circ$ of their setpoint.

The alarms were tested by opening and closing breakers associated with the circuits.

III. RESULTS

All acceptance criteria as provided in the test were met.

A comparison of the measured temperatures with the acceptance criteria is as follows:

<u>Thermocouple</u>	<u>Measured Temperature ($^\circ$F)</u>	<u>Acceptable Temp. Range ($^\circ$F)</u>
1S-NV-12T	96	70-175
1S-NV-13T	88	70-175
1S-NV-14T	89	70-175
1S-NB-21T	175	150-200
1S-NB-22T	183	150-200
1S-NB-23T	177	150-200
1S-WL-31T	163	150-200
1S-WL-32T	159	150-200
1S-WL-33T	176	150-200
1S-WS-01T	175	150-200
1S-WS-02T	173	150-200
1S-WS-03T	179	150-200

A comparison of the measured temperatures at which the temperature controllers energized and deenergized with the acceptance criteria is as follows:

<u>Temp. Controller</u>	<u>Energized (°F)</u>	<u>De-energized (°F)</u>	<u>Acceptable Temperature Range (°F)</u>
1P-NV-01	86.6	90.0	85 ± 5
1B-NV-01	72.5	74.8	70 ± 5
1P-NB-02	175.5	178.6	175 ± 5
1P-NB-02	157.9	161.0	160 ± 5
1P-WL-01	177.8	180.0	175 ± 5
1B-WL-01	155.3	157.8	160 ± 5
1P-WS-01	170.5	172.6	175 ± 5
1B-WS-01	156.7	159.6	160 ± 5

The Electric Heat Tracing System functioned as designed without any outstanding discrepancies that could affect the system's operability.

IV. CORRECTIVE ACTION

Major interruptions were caused by having to take off insulation, reposition thermocouples and heater cables and reinsulate EHT circuits in which the acceptable temperature range could not be obtained. Replacement of the heater cable to WS-15 was also required.

12.13

PRESERVICE INSPECTION OF PSA MECHANICAL SNUBBERS - TP/1/A/1150/09

Date(s) Performed: 3/5/84 through 9/17/84

I. PURPOSE

The purpose of this test was to ensure that all safety-related PSA Mechanical Snubbers are operable.

II. METHOD

Snubber operability was ensured by verifying that there were no visible signs of damage or impaired operability; that adequate swing clearance was available to allow snubber movement; that the snubber was not seized or jammed; and that structural connections were properly installed.

III. RESULTS

All safety-related PSA Mechanical Snubbers were verified to be operable.

IV. CORRECTIVE ACTION

Snubbers which failed to meet the criteria for operability were repaired and/or replaced. The repaired/replaced snubbers were reinspected and verified to be operable.

Date(s) Performed: 8/1/84 through 8/17/84

I. PURPOSE

The objective of this test was to show the in-place efficiency of the two trains of Auxiliary Building Filtered Exhaust.

II. METHOD

The total capacity of each fan in the system was measured after filter installation and verified to be within acceptable limits. The Pressure drop across the combined HEPA filters and charcoal adsorber banks was also verified to be acceptable. An air distribution test was performed for both the HEPA filters and the carbon beds to show that no dead spots are present in the housing.

An air-aerosol mixing capacity test was conducted for each injection port to demonstrate that a uniform mixture of air and aerosol (the test gas) could be assumed at up-stream test points.

The HEPA filter banks were then tested by injecting the test gas and taking concentration samples upstream and downstream of the bank to determine the penetration. The absorber banks were tested in a similar fashion.

III. RESULTS

The capacity of all fans were within the acceptance criteria of 30,000 cfm \pm 9%. The pressure drops across the filter assemblies were acceptable at the specified flow rate. Measurements of velocity and tracer gas concentrations were within 20% of the average for each filter bank. All HEPA filter banks demonstrated an efficiency greater than 99%, and the charcoal absorbers removed more the 99% of the test gas.

IV. CORRECTIVE ACTIONS

Air-Aerosol mixing was inadequate on both upstream and downstream HEPA filter banks. Spargers had to be constructed and placed in the duct work and the filter housing to achieve adequate mixing.

Date(s) Performed: 6/13/84 - 7/4/84

I. PURPOSE

The objectives of the Process Radiation Monitoring System Functional Test were as follows:

- A. To demonstrate the capacity of the Process Radiation Monitoring System to detect and record high radiation levels in both process systems and plant effluents.
- B. To demonstrate that plant operators will be alerted to either high radiation or monitor system failure.
- C. To demonstrate that the proper control actions will be initiated by the monitors following the detection of abnormally high radiation levels.

II. METHOD

Each of the following monitor channels was tested on an individual basis:

1EMF31	1EMF33	1EMF34(L)	1EMF34(H)
1EMF35(L)	1EMF35(H)	1EMF36(L)	1EMF36(H)
1EMF37	1EMF38(L)	1EMF38(H)	1EMF39(L)
1EMF39(H)	1EMF40	EMF41	1EMF42
EFM43	EM43B	1EMF44(L)	1EMF44(H)
1EMF45A(L)	1EMF45A(H)	1EMF45B(L)	1EMF45B(H)
1EMF46A	1EMF46B	EMF47	1EMF48
EMF49(L)	EMF49(H)	EMF50(L)	EMF50(H)
1EMF52	1EMF53A	1EMF53B	1EMF54
1EMF55A	1EMF55B		

The following tests were performed on each channel:

- A. Demonstration of sample flow rates and/or loss of flow alarms to prove the performance of the flow units.
- B. Electronic modules were tested for proper operation and calibration by use of the built-in checksource and a pulse generator.
- C. Demonstration that the output of each module was being indicated on the trend and multipoint chart recorders, as well as the Operator Aid Computer.
- D. Verification of the operation of the HIGH RADIATION and CABINET TROUBLE annunciators.

- E. Verification of proper control action execution by use of a simulated high radiation signal from a pulse generator to trip the modules.
- F. Accuracy of the setpoints was checked by use of a pulse generator or the use of the TRIP ADJUST feature.

III. RESULTS

- A. All of the flow units except for 1EMF48 (Reactor Coolant Radiation Monitor) were verified to meet the acceptance criteria of the FSAR and the procedure. 1EMF48 was not verified by the procedure to have a flow rate of 2 GPM \pm 10% as specified by the FSAR. This flow rate of 2 GPM \pm 10% was verified by Work Request 164710PS to meet the FSAR acceptance criteria.
- B. All electronic modules operated properly according to the acceptance criteria and were in calibration.
- C. The output of each module was indicated on the proper recorders and the Operator Aid Computer.
- D. All annunciators operated properly.
- E. All modules executed the proper control actions, upon a simulated high radiation signal, as specified by the procedure.
- F. All setpoints were found to be set accurately as specified by the procedure.

IV. CORRECTIVE ACTION

The procedure did not check the FSAR acceptance criteria for 1EMF48 (Reactor Coolant Radiation Monitor). Work Request W.R. 164710PS verified that the flow rate on 1EMF48 met the acceptance criteria of the FSAR.

13.0 SPECIAL REPORTS

As required by the FSAR the following Special Reports are provided.

Loose Parts Monitoring System
Post Accident Liquid Sampling System

13.1 LOOSE PARTS MONITORING SYSTEM DESCRIPTION

I. SYSTEM DESCRIPTION

The Vibration and Loose Parts Monitoring System (LPMS) consists of 12 vibration and loose parts channels, 4 vessel channels and 2 channels per Steam Generator. Each LPMS channel consists of a piezoelectric transducer, pre-amplifier, and signal processor unit. The signal processor units, alarm modules, signal status indicator lights, vibration level meter (db indication), audio cassette tape recorder with associated speaker and operator selector switches are mounted in the Operator Interface Panel located in the Control Room. This panel provides both visual and audible indication to the Control Room Operator. The indicating lights on the operator interface panel provide notification when a loose parts signal setpoint is exceeded. Upon incidence of any of these alarms a common control board indicator is activated along with its associated audible alarm. Both the interface panel and control board alarms are latched in the alarm state and require operator action to reset.

The four channel tape recorder has two modes of operation: manual or automatic. In the manual mode the operator selectively records or plays back desired information. The auto-tape turn-on feature automatically starts the recording of the appropriate signals upon receipt of a loose parts alarm signal. The selection of the four signals to be recorded is pre-determined by a switching sequence that is programmed to key off of the first alarm; that is, the signal which first exceeds its alarm setpoint determines which other transducer outputs will be recorded. The panel status indicator light associated with the first alarm detected will blink, the other lights will be indicated by a non-blinking light when alarmed. If a vessel head transducer is the first to sense a loose part, the signals from the upper and lower vessel heads (including the alarmed channel) and from two of the steam generators are recorded. When a steam generator transducer first detects a loose part, the signals from the following transducers are recorded: one of the reactor vessel upper head transducers, one of the reactor vessel lower head transducers, the transducer which generated the first alarm level and the transducer of one other steam generator. This keying assures that the signals of most importance are recorded during the initial alarm period. Upon receipt of a loose parts alarm signal a Control Room annunciator is sounded.

The operator, after the incidence of a loose part alarm, can then check all the inputs individually. The operator, after listening and observing each input may then re-select the appropriate signals to be recorded if another area is deemed to be of greater importance.

