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NYN-90178

September 2[illegible] 1990

United States Nuclear Regulatory Commission
Washington, D.C. 20555

Attention: Document Control Desk

References: (a) Facility Operating License No. NPF-86, Docket No. 50-443

(b) NHY Letter NYN-90070, "Initial Startup Report" dated March 13, 1990, T. C. Feigenbaum to USNRC

(c) NHY Letter NYN-90126, "Supplement 1 to the Initial Startup Report" dated June 13, 1990, T. C. Feigenbaum to USNRC

(d) NHY Letter NYN-90167, "Supplement 2 to the Initial Startup Report" dated September 13, 1990, T. C. Feigenbaum to USNRC

Subject: Supplement 3 to the Initial Startup Report

Gentlemen:

In accordance with the requirements of Technical Specification 6.8.1.1 enclosed is Supplement 3 to the Initial Startup Report submitted via Reference (b) and supplemented via References (c) and (d). Supplement 3 to the Initial Startup Report covers the period from June 1990 through August 1990. The Power Ascension Test Program was completed on August 18, 1990.

Should you have any questions regarding this report please contact Mr. James M. Peschel, Regulatory Compliance Manager at (603) 474-9521 extension 3772.

Very truly yours,

Bruce L Drawbridge for

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Enclosure

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NEW HAMPSHIRE YANKEE

SEABROOK STATION

SUPPLEMENT 3

to

INITIAL STARTUP REPORT

to the

UNITED STATES

NUCLEAR REGULATORY COMMISSION

OPERATING LICENSE: NPF 86

NRC DOCKET NO. 50-443

For the Period

June, 1990

through

August, 1990

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LIST OF ACRONYMS

ACOT - Analog Channel Operational Test

AFD - Axial Flux Difference

AO - Auxiliary Operator

ARI - All Rods Inserted

ARO - All Rods Out

ASDV - Atmospheric Steam Dump Valve

CB_ - Control Bank ______ (A,B,C, or D)

CE - Combustion Engineering

CIV- Combined Intermediate Valve

CRD - Control Rod Drive

CRDM - Control Rod Drive Mechanism

CV - Control Valve

CVCS - Chemical and Volume Control System

DRPI - Digital Rod Position Indication

EBOP - Emergency Bearing Oil Pump

ECCS - Emergency Component Cooling System

EFPM - Effective Full Power Minute

EFW - Emergency Feedwater

EHC - Electrohydraulic Control

ELOP - Emergency Lube Oil Pump

ESOP - Emergency Seal Oil Pump

FCFM - Full Core Flux Map

EPS - Emergency Power Sequencer

FSAR - Final Safety Analysis Report

FTC - Fuel Temperature Coefficient

GETARS - General Electric Transient Analysis Recording System

HFT - Hot Functional Test

LIST OF ACRONYMS (Continued)

HSB - Hot Standby

HX - Heat Exchanger

HZP - Hot Zero Power

I&C - Instrumentation and Control

ICRR - Inverse Count Rate Ratio

IV - Intercept Valve

IR - Intermediate Range

ITC - Isothermal Temperature Coefficient

LPMS - Loose Parts Monitoring System

LPDS - Loose Part Detection System

MCB - Main Control Board

MIDS - Movable Incore Detector System

MSIV - Main Steam Isolation Valve

MTC - Moderator Temperature Coefficient

MWE - Megawatts Electric

MWT - Megawatts Thermal

NDR - Nuclear Design Report

NIS - Nuclear Instrumentation System

NPDES - National Pollutant Discharge Elimination System

NR - Narrow Range

NRC - Nuclear Regulatory Commission

NSSS - Nuclear Steam Supply System

OTDeltaT - Over Temperature Delta-Temperature

PATP - Power Ascension Test Program

PCV - Pressure Control Valve

PCCW - Primary Component Cooling Water

PLS - Precautions, Limitations and Setpoints

LIST OF ACRONYMS (Continued)

PLU - Power Load Unbalance

PORV - Power Operated Relief Valve

PR - Power Range

RAT - Reserve Auxiliary Transformer

RCCA - Rod Cluster Control Assembly

RCS - Reactor Coolant System

RDMS - Radiation Data Management System

RHR - Reactor Heat Removal

RMO - Remote Manual Operation

RSS - Remote Safe Shutdown

RTD - Resistance Temperature Detector

RTP - Rated Thermal Power

RVLIS - Reactor Vessel Level Indication System

SB_ - Shutdown Bank ______ (A,B,C,D and E)

SG_ - Steam Generator _____ (A,B,C and D)

SGFP - Steam Generator Feed Pump

SORC - Station Operating Review Committee

SR - Source Range

SSPS - Solid State Protective System

SSCP - Seabrook Station Chemistry Program Manual

TBWD - Thrust Bearing Wear Detector

TEC - Technology for Energy Corp.

UAT - Unit Auxiliary Transformer

UE&C - United Engineers and Constructors

UPS - Uninterruptible Power Source

URAL - Underexcited Reactive Ampere Limit

WR - Wide Range

1.0 INTRODUCTION

The Initial Startup Report was submitted to the Nuclear Regulatory Commission in March, 1990 and covered startup activities through completion of low power physics testing (June 1989). Supplement 1 reported testing which took place in the interval July 1989 through May 1990. Supplement 2 was a summary document covering the tests reported in detail in Supplement 3. Supplement 3 covers the remainder of the Power Ascension Test Program from June 1990 through August 1990, and completes the initial startup documentation as required by NRC Reg. Guide 1.16, Section C, Part 1a.

Approximately eight months elapsed between completion of the low power physics tests and receipt of a full power license; the full power license was received on March 15, 1990 and the power ascension test program was undertaken promptly thereafter.

After completion of preparations to begin the ascension to the 30% power level test plateau, the first section of procedure ST-48, Turbine Generator Startup Test was run and ST-48.1, Turbine Generator Torsional Response Test started. An unsatisfactory resonance was detected by the latter test, and the PATP was suspended for approximately one month to allow GE turbine personnel to modify low pressure turbine rotor "C". Upon completion of the modifications, a retest of the turbine, using a revised ST-48.1, was conducted and satisfactory performance obtained.

Supplement 1 to the Initial Startup Report covers testing through ST-48.1. The turbine was synchronized onto the grid, but actual ascension to 30% power was not included.

A summary of testing was reported in Supplement 2 which covered entrance into the 30% power level test plateau through the NSSS Acceptance Test (ST-40), the final test in the PAT sequence.

Supplement 3 covers the Supplement 2 test period in detail, and is the final supplement to the Initial Startup Report.

The following tests are included in Supplement 3:

- ST-13, Operational Alignment of Nuclear Instrumentation
- ST-14.1, Operational Alignment of the Process Temperature Instrumentation
- ST-15, Reactor Plant System Setpoint Verification
- ST-22, Natural Circulation Test
- ST-24, Automatic Reactor Control
- ST-25, Automatic Steam Generator Level Control
- ST-26, Thermal Power Measurement and Statepoint Data Collection
- ST-27, Startup Adjustments of Reactor Control System
- ST-28, Calibration of Steam and Feedwater Flow Instrumentation
- ST-29, Core Performance Evaluation
- ST-30, Power Coefficient Measurement
- ST-33, Shutdown from Outside the Control Room
- ST-34, Load Swing Test
- ST-35, Large Load Reduction
- ST-36, Axial Flux Difference Instrumentation Calibration

1.0 INTRODUCTION (Continued)

Tests included in Supplement 3 (Continued):

ST-37, Steam Generator Moisture Carryover Measurement
ST-38, Unit Trip from 100% Power
ST-39, Loss of Offsite Power Test
ST-40, NSSS Acceptance Test
ST-41, Radiation Survey
ST-42, Water Chemistry Control
ST-43, Process Computer
ST-44, Loose Parts Monitoring
ST-45, Process Effluent Radiation Monitoring System
ST-46, Ventilation System Operability Test
ST-48, Turbine Generator Startup Test
*ST-49, Circulating Water System Thermal-Hydraulic Test
ST-51, Power Ascension Dynamic Vibration Test
ST-52, Thermal Expansion
ST-56, Piping Vibration Testing

* Test deferred. Testing will be performed prior to operation of the circulating water heat treatment.

2.0 STARTUP TEST PROGRAM OVERVIEW, SUPPLEMENT 3

The Initial Startup Report, submitted to the Nuclear Regulatory Commission on March 13, 1990, covered that portion of the startup sequence through low power physics testing. Supplement 1, was prepared to report testing which took place in the three months after the initial report. Supplement 2 was a summary document covering the tests reported in detail in Supplement 3. Supplement 3 reports those startup activities after Supplement 1, through completion of the Power Ascension Test Program.

Startup tests which had sections previously reported, but carry over into the test sequences covered in Supplement 3, are included as complete tests.

A full power license was received on March 15, 1990.

Portions of several startup tests involving alignments and setpoints were required in the startup sequence prior to increasing power to the 30% power level plateau. These tests were completed and the Station Operating Review Committee (SORC) authorized entry into the 30% test sequence on March 25, 1990. During power ascension, the initial turbine generator tests were scheduled in the 8%-20% power range. A turbine generator torsional response test had been recommended by General Electric (GE), the turbine vendor, and after turbine rolls to validate turbine protective systems and make adjustments to control systems, the torsional response test was undertaken.

The torsional response test disclosed an undesirable resonance, requiring modification to the "C" low pressure turbine. Power ascension testing was interrupted on April 27, 1990 for turbine modification and resumed on May 25, 1990.

When the turbine was reassembled after modification, a revised torsional response test was performed to verify correction of the resonance problem. Testing through completion of ST-48.1, Turbine Generator Torsional Response Test, was covered in Supplement 1. Supplement 2 was a summary document covering the remainder of the PAT sequence; reported in detail in Supplement 3, the final supplement in the Initial Startup Report.

Supplement 3 reports ascension to the 30% power level test plateau and the remaining tests in the power ascension sequence. The 10%-30% turbine tests, preliminary to testing at 30% power, were completed on June 4, 1990.

Section 5.0, Power Ascension Testing, includes those tests related to primary and secondary plant performance. Many tests in this sequence, usually those identified with one aspect of plant systems, such as S/G level control, were conducted at every plateau, and related control systems readjusted as required. Others, such as the loss of offsite power, and unit trip from 100%, were performed only once, and evaluated many plant systems and system interactions.

At the 30% and subsequent power level test plateaus, several startup tests were conducted to verify or readjust instrument ranges or setpoints. In some cases data from these tests were entered into other tests; in general, tests in this category are found in Section 6.0, Instrument Calibration and Alignment.

Other tests to validate plant support systems, such as the radiation shield design and ventilation systems; are included in Section 7.0, General Plant Testing, and were sequenced at appropriate test plateaus. For example, ST-41, Radiation Survey, was performed at the 50% and 100% power level test plateaus, but not at 30%, where reduced radiation levels would mean fewer useful data points.

In general, testing of the primary and secondary plant systems went smoothly, with only minor problems and interruptions. Turbine adjustment and fine tuning, primarily at the 30% plateau, was the major challenge early in the test sequence. Later, condensate pump heater drain pump, and feedwater heater interactions contributed to control problems in the secondary plant, limited progress.

Testing of the NSSS portion of the plant yielded expected results, almost without exception. The availability of GETARS for data acquisition provided test personnel with fast and very accurate information. At no time did data analysis contribute significantly to delays in the program sequence.

The NSSS Acceptance Test, ST-40, was started on August 5, 1990 at 1700 hours, and completed on August 17, 1990 at 1800 hours. The test was interrupted for two hours on August 6 for stop and control valve surveillance tests at a power level of < 95%, and for approximately 1½ days on August 13 for repair of a leak in an EHC line. Later, on August 16, a steam leak in a S/G blowdown line required isolation of the flash tank, but the line was repaired without reducing power.

Results of the acceptance test were reviewed following the warranty run, and the Power Ascension Test Program was officially completed at 2400 hours, August 18, 1990.

3.0 SEABROOK STARTUP CHRONOLOGY, SUPPLEMENT 3

This chronology documents the Power Ascension Test Program (PATP) from test resumption following main turbine modifications to completion of PATP.

Alignment and setpoint adjustments per ST-13, ST-14.1 and ST-15 occur frequently in the test sequence. These entries have not been included in the chronology.

Date	Event
5/30/90	Testing underway; transfer from bypass to main feedwater valves (ST-25), overspeed testing (ST-48).
	Test interruption for turbine overspeed trip rework.
6/4/90	Completion of 10%-30% load data collection (ST-48).
	Ascension to 30% power.
6/5/90	Verification of S/G feedwater pump auto speed control (ST-25); completion of 30% thermal expansion data (ST-52).
6/6/90	Turbine shutdown; arcing in generator isophase bus duct.
6/7/90	Verification of main feedwater reg valve stability (ST-25); preparation of MIDS for flux mapping (ST-29).
6/9/90	Water chemistry sampling (ST-42); flux mapping at 30% completed (ST-29) and first power coefficient determination (ST-30); completed valve and feedwater pump testing at this plateau (ST-25); initial MPCS data taken (ST-43).
6/10/90	Load swings of 10% (ST-34); automatic reactor control verification (ST-24).
	SORC approval received for ascension to the 50% power level test plateau.
6/11/90	Completion of 30%-50% load data collection (ST-48); shield survey underway (ST-41).
	Fifteen hours S/G chemistry holdup.
6/12/90	Statepoint data collection (ST-26); steam feedwater flow calibration (ST-28).
6/13/90	Adjustments to reactor control system (ST-27); completed thermal expansion observations (ST-52); core performance evaluation completed (ST-29); radiation surveys (ST-41) and chemistry sampling (ST-42) underway.
	Feedwater heater level fluctuations prevent operation of MFP-B per ST-25. Operations starts second heater drain pump to improve heater drain tank level control.

6/14/90 Calibration of AFD instrumentation (ST-36); loose parts monitoring (ST-44) and ventilation system operability test (ST-46) underway.

Power decrease of 100 MWe transient due to turbine control problem.

6/15/90 Power coefficient determination (ST-30); 10% load swings (ST-34).

6/16/90 Shutdown from outside the control room (ST-33); MPCS data acquisition complete (ST-43).

6/18/90 Completed 50% power level AFD calibrations (ST-36).

6/20/90 SORC approval received for ascension to 75% power level plateau.

Unplanned turbine trip/reactor trip due to a fault in generator protective relaying.

6/25/90 Restored criticality.

6/27/90 Repeated ST-48 30%-50% load data collection.

Oscillations of feedwater heater level when second MFP (A) placed in service (second heater drain pump placed in service to dampen oscillations).

6/28/90 Additional flux map performed.

6/29/90 Flux mapping in progress.

6/30/90 Heater drain piping leak, manual turbine shutdown.

7/5/90 Commenced testing at 75% power; statepoint data (ST-26) and S/G level control (ST-25) completed.

Unplanned reactor trip due to high vibration on EHC system pressure switches.

7/7/90 Reactor entered Mode 1.

7/8/90 Returned to 75% power.

7/9/90 Completed 50%-70% load data collection (ST-48); thermal expansion evaluation (ST-52); steam and feedwater flow calibration (ST-28); water chemistry sampling (ST-42).

7/10/90 Flux mapping (ST-29 and ST-36).

7/11/90 During power coefficient measurement (ST-30), throttle pressure limiter interference caused megawatt reduction, and a test interruption.

Completed 10% load swing at 75% (ST-34).

7/12/90 Power coefficient determination (ST-30); large load reduction (ST-35).

7/13/90 AFD calibration (ST-36); plant computer validation (ST-43).

7/14/90 T_{REF} program change completed; reactor control system adjustments completed for 75% (ST-27).

SORC approval received for ascension to the 100% power level test plateau. PATP Management hold for 90% testing.

70%-90% load data collection (ST-48); flux mapping (ST-29).

7/16/90 Statepoint data collection at 90% power (ST-26).

Reduced power to 75% temporarily, for Operations to test Main Steam Control Valve; process temperature alignment (ST-14.1) and steam/feedwater flow calibration (ST-28).

7/19/90 Returned power level to 90% and conducted additional process temperature alignment (ST-14.1).

PATP Management approval received for ascension to the 100% power level test plateau.

7/20/90 Feedwater heater oscillations forced power reduction to 90%. After approximately six hours, return to 100% started; high feedwater flow oscillations required manual control of S/Gs A, B and D. New gain settings requested for controllers.

7/21/90 Returned to 100% power level test plateau.

7/22/90 100% load data collection (ST-48); verification of feedwater pump speed control (ST-25); statepoint data collection (ST-26); chemistry sampling (ST-42); shield survey (ST-41); moisture carryover test (ST-37); baseline loose parts monitoring (ST-44); ventilation system (ST-46).

7/23/90 Thermal expansion (ST-52) and piping vibration (ST-56) completed; process effluent data collection (ST-45); core performance flux maps (ST-29).

7/26/90 Load swings (ST-34) and Large Load Rejection (ST-35) accumulated > 60 AFD penalty minutes required power to remain below 50% for 24 hours.

7/28/90 Returned to 100% power level; preparations for unit trip from 100% (ST-38).

7/29/90 Unit trip from 100% power (ST-38) coordinated with process computer (ST-43), LPMS (ST-44) and vibration measurements (ST-51); natural circulation test (ST-22).

7/31/90	During restart, high vibration on turbine bearings required power reduction and then a manual turbine shutdown. After four hours on turning gear, turbine was resynchronized.
8/1/90	Loss of offsite power test from 20% RTP (ST-39).
8/3/90	Returned plant to service; preparation for ST-25.1, single feedwater pump capacity test.
8/4/90	ST-25.1 (Main Feed Pump Flow Capacity Test) interrupted (and later cancelled) when MFP A suction pressure dropped and was accompanied by feedwater heater level instabilities.
8/5/90	Attempt to reach 100% RTP on two condensate pumps; third pump (in automatic) started on low suction pressure at 88% RTP.
	Commenced 250 hour warranty run (ST-40).
8/7/90	Control system adjustments (ST-27).
	Precision calorimetric measurement (ST-40).
8/13/90	EHC piping leak; turbine shutdown to effect repair.
8/14/90	Generator back on line.
8/15/90	Recommenced warranty run.
8/16/90	Two inch crack found in S/G C blowdown piping reducer; flash tank isolated for repair.
	Completed process computer data (ST-43).
8/17/90	Blowdown returned to service.
	Warranty run completed (ST-40).
8/18/90	Review of warranty run; startup program officially completed.

4.0 SUMMARY OF INITIAL STARTUP REPORT, SUPPLEMENT 3

The Power Ascension Test phase of initial startup was initiated with the receipt of a full power license on March 15, 1990. Supplement 3 reports testing beginning with the approach to the 30% power level test plateau and continuing to the completion of the PAT sequence.

Supplement 2 was a summary document covering the tests reported in detail in Supplement 3.

Those tests which were conducted at several power level test plateaus, and were reported in the original startup report and/or in Supplement 1, are reported in Supplement 3 completely, for continuity. In some cases, less detail is included for sections previously reported. Following is a list of startup tests which were completed prior to the time frame covered by Supplement 3; these tests are discussed in the earlier documents and are not discussed herein:

ST-2,	Primary Source Installation
ST-3,	Core Loading Prerequisites
ST-4,	Initial Core Loading
ST-5,	Control Rod Drive Mechanism Operational Test
ST-6,	Rod Control System
ST-7,	Rod Drop Time Measurements
ST-8,	Rod Position Indication
ST-9,	Pressurizer Spray and Heater Capability
ST-10,	RTD Bypass Loop Flow Verification
ST-11,	Reactor Coolant System Flow Measurement
ST-12,	Reactor Coolant System Flow Coastdown
ST-14.2,	Resistance Temperature Detector and Incore Thermocouple Cross Calibration
ST-16,	Initial Criticality
ST-17,	Boron Endpoint Measurement
ST-18,	Isothermal Temperature Coefficient
ST-19,	Flux Distribution Measurements at Low Power
ST-20,	Control Rod Worth Measurements
ST-20.1,	Additional Control Rod Worth Measurements
ST-21,	Pseudo Rod Ejection Test
ST-23,	Dynamic Automatic Steam Dump Control
ST-48.1,	Turbine Generator Torsional Response Test
ST-50,	Movable Incore Detector System
ST-53,	Turbine Driven Emergency Feedwater Start Verification
ST-55,	Steam Dump System Test

In the 30% sequence, reactor power was increased enough to bring the turbine on line (8-10%), and after a detailed checkout, to allow synchronization of the generator to the grid. Only after completion of the turbine generator qualification testing did actual ascension to and testing at 30% power begin. As was reported in Supplement 1, power ascension testing was interrupted on April 27, 1990 for turbine modification, and resumed on May 25, 1990.

4.0 SUMMARY OF INITIAL STARTUP REPORT, SUPPLEMENT 3 (Continued)

Power ascension testing moved rapidly through the several test plateaus; the longest interruption was about four days. At the start of 30% testing, turbine overspeed problems required three days, and a few days later, as power escalation to the 50% plateau was underway, a chemistry hold of two shifts was necessary as turbine drains and other piping systems previously not in service received steam and were flushed of residual foreign matter. Water chemistry testing (ST-42) results were typical of systems just coming on line. Cleanup of secondary systems was accelerated by the use of auxiliary, trailer-mounted carbon filter and resin beds, which were frequently changed as needed.

Two unplanned reactor trips, the first at approximately 30% power and the second at the 75% power level test plateau, interrupted testing for 4 days, due to a fault in generator protective relaying, and 3 days as a result of high vibration in EHC system pressure switches, respectively. Turbine setbacks were responsible for test delays of 1-2 shifts.

Fine tuning of the secondary plant, particularly the feedwater train, was the primary activity outside the testing sequence itself. Initially feedwater valve instability required attention, then feedwater heater level-heater drain tank level fluctuations, and problems in bringing two heater drain pumps into service. Smooth operation with two main feedwater pumps required additional care, and for full power operation, three condensate pumps were needed, to maintain feedwater pump suction pressure, although the system was designed for two pump operation.

Instrumentation adjustments such as the T_{AVG} program, based on extrapolations to full power, were ongoing with few problems. Operation, after the 90% power level test plateau, with the MSRs in service improved plant efficiency and lowered turbine impulse pressure down to design specifications.

The major plant transient tests, such as large load reduction, and loss of all offsite power, were completed as planned with only minor problems. When attempted in the low power physics test sequence (Startup Report, Initial document), natural circulation (ST-22), failed due to problems with a steam dump valve. The test was successfully performed in conjunction with the unit trip from 100% power (ST-38).

Testing of the NSSS portion of the plant yielded expected results and availability of GETARS for data acquisition provided test personnel with fast and very accurate information. At no time did data analysis contribute significantly to delays in the program sequence.

Tests to validate plant support systems, such as the radiation shield design and ventilation systems were sequenced at appropriate test plateaus.

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Tests to validate plant support systems, such as the radiation shield design and ventilation systems were sequenced at appropriate test plateaus.

The NSSS Acceptance Test, ST-40, was started on August 5, 1990 at 1700 hours, and completed on August 17, 1990 at 1800 hours. The test was interrupted for two hours on August 6 for stop and control valve surveillance tests at a power level of < 95%, and for approximately 1½ days on August 13 for repair of a leak in an EHC line. Later, on August 16, a steam leak in a S/G blowdown line required isolation of the flash tank, but the line was repaired without reducing power.

Results of the acceptance test were reviewed following the warranty run, and the Power Ascension Test Program was officially completed at 2400 hours, August 18, 1990.

5.0 POWER ASCENSION TESTING

Contents:

5.1 ST-22, NATURAL CIRCULATION TEST

Objective

The objective of this test was a demonstration of heat removal from the reactor coolant system using natural circulation and determination of several natural circulation characteristics.

The natural circulation test is described in FSAR, Section 14, Table 14.2-5, Sheet 25.

Discussion

Natural circulation requires residual heat in the reactor core to establish convection flow. ST-22 was initially scheduled in the low power physics test sequence, with the necessary core heat provided by maintaining reactor power at approximately 3%. Failure of a steam dump valve to function properly prior to reaching natural circulation conditions required rescheduling the test in the PAT sequence.

ST-22 was conducted in conjunction with ST-38, Unit Trip from 100% Power.

The natural circulation test demonstrated the following:

- The reactor coolant system can transition from forced to natural circulation.
- Natural circulation is established and maintained as indicated by stable RCS temperature indications.
- A uniform reactor coolant flow distribution under natural circulation conditions as indicated by incore thermocouple temperature data.
- A determination of the length of time necessary to stabilize natural circulation.

In addition, the test provided data to verify simulator modeling and to support results of transient analysis.

The test was initiated following the trip from 100% power.

A loss of forced flow was simulated by simultaneously tripping all reactor coolant pumps. Manual manipulation of the pressurizer pressure control system; utilizing auxiliary spray; adjustments of charging and letdown flow; and ASDV use ensured stable plant conditions during natural circulation operation.

Results

The acceptance criteria were met; demonstration of natural circulation by a stable RCS temperature, and a coolant flow distribution as shown by incore thermocouple maps.

Following RCP trip, the transition to natural circulation went smoothly, requiring approximately eleven minutes.

5.1 ST-22, NATURAL CIRCULATION TEST (Continued)

Auxiliary spray for pressurizer control was set up three minutes after RCP trip, and as shown by Figures 1,2, 3 and 4, prepared from GETARS data; a smooth transition to a natural circulation state was obtained. Pressurizer pressure peaked at T_{HOT} increased, and then stabilized at a value about 20 psig higher than the initial value. ASDVs cycled while natural circulation conditions existed, and as noted, provided a stable means of automatic heat rejection. RCPs were restarted at approximately 50.3 minutes after pump trip.

The final conditions specified for the test were met as follows:

(1) Level of Decay Heat(Q) - Q was calculated to be 56 MWth prior to trip of the RCPs, using the known flow rate and T_{AVG}(NR).

(2) RCS Flow Rate under Natural Circulation - The RCS flow rate under natural circulation was calculated using Q (56 MWth) and T_{AVG}(WR). A flow rate of 4.9% of full flow resulted. It is noted that the use of T_{AVG}(WR) introduces a relatively large uncertainty in the result. T_{AVG}(NR) was not used because of its unreliability under natural circulation conditions.

(3) Uniformity of Coolant Flow within the Reactor Core - From the incore T/C temperature distributions taken with pumps running and after establishing natural circulation, a uniform coolant flow with natural circulation was verified.

(4) Time for Natural Circulation to Stabilize - Approximately eleven minutes after RCP, natural circulation had been established.

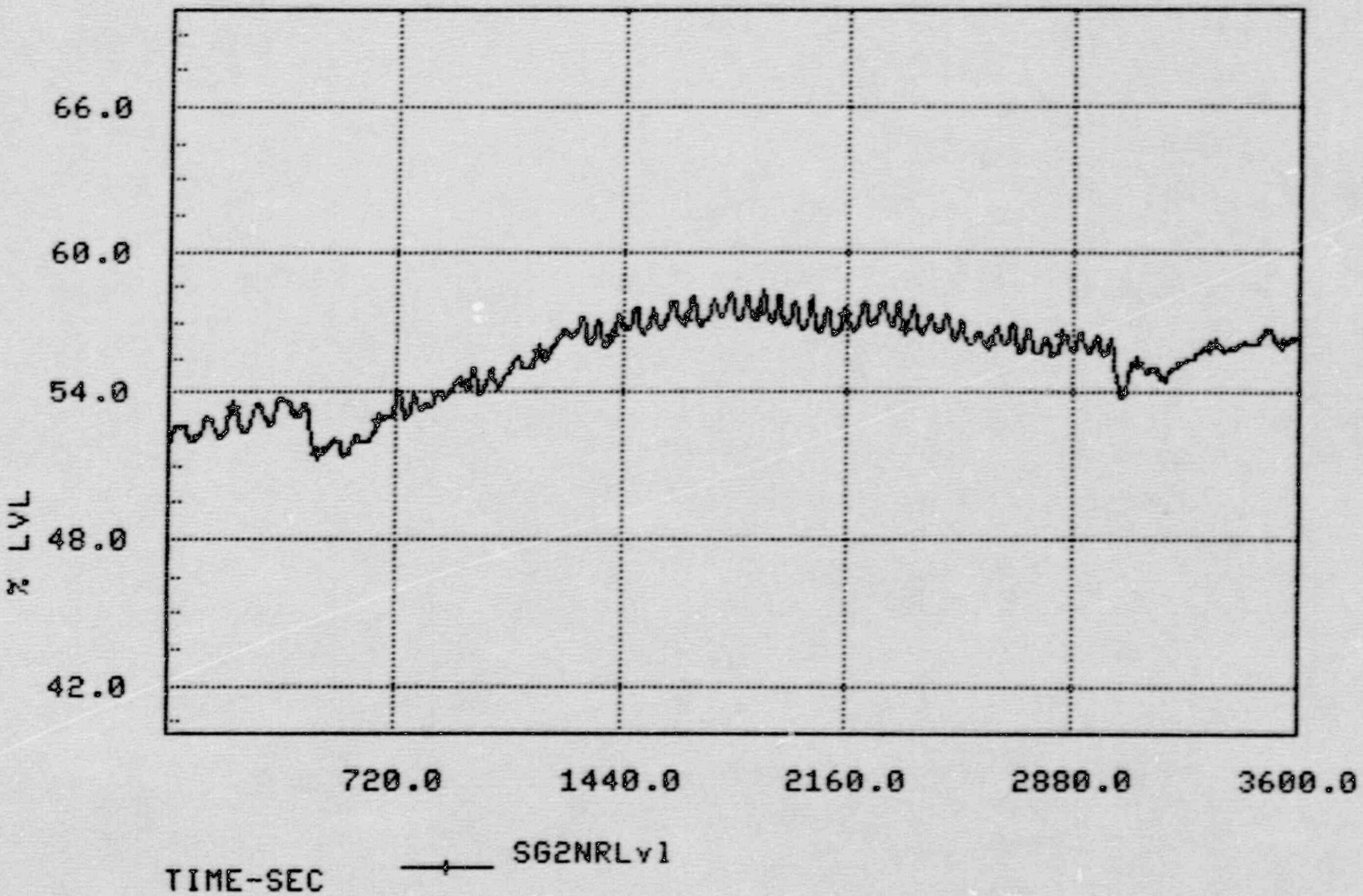
(5) Ability of Subcooling Monitors to Accurately Display Saturation- As shown in Table 1, the subcooling margin displayed was in agreement with the margin determined using RCS (WR) pressure and Core Quad Max Temp.

5.1 ST-22, NATURAL CIRCULATION TEST (Continued)

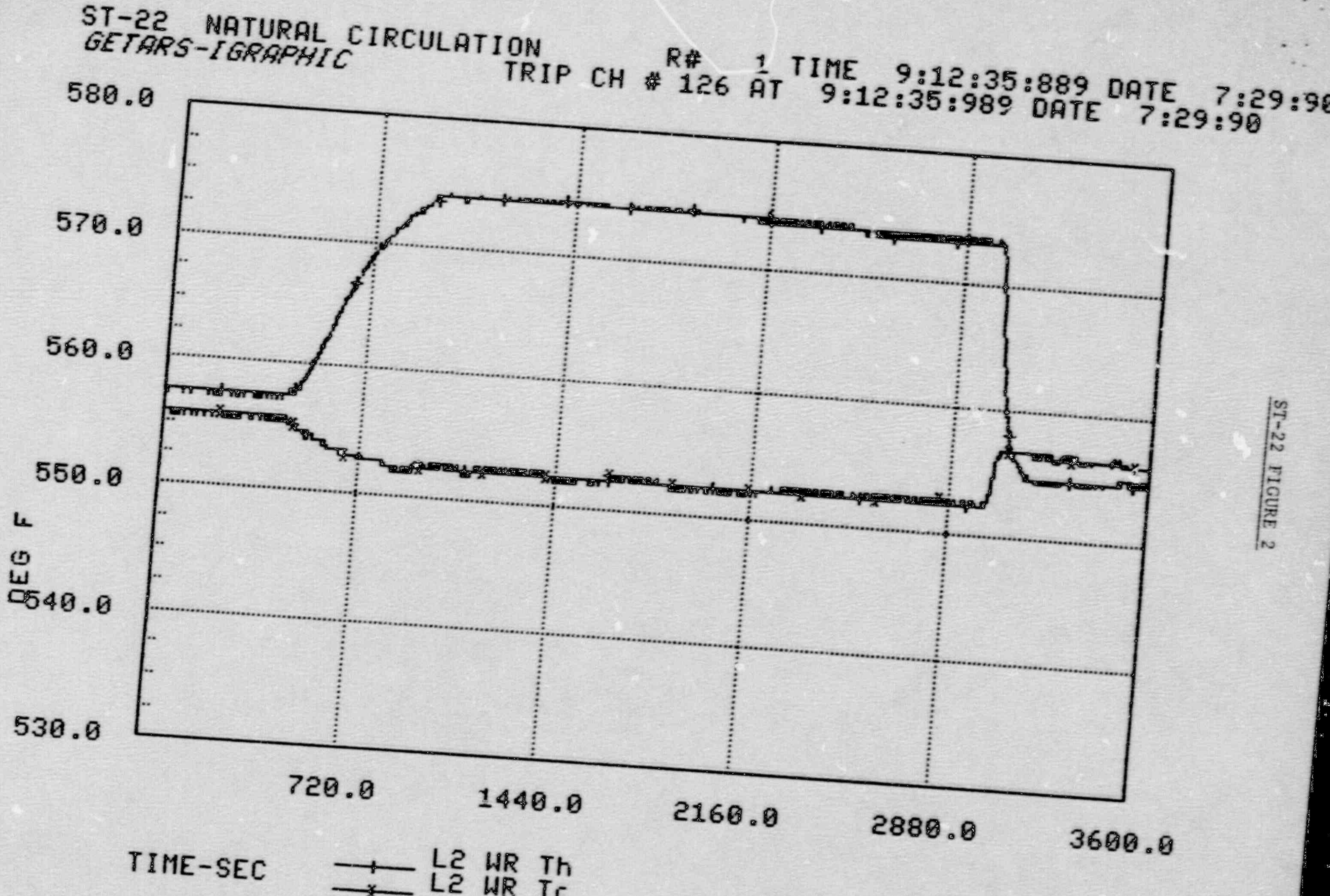
TABLE 1
ST-22 NATURAL CIRCULATION SUBCOOLING MARGIN

GETARS Time	RCS (WR) Pressure (A2970) (psig)	Core Quad Max Temp (A2927) (°F)	T (Sat. Curve) (°F)	RVLIS Sub. Margin (°F)	RVLIS Sub. Margin (°F)	Calculated Margin from Curve (°F)
9:30	2285.0	579.9	655.8	76.6	78.4	75.9
9:32	2281.9	579.5	655.5	77.1	79.2	76.0
9:34	2283.5	581.19	655.6	76.9	78.7	74.4
9:36:15	2286.6	579.9	655.9	76.8	79.1	76.0
9:38:15	2286.6	579.5	655.9	76.9	80.0	76.4
9:39:15	2288.2	579.9	656.0	76.7	79.1	76.1
9:42:10	2286.6	579.5	655.9	77.0	79.8	76.4
9:44:15	2286.6	579.5	655.9	77.3	79.7	76.4
9:46:20	2286.6	579.9	655.9	76.9	80.0	76.0
9:48	2286.6	580.2	655.9	76.6	79.7	75.7
9:50:10	2286.6	579.9	655.9	77.3	79.7	76.0
9:52:05	2285.0	579.9	655.7	77.5	80.3	75.8
9:54:20	2285.0	579.5	655.7	77.2	79.6	76.2
09:56:05	2285.0	578.2	655.7	77.5	80.0	77.5
9:58	2285.0	578.6	655.7	77.3	77.9	77.2
10:00:10	2285.0	579.9	655.7	77.2	79.4	75.8

ST-22 NATURAL CIRCULATION R# 1 TIME 9:12:35:889 DATE 7:29:90
GETARS-IGRAPHIC TRIP CH # 126 AT 9:12:35:989 DATE 7:29:90



ST-22 FIGURE 1



ST-22 FIGURE 2

5.2 ST-24, AUTOMATIC REACTOR CONTROL

Objective

The objective of this test was a demonstration that the Automatic Reactor Control System is able to maintain the average reactor coolant temperature within acceptable steady state limits.

The procedure is described in FSAR, Section 14, TAble 14.2-5, Sheet 27.

Discussion

Procedure ST-24 was performed with the reactor at the 30% power level test plateau, in three sections:

1) Stable operation, following a switch to automatic control.

2) Automatic restoration of stable operation from a temperature imbalance of $T_{AVG} > T_{REF}$ by approximately 6°F.

3) Automatic restoration of stable operation from a temperature imbalance of $T_{AVG} < T_{REF}$ by approximately 6°F.

The change in T_{AVG} was produced by moving Control Bank D and stability determined by monitoring Hi T_{AVG}/T_{REF}. A maximum deviation of ± 1.5°F was permitted for satisfactory performance.

Results

All acceptance criteria were met;

- Automatic plant control systems respond properly.

  Test Result; Satisfactory performance.

- No manual intervention required to bring T_{AVG} to and maintain within ± 1.5°F of T_{REF}.

  Test Result; No manual intervention.
  On increase, T_{AVG} returned to 1.0°F of T_{REF}
  On decrease, T_{AVG} returned to 0.4°F of T_{REF}

- Combined T_{ERROR} signal returned to ± 1•F of T_{REF}.

  Test Result; On increase, T_{AVG} returned to 0.8°F of T_{REF}
  On decrease, T_{AVG} returned to 0.8°F of T_{REF}

There were no test exceptions.

5.3 ST-25, STEAM GENERATOR AUTOMATIC LEVEL CONTROL

Objective

This procedure demonstrated the stability of the Automatic Steam Generator Level Control System under simulated transient conditions, and proper operation of the main feedwater pump speed control. System stability during transfer from bypass feedwater regulating valves to main feedwater regulating valves was included.

The steam generator automatic level control tests are described in FSAR Section 14, Table 14.2-5, Sheet 28.

Discussion

Level control of the steam generators and speed control of feedwater pumps is validated in ST-25 by testing at reactor power levels of 1%-4%, 8%-10%, 8%-20%, 30%, 50%, 75%, and 100%.

At a power level of approximately 3%, each feedwater bypass valve automatic controller was tested, using station operating procedures, to demonstrate stability during a steady state manual to automatic transfer, as well as stability during steady state operations. The changeover from startup feedwater pump to one main feedwater pump was next demonstrated using station procedures. Again, stability during a steady state manual to automatic transfer, and steady state operation in automatic control was verified for main feedwater pump operation. The changeover from startup feedwater pump to the other main feedwater pump, and the stability verifications was demonstrated at the end of the test, by repeating a section of the procedure.

At 8%-10% power, after the feedwater regulating block valves were opened, and stable level control demonstrated with feedwater regulating bypass valves in automatic, the ability of each of these valves to restore and maintain steam generator narrow range level was demonstrated when the narrow range level was successively raised and lowered.

A transfer from bypass to main feedwater regulating valves was made at 8%-20% power, and after stable level control was demonstrated in automatic again, and the power level raised to 30%, the narrow range level was succssively raised and lowered and the ability to restore and maintain level demonstrated under these conditions.

Main feedwater pump (MFP) automatic speed control was demonstrated at the 30%, 50%, 75% and 100% power level test plateaus by monitoring feedwater pump parameters and steam generator levels.

Results

All acceptance criteria were met;

- No manual intervention was required after initiating automatic control.

- Steam generator level returned to and remained within ± 2% of the reference level, within 3 times the level controller time constant, following transfers and simulated level transients.

- Steam generator level overshoot (undershoot) was less than 4% following a level increase (decrease).

- Feedwater pump discharge pressure oscillations were less than ± 3% of the final value.

- At the 100% power level test plateau, the main feedwater regulating valve stem position stabilized at less than 85% open.

A test data summary is given in Table 1. System behavior at full power is shown in Figures 1 and 2.

Two test exceptions were taken. One excepted a missing data sheet which was located later; the second was taken to account for an erroneous valve position indication. The reading failed to meet acceptance criteria, but the actual position, as determined by valve positioner, was 75%, meeting the criterion.

TABLE 1
TEST DATA, ST-25 STEAM GENERATOR
AUTOMATIC LEVEL CONTROL

Automatic Level Control on Bypass Valves: SG Level at 50%

Controllers (% Open):	Acceptance Criterion:
Maximum 39%	No manual intervention.
Minimum 36%	None required.

Main Feedwater Pump A (or B) operating in automatic:

S/G Level at 50%	Acceptance Criterion: No manual intervention. None required.

Feedwater regulating block valve open; feedwater regulating bypass valves in automatic:

S/G Level at 50%	Acceptance Criterion: No manual intervention. None required.