The digital panel meter displays the vibration level (g) in decibels (dB). The 0 dB indication is calibrated to represent the amplitude level of the largest signal within the bandwidth of the vibration amplifier. Signal selection for normal monitoring is a manual function.

Twelve transducers are located in the areas where loose parts are most likely to become entrapped. These are:

- two on the reactor vessel lower head, diametrically opposed.
- two on the reactor vessel upper head
- two on the lower head of each steam generator.

Experience has shown that the exact location of these transducers is not critical since the acoustic wave that results from an impact propagates throughout the reactor coolant system piping and vessel.

The accelerometers in the lower reactor vessel region are positioned on the incore instrument guide tubes as close to the reactor vessel as practical for inspection and maintenance. The accelerometers are located not more than two feet from the vessel bottom. Mounting blocks are provided to physically clamp the accelerometer and associated junction box to the guide tube.

The accelerometers in the upper reactor vessel region are stud mounted directly to the vessel head lifting lugs. The associated junction boxes are also stud mounted near the lugs.

The accelerometers for the steam generators are stud mounted to the steam generator inlet-side support structures, and the associated junction boxes are also stud mounted at the same location.

Details of the sensor locations are provided in Figure 13.1-1.

Segments of high temperature coaxial cabling are used in the high temperature region near the accelerometers. Twisted shielded pair cable is used throughout the rest of the system to provide noise immunity. The pre-amplifiers are located in the Containment as near as practical for inspection and maintenance to their associated transducers to insure good signal-to-noise transmission and located in a position to minimize radiation exposure. The pre-amplifiers used are the charge to voltage type converters; i.e., the voltage output is proportional to the charge input. This voltage output is not affected by capacitance in the input cable, rendering the system accuracy largely independent of input cable length or capacitance.

Detailed specifications for the transducers, pre-amps and recorder are included in Tables 13.1-2, 13.1-2 and 13.1-3.

The LPMS is capable of detecting an impact energy of 0.05 ft-lb. The initial alarm settings corresponds to an 0.5 ft-lb impact. An input calibration was performed during system installation. This calibration established the relationship between the output signal strength and impact energy. It also determined the frequency of the clamped ringing signal that is characteristic of all large steel structures when struck. To perform this impact calibration, the LPMS was activated and the reactor vessel and steam generators were then struck with a known impact energy.

II. SYSTEM CALIBRATION PROCEDURES

A. Initial and Subsequent Calibrations

During the pre-operational test program, the channels of the LPMS were calibrated and functionally tested to insure the ability of each channel to respond to the appropriate level of signal from its sensor. The ability of each sensor to detect the appropriate level of vibration was verified by local stimulation using a controlled mechanical input. The ability of the alert actuation circuitry to respond to the desired signal level and actuate the appropriate automatic recording mode was verified. These tests assured that the system was capable of detecting appropriate levels of vibration, responding and recording quality data.

Prior to initial startup with the Reactor Coolant System filled/vented and no pumps operating, impact data was collected. Spherical balls (1 inch, 2 inch, 4 inch diameter) were impacted at various locations throughout the Reactor Coolant System. All twelve channels were simultaneously recorded. From this data impact amplitudes and sensor time delays were determined. From this data actual loose parts can be located and its size approximated. Figures 13.1-2 thru 13.1-4 provide typical sensor responses to impact.

During the startup test program, baseline data was recorded from each channel at each of the following conditions: Hot Zero Power, 30% power, 50% power, 75% power, and 100% power. The data was recorded at each power level with all Reactor Coolant System loops in operation, since less than four-pump operation is not allowed by Catawba Technical Specifications. The baseline data was analyzed and a relative spectral density function ("signature") was determined for each channel. These were examined to verify that the level of background noise on each channel was not so high as to interfere with the detection of loose parts, and also to determine the existence of resonance peaks in the baseline spectra which will serve as reference points for comparison with future spectra. Refer to Figure 13.1-5 thru 13.1-7 for typical data at 100% power.

During power operation following initial startup testing, recordings will be taken from each operating channel and analyzed. The resulting spectra will be compared with baseline spectra and any significant changes indicative of problems with the LPMS or the Reactor Coolant System will be investigated. This recording and analysis will be performed as a quarterly surveillance item.

B. Functional Check

A monthly functional check will be performed to verify the proper response of the channel circuitry to a simulated input signal, including verification of alert alarm actuation and initiation of automatic recording at the proper signal level.

C. Channel Check

A qualitative check of each channel will be performed twice each day by monitoring the audio output from each channel to verify that a signal is present on each channel and that there are not unusual noises present which could indicate a problem with the LPMS or the Reactor Coolant System.

III. CONFORMANCE TO REG.GUIDE 1.133

A. Loose Parts Detection Program (LPDP)

1. Description of Deviations

None, the LPDP is in conformance with Reg. Guide 1.133.

2. Description of Program Modifications

None necessary, the LPDP is in conformance with Reg. Guide 1.133.

3. Justification for Remaining Deviations

Not applicable, no remaining deviations.

B. Loose Part Monitoring System (LPMS)

1. Description of Deviations

The Catawba LPMS was not specified to the operating basis earthquake requirements of the Regulator Guide 1.133, and therefore no seismic test reports for this system are available.

2. Description of Needed Modifications

None needed since the LPMS for Catawba was designed and procured in early 1975. At that time there were no known requirements for the seismic design qualification of this system. The Regulator Guide 1.133 guidance on operability for seismic and environmental conditions was issued six years later in May, 1981.

Additionally, the in-containment portion of the Catawba LPMS consists of accelerometers, pre-amplifiers and interconnection cables. These components have been reviewed for seismic capability and it has been determined that:

- (i) The design of accelerometers inherently provides some seismic capability. The accelerometer design at Catawba is similar to designs used in systems that have been seismically tested and qualified.
- (ii) The pre-amplifiers used are solid state, light weight devices similar to other type amplifiers which have been seismically tested and qualified in safety related systems.
- (iii) The cables used for LPMS are high quality, low noise cables protected by an overall armor. These cables inside containment are installed in seismically qualified cable support systems.

3. Cost/Benefit Evaluation of Considered Modifications

No modifications are considered necessary.

4. Plans for Upgrade or Backfit

Duke has no plans for upgrading or backfitting the Catawba LPMS.

LOOSE PARTS MONITORING SYSTEM
SENSOR SPECIFICATIONS

Manufacturer: ENDEVCO
 Model: 76M1
 Type: High Temperature piezoelectric accelerometer

DYNAMIC

Charge Sensitivity	: 10.0 ± 1 pC/g
Voltage Sensitivity	: 12 mV/g nominal
Mounted Resonance Frequency	: 27.000 ± 3000 Hz
Frequency Response	: 0%/0%/+5% nominal at : 5/100/5000 Hz; ± 2% .20 Hz : to 2000 Hz; reference 100 Hz
Transverse Sensitivity	: 3% maximum, 1% typical
Amplitude Linearity, Range	: Sensitivity increases : approximately 1% per 1000 g. : 0 to 3000 g.

ELECTRICAL

Transducer Capacitance	: 660 ± 100 pF at 72°F (22°C)
Transducer Resistance	: 100 MΩ minimum at 72°F (22°C) : 10 MΩ minimum at 700°F (370°C) : 100 kΩ minimum at 900°F (482°C)

ENVIRONMENTAL

Acceleration Limits	Vibration	: 500 g pk in any direction
Temperature Range	Shock	: 3000 g pk in any direction : -65°F to 900°F (-54°C to 482°C) : (short connector during storage at elevated temperatures)
Altitude		: Not affected
Humidity		: Sealed by glass-to-metal fusion and welding
Strain Sensitivity		: 0.5 equivalent g. nominal, at 250 μ in./in. strain
Radiation		: 10 ¹⁸ nvt : 10 ¹⁰ R
Pressure		: 60 psig
Vibration		: 100 g's

LOOSE PARTS MONITORING SYSTEM
REMOTE CHARGE CONVERTER (PRE-AMPLIFIER) SPECIFICATIONS

Manufacturer: Endevco

Model: 2652M4

DYNAMIC

Sensitivity	: 1mV per pC
Accuracy	: $\pm 1\%$ full scale
Frequency Response	: $\pm 5\%$ 3 Hz to 16 KHz (with reference to 1 KHz response) : -29.3% 1 Hz to 50 KHz (with reference to 1 KHz response)
Linearity	: 0.5% of reading from best fit straight line approximation to the curve of output
Harmonic Distortion	: 0.2% maximum
Gain Stability	: Less than 2% over the operating temperature range

ENVIRONMENT

Operating Temperature	: -67°F to 185°F (-55°C to 85°C)
Pressure	: 60 psig
Humidity	: 0-100%
Radiation	: 10^{10} nvt 10*R

LOOSE PARTS MONITORING SYSTEM
TEAC MODEL R-80 CASSETTE DATA RECORDER

Recording/reproducing method	FM wide band
Magnetic tape	TEAC cassette tape CT-90
No. of channels	4
Channel configuration	CH-1 Measuring signal CH-2 Measuring signal CH-3 Measuring signal, noise compensating reference signal (switch selection) CH-4 Measuring signal, voice for memo (switch selection)
Heads	Record and reproduce head x 1 Erase head x 1
Controls	REWIND, FAST FORWARD, STOP, START and RECORD are selected by a push-button switch. Tape speed is selected by a rotary switch.
Tape speeds	19.05 cm/s (7 1/2 in/s) 9.52 " (3 3/4 ") 4.76 " (1 7/8 ") 2.38 " (15/16 ")
Tape speed deviation	Within $\pm 0.3\%$
Flutter	19.05 cm/s 0.5% p-p 9.52 " 0.5 " 4.76 " 0.6 " 2.38 " 0.6 "
Starting time	Constant speed is reached within 1 sec.
Fast forward/rewind time	Within 150 sec for CT-90 tape
Input impedance and input/output levels	
Input impedance	100 K Ω , unbalanced to ground ± 1 to ± 10 volts for $\pm 40\%$ deviation

LOOSE PARTS MONITORING SYSTEM
TEAC MODEL R-80 CASSETTE DATA RECORDER

Input voltage	± 1.5 to ± 10 volts for $\pm 60\%$ deviation
Output voltage	$0 \pm 2V$ or more (at rated input load impedance of 600Ω)
Output current	20 mA or more (at rated input load impedance of 20Ω)

DC level (zero point) can be shifted by $\pm 150\%$ for both input and output.