Main Feedwater Regulating Valve Controller Stability, in Automatic:

30% Power Test Plateau Valve, FW-LK-	Level Change	(%) Initial	Level /Final	Recovery Time (Seconds)	Level Overshoot-Undershoot	FW Pump Discharge Pressure Osc.
510	Raised	50.8	51.6	2340	-1.5	None
	Lowered	49.5	49.7	2400	0.4	None
520	Raised	49.1	50.0	2190	1.8	None
	Lowered	49.7	48.5	2240	-2.4	None
530	Raised	49.6	50.2	2310	1.2	None
	Lowered	50.0	48.6	2170	-2.8	None
540	Raised	49.3	50.3	2460	2.0	None
	Lowered	50.0	48.6	1860	-2.8	None
Acceptance Criteria		Max. 52	Min. 48	3000	<4%	< ±3%

Computed Difference Between Feedwater Pump Discharge Pressure and Steam Header Pressure:

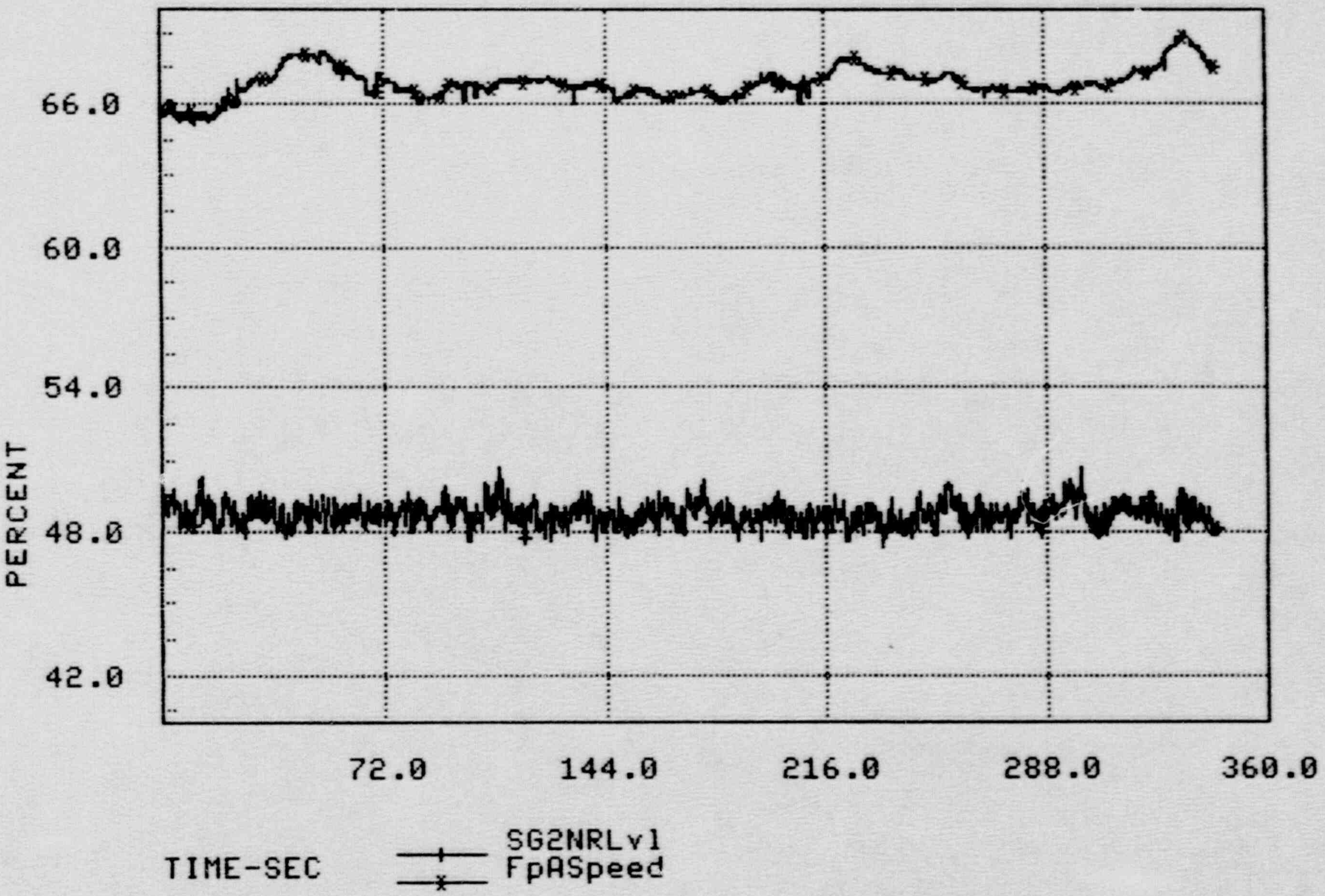
Power Level	Computed Delta-P:	Master Speed Controller Setpoint
30%	100 psi	90 psi
50	125	110
75	163.4	162.4
100	196.5	195.6

Main Feedwater valve position (reg. valve controller M/A station):

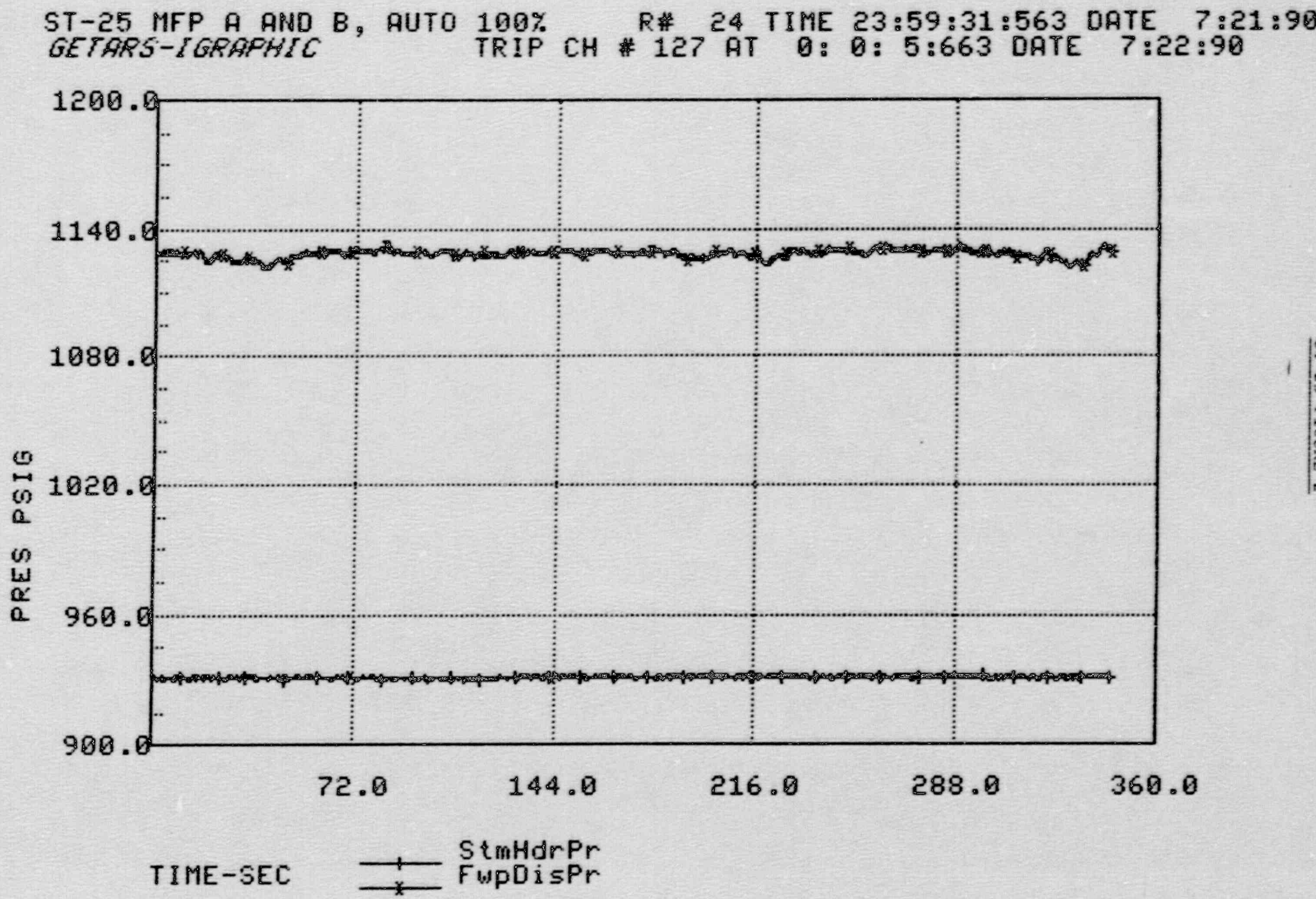
Valve ID	Controller Output (% Open)
FW-FCV-510	77
FW-FCV-520	76
FW-FCV-530	78
FW-FCV-540	75*

*Control Room reading 85%, see Results for explanation.

ST-25 MFP A AND B, AUTO 100%   R# 24 TIME 23:59:31:563 DATE 7:21:90
GETARS-IGRAPHIC   TRIP CH # 127 AT 0: 0: 5:663 DATE 7:22:90

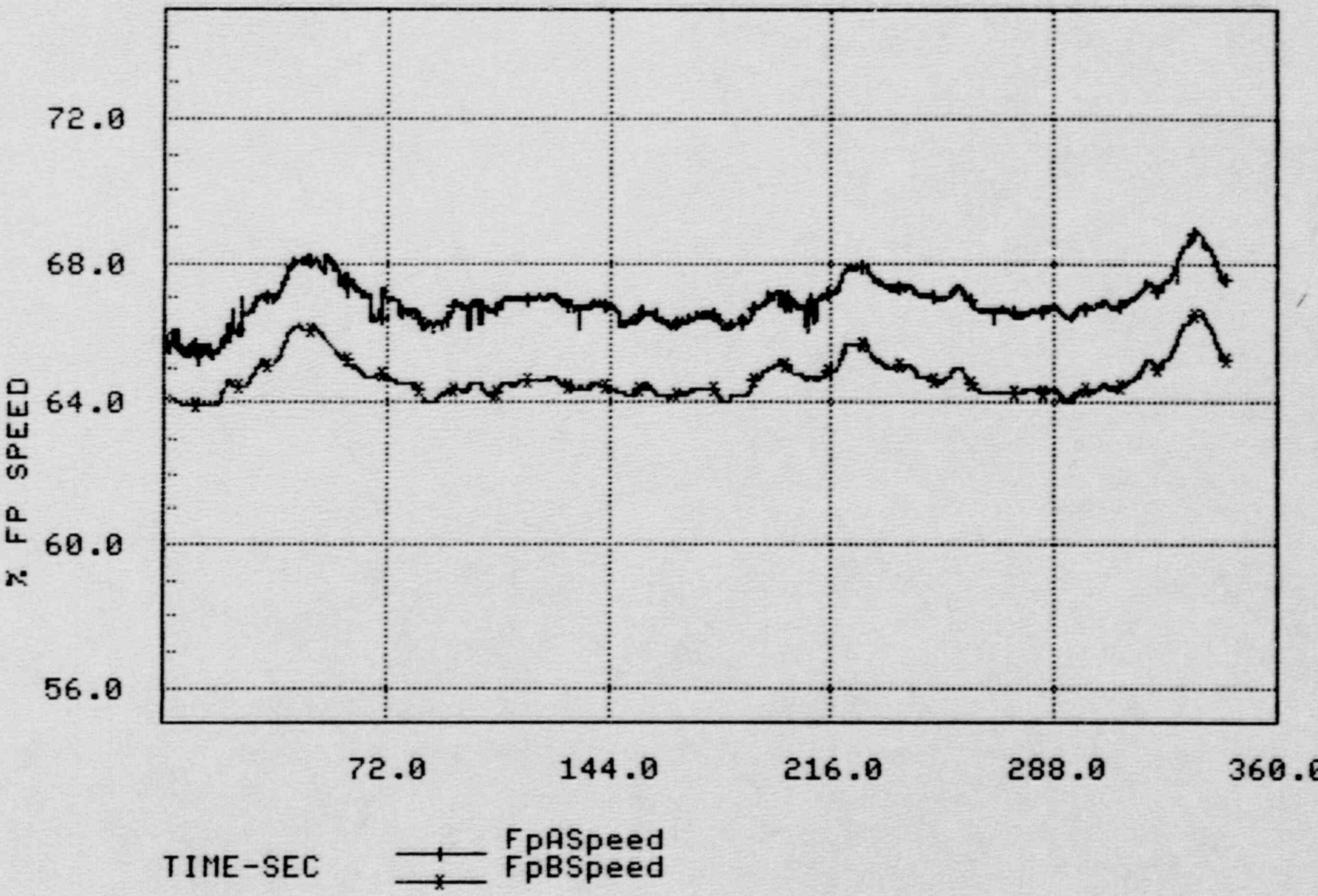


ST-25 FIGURE 1



ST-25 FIGURE 2

ST-25 MFP A AND B, AUTO 100%   R# 24 TIME 23:59:31:563 DATE 7:21:90
GETARS-IGRAPHIC   TRIP CH # 127 AT 0: 0: 5:663 DATE 7:22:90



ST-25 FIGURE 3

5.4 ST-29, CORE PERFORMANCE EVALUATION

Objective

The objective of this procedure was verification of proper core performance through acquisition and analysis of incore flux and thermocouple maps.

The procedure is described in FSAR, Section 14, Table 14.2-5, Sheet 32.

Discussion

ST-29 was performed in its entirety at the 30%, 50%, 75%, 90%, and 100% power level test plateaus and utilized Reactor Engineering surveillance procedures throughout.

Full core flux maps (FCFM) were taken and analyzed using Reactor Engineering procedures. In the analysis, the resulting measured versus predicted assembly power distribution was compared within the INCORE computer code.

A quadrant power tilt ratio (QPTR) surveillance was taken with each flux map at greater than 50% power to ensure that technical specification requirements were met.

Analysis using additional Reactor Engineering procedures yielded the heat flux hot channel factor, $F_Q(Z)$, and the nuclear enthalpy rise hot channel factor, $F_{DELTA-H}$. The peaking factors were then used to support power ascension to the next power level test plateau.

Departure from Nucleate Boiling (DNB) data was transcribed from the results obtained at the same test plateau by ST-26, Thermal Power Measurement and Statepoint Data Collection.

Results

All acceptance criteria were met:

- The core performance parameters of $F_Q(Z)$, $F_{DELTA-H}$, QPTR, and DNB parameters meet technical specification requirements.
- Discrepancies in the measured to predicted assembly power distribution shall be less than 10%.

Results of the core performance parameter measurements are given in Table #1.

Core performance results at the 50% power level test plateau were acceptable for escalation to the 75% plateau. However, a technical specification requirement in the station surveillance procedure used in ST-29, required a remeasurement of F_{XY} at 65% RTP. The necessary flux map was taken at 65% with acceptable results.

5.4 ST-29, CORE PERFORMANCE EVALUATION (Continued)

The value of F_{XY} at the 100% power level test plateau exceeded the full-power limit of 1.55 (Measured Value, 1.572). The value was consistently high at lower power levels also; however, at lower levels it did not limit escalation to the next test plateau.

A test exception was taken to accept the result based on the value of F_Q (Measured F_Q = 2.08, Upper Limit = 2.32), and a Beginning of Cycle 1 F_Q/F_{XY} Evaluation which determined that there was sufficient margin between the design F_Q for all possible operating conditions and the technical specification limit.

TABLE 1
ST-29 CORE PERFORMANCE PARAMETERS

Power Level	30%	50%	75%	90%	100%
Bank D Position	210	210	191	201	212
F_{XY}	1.596	1.567	1.558	1.565	1.572
F_Q	2.189	2.120	2.089	2.080	2.077
$F_{DELTA-H}$	1.454	1.430	1.390	1.396	1.404
Maximum Incore Tilt	1.014	1.017	1.016	1.014	1.016

5.5 ST-30, POWER COEFFICIENT MEASUREMENT

Objective

The Power Coefficient Measurement verified the nuclear design predictions of the "Doppler-only" power coefficient, through correlative measurements of RCS temperatures and core thermal power output.

The measurement is described in FSAR, Section 14, Table 14.2-5, Sheet 33.

Discussion

The Doppler coefficient of reactivity is that portion of the reactivity feedback due to temperature changes in the fuel. The Doppler temperature coefficient, pcm/°F, relates the reactivity change to the change in average temperature of the fuel. The "Doppler-only" power coefficient, pcm/% pwr, relates to the change in power which produced the temperature change.

ST-30 was performed at the 30%, 50%, 75% and 100% power level test plateaus.

In the test, T_{AVG}, delta-T, and reactor power were measured for a series of three small (≈ 3%) load increases (and decreases) by changing the turbine generator output with control rod position held constant. The power coefficient was then inferred from a quantity called the Doppler coefficient verification factor, C^P, defined as the ratio of the change in core average temperature to the change in core power due to the Doppler effect.

The predicted value of C^P was determined from the Nuclear Design Report (NDR), WCAP-10982.

Results

At all test plateaus the acceptance criterion was met; the average measured Doppler coefficient verification factor shall be within ± 0.5°F/% of the predicted Doppler coefficient verification factor.

TABLE 1
ST-30 POWER COEFFICIENT MEASUREMENT

Power Level	Difference Between Doppler Coefficient Measured and Predicted Verification Factors
30%	0.2478°F/%
50%	0.1732
75%	0.0116
100%	0.1121

5.6 ST-33, SHUTDOWN FROM OUTSIDE THE CONTROL ROOM

Objective

The procedure demonstrated that the reactor could be tripped from a location external to the control room, that operations could be transferred to the remote safe shutdown (RSS) facility, and the plant brought to hot standby with the normal shift compliment of personnel.

The procedure is described .n FSAR, Section 14, Table 14.2-5, Sheet 36.

Discussion

ST-33 was performed at the end of the 50% power level test plateau. The power level was reduced to approximately 20% rated thermal power as an initial condition prior to shutdown.

Two operating crews, under the direction of a single Shift Superintendent, performed the test. Ultimate command and control authority remained in the control room with the normal watch crew. The second crew initiated the trip and performed the shutdown from the RSS facility, while the control room crew observed and monitored plant status using all of the instrumentation available at the main control board (MCB). The reactor coolant pumps remained in operation throughout the procedure.

The remote shutdown utilized Train B and selected portions of Train A RSS controls. Some Train A controls were selected for use to minimize the amount of protection and control functions bypassed when local control was established. Communications were established using the RSS sound-powered phone channel and two-way radios.

Results

The acceptance criteria for the test;

1. The reactor has been tripped from outside the control room.

2. The unit has been maintained at stable (T_{AVG} > 480°F), hot standby conditions for at least 30 minutes.

3. Operations to control RCS temperature and to maintain the reactor in a safe shutdown condition were performed without assistance from the Control Room Crew.

All acceptance criteria were met.

5.6 ST-33, SHUTDOWN FROM OUTSIDE THE CONTROL ROOM (Continued)

The reactor trip was initiated from the Train A Switchgear Room by depressing the trip levers for reactor trip and bypass breakers. The MSIV's were closed from the B RSS panel; the resulting transient was easily controlled and plant parameters remained relatively stable:

SG Pressure = Small change, -20 psi to +30 psi
T_{HOT} (WR) - Decrease, 5-15°F
Pressurizer Level - Decrease, 10%
Steam Generator Levels - Small change -2% to +4% (NR)
Automatic EFW Actuation - No

The Train A steam driven EFW pump was manually started from the RSS panels, but the motor driven EFW pump was not required and was never started. Steam generator levels were controlled at the B RSS panel (B and D generators) and the A RSS panel (A and C generators); the atmospheric steam dumps (ASDVs) were not used until about 25 minutes into the recovery, and then only minimal jogging was required.

Plant conditions at the RSS panels on completion of the thirty minute stability period were as follows:

T_{HOT} - L1 = 575°F;
L4 = 560°F
T_{COLD} - L1 = 570°F;
L4 = 560°F
SG Pressure - SGA 1095 psig,
SGB 1080 psig,
SGC 1100 psig,
SGD 1070 psig.
SG Levels - SGA 86%,
SGB 80%,
SGC 86%,
SGD 85%.
Pressurizer Level - 24-25%
RCS Pressure - 2195-2200 psig.
RCS Cooldown Rate - Approximately 10°/hour.

Two minor problems were observed during the test. Three computer points were not recorded by the MPCS, and operation of the RSS Panel PCCW HX temperature control valve hand controller was reverse acting from what had been expected, but posed no control problem.

There were no test exceptions.

5.7 ST-34, LOAD SWING TEST

Objective

The objective of this procedure was a demonstration of proper plant transient and automatic control system performance for a 10% step load change introduced at the turbine generator.

The procedure is described in FSAR, Section 14, Table 14.2-5, Sheet 37.

Discussion

The load swing test was performed at the 30%, 50%, 75% and 100% power level test plateaus.

With the plant stable and control systems in automatic, a step load change was introduced by manual manipulation of turbine generator controls associated with the load set reference signals between primary and standby controls. By use of the standby load set potentiometer, a deviation between the standby and primary control signals was introduced, which developed an equivalent 10% turbine power signal mismatch. Once the mismatch had been developed, transfer to standby control caused a stepwise load change.

The transient (initially a load decrease) was recorded on GETARS and an MPCS trend block, and after system stability had been observed for approximately ten minutes, transfer back to primary control produced a reverse load change (increase).

Results

Successful completion of the load swing test is based on the following acceptance criteria:

- No reactor or turbine trip.
- No safety injection (SI).
- No lifting of steam generator ASDVs or safety valves.
- No lifting of pressurizer PORVs or safety valves.
- No manual intervention to reach steady state.
- No sustained or diverging oscillations in plant parameters.
- Nuclear power overshoot (undershoot) < 3% rated thermal power.
- No manual intervention for T_{AVG} ± 1.5°F T_{REF} after load swing.
- During data evaluation, the combined T_{ERROR} signal, which is T_{AVG} - Power Mismatch (°F), was returned to within ± 1°F of T_{REF} after the load swing.
- Feedwater pump discharge pressure oscillations are less than ± 3% of the final value, two minutes after a steam flow change.

All acceptance criteria were met except for the following:

At the 50% power level test plateau:

T_{AVG} returns to ± 1.5°F of T_{REF}
Actual return (load increase) approximately 1.6°F
T_{ERROR} returns to ± 1°F of T_{REF}
Actual return (load increase) approximately 1.5°F

At the 100% power level test plateau:

No manual intervention;
Loop 2, S/G Level Control placed in manual (load decrease).
T_{AVG} returns to ± 1.5°F of T_{REF};
Actual return (load increase) approximately 3.2°F.

Test exceptions were written to address the deviations. At the 100% condition; manual control was taken, but GETARS data showed that the oscillation was converging at the time; the failure of T_{AVG} to return, was due to control rods reaching a fully withdrawn position.