Frequency characteristics and SN ratio

Speed	Freq. Range	+0.5	w/o noise compensation	with noise compensation
19.05 cm/s	DC - 5 KHz	-1.0 db	46dB	50 dB
9.52 "	DC - 2.5 "	"	46dB	50 dB
4.76 "	DC - 1.25 "	"	44dB	48 dB
2.38 "	DC - 0.625 "	"	44dB	48 dB

Total harmonic distortion less than 1%

Linearity $\pm 0.5\%$ of full deviation

Drift Less than $\pm 0.5\%$ of full deviation
(using CT-90 tape after 15 min. warm-up)

Continuous record/reproduce time Approx. 90 min. (using CT-90 tape at speed of 2.38 cm/s)

Calibration voltage DC : $\pm 1V$
AC : 2 Vp-p (250 Hz)

Remote control output signals

Unless otherwise specified, all inputs/outputs are H (+ 5V) and L (0V) and outputs are all open collector.

1. EOT/BOT LAMP EOT/BOT lamp lights when 33 mA current

LOOSE PARTS MONITORING SYSTEM
TEACH MODEL R-80 CASSETTE DATA RECORDER

2.	<u>SEARCH SPEED</u>	Regardless of speed selection at the recorder, the search speed becomes 7 1/2 in/s when this signal is low level.
3.	START IND	START indicating output
4.	REC IND	REC indicating output
5.	<u>STOP</u>	STOP mode input
6	STOP IND	STOP indicating output
7	<u>REWIND</u>	REW mode input
8	FAST FWD IND	F. FWD indicating output
9	<u>FAST FWD</u>	F. FWD mode input
10	REWIND IND	REWIND indicating output
11	<u>RECORD</u>	RECORD mode input
12	<u>START</u>	START mode input
13	END DETECT CLOCK	END detect clock output
14	REMOTE SERVO CLOCK	REMOTE servo clock output
15	6V RETURN	6V return
16	+ 6V	+ 6V
17	END DETECT PULSE	END detect pulse output
18	EOT/BOT	Detection of tape end
19	<u>MEMORY COUNTER SW</u>	Shorted to 0 V when 10 ³ digit in counter becomes 9.
20	12V RETURN	12V return
21	MEMO INPUT	MEMO MIC input
22	GND	MEMO MIC input ground terminal
23	<u>MEMO SW</u>	MEMO MIC becomes ON when shorted to 0 V.

TABLE 13.1-3 (cont)

LOOSE PARTS MONITORING SYSTEM
TEACH MODEL R-80 CASSETTE DATA RECORDER

24 12V	12V for EOT/BOT LAMP, unregulated
Operating condition	Horizontal or vertical
Ambient temperature and humidity	0 ~ 40°C, 20 ~ 80%
Power requirements	DC 11 ~ 15V approx. 1.8A (stable state) AC 100, 117, 200, 220 or 240V ± 10%
Dimensions	430(W) x 133(H) x 310(D)mm Approx
Zero drift	Less than ± 0.5%
Weight	Approx. 12 kg (recorder only)

LOOSE PARTS MONITORING SYSTEM

ACCELEROMETER LOCATIONS

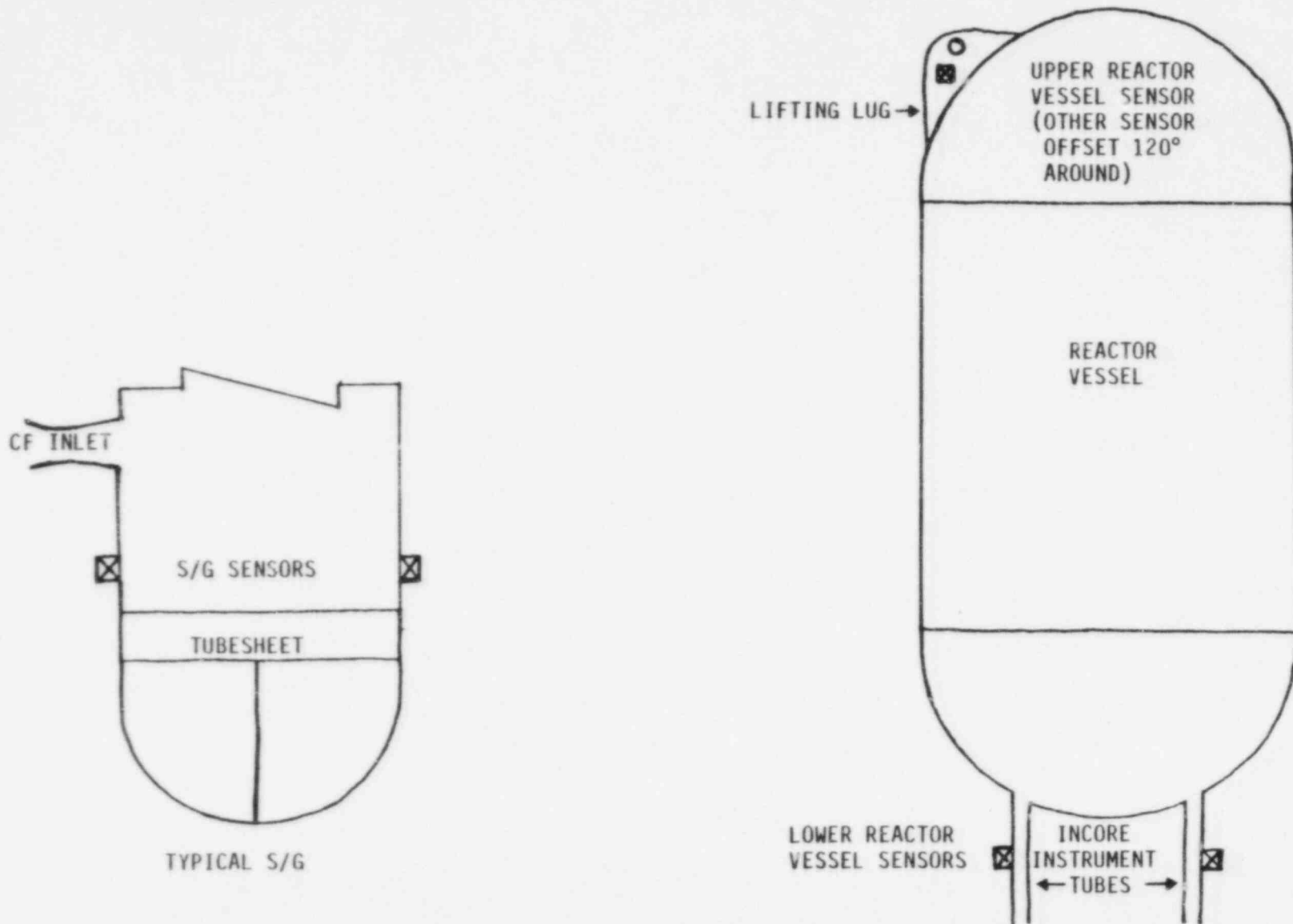


Figure 13.1-1

TYPICAL LPMS IMPACT BASELINE -- IMPACT AT CHANNEL 12 (S/G D)