In addition to meeting acceptance criteria, the procedure requires certain conditions to be reported to Westinghouse for review. The following discrepancies, at the specified power levels, were reported to Westinghouse. Westinghouse, after review, found the values acceptable:

At the 30% power level test plateau:

SG (NR) Level Variations:
Range 6.8% to 7.8% (Load Increase)
6.8% to 7.4% (Load Decrease)
Reporting Requirement: ≥ 5%

SG Press. Over/Undershoot:
Range 36.2 to 37.5 psig (Increase)
Reporting Requirement: > 25 psi

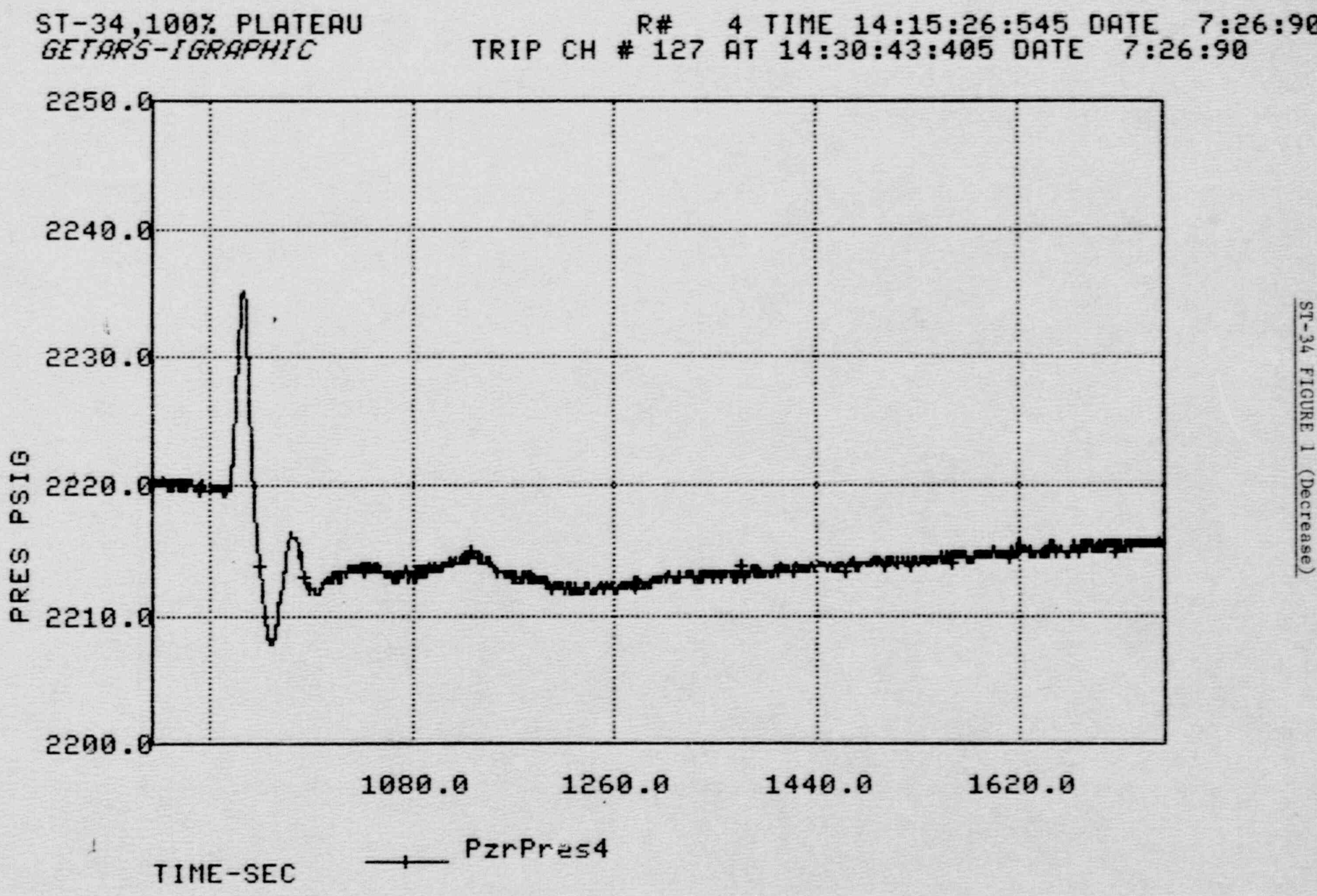
At the 50% power level test plateau:

The above parameters again exceeded the Westinghouse limits.

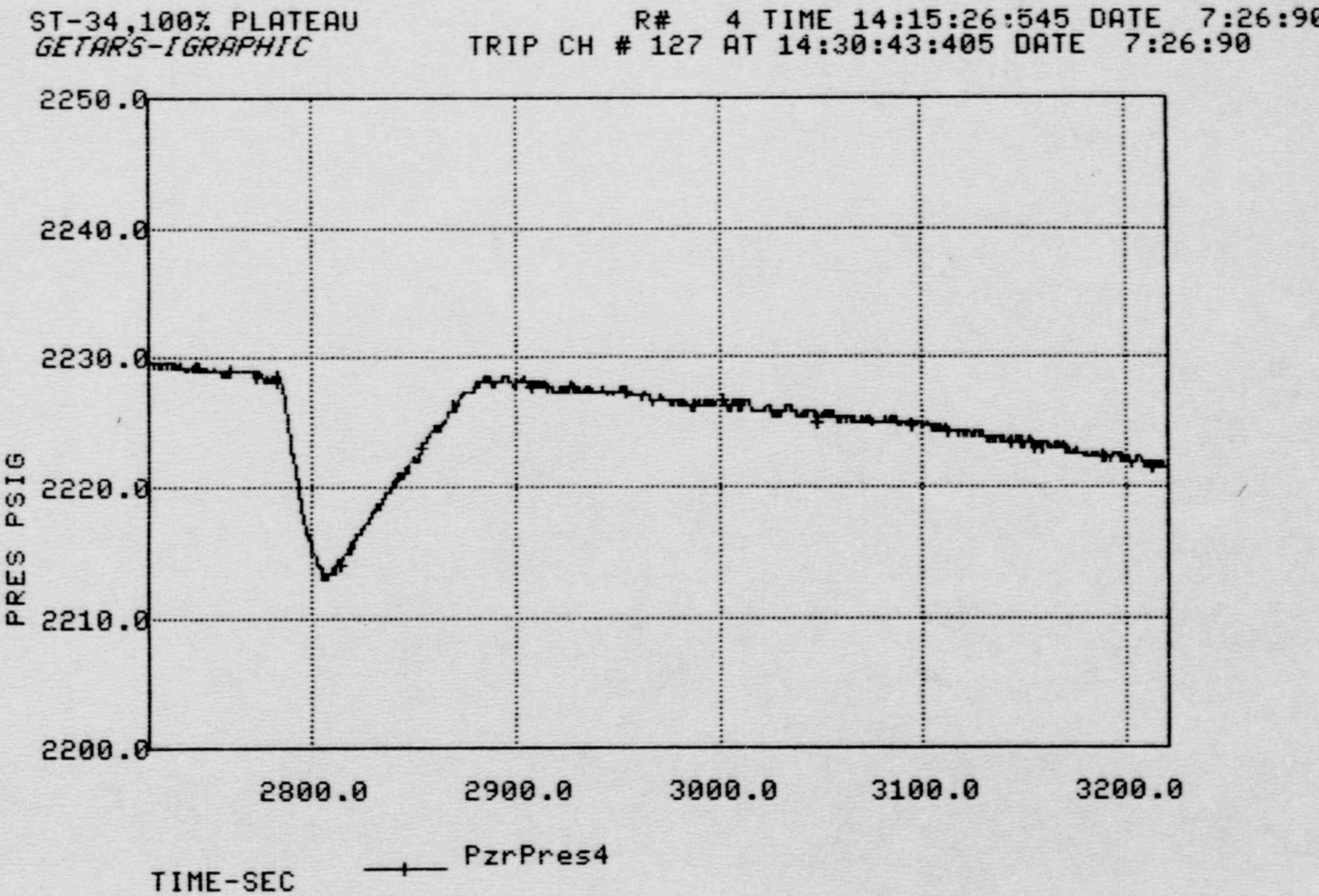
At the 75% power level test plateau:

SG (NR) Level Variations:
Range 12.5% (Load Increase)*
Reporting Requirement: ≥ 5%
* The power level swing was specified to be 10%.
Actually, the value at this plateau was 16%.

Typical transient behavior at the 100% power level test plateau is shown in Figures 1-8.

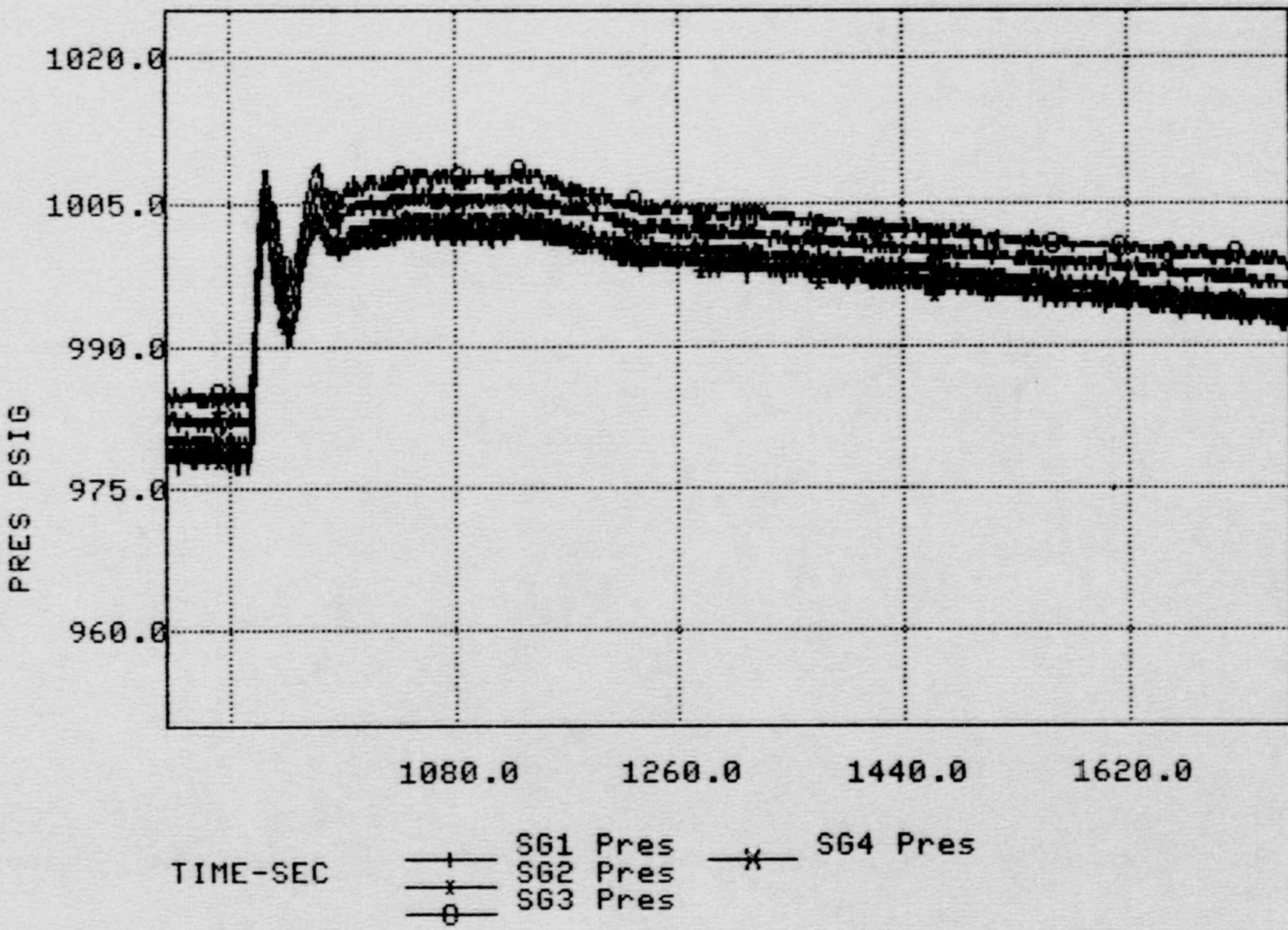


ST-34 FIGURE 1 (Decrease)

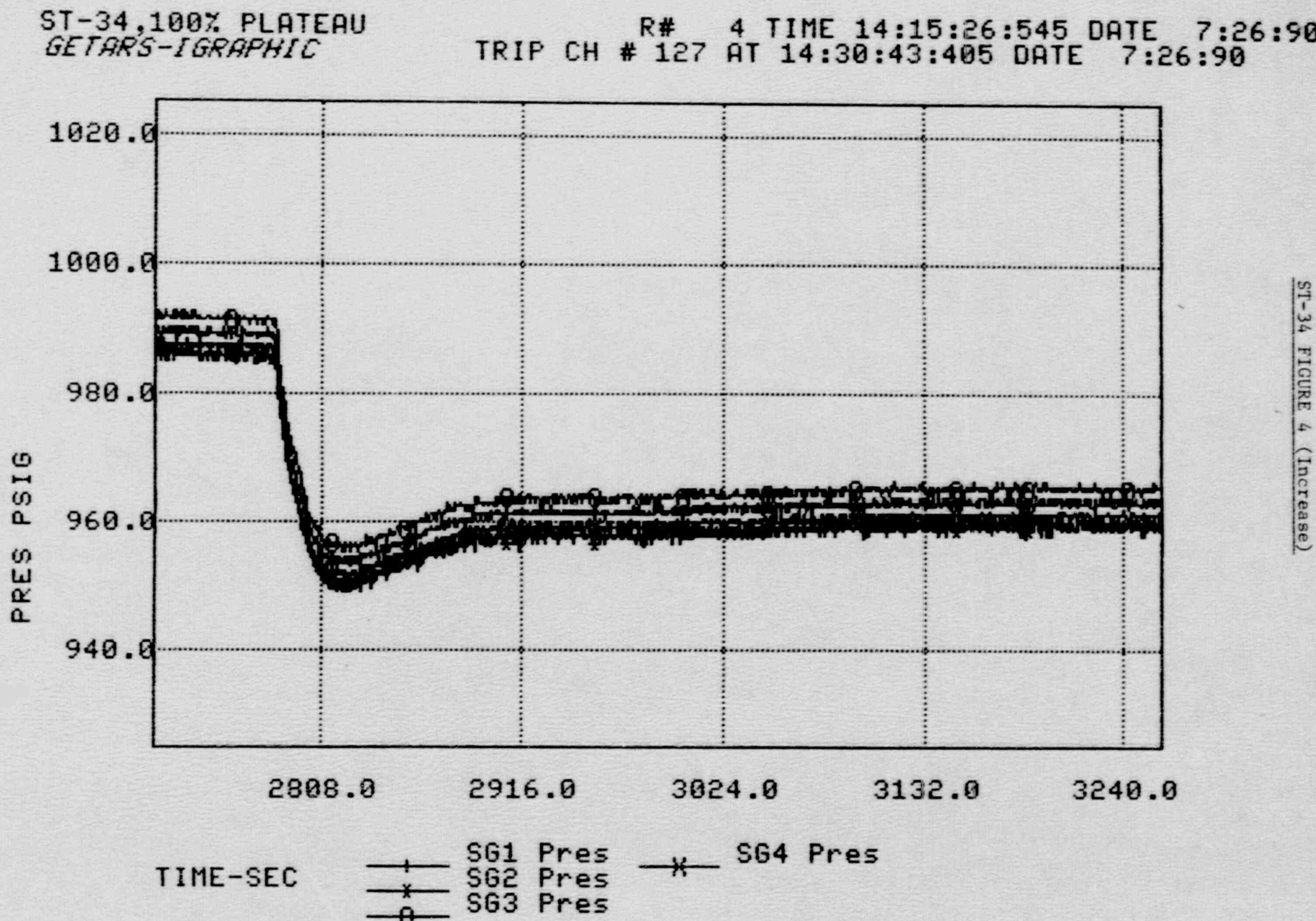


ST-34 FIGURE 2 (Increase)

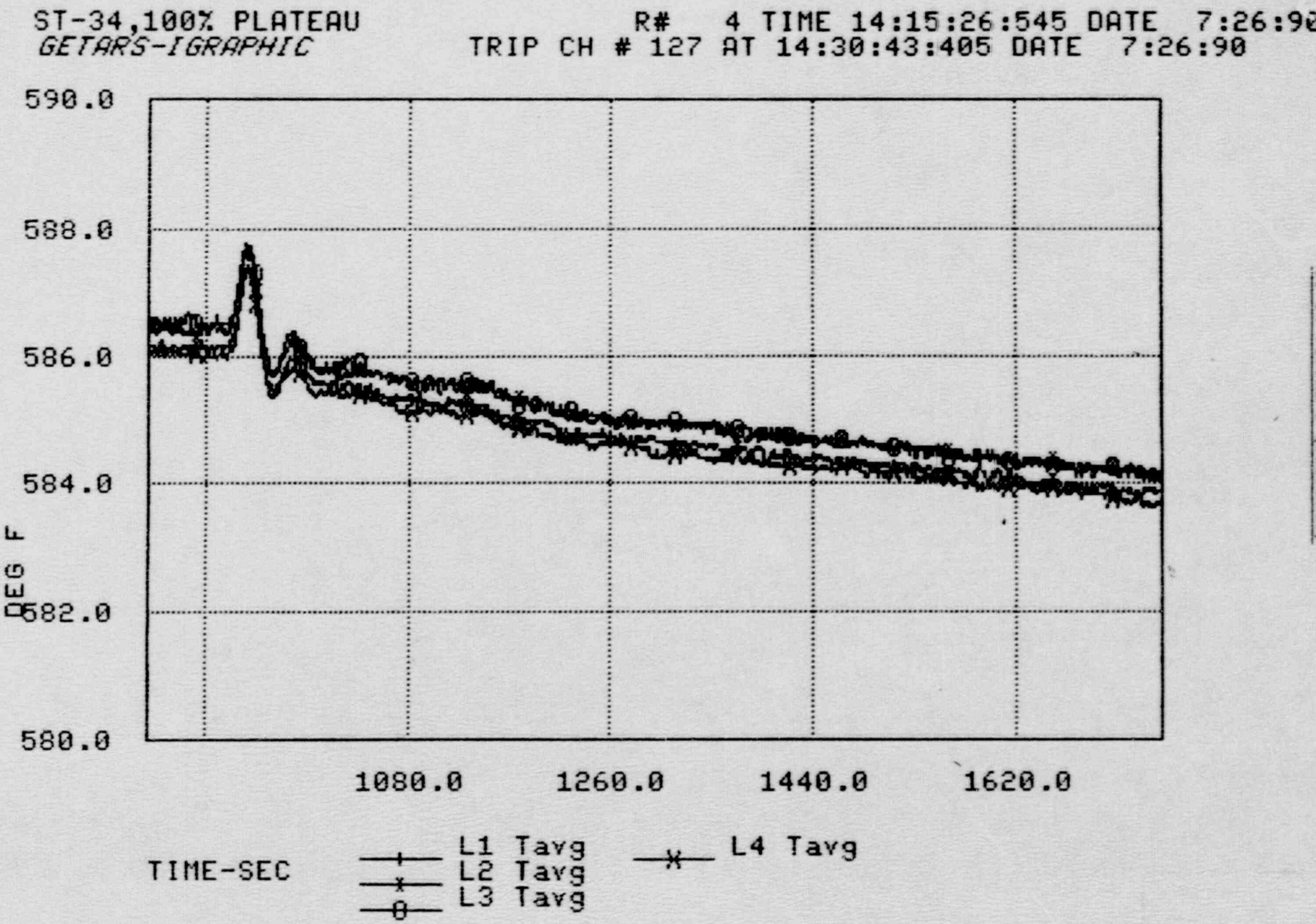
ST-34,100% PLATEAU R# 4 TIME 14:15:26:545 DATE 7:26:90
GETARS-IGRAPHIC TRIP CH # 127 AT 14:30:43:405 DATE 7:26:90



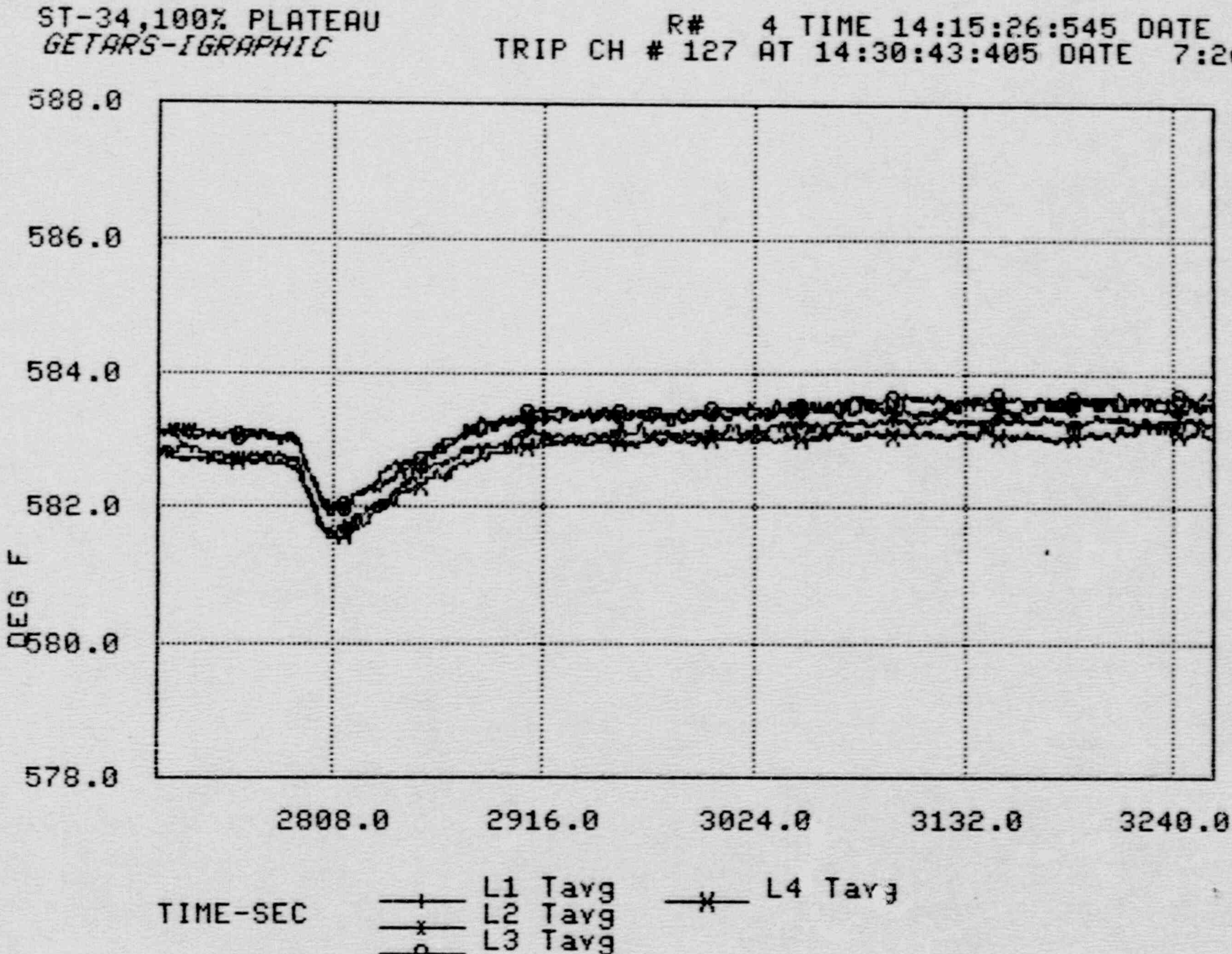
ST-34 FIGURE 3 (Decrease)



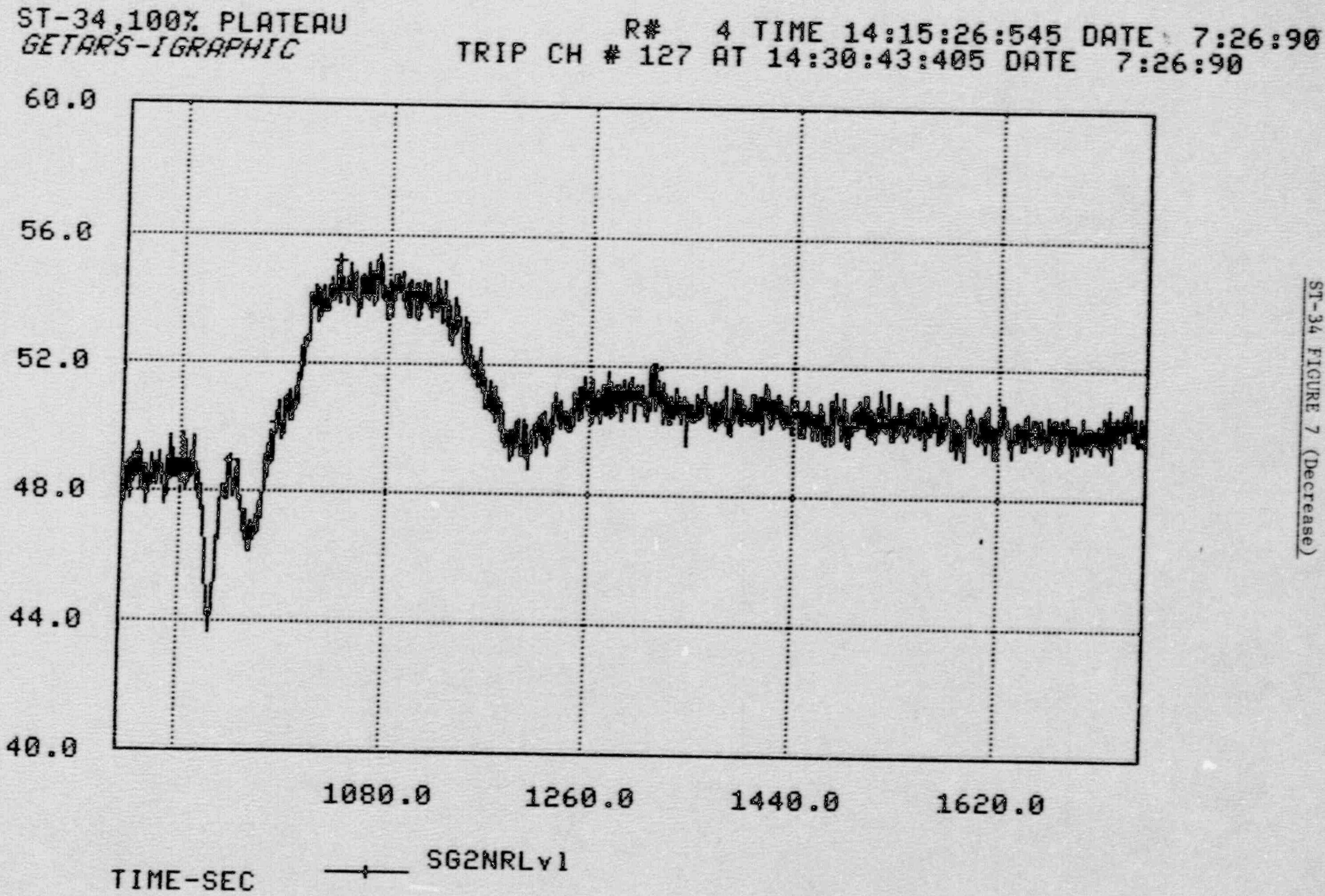
ST-34 FIGURE 4 (Increase)



ST-34 FIGURE 5 (Decrease)

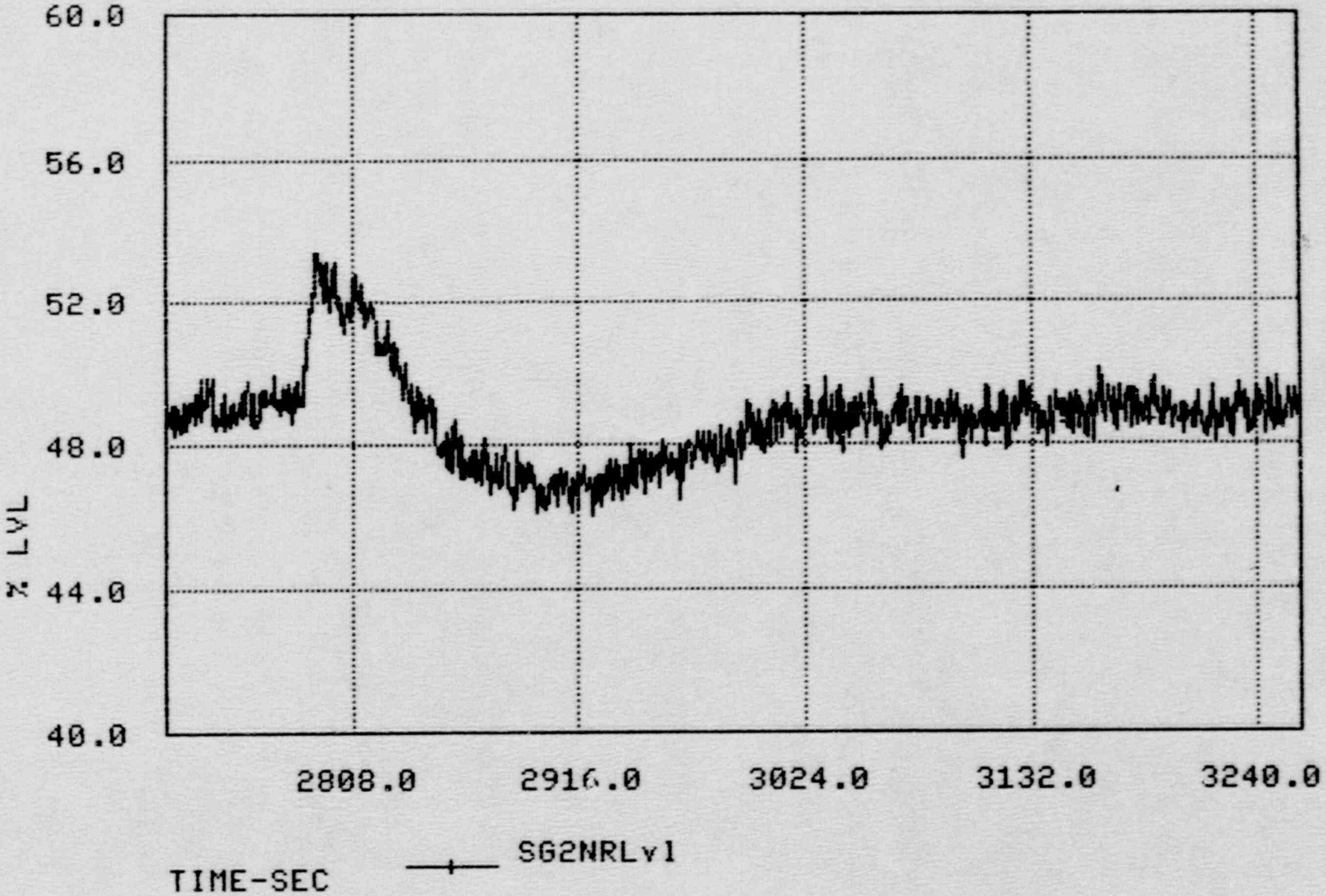


ST-34 FIGURE 6 (Increase)



ST-34 FIGURE 7 (Decrease)

ST-34,100% PLATEAU R# 4 TIME 14:15:26:545 DATE 7:26:90
GETARS-IGRAPHIC TRIP CH # 127 AT 14:30:43:405 DATE 7:26:90



ST-34 FIGURE 8 (Increase)

5.8 ST-35, LARGE LOAD REDUCTION

Objective

The large load reduction test demonstrated proper automatic response of the plant control systems and proper plant transient response to a large load reduction of approximately 50%.

ST-35 is described in FSAR, Section 14, Table 14.2-5, Sheet 38.

Discussion

The large load reduction test was conducted with the plant operating at the 75% and 100% power level test plateaus.

A large load reduction, subjecting the plant to the design ramp rate (133%/minute), utilizes the "Decrease Load" pushbutton on the turbine load selector. With plant systems stable, control systems in automatic, and the turbine generator in the manual mode, the "Decrease Load" pushbutton was depressed and held until the load reduction setpoint (indicated on the Load Set Meter) was decreased by 45%. The 45% decrease was selected to minimize the possibility of an overshoot beyond 50%.

In addition to the acceptance criteria for the test (Results below), deviation of parameters beyond stated limits required notification of the Westinghouse representative.

Results

All acceptance criteria were met:

- The reactor and turbine did not trip.
- Safety injection was not initiated.
- Pressure response and visual observation verify that the steam generator safety valves did not lift.
- No manual intervention was required to bring plant conditions to steady state.
- Plant parameters did not incur sustained or diverging oscillations.
- The feedwater pump discharge pressure oscillations are less than ± 3% of the final value, two minutes after a steam flow change.

Plant performance during the two load reductions was as anticipated. The load rediction at the 75% power level test plateau was from 77% to 31% (Reduction = 540 mwe), and at 100%, from 100% to 48% (Reduction = 620 mwe).

During the load reduction from 100%, feedpump suction pressure during the transient fell below the feedpump trip setpoint, but no trip occurred. The trip circuit, which utilizes 2 out of 3 logic, was examined after the test, and the failure determined to have resulted from two problems; the trip setpoint was adjusted to below the desired point, and an incorrect head pressure correction was used. Necessary changes in the adjustment process have been made to prevent a recurrence and the setpoint is being lowered.

A test exception was written to address the failure to trip. Test results are given in Table 1.

5.8 ST-35, LARGE LOAD REDUCTION (Continued)

TABLE 1
ST-35 LARGE LOAD REDUCTION

Parameter	Measured Values 75%	100%	Acceptable Deviation
Primary Pressure Swings (from initial value).	+30psig/-93psig	+69psig/-99psig	+100/-150 psig
SG (NR) Level	+09%/-11%	+12%/-10%	± 25%
T_{AVG} undershoot below final, steady state value	None	None	3°F
T_{AVG} peaked above initial value	3°F	3°F	8°F
TAVG oscillations during steam dump operations	Approx. 0.5°F	Approx. 0.5°F	No Oscillations
T_{AVG} oscillations after steam dump operation ended	None	None	5°F Peak/Valley
T_{AVG} within 1°F T_{REF} after transient	< 0.8°F	< 0.5°F	No manual intervention
With T_{AVG} restored to 1°F of T_{REF}, auto control maintains T_{AVG} with respect to T_{REF}.	Approx. ± 0.5°F	Approx. ± 0.5°F	± 1.5°F
Automatic rod control inserts rods at maximum speed	Approx. 70 secs.	Approx. 80 secs.	*Approximately 30 seconds
Steam dump system	Yes	Yes	Did not cycle on and off
Duration of steam dump system operation	7.3 min.	12 min.	*Approximately 8 minutes
Feedwater pump discharge pressure oscillations	1.2%	2.0%	** < ± 3% final value, 2 min, after change

Note: Parameters in Table 1 require vendor notification only, except for the last item.

* Westinghouse notified
** Acceptance criterion

5.9 ST-37, MOISTURE CARRYOVER MEASUREMENT

Objective

Steam generator moisture carryover was determined in this test by accurately measuring the moisture leaving each steam generator after injecting a nonradioactive chemical tracer, lithium-6, directly into the steam generators via the main feedwater system.

The moisture carryover measurement is described in FSAR, Chapter 14, Table 14.2-5, Sheet 40.

Discussion

The traditional method for measuring moisture carryover utilizes a short-lived radioactive tracer, Na-24. The problems associated with transport, handling, and exposure to a high level radioactive source are eliminated using a nonradioactive isotopic tracer (lithium-6 hydroxide monohydrate). Isotopic dilution mass spectroscopy was used by Combustion Engineering (CE), under contract, to measure the concentration of Li-6 carried over.

A valid or representative sample of moisture carryover cannot be obtained from the main steam line sampling nozzles, as moisture droplets tend to flow along the walls of the steam lines. Isokinetic sampling nozzles are not normally available, and were not in this instance. In ST-37, samples were drawn from a common header in the feedwater line, and by proportioning the feedwater tracer concentration between the steam generators, using the individual steam line samples, the performance of each steam generator was estimated. A temporary modification was made to permit individual sampling of the main steam lines.

The tracer was injected after a preconditioning purge of sample lines for twelve hours, and a one-hour stable power condition. After a half-hour wait to ensure an equilibrium condition, samples were taken from each S/G blowdown sample point. The blowdown sample from each generator was required to yield at least 5 ppb. lithium-6, prior to general sampling.

Six sample sets were drawn from each S/G blowdown line, each main steam line and the common main feedwater line. The samples were transferred to CE analysis.

5.9 ST-37, MOISTURE CARRYOVER MEASUREMENT (Continued)

Results

The acceptance criterion was met; moisture carryover of each steam generator has been calculated by CE and determined to be ≤ 0.25%.

In Table 1, the sample concentrations and moisture carryovers reported by CE are given.

TABLE 1
ST-37 MOISTURE CARRYOVER RESULTS

	Sample 1	Sample 2
S/G A Concentration -		
Blowdown	17.425 ppb	16.715 ppb
Mainsteam	0.004 ppb	0.003 ppb
Moisture Carryover -	0.03 %	0.01 %
S/G B Concentration -		
Blowdown	16.769 ppb	16.049 ppb
Mainsteam	0.008 ppb	0.008 ppb
Moisture Carryover -	0.06 %	0.03 %
S/G C Concentration -		
Blowdown	17.882 ppb	16.912 ppb
Mainsteam	0.004 ppb	0.001 ppb
Moisture Carryover -	0.03 %	0.00 %
S/G D Concentration -		
Blowdown	14.686 ppb	14.396 ppb
Mainsteam	0.002 ppb	0.003 ppb
Moisture Carryover -	0.02 %	0.01 %
Main Feedwater Concentration -	0.007 ppb	0.003 ppb

There were no test exceptions.

5.10 ST-38, UNIT TRIP FROM 100% POWER

Objective

The procedure demonstrated proper plant response to a trip from 100% power, and verified that the actual overall hot leg resistance temperature detector (RTD) response time is conservative with respect to the value used in the accident analysis.

The test is described in FSAR, Chapter 14, Table 14.2-5, Sheet 41.

Discussion

With the plant in steady state operation at the 100% power level test plateau, a unit trip was initiated by manually opening the generator breaker from the main control board (MCB). This action caused the main generator breaker to trip open and the turbine to trip with a resultant reactor trip.

Prior to test initiation, the following systems were placed in automatic:

- Steam Generator Level Control
- Feedwater Pump Speed Control
- Steam Dump Control (T_{AVG} Mode)
- ASDVs (Set Point 1125 psig)
- Pressurizer Pressure Control (Set Point 2235 psig)

The UATs were aligned as the source of power for the onsite distribution system.

Results

All acceptance criteria were met:

- Pressurizer safety valves do not lift
- Steam generator safety valves do not lift
- Safety injection was not initiated
- Overall hot leg RTD response time* ≤ 6.7 seconds
  Measured Response Time = 5.5 seconds.

  * The interval of time measured between the point where the neutron flux has decreased by 50% from its initial value to the point where the hot leg temperature signal has decreased by one-third of the initial loop delta-T value (°F).

Plant performance during the test was generally as expected. Representative system behavior during the transient is shown in Figures 1, 2, 3, 4 and 5.

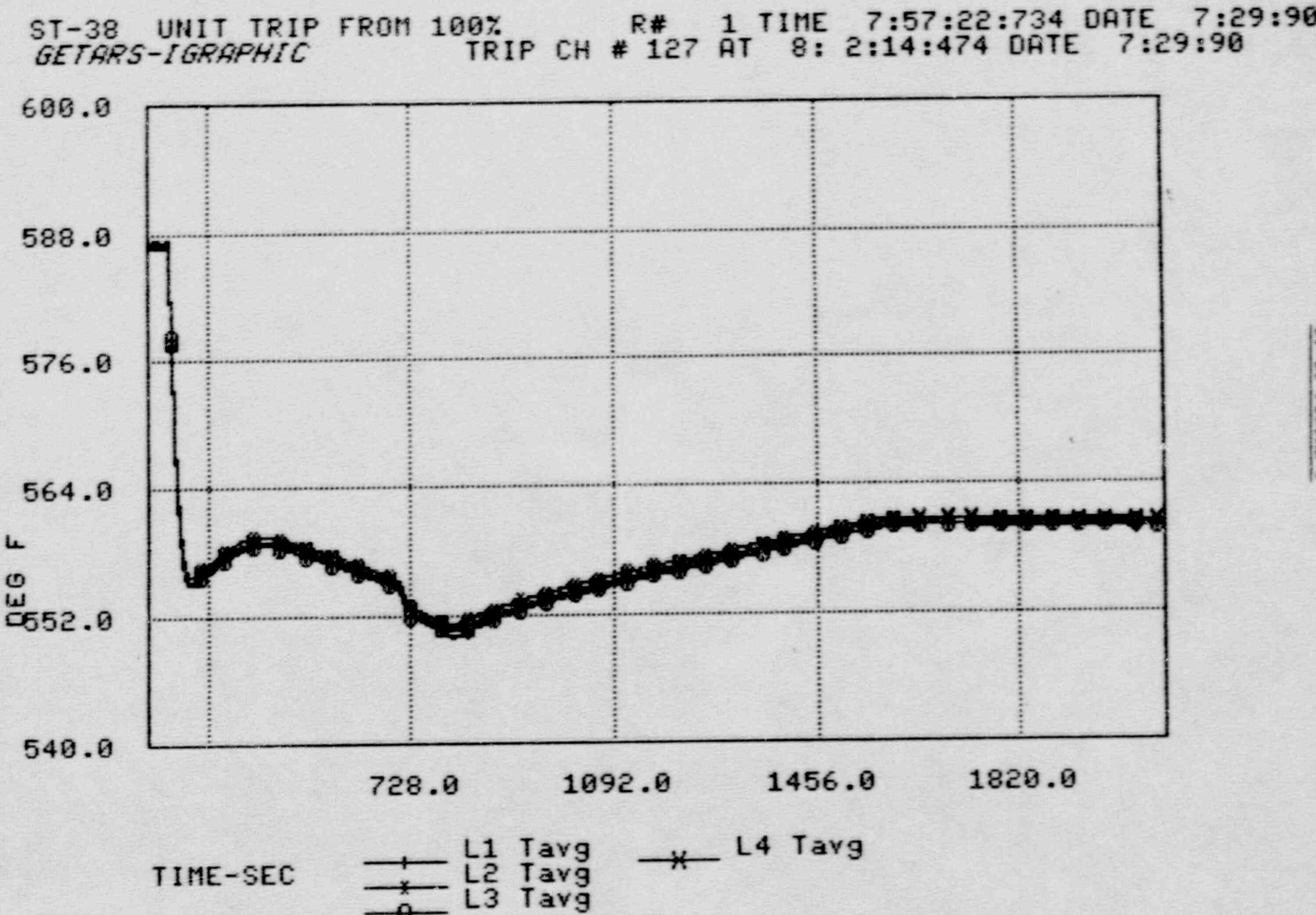
Two operational responses requiring evaluation were identified:

- A P-14 feedwater isolation signal occurred due to S/G narrow range level spiking high. The action caused the feedwater regulating valves to rapidly close, causing a number of feedwater heater relief valves to lift. A design change to address the problem is in preparation.

- When transferring the steam dump system to the steam pressure mode from the T_{AVG} mode, a 35% demand was observed which lasted 5 seconds. A procedural change has been initiated to ensure a bumpless transfer.