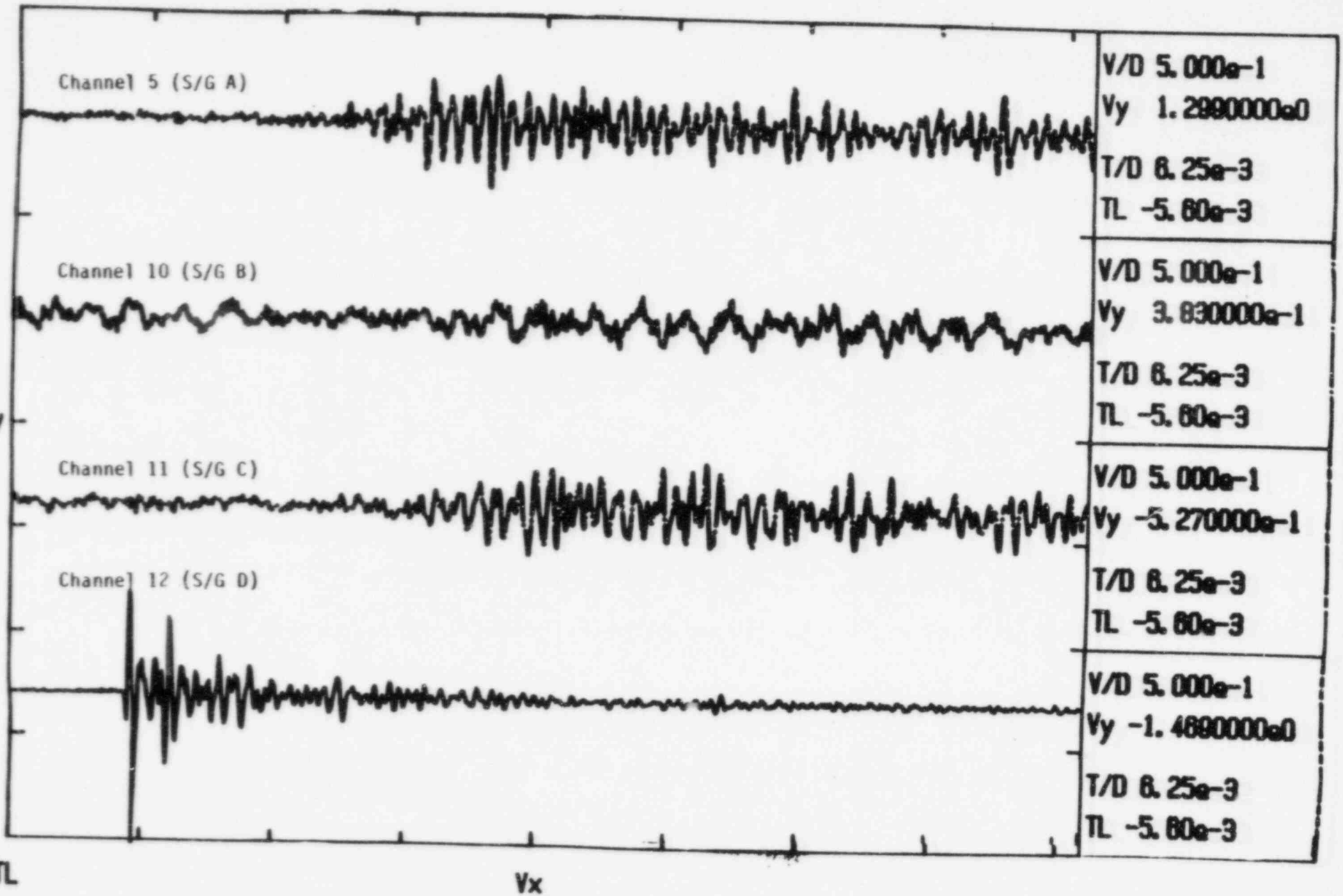


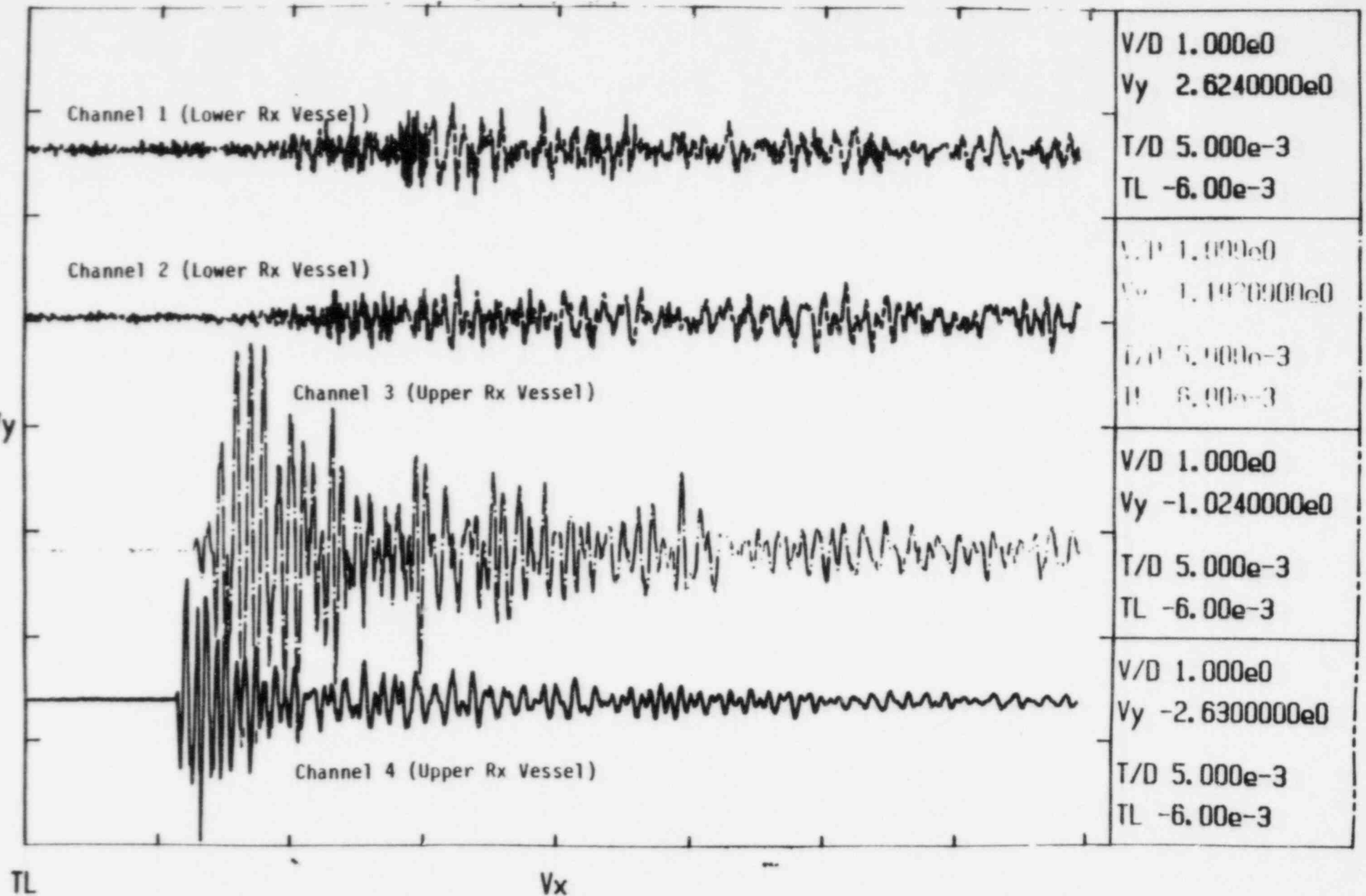
Figure 13.1-2

CATAMBA 1

TYPICAL LPMS IMPACT BASELINE -- IMPACT AT CHANNEL 4 (UPPER RX VESSEL)

Figure 13.1-3

CATAMBA 1



TYPICAL LPMS IMPACT BASELINE -- IMPACT AT CHANNEL 4 (UPPER RX VESSEL)