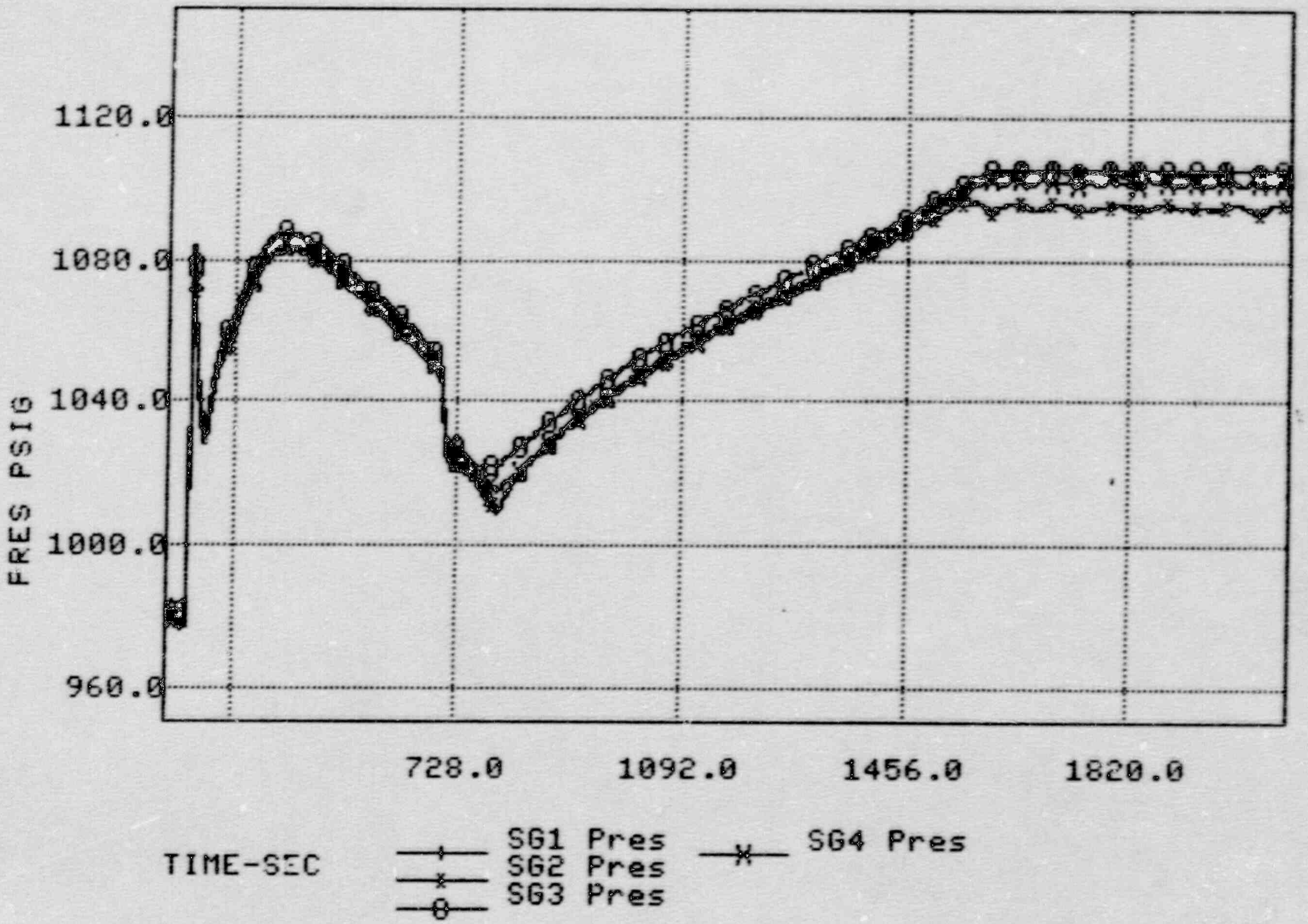
In addition, the following operational response was reported to Westinghouse:

- Pressurizer pressure dropped 31 psi below the expected minimum of 2000 psig.

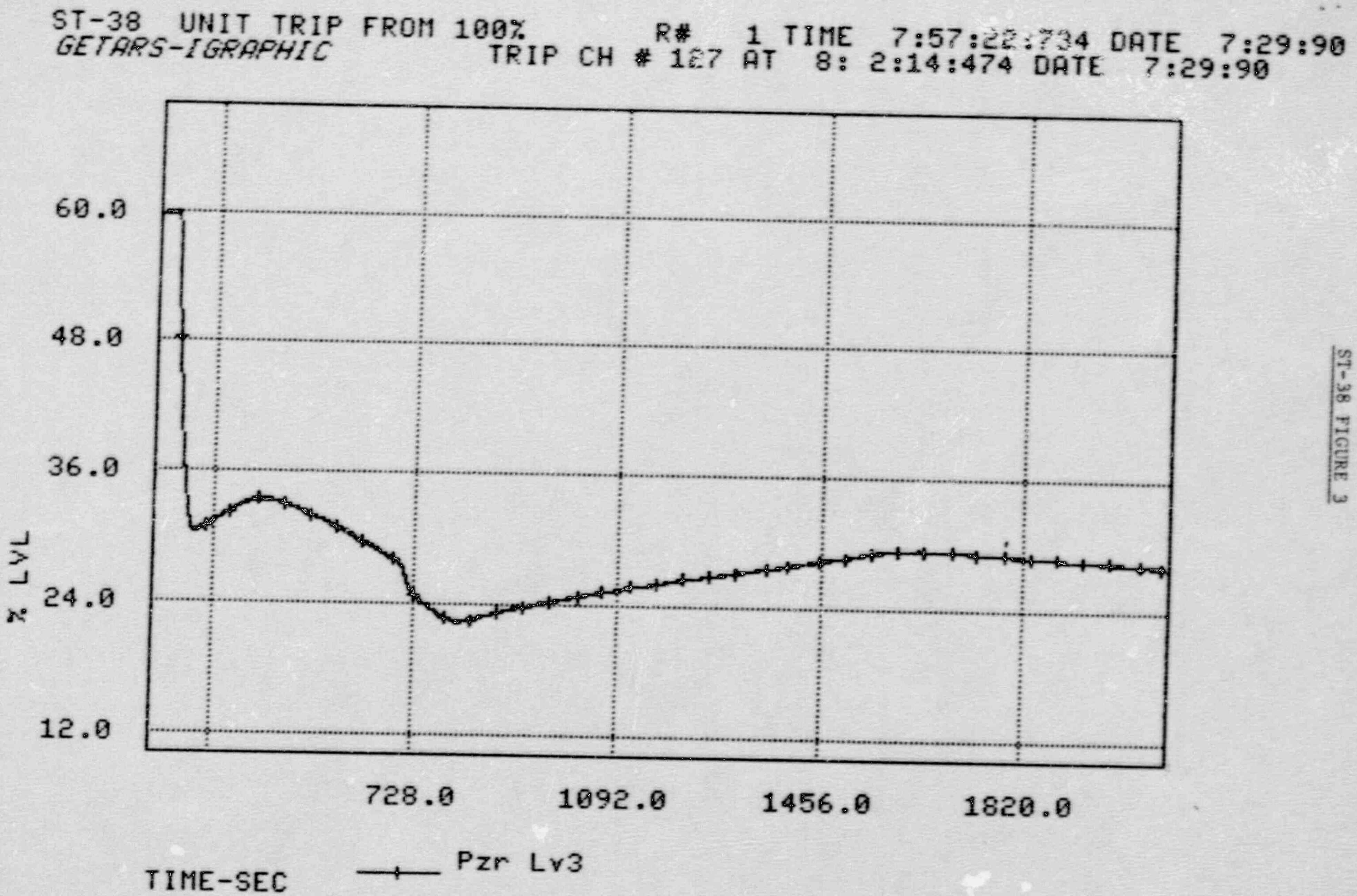


ST-38 FIGURE 1

ST-38 UNIT TRIP FROM 100% R# 1 TIME 7:57:22:734 DATE 7:29:90
GETARS-IGRAPHIC TRIP CH # 127 AT 8: 2:14:474 DATE 7:29:90

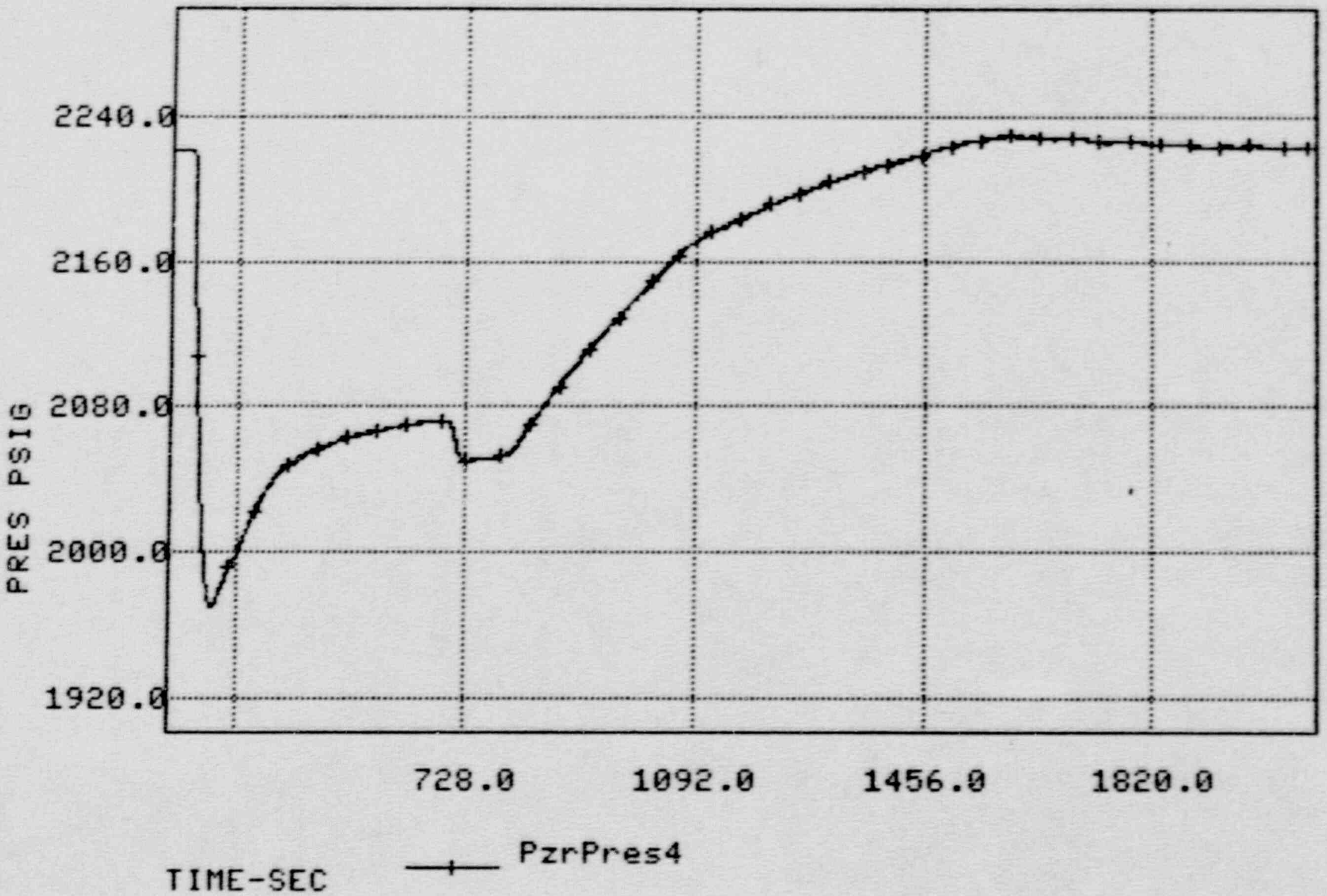


ST-38 FIGURE 2



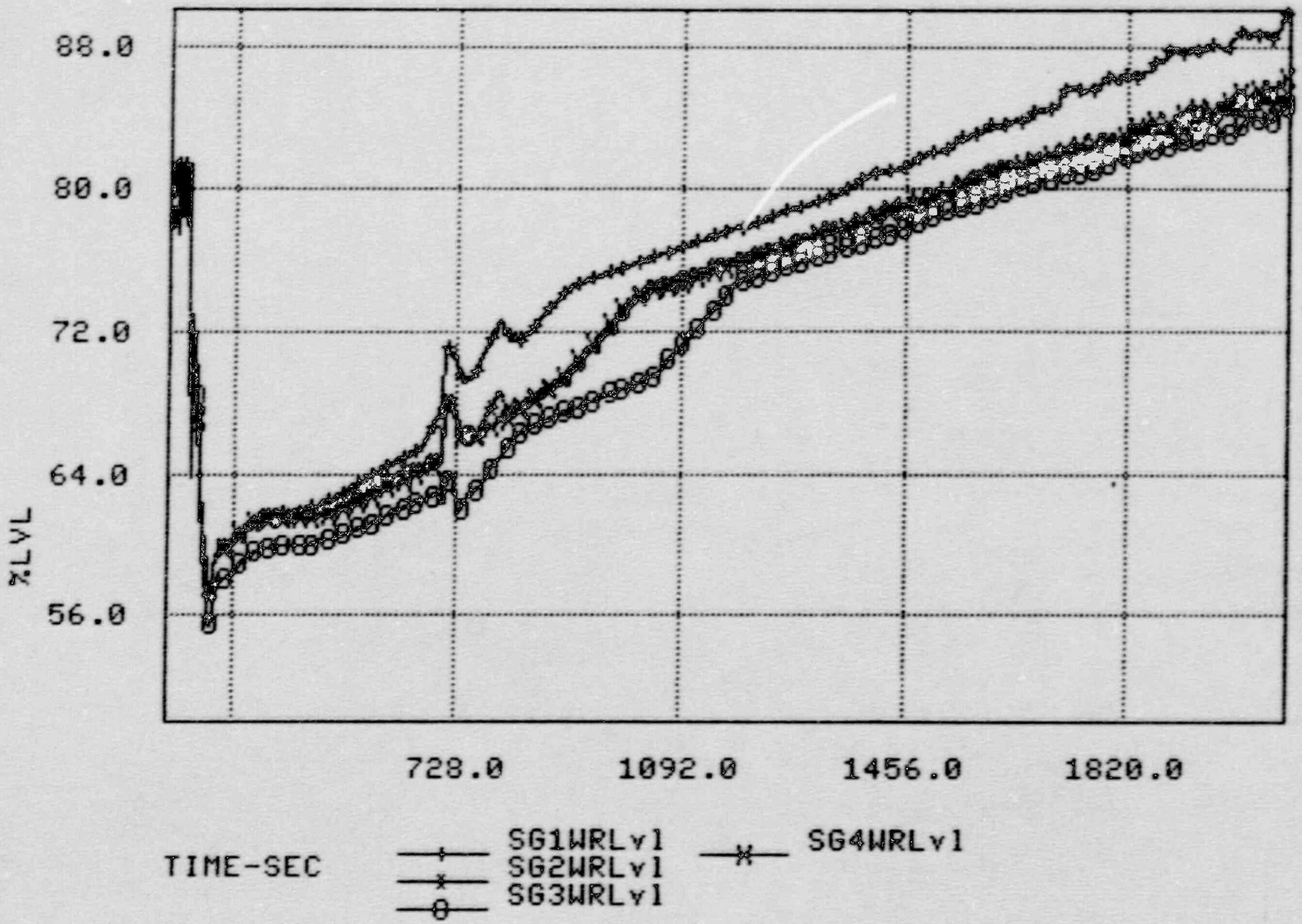
ST-38 FIGURE 3

ST-38 UNIT TRIP FROM 100% R# 1 TIME 7:57:22:734 DATE 7:29:90
GETARS-IGRAPHIC TRIP CH # 127 AT 8: 2:14:474 DATE 7:29:90



ST-38 FIGURE 4

ST-38 UNIT TRIP FROM 100% R# 1 TIME 7:57:22:734 DATE 7:29:90
GETARS-IGRAPHIC TRIP CH # 127 AT 8: 2:14:474 DATE 7:29:90



5-40

ST-38 FIGURE 5

5.11 ST-39, LOSS OF OFFSITE POWER

Objective

The Loss of Offsite Power Test demonstrated response of the plant and emergency electrical power system to meet design performance requirements under the condition of a loss of all offsite power, coincident with the loss of the main generator. Also, stable, shutdown maintenance of the reactor plant under natural circulation conditions with a loss of offsite power was shown.

Discussion

The loss of offsite power was initiated, from a power level just above the P-9 setpoint (approximately 20% RTP), by simultaneously tripping the turbine generator and opening a 345 KV control breaker. The incoming supply breakers for the reserve auxiliary transformers (RATs) had been placed in "pull-to-lock" to prevent an automatic transfer. A reactor trip resulted and the reactor coolant pumps tripped on underfrequency.

The test method permitted bus voltages to decay and be sensed by all of the various undervoltage protection circuits. First level undervoltage protection (less than 72% rated voltage for > 1.2 seconds), caused emergency power sequencer (EPS) actuation, which then initiated an emergency start of the diesel generators.

The emergency power sequencers (EPS) sequenced the required loads onto the diesel generators. The uninterruptible power sources (UPS) shifted to the DC backup supply with the trip and loss of offsite power (LOP), and then back to AC from the diesel generators.

During recovery, pressurizer backup heaters, auxiliary spray and PORVs were available to control primary pressure. Atmospheric steam dump valves (ASDVs) were available, but not the steam dumps to the condenser.

Results

All acceptance criteria were met. The criteria and related performance are given as follows:

- First level undervoltage protection causes the start of both emergency diesel generators, and both emergency power sequencers (EPS) perform their automatic sequencing correctly, as recorded on Form ST-39A.

  Main plant computer system (MPCS) logger data is used to complete the above form, which requires the start time for the diesels and sequencing steps. In the two-second period following test initiation, the MPCS degraded which caused the prime host to fail over to the backup host. During the failover (2-6 minutes depending on the MPCS feature) diesel and sequencer information was lost or unreliable.

The possibility of failover was anticipated and, as a failover contingency, a trained observer with a calibrated stopwatch had been stationed at the MCB electrical section to visually verify startup of the diesels and the sequencer steps. The diesels started and sequenced essentially together; the bounding start time for the diesels was 8.44 seconds and panel lights verified that all sequence steps occurred in order.

A test exception was written to cover the loss of MPCS data during the failover.

- The diesel generator starting times are less than 10 seconds.

  Maximum starting time was 8.44 seconds.

- Satisfactory operation without offsite AC electrical power was achieved for a minimum of 30 minutes.

  The plant was tripped at 0941 hours and recovery began at 1014 hours.

- Evaluation of recorded data and plant responses confirm proper dynamic system responses resulting from a loss of offsite AC electrical power.

  Proper dynamic system response was confirmed. Representative plant response is shown on Figures 1, 2 and 3. Plant responses which were not anticipated are noted below.

Vital Busses 1A, 1B and 1C properly shifted to the DC backup supply, but did not automatically shift back to AC when the diesels had powered the emergency busses (E-5 and E-6). This is a characteristic of these units. The inverters were manually transferred.

DC Voltage and current indications on MCB were lost. An investigation of the problem disclosed that the transducers feeding the indicators are AC supplied. An engineering evaluation has been requested.

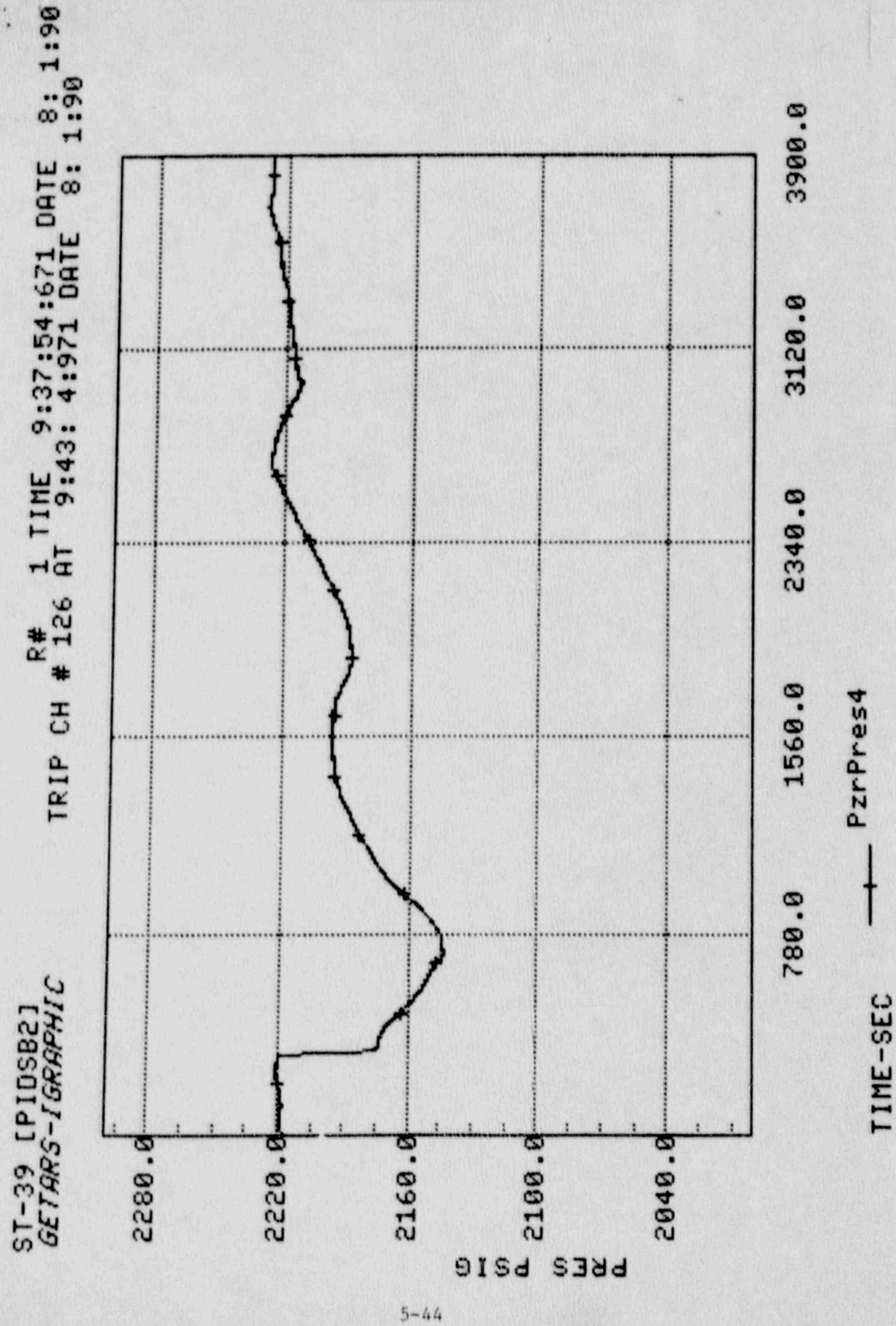
The current limit circuit of non-vital Battery Charger 2A failed early into the test, causing the Bus 12A supply breaker from the charger to overload and open. As a result, Bus 12A voltage decreased from approximately 150 VDC to 120 VDC due to the load of the DC powered bearing, seal and MFP lube oil pumps. The DC pumps were no longer required when the emergency busses were powered, and the voltage rate of decrease slowed significantly. With the restoration of offsite power, and no current limit, the bus was recharged without problem.