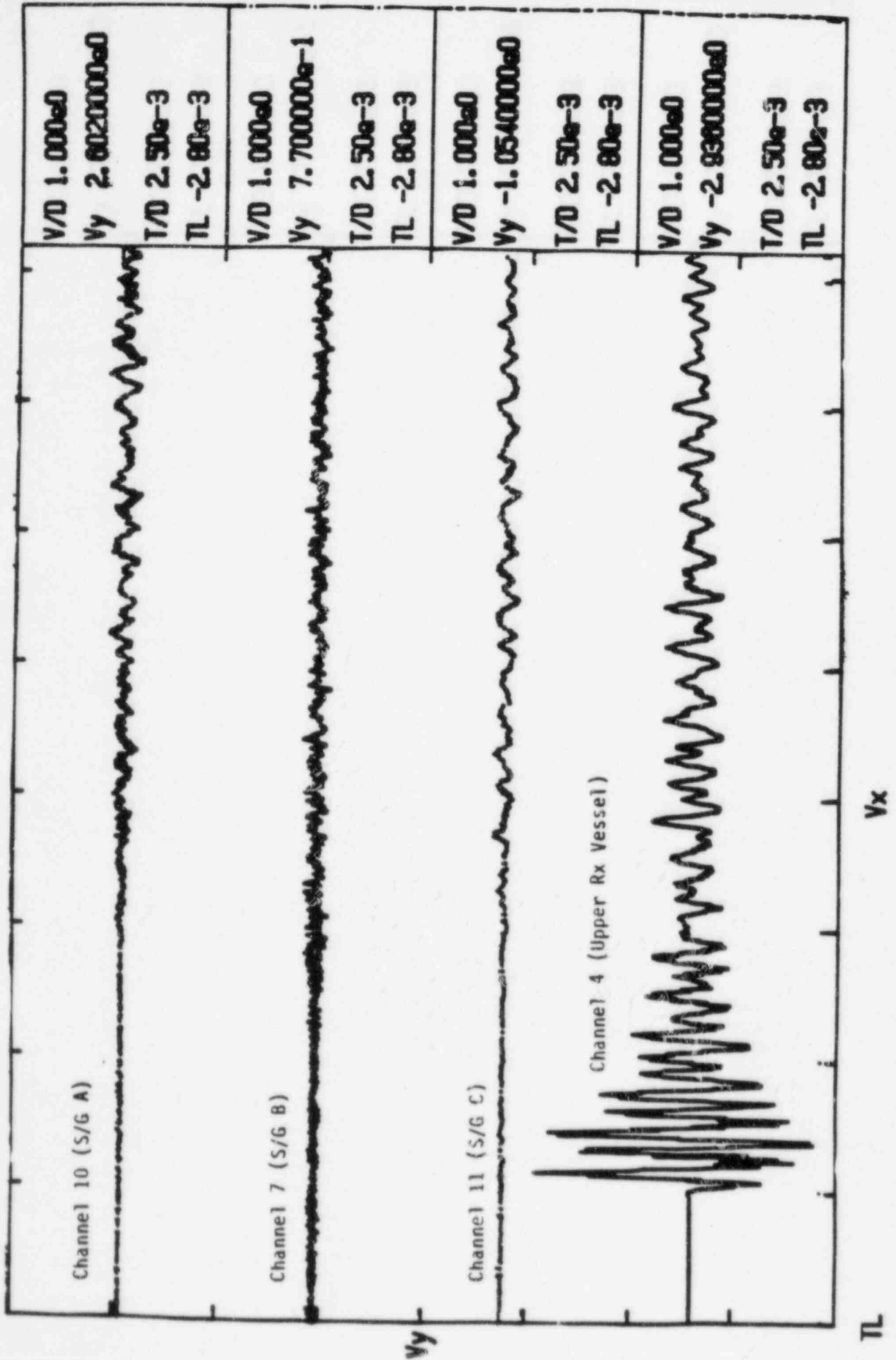


Figure 13.1-4

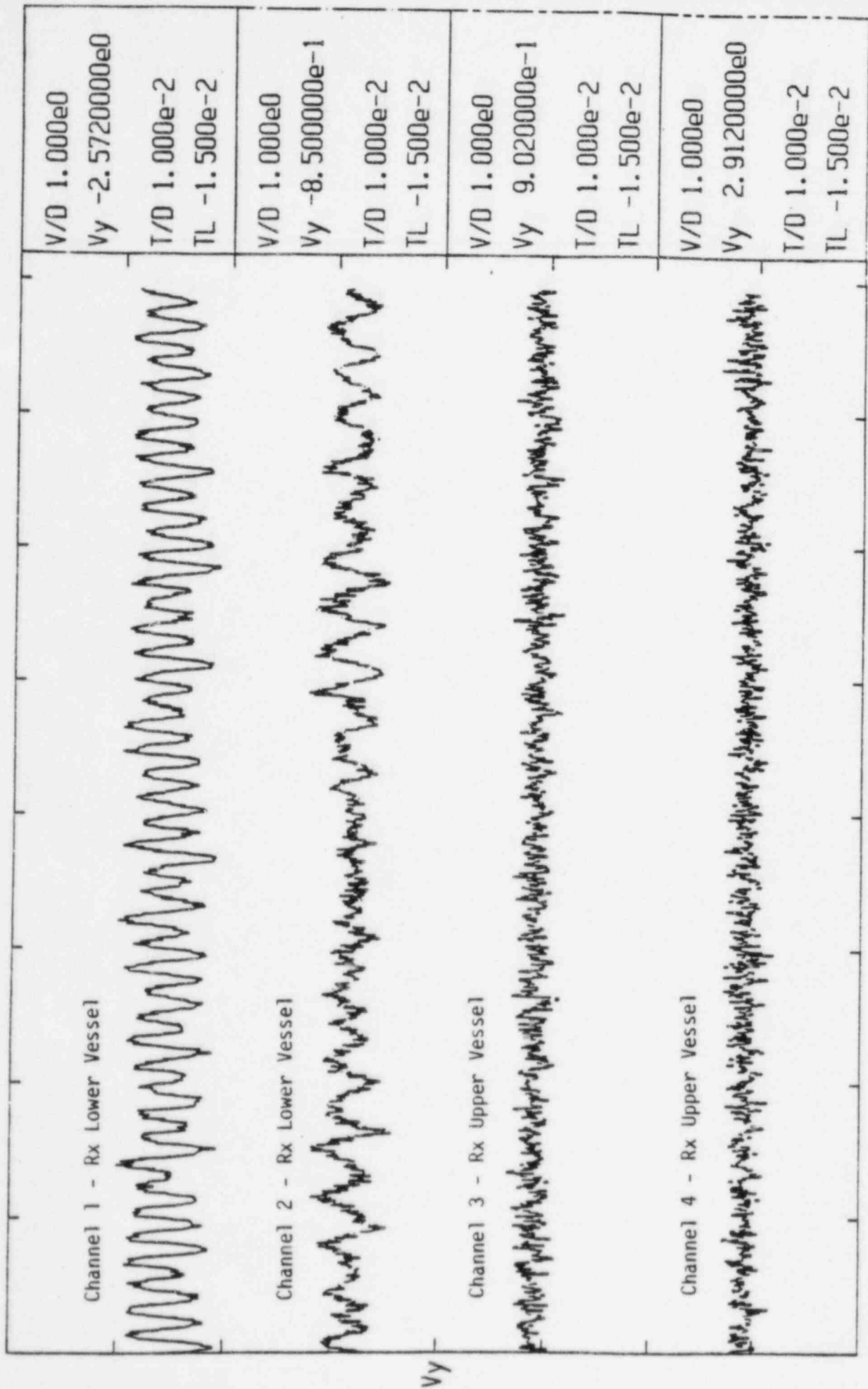


Figure 13.1-5

TYPICAL LPMS OPERATIONAL BASELINE AT 100% POWER

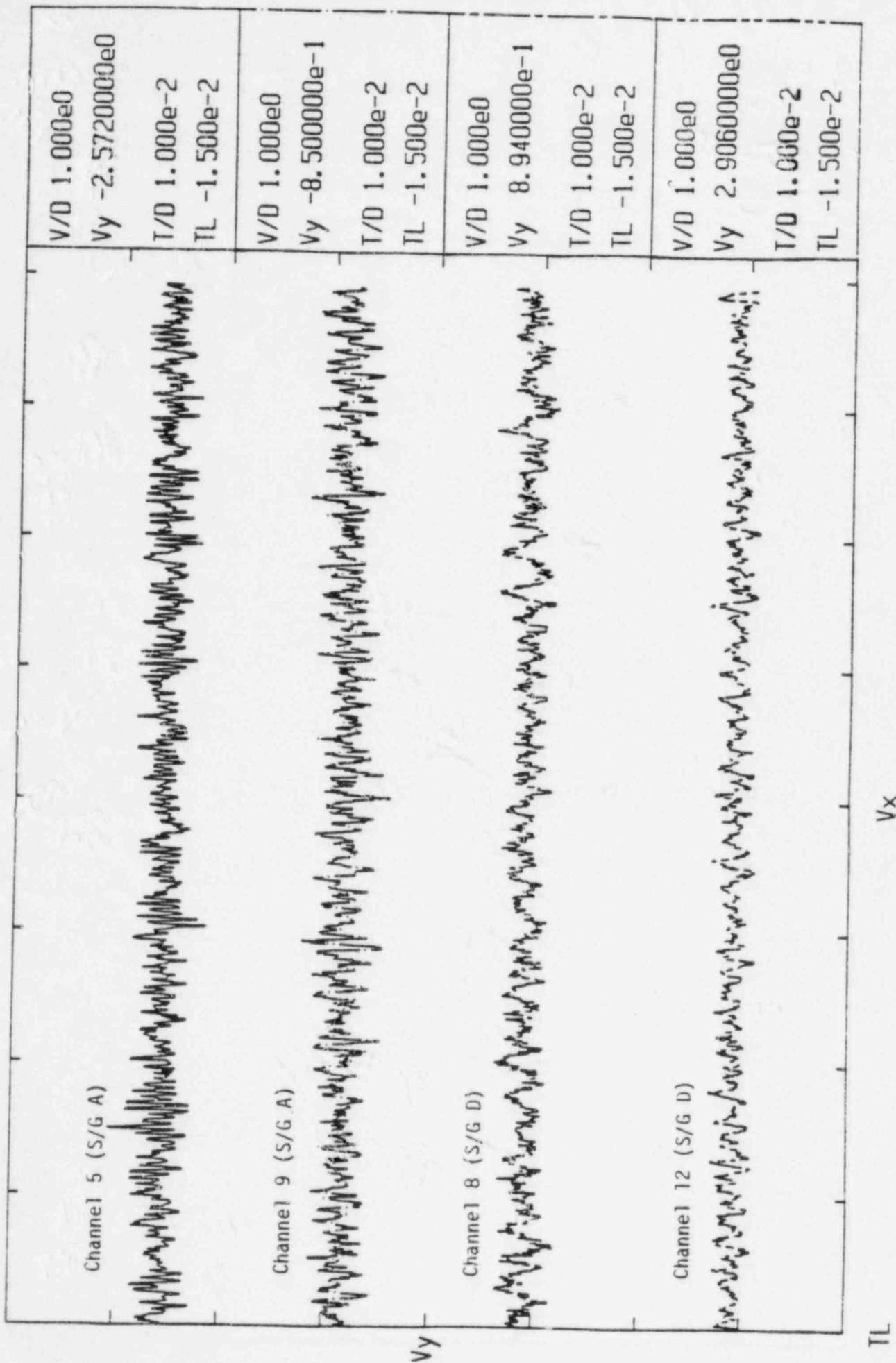


Figure 13.1-6

TYPICAL LPMS OPERATIONAL BASELINE AT 100% POWER

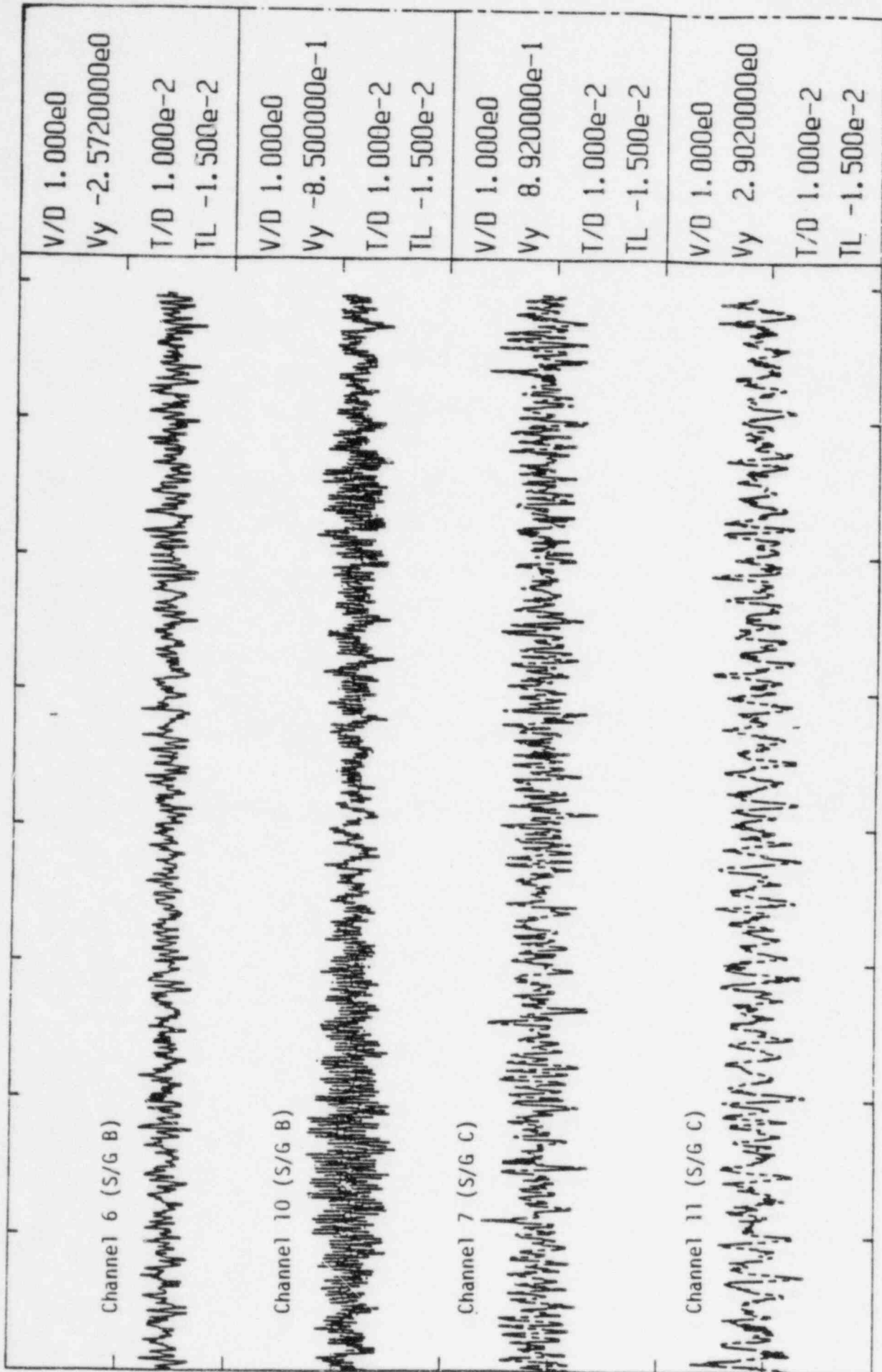


Figure 13.1-7

Vx

TL

13.2 POST ACCIDENT LIQUID SAMPLING SYSTEM

I. PURPOSE

The Post Accident Liquid Sampling System (PALS) provides the station with the capability to obtain reactor coolant samples under accident conditions in accordance with NUREG-0737. Refer to Figure 13.2-1 for typical system schematic.

II. GENERAL SYSTEM DESCRIPTION

A. Control Panel

1. All outputs from the control panel are 115 VAC, 1.5A.
2. To keep the radiation exposure as low as reasonably possible, the control panel can be located approximately 250 cabling feet away from the sampling panel.
3. Instrumentation is provided to measure the pH, temperature, and sample flow of the cooling water and sample, flow of dilution water, and radiation level within the sample panel.
4. The control panel weighs approximately 220 pounds.

B. Sample Panel

1. Reactor coolant is brought into the sample panel through a cooler, which brings the sample temperature below 212°F, and a 95 ml sample is trapped at system pressure.
2. The sample trap is depressurized by opening a valve between the trapped area and an evacuated volume. An inert gas is then bubbled through the depressurized sample to aid in collecting the gases, which come out of solution.
3. pH measurements are then made on the full strength sample.
4. 5.4 milliliters of the full strength sample are trapped for dilutions of up to 600:1.
5. Approximately 20 ml of the diluted gas and 150 ml of the diluted liquid sample are trapped for retrieval by the operator.
6. Before retrieving the samples the sample panel is flushed of any excess sample.
7. A spray head is provided so that the insides of the sample panel can be remotely washed down in the event of a leak.

8. The sample panel sump collects any liquids which are discharged by the sample panel, and the sump pump returns the liquid to Waste Evaporator Feed Tank Sump A.
 9. The sample panel weighs approximately 800 lbs.
- C. The PALS is designed to handle two (NC, ND or containment sump) sampling points. Each sampling point will have its own inlet valve and demineralized water flush valve, which are external to the sample panel. These external valves must be able to handle the highest system pressure that will be seen by the panel.
- D. The motive force for the sample is supplied by the pressure of the system being sampled or by pumps which are external to the sample panels.
- E. Flow through the sampling system is controlled by a high pressure regulator and needle valve. In the event of failure of the high pressure regulator or the needle valve flow through the system is throttled by the sample cooler.

III. ITEMS TO BE PROVIDED FOR THE PALS