The Control room non-vital lighting inverter did not operate and was found to be misaligned. Loss of the inverter did not affect operator performance during the test. A procedure upgrade will correct the problem.

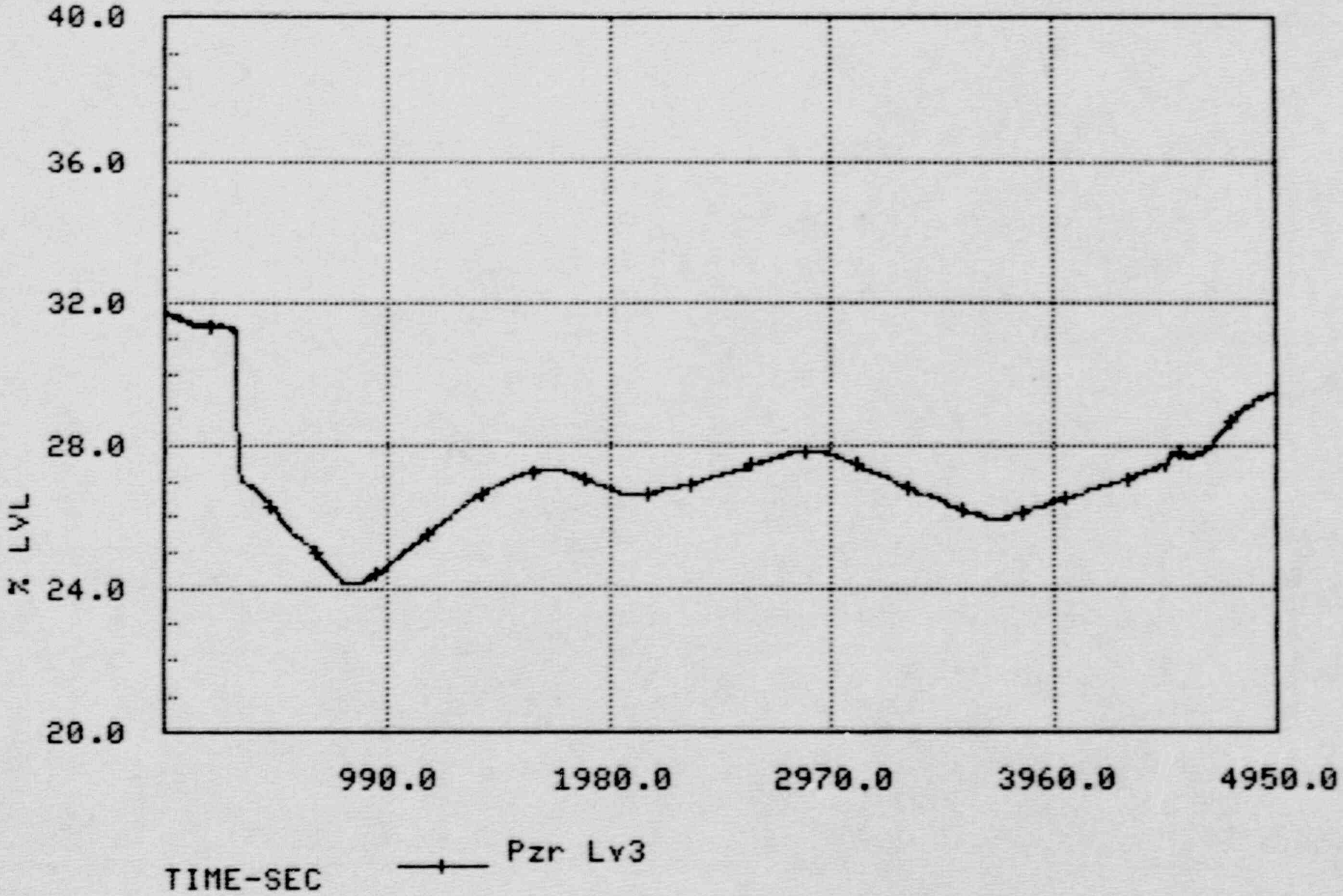
The pressurizer Group A backup heaters could not be manually re-energized from the MCB following Remote Manual Operation (RMO) reset. A minor wiring discrepancy was found to be the cause and was corrected.

During restoration of offsite power, a Train A RAT closing circuit inop alarm was received. A blown secondary PT fuse was the cause.

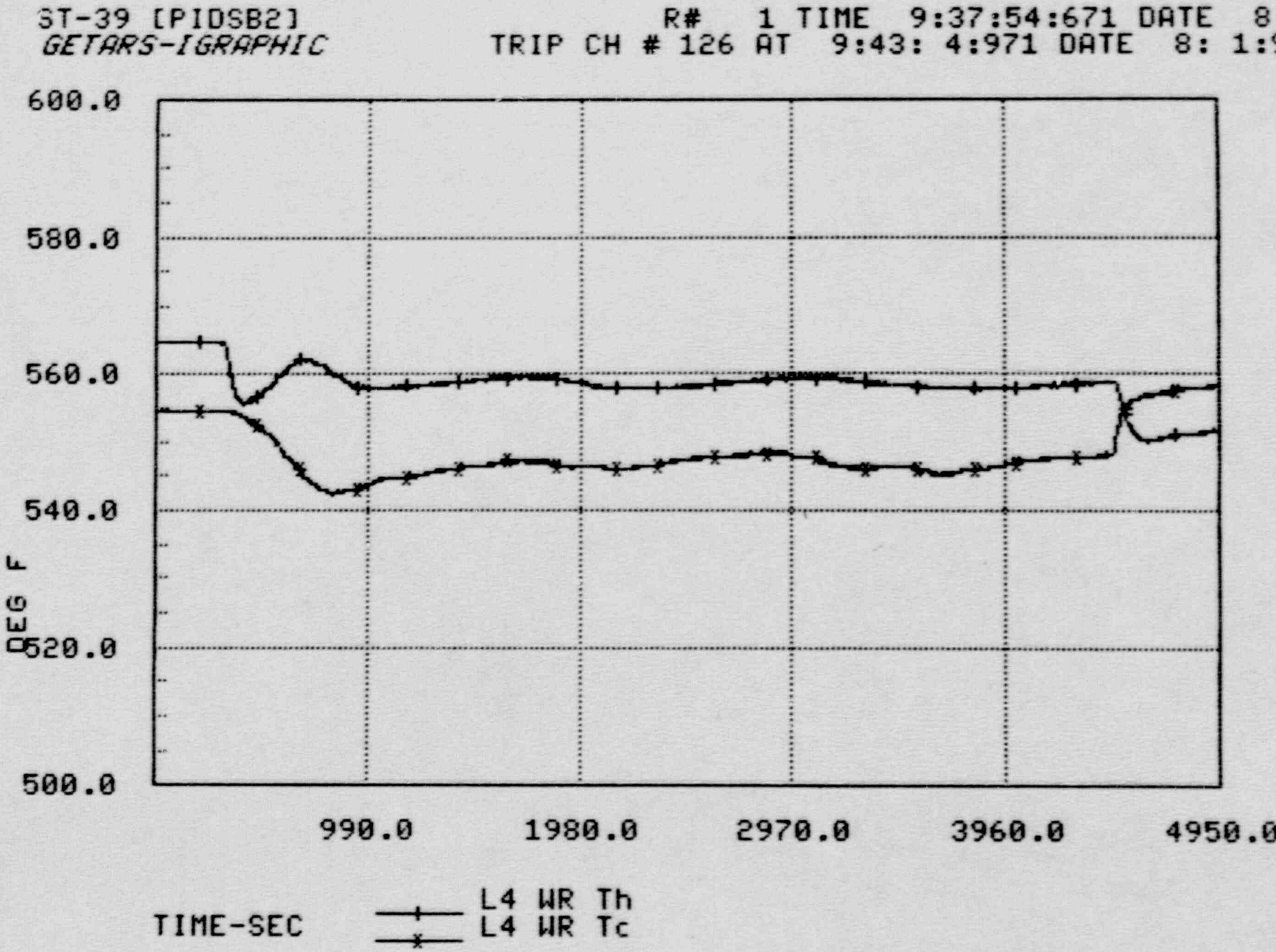
ST-39 FIGURE 1



ST-39 [PIDSB2]   R# 1 TIME 9:37:54:671 DATE 8: 1:90
GETARS-IGRAPHIC   TRIP CH # 126 AT 9:43: 4:971 DATE 8: 1:90



ST-39 FIGURE 2



ST-39 FIGURE 3

5.12 ST-40, NSSS ACCEPTANCE TEST

Objective

The Nuclear Steam Supply System (NSSS) test demonstrated the reliability of the NSSS by maintaining the plant, at 100% (+0%/-5%) reactor power, for 250 hours without a load reduction or a plant trip resulting from an NSSS malfunction. Also, the NSSS output was measured at (or near) its warranted rating.

The acceptance test is described in FSAR, Chapter 14, Table 14.2-5, Sheet 43.

Discussion

To demonstrate acceptable NSSS performance, the plant was required to maintain 95% to 100% of full power for 250 hours. Full power rated conditions were stated as:

- Thermal Output of NSSS: 3425 Mwt
- Thermal output of reactor core: 3411 Mwt
- Steam flow from NSSS (No blowdown): 15,140,000 lb/hr
- Steam generator outlet pressure: 1000 psia
- Assumed feedwater temperature: 440°F
- Maximum moisture content: 0.25%

The warranted NSSS thermal output was 3425 Mwt, and the acceptable performance was 98%, or 3357 Mwt.

On an hourly basis, during the 250 hours, power level data was taken using the NIS power range channels, station recorder charts and data log printouts. After approximately 60 hours, the performance measurements were taken, to verify the warranted thermal output. Four sets of hourly data were taken, with blowdown secured, for the precision calorimetric station procedure.

Results

The acceptance criteria were met; plant operated at 100% (+0%/-5%) for 250 hours without a load reduction or plant trip resulting from a NSSS malfunction, and NSSS thermal output as determined by a performance measurement, is ≥ 3357 Mwt.

Performance measurements yielded the following:

- Average reactor power level from calorimetrics = 99.78%
- Power deviation during performance measurements = 0.21%
- Measured NSSS thermal output* = 3422 Mwt

* Includes 18.4 Mwt net input from the RCS

5.12 ST-40, NSSS ACCEPTANCE TEST (Continued)

The test began on August 5, 1990 at 1700 hours, and ended on August 17, 1990 at 1800 hours. The test was interrupted for two hours on August 6 for stop and control valve surveillance tests at a power level of < 95%, and for approximately 1½ days on August 13 for repair of a leak in an EHC line. Power was reduced and the turbine tripped for the latter interruption. On August 16, a steam leak in a S/G blowdown line required isolation of the flash tank, but the line was repaired without reducing power.

There were no test exceptions.

5.13 ST-48, TURBINE GENERATOR STARTUP TEST

Objective

The objective of the turbine generator startup test was acquisition of baseline operating parameters for the turbine generator and associated components, and operational data at each of the power level test plateaus for evaluating unit performance.

This procedure is described in FSAR, Chapter 14, TAble 14.2-5, Sheet 51.

Discussion

The turbine generator startup test demonstrated the following:

- The loss of primary or backup speed signals will not trip the turbine, but loss of both speed signals causes a turbine trip.

- The Backup Overspeed Trip and Emergency Trip circuits function as designated.

- The turbine-generator is capable of operating at various loads without exceeding any manufacturers' design limitations.

GE Startup Engineers assisted PAT personnel throughout the test, and on occasion requested that additional measurements be made, or extended the time for gathering data. As was reported in Supplement 1, Startup Test Report, ST-48 was interrupted, after rolling the turbine and validating protective systems, to conduct ST-48.1, Turbine Generator Torsional Response Test. An undesirable resonance was found, necessitating a month-long PATP interruption for modifications to the "C" low pressure turbine.

No-load data was recorded for turbine steam conditions, lubrication and control systems, and generator parameters.

Following initial synchronization and overspeed testing, the load was increased, and at selected power levels, subsystems of the turbine generator such as EHC and Alterex were adjusted and operation of protective systems verified.

Turbine generator tests were conducted at the 30%, 50%, 75%, 90% and 100% power level test plateaus. At each level, steady-state data was collected; where necessary, power load unbalance (PLU) checks, underexcited reactive ampere limit (URAL) checks and trip tests of the thrust bearing wear detector (TBWD) were included.

Results

The acceptance criteria were met; the turbine generator has been synchronized to the grid, and all required operational data has been collected and evaluated as satisfactory.

5.13 ST-48, TURBINE GENERATOR STARTUP TEST (Continued)

In the initial tests, after no-load data was taken, and initial synchronization and overspeed testing completed, some corrective measures were taken to address problems identified. These included backseating intercept valves (IV), readjusting the EHC speed error signal, resolution of thermal expansion concerns, repair of the mechanical overspeed trip, and installation of flow orifices in the fast acting solenoids of the EHC trip circuit.

At the 50% and higher power level test plateaus, the required turbine generator performance data was collected, and where indicated, PLU, TBWD and URAL checks made.

Eight test exceptions were taken; nine RTDs embedded in generator windings are defective, but do not prevent adequate monitoring of winding temperatures; six exceptions were written against inoperative instruments or readouts and one excepted trend data which was misplaced.

6.0 INSTRUMENT CALIBRATION AND ALIGNMENT

Contents:

6.1 ST-13, OPERATIONAL ALIGNMENT OF NUCLEAR INSTRUMENTATION

Objective

The objective of this procedure was determination of various voltage, trip, alarm, operational and overlap settings for the source, intermediate and power range nuclear instrumentation. Portions of ST-13 are performed at a number of power levels and at each of the power level test plateaus in Power Ascension Testing.

The procedure is described in FSAR, Section 14, Table 14.2-5, Sheet 16.

Discussion

Calibration of nuclear instrumentation, including alarm settings, trip points, and operational ranges, cannot be properly completed until the system is functioning on line, in its intended operational ranges. At each test condition, the nuclear instruments are adjusted, using the best available conservative information. Initially, setpoint data furnished by Westinghouse was used, and, as higher neutron fluxes became available, these values were superseded by actual measured data.

Test conditions and the adjustments made during ST-13 were:

Prior to Criticality -

- IR Channels; high flux rod stop bistable and high flux bistable.
- PR Channels; scaled for total full power using 400 microamps., verified/scaled f(delta-I) summing amplifiers, scaled AFD (from Reactor Engineering).

Approach to Criticality -

- MCB shutdown monitor test

10% to 15% Power -

- PR Channels - Indications checked against heat balance.

30% Power Level Test Plateau -

- IR-PR overlap data (from ST-26)
- PR Channels, total full power detector currents by extrapolation using calorimetric power data (from ST-26).

50% Power Level Test Plateau -

- IR-PR overlap data (from ST-26)
- PR Channels, total full power detector currents by extrapolation using calorimetric power data (from ST-26); rescaling using AFD data (from ST-36).
- Gammametric Channels, calorimetric adjustment

  Note: Gammametrics is the trade name for the post accident excore detectors.

75% Power Level Test Plateau -

- Same as 50% without Gammametric adjustment

100% Power Level Test Plateau -

- IR and PR saturation curves, IR-PR overlap
- PR Channels, total full power currents by measurement.
- IR Channels, bistable adjustment if required
- Gammametric Channels, adjustment

Shutdown from 100% Power In Conjunction With (ST-38) -

- SR Channels, saturation and integral bias curves
- IR Channels, compensating voltage and bistable adjustment
- Gammametric Channels, discriminator adjustment, if required.

Results

All acceptance criteria were met with certain exceptions (indicated by asterisks);

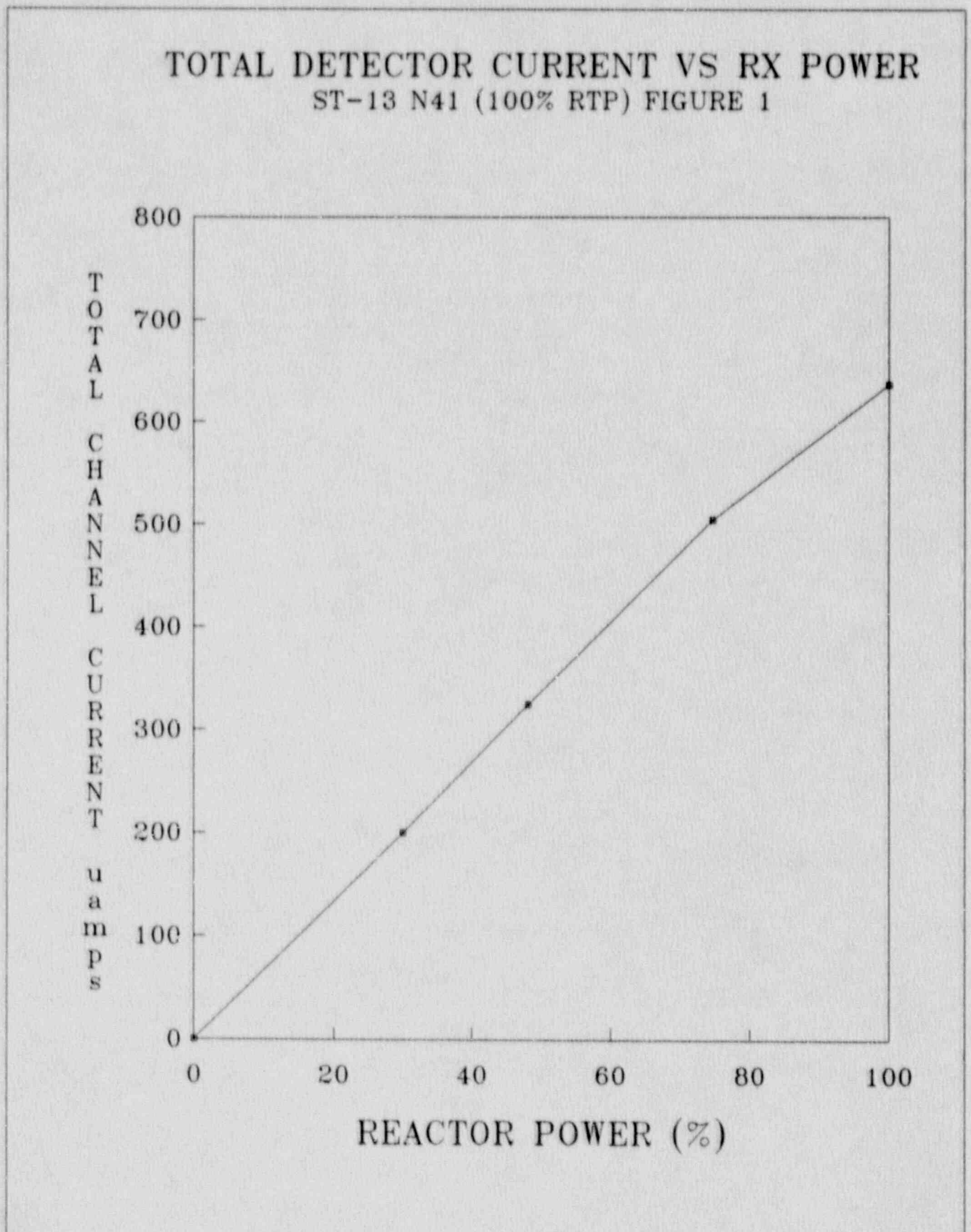
- Shutdown Monitor Alarm setpoints were 1.5 times the previously recorded countrate ± 10%.*
- Shutdown monitor countrates at alarm were equal to the previous alarm ± 10%.*
- Overlap data was obtained between IR and PR channels at the individual test plateaus.
- Plots of PR channel total detector currents vs calorimetric power exhibit linear response from 0% to 100% power.*
- Final operational settings have been documented for SR, IR and PR channels and meet the range limitations of T.S. 3.3.1 and the Westinghouse NI Manual.
- Gammametric detectors have been adjusted at the 100% power level test plateau.

The failure to meet acceptance criteria for the shutdown monitor resulted from testing requirements which did not simulate actual plant conditions for normal service. A detailed discussion of the test, conducted during the initial PAT criticality, is given in Supplement 1 to the Initial Startup Report. A test exception was taken.