A. Services

1. Instrument Air, 1.4 SCFM at 100 PSIG.
2. Demineralized Water, 10 GPM at 80 PSIG.
3. Nitrogen, 2 SCFM at 110 PSIG.
4. Cooling Water, 10 GPM at 250 PSIG maximum. If the sample cooler should fail the cooling water could be contaminated.
5. 115 VAC, 20 A single phase service to the control panel.

B. Interconnecting wiring

1. No. 14 AWG, 51 conductors, (solenoid valves).
2. 2 pair, shielded, Type J thermocouple wire.
3. 3 twisted pair, shielded, (transmitters, flow meters).
4. 12 conductors, shielded, instrument wire (Radiation Monitor, Conductivity and PH Probes).
5. 115 VAC, 1.5 A control wires for valves and pumps external to the sample panel.

- C. Valves and Pumps not supplied with the sample panel:
 - 1. All sample inlet valves external to the sample panel.
 - 2. All demineralized water flush valves external to the sample panel.
 - 3. All pumps for moving the sample to a sample panel.
- D. Inlet and Outlet Sample Lines for the Sample Panel
- E. A duct for venting away any possible radioactive gases that could escape during sampling.
- F. The control panel and the sampling panel must be seismically anchored.

IV. SYSTEM OPERATION DESCRIPTION

A. Control Switches

- 1. These switches are located behind the front upper left control panel.

PC Power - Disconnects the hot leg from the P.C., but not the racks.

Dilute Samplers Purge - Energizes valves 109 and 128 so that the liquid and gas samplers may be cleaned in Flush and Drain steps.

Panel Spray - Energizes the valve which admits water to the spray head. Also prevents operation of any other function when energized.

System Power - Disconnects both line and neutral from the system.

- 2. These switches are located on the bottom section of the left front control panel.

System Power - The left hand position operates the sump pump and the right hand position supplies power to the control panel.

Selection Power - Initiates selected function operation when depressed.

- 3. This rotary switch is located in the center of the left front control panel.

Selector Switch - Indicates which function will be implemented when "Selection Power" pushbutton is depressed. Stops all function operations when switched from one function to another.

B. Functions

1. Panel Prep.

When the system is shut down, the pH probe is stored in Demineralized (D.M.) water. This water must be removed and replaced with a calibrating fluid, so that operation of the probe may be checked. The calibration fluid must then be cleared out so that the probe will be ready to receive the sampling fluid. Evacuation of the 150ml cylinder is started. Cooling water to the sample panel is controlled by manual valves external to the sample panel.

- a. Calibrate button - Adds calibration fluid to the pH probe.
- b. Purge button - Uses N_2 to push fluid out of the pH probe into the panel sump.
- c. Flush button - Flows D.M. water through the pH probe.
- d. Drain button - Relieves pressure within the pH probe.

2. Sample Trap

Sample fluid flow is started. Evacuation of the 150ml cylinder down to 25 ± 2 inches Hg is completed and sample fluid is trapped at system pressure in the 95ml cylinder.

- a. Pressurize - Stops sample flow and allows the 95ml cylinder to reach full system pressure.
- b. Trap - Traps sample fluid in the 95ml cylinder at system pressure.

3. Depressurization

The valve between the 95ml cylinder and the 150ml cylinder opens, which instantly depressurizes the sample fluid. N_2 is bubbled up through the bottom of the 95ml cylinder to help in stripping the gases, which come out of solution, and to bring the 150ml cylinder up to 15 ± 1 inches Hg.

- a. Gas Stripping Start - Admits N_2 through the bottom of the 95ml cylinder.

b. Gas Stripping Stop - Stops N_2 from entering the 95ml cylinder.

4. Liquid Sample

D.M. water flow through the turbine flow element into the sump is started. This is done because it takes 10 seconds for the flow linearizer to respond when the flow has been stopped. Once the flow linearizer is responding the flow can be totalized.

N_2 is used to move the sample fluid from the 95ml cylinder through the 5.4 ml sample loop and into the pH chamber. This is allowed to happen for two seconds which is just enough time for the sample loop to be full and free of any bubbles. At the end of two seconds the three valves at the ends of the sample loop are energized, thus lining the sample loop up with the turbine flow meter and the 3,785ml cylinder.

D.M. water is used to dilute and push the 5 milliliters of sample out of the sample loop and into the 3.785l cylinder. The flow of D.M. water is controlled and measured by the totalizer. N_2 is used to push all of the dilution water out of the sample loop and to bubble through the bottom of the 3,785ml cylinder to ensure complete mixing of the sample with the D.M. water.

The gas sampler is opened and given time to fill with sample gas from the 150ml cylinder. Pressure in the system will drop from 10"Hg to 0"Hg.

When the Gas Sampler is opened, a flush of the sample line is started. The valve that actuates to admit the D.M. water is delayed until all of the valves to the drain are opened. This prevents anything from being pushed into the D.M. water system by the pressure trapped in the sample line.

- a. Sample Grab - Allows flow through the sample loop for two seconds, thus filling the loop.
- b. D.M. Flow Meter Start - Admits dilution D.M. water.
- c. D.M. Flow Meter Stop - Stops the flow of D.M. water.

- d. Mixing - Admits N_2 through the sample loop and into the bottom of the Dilute Tank.
- e. pH - Admits the remaining sample fluid from the 95ml cylinder to the pH probe.
- f. Trap Open - Opens the Gas Sampler and starts the sample line flush.
- g. Trap Close - Closes the Gas Sampler.

5. Liquid Sample

N_2 causes the diluted sample to flow from the 3.785ℓ cylinder through the Liquid Sampler. After enough time has elapsed to fill the Liquid Sampler, the flow is stopped and the Liquid Sampler and Dilute Tank are depressurized. The Liquid Sampler is then closed.

- a. Trap Open - When held in, allows N_2 to enter the top of the Dilute Tank and the Liquid Sampler to fill. Released, the flow is stopped and the Dilute Tank is depressurized along with the liquid Sampler.
- b. Trap Close - Closes the Liquid Sampler.

6. Flush

D.M. water is used in four steps to flush excess sample out of the Sample Panel. The "Step" pushbutton is depressed to advance to the next flush step.

7. Drain

N_2 is used in three steps to push remaining flush water into the Sample Panel Sump. The fourth step fills the pH probe with D.M. water and vents the 150ml cylinder and 3.785ℓ cylinder. When depressed, the "Step" pushbutton advances the system to the next drain step.

V. SYSTEM OPERABILITY AND MEASUREMENT RESULTS

A. Basis for Acceptable Operation

The basis for acceptable PALS operation were established in an October 5, 1984 letter from NRC/OIE Region II to Duke Power Company (Reference Report Nos. 50-369/84-07 and 50-370/84-07). The criteria are summarized below:

1. "Gross Activity (gamma spectrum of the fission product gases or fission products in the coolant measured to estimate core damage): These analyses shall be accurate within a factor of two over the range of coolant activity 1 μ Ci/gm - 10 Ci/gm."
2. "Boron: This analysis shall be accurate within $\pm 10\%$ of the measured value. The $\pm 10\%$ accuracy for this analysis accounts for inaccuracies in sample dilution and laboratory analysis techniques. For concentrations below 500 ppm, the tolerance band remains at ± 50 ppm."
3. "Chlorides: For concentrations between 0.5 and 20.0 ppm chloride, the analysis shall be accurate within $\pm 10\%$ of the measured value. At concentrations below 0.5 ppm, the tolerance band remains at ± 0.05 ppm".
4. "Hydrogen: For concentrations less than or equal to 50 cc/kg, the analysis shall be accurate within ± 15 cc/kg. For concentrations between 50 cc/kg and 2000 cc/kg, the analysis shall be accurate within ± 20 percent."
5. "Oxygen: The smallest concentration measurable should be at least 250 ppb. Concentrations above 250 ppb should be accurate within ± 10 percent. Analysis for dissolved oxygen is recommended by NUREG 0737 but is not a requirement."
6. "pH: Between a pH of 5 to 9, the reading shall be accurate within ± 0.3 pH units. For all other ranges, ± 0.5 pH units in acceptable."
7. "When parameters such as oxygen or chlorides are not present in the undiluted sample, an evaluation shall be performed to ensure the indicated accuracy could be met for the diluted sample."

B. Discussion of Methods/Results

PALS testing conducted from March 17 to March 27, 1985 yielded the data in Table 13.2-1. These data indicate that the PALS provides results within the required limits, except for oxygen, as stated above.

The following is a brief description of the methods employed to obtain the results in Table 13.2-1.

1. Gross Activity: Presently, the isotopic levels encountered in the reactor coolant are very low. This is due to Catawba's relatively young reactor core. Therefore, the counting statistical error for the diluted PALS sample is high. In this instance, an assessment of

the achievable accuracy was made. The major source of counting error is in the dilution of the collected sample by PALS. However, the data obtained from the boron and hydrogen analyses were well within the limits specified. Therefore, it can be concluded that the dilution error for both the gas and liquid isotopic samples are acceptable. We feel that once reactor coolant activities reach sufficient levels, the "factor of two" accuracy criterion can be attained. At present, F-18 exhibits the highest activity in our reactor coolant. The F-18 data for the gas and liquid samples can be found in Table 13.2-1. In all cases, the "factor of two" accuracy was obtained for the liquid samples (highest activity).

2. Boron: Boron is determined by a pH titration of boric acid in the presence of mannitol. As can be seen from Table 13.2-1, the boron analysis accuracies are within the required limits.
3. Chlorides: Chloride levels in the reactor coolant during power operations were less than detectable. Therefore, at present, only an assessment of the analytical accuracy can be made. The acceptable criteria for this analysis has been set at less than detectable. Chloride determinations are made using a chloride ion selective electrode with a sensitivity of 10 ppb. All ten PALS samples exhibited less than detectable amounts (<10 ppb) of chlorides (see Table 13.2-1). These results also verify the absence of any chloride contamination in the PALS. In an accident, a portion of the diluted PALS sample would be sent to the Physical Sciences Building for chloride determination by IC (ion chromatography). The method is well established for its reliability and the method has been verified as acceptable for a typical post accident matrix. (Reference: Evaluation of GE and SEC Chemical Procedures for Post Accident Analysis of Reactor Coolant Samples- Prepared by Exxon Nuclear Idaho Co., Inc. for the NRC).
4. Hydrogen: Hydrogen is determined by analyzing a portion of the stripped gas sample by gas chromatography. Sample accuracy was within the ± 15 cc/kg limit specified (see Table 13.2-1).
5. Oxygen: Oxygen analyses were attempted on portions of the PALS stripped gas sample by gas chromatography during McGuire's PALS testing. The results that McGuire obtained during their testing were unacceptable. It was therefore concluded that the current design of McGuire's PALS makes it impossible to perform accurate oxygen analyses. Since Catawba's PALS design is similar to McGuire's PALS, there are no plans to pursue oxygen analysis due to the poor performance encountered during McGuire's PALS testing. It should be noted that oxygen analysis is not a NUREG 0737 requirement.

6. pH: The pH of the PALS samples are taken on a portion of the undiluted sample by an in-line pH probe. The specified accuracy of ± 0.3 pH units was met by all the samples analyzed (see Table 13.2-1).

POST ACCIDENT LIQUID SYSTEM
MEASUREMENT RESULTS

Date 3-17-85
Time 1116

	H ₂ cc/kg	BORON ppm	Cl ⁻ ppb	pH	F-18 μCi/cc	
					Gas	Liq
PALS	29.5	759	*LTD	6.3	3.52E-3	3.00E-2
NC _{hot leg A}	33.9	721	*LTD	6.2	1.06E-3	5.6E-2
**DEVIATION	4.4	5.3%	-	0.1	3.3	1.9

Date 3-19-85
Time 1044

	H ₂ cc/kg	BORON ppm	Cl ⁻ ppb	pH	F-18 μCi/cc	
					Gas	Liq
PALS	24.1	692	*LTD	6.4	LTD	2.04E-2
NC _{hot leg A}	34.0	693	*LTD	6.1	LTD	1.50E-2
**DEVIATION	9.9	0.1%	-	0.3	-	1.4

Date 3-22-85
Time 0906

	H ₂ cc/kg	BORON ppm	Cl ⁻ ppb	pH	F-18 μCi/cc	
					Gas	Liq
PALS	21.4	697	*LTD	6.4	1.64E-3	3.86E-2
NC _{hot leg A}	18.9	705	*LTD	6.3	1.06E-3	5.19E-2
**DEVIATION	2.5	1.1%	-	0.1	1.5	1.3

Date 3-22-85
Time 1053

	H ₂ cc/kg	BORON ppm	Cl ⁻ ppb	pH	F-18 μCi/cc	
					Gas	Liq
PALS	22.8	692	*LTD	6.5	2.58E-3	3.88E-2
NC _{hot leg A}	18.9	705	*LTD	6.3	1.06E-3	5.19E-2
**DEVIATION	3.9	1.8%	-	0.2	2.4	1.3

*LTD = Less than detectable

Table 13.2-1

CATAWBA 1

POST ACCIDENT LIQUID SYSTEM
MEASUREMENT RESULTS

Date 3-25-85
Time 1054

	H ₂	BORON	Cl ⁻	pH	F-18 μ Ci/cc	
	cc/kg	ppm	ppb		Gas	Liq
PALS	31.7	715	*LTD	6.4	1.52E-3	4.36E-2
NC _{hot leg A}	22.9	695	*LTD	6.5	4.16E-3	6.14E-2
**DEVIATION	8.8	2.9%	-	0.1	2.7	1.4

Date 3-25-85
Time 1310

	H ₂	BORON	Cl ⁻	pH	F-18 μ Ci/cc	
	cc/kg	ppm	ppb		Gas	Liq
PALS	24.9	642	*LTD	6.5	6.80E-3	3.68E-2
NC _{hot leg A}	22.9	695	*LTD	6.5	4.16E-3	6.14E-2
**DEVIATION	2.0	7.6%	-	0.0	1.6	1.7

Date 3-25-85
Time 1353

	H ₂	BORON	Cl ⁻	pH	F-18 μ Ci/cc	
	cc/kg	ppm	ppb		Gas	Liq
PALS	22.1	726	*LTD	6.4	4.45E-3	4.34E-2
NC _{hot leg A}	22.9	695	*LTD	6.5	4.16E-3	6.14E-2
**DEVIATION	0.8	4.5%	-	0.1	1.1	1.4

Date 3-27-85
Time 0915

	H ₂	BORON	Cl ⁻	pH	F-18 μ Ci/cc	
	cc/kg	ppm	ppb		Gas	Liq
PALS	25.1	665	*LTD	6.4	5.40E-3	4.76E-3
NC _{hot leg A}	21.9	702	*LTD	6.5	4.39E-3	7.84E-3
**DEVIATION	3.2	5.3%	-	0.1	1.2	1.6

*LTD = Less than detectable

Table 13.2-1 (continued)

CATAWBA 1

POST ACCIDENT LIQUID SYSTEM
MEASUREMENT RESULTS

Date 3-27-85
Time 1015

	H ₂	BORON	Cl ⁻	pH	F-18 μ Ci/cc	
	cc/kg	ppm	ppb		Gas	Liq
PALS	25.1	738	*LTD	6.4	5.07E-3	5.23E-2
NC _{hot leg A}	21.9	702	*LTD	6.5	4.39E-3	7.84E-2
**DEVIATION	3.2	5.1%	-	0.1	1.2	1.5

Date 3-27-85
Time 1052

	H ₂	BORON	Cl ⁻	pH	F-18 μ Ci/cc	
	cc/kg	ppm	ppb		Gas	Liq
PALS	22.3	726	*LTD	6.4	6.50E-4	6.44E-2
NC _{hot leg A}	21.9	702	*LTD	6.5	4.39E-3	7.84E-2
**DEVIATION	0.4	3.4%	-	0.1	6.8	1.2

*LTD = Less than detectable

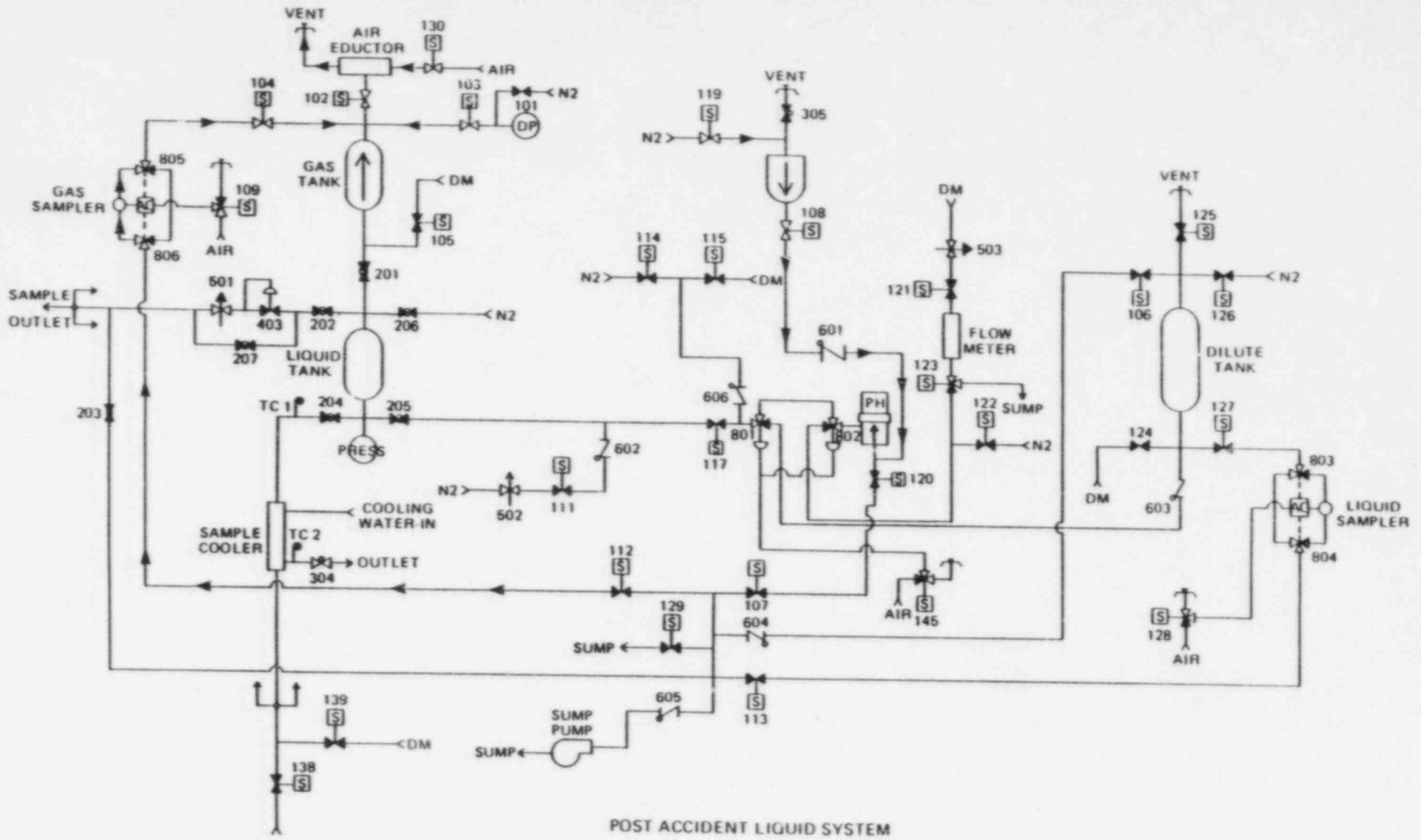
**DEVIATION = H₂ deviation is expressed in cc/kg.

Boron deviation is expressed in percent.

pH deviation is expressed in pH units.

Both gas and liquid F-18 deviations are expressed as a factor.

POST ACCIDENT LIQUID SAMPLING SYSTEM SCHEMATIC



POST ACCIDENT LIQUID SYSTEM

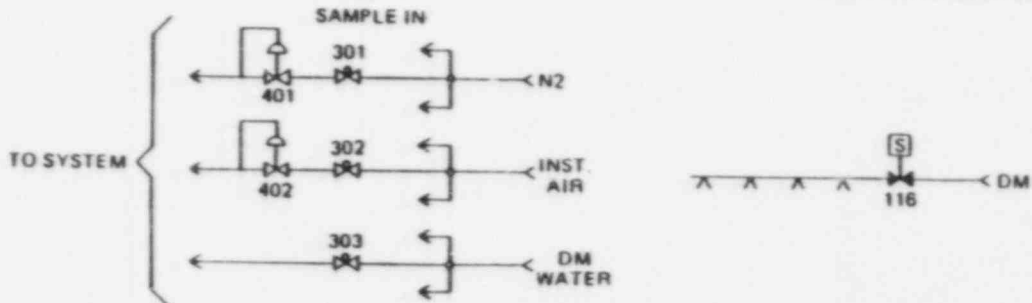


Figure 13.2-1