Per procedure, following the shutdown from 100% power (ST-38), the compensating voltages and bistable alarm settings for the IR channels were to be adjusted. The adjustments were not successful, and were attempted a second time following the loss of offsite power test (trip from 20% power, ST-39). Westinghouse was consulted after the first attempt, and again, after the second failure to adjust. Since the P-6 permissive function (Energization of SRs) by the IRs, and proper SR to IR overlap was observed during the two trips and subsequent startups, the instrument settings were left as is, and a test exception taken.

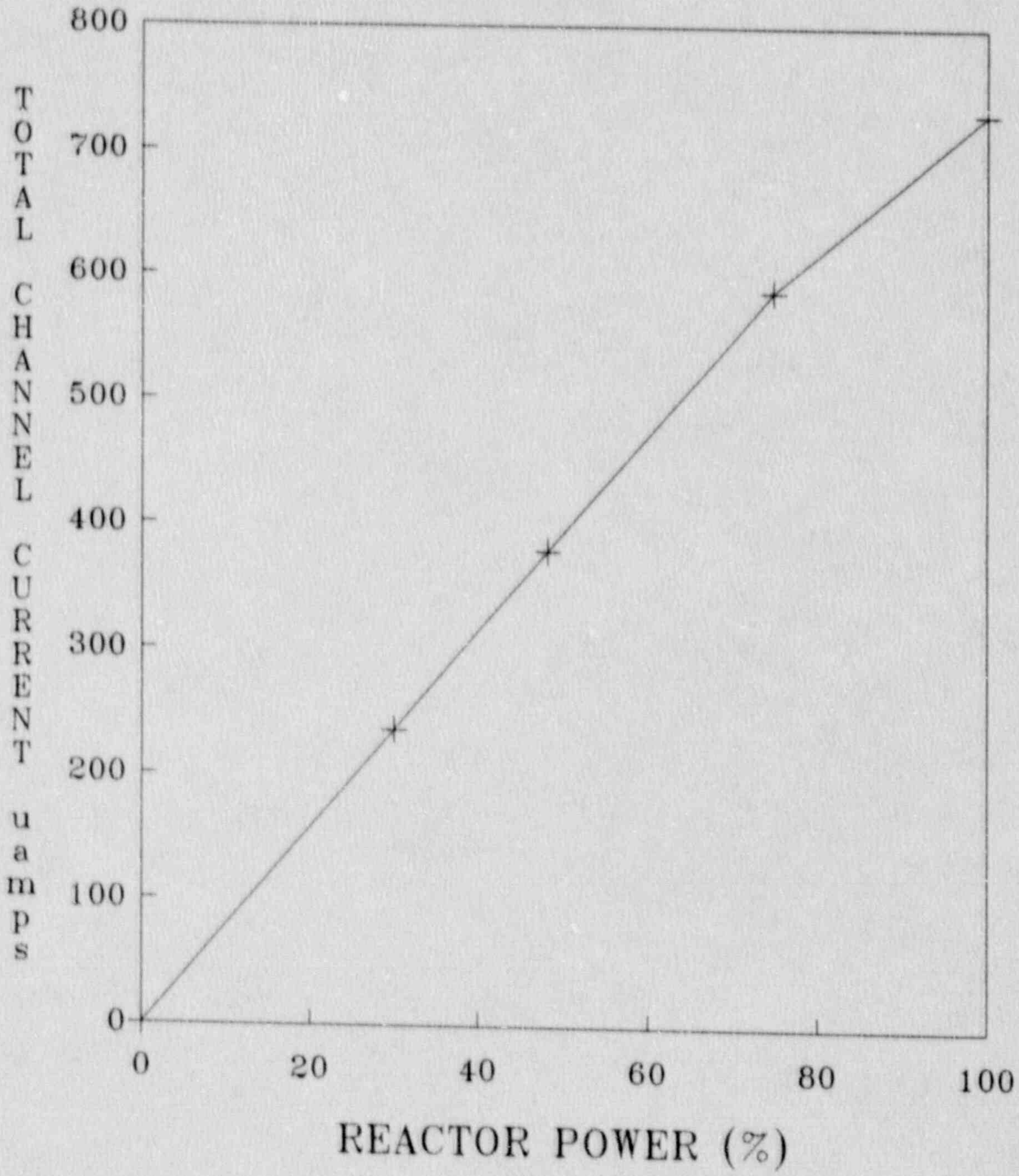
6.1 ST-13, OPERATIONAL ALIGNMENT OF NUCLEAR INSTRUMENTATION (Continued)

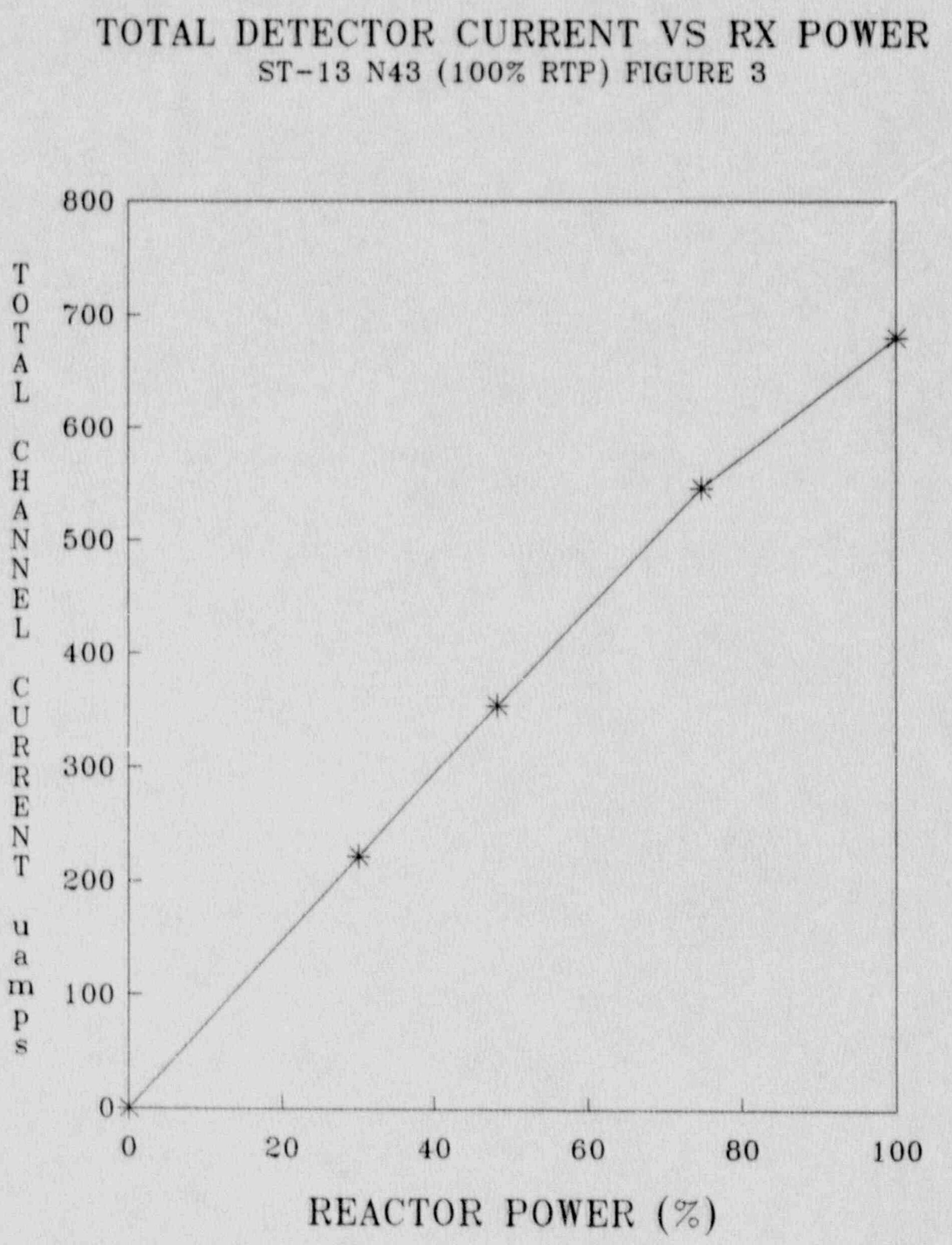
Figures 1, 2, 3 and 4 are graphs of the PR channels versus reactor power. A linear behavior was observed until the 100% power level test plateau; at 100%, a lower value of current for each detector was noted than would have been expected. The non-linearity was attributed to adjustments made in the turbine impulse pressure and T_{AVG} program at 75% power (ST-27) which caused T_{AVG} to be lower than expected at 100% reactor power. A test exception was taken.

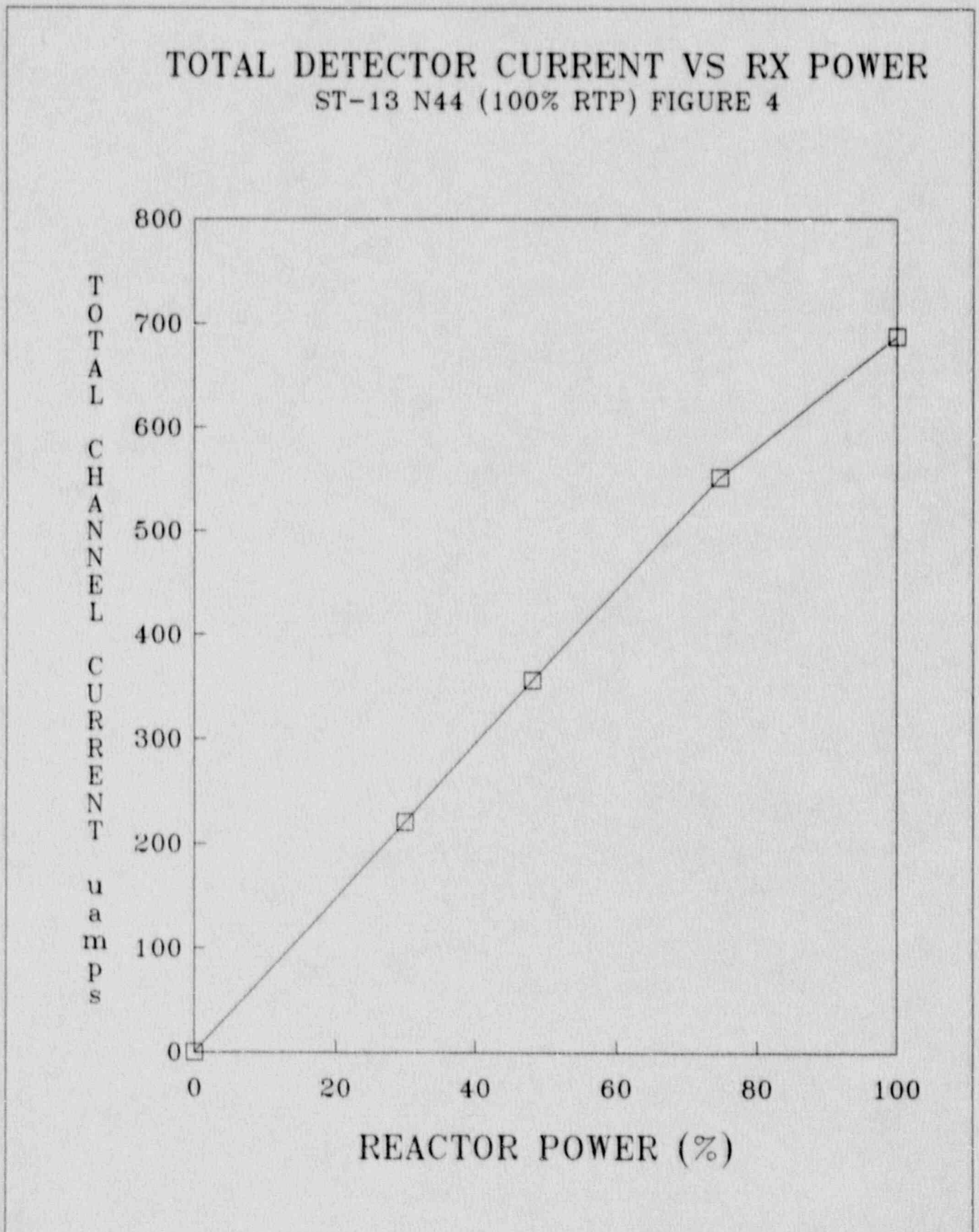

TOTAL DETECTOR CURRENT VS RX POWER
ST-13 N41 (100% RTP) FIGURE 1
TOTAL CHANNEL CURRENT uamps
800
700
600
500
400
300
200
100
0
0
20
40
60
80
100
REACTOR POWER (%)

TOTAL DETECTOR CURRENT VS RX POWER

ST-13 N42 (100% RTP) FIGURE 2




TOTAL DETECTOR CURRENT VS RX POWER
ST-13 N43 (100% RTP) FIGURE 3
TOTAL CHANNEL CURRENT uamps
800
700
600
500
400
300
200
100
0
0
20
40
60
80
100
REACTOR POWER (%)



TOTAL DETECTOR CURRENT VS RX POWER
ST-13 N44 (100% RTP) FIGURE 4
800
700
600
500
400
300
200
100
0
T O T A L C H A N N E L C U R R E N T u a m p s
0
20
40
60
80
100
REACTOR POWER (%)

6.2 ST-14.1, OPERATIONAL ALIGNMENT OF PROCESS TEMPERATURE INSTRUMENTATION

Objective

The objective of the procedure was proper alignment of the delta-T and T_{AVG} instrumentation channels at all test plateaus.

The procedure is described in FSAR, Section 14, Table 14.2-5, Sheet 17.

Discussion

Initial alignments were carried out during Low Power Physics Testing. With the plant in the hot standby condition and temperature stabilized, the T_{HOT} and T_{COLD} R/E Converters, and the T_{AVG} and delta-T Summing Amplifiers were aligned using station procedures. Temperatures were then measured; values converted to corresponding engineering units; and temperature and delta-T values calculated. Results were checked against allowable tolerances.

Further alignment of the process temperature instrumentation was dependent on data obtained during the performance of ST-26, Thermal Power Measurement and Statepoint Data Collection. At power level test plateaus of 30%, 50%, 75%, 90% and 100%, the following data was obtained from ST-26:

- Total reactor power (%) from the calorimetric analysis.
- T_{HOT} and T_{COLD} (°F) from the operating and installed spare RTD's.
- T_{AVG} (°F) from the summing amplifier output.
- Delta-T (%) from the summing amplifier.

The data was then used to calculate the T_{HOT} and T_{COLD} difference, T_{AVG}, and delta-T_{AVG} for all four loops. If values obtained failed to meet specified tolerances, then corrective actions were taken using appropriate I&C Department procedures.

At the 75% power level test plateau, the fluid specific enthalpies for each of the T_{HOT}/T_{COLD} temperatures were determined for each of the four loops at a nominal RCS pressure value of 2235 psig. The enthalpies were correlated with the calorimetric values of power and extrapolated to 100% power. From these results, extrapolated full power delta-T and T_{AVG} values were calculated and the delta-T summing amplifiers rescaled.

The procedure provided for additional rescaling, as necessary, at the 90% and 100% power level test plateaus.

Results

All acceptance criteria (applies to full power results only), were met:

- The T_{AVG} (summing amp) from each channel is within ± 0.5°F of the value calculated from the T_{HOT} and T_{COLD} R/E converter outputs.
- The delta-T (%) from each channel is within ± 1% of the calorimetric power.
- The T_{HOT} and T_{COLD} from the R/E converters are within ± 1.2°F of the installed spare RTD values.

The measured full-power average core delta-T was determined to be 56.2°F.

Process temperature parameters which were found to be out-of-tolerance, and the actions taken, are listed in Table 1.

TABLE 1
ST-14.1 OUT-OF-TOLERANCE CONDITIONS

Power Lvl	Parameter/Problem	Value	Tolerance	Action
30%	Loop 2 (NR) T_{COLD} Difference	-1.636°F	± 1.2°F	None - Test Exception Written 30% Results*
	Loop 2 (NR) Calculated Delta-T_{AVG}	0.961°F	± 0.5°F	
50%	None			
75%	Loop 3 Extrapolated Full Power T_{AVG}	589.62°F	≤ 588.5°F	Rescale T_{AVG} and Turbine Impulse Pressure (per ST-27) Test Exception Written
90%	Loop 2 Calorimetric Power - Delta-T (% Power)	-1.125%	± 1%	Rescaled
	Loop 4 Calorimetric Power - Delta-T (% Power)	-2.542%	± 1%	Rescaled
100%	Loop 2 Calorimetric Power - Delta-T (% Power)	1.154%	± 1%	Rescaled
	Loop 3 Calorimetric Power - Delta-T (% Power)	1.102%	± 1%	Rescaled

* Difference not observed at higher test plateaus.

6.3 ST-15, REACTOR PLANT SYSTEMS SETPOINT VERIFICATION

Objective

Procedure ST-15 provides verification that the initial setpoint adjustments have been made prior to startup, and serves to document setpoint modifications made during startup testing.

This procedure is described in FSAR, Section 14, TAble 14.2-5, Sheet 18.

Discussion

Initial setpoints were verified for the following plant components and systems:

1. Safeguards
2. Reactor Coolant Pumps
3. Nuclear Instrumentation (Excore)
4. Delta T - T_{AVG}
5. Pressurizer Pressure
6. Cold Overpressurization Mitigation System (COMS)
7. Pressurizer Level
8. Charging Flow
9. Rod Control
10. RCS Flow
11. Feedwater Flow
12. Steam Generator Level
13. Steam Line Pressure
14. Steam Dump System
15. Steam Generator Relief Valve Control (ASDVs)
16. Turbine Impulse Chamber Pressure

Prior to criticality, ST-15 served as a detailed listing of setpoints for plant instrumentation to be compared to the Westinghouse Precautions, Limitations and Setpoints (PLS) values, and Technical Specifications.

During Low Power Physics Tests, had any setpoint changes been required, these would have been documented by this procedure. No changes were required.

All setpoint changes made during Power Ascension Testing were documented by ST-15, and the final PAT requirement prior to the 100% power plateau results review was a verification that any changes made were incorporated into both the plant hardware and station procedural text.

Results

The acceptance criterion was met; all setpoint changes during PAT were documented in the procedure.

A total of 17 setpoint changes were necessary and were entered into ST-15; of those, 7 changes were later superseded by test results at higher power level test plateaus.

6.4 ST-26, THERMAL POWER MEASUREMENT AND STATEPOINT DATA COLLECTION

Objective

The objective of this procedure was a calorimetric determination of reactor power, and verification of main steam and feedwater performance from various primary and secondary process data.

The procedure is described in FSAR, Section 14, Table 14.2-5, Sheet 29.

Discussion

Primary and secondary process parameters were measured, using station procedures, and from these data a calorimetric determination of power was made. Calorimetric determinations of thermal power were made at the 30%, 50%, 75%, 90% and 100% power level test plateaus.

ST-26 specified the following stability requirements for a calorimetric power level determination:

- RCS temperature (T_{AVG}) changing less than 1°F/hour.
- Core power (power range) changing < 0.5%/hour.
- Steam generator water level at 50% (48-52%).
- Pressurizer pressure at 2235 psig (2210-2260 psig).
- Pressurizer level +0/-2% of programmed level.
- Blowdown secured.
- Charging and letdown flow constant.

This procedure served as the data gathering procedure for a number of the instrumentation calibration and alignment procedures.

Operation of main steam and feedwater systems were observed throughout the entire test program and demonstrated satisfactory performance.

Results

All acceptance criteria were met.

At hot zero power (HZP) an initial set of process data was taken using GETARS.

At each of the power level test plateaus, data was collected and analysis performed using station procedures RN 1730, Precision Calorimetric/RCS Flow Rate Measurement, and RN 1731, Secondary Heat Balance.

A problem was encountered at 30% power with the method for locally determining feedwater temperatures; a wiring change in the test instrumentation circuitry was implemented to correct the problem.

6.4 ST-26, THERMAL POWER MEASUREMENT AND STATEPOINT DATA COLLECTION (Continued)

The calculated RCS flow rate, (required only at the 50% power level test plateau) was:

RCS Flow Rate Measurement Surveillance = 416,771 gpm
Technical Specification Limit = > 391,000 gpm

Calculated results based on the collected data were consistent with plant conditions at the time. There were two test exceptions; both were of a minor procedural nature.

6.5 ST-27, STARTUP ADJUSTMENTS OF THE REACTOR CONTROL SYSTEM

Description

The procedure obtained and evaluated the data necessary to determine the T_{AVG} program that will result in the highest possible steam pressure to assure optimum plant efficiency, while maintaining pressure for the turbine and T_{AVG} within required limitations. The system pressure and temperature data provided a basis for adjustments of the reactor control system (T_{REF} on the T_{AVG} programming module).

The procedure is described in FSAR, Section 14, Table 14.2-5, Sheet 30.

Discussion

The following data, obtained from ST-26, Thermal Power Measurement and Statepoint Data Collection, at hot zero power, 30% power and all subsequent power level test plateaus, were used in this procedure:

Loops 1,2,3,4 - T_{HOT}, T_{COLD} and T_{AVG}.
SG Pressure
Calorimetric Reactor Power
Turbine Impulse Pressure

The data were averaged and used to generate graphs of:

RCS Temperature vs Power
S/G Pressure vs Power
Turbine Impulse Pressure vs Power

At the 50% power level test plateau, extrapolated full power values, obtained from the graphs, were compared to the design full power steam generator pressure and the design (or current span) turbine impulse pressure. If the allowable tolerances (± 10 psi and ± 50 psi, respectively) were exceeded, the turbine impulse pressure and full power T_{AVG} were adjusted to the extrapolated values. The process was repeated at each subsequent power level test plateau (75%, 90% and 100%). The allowable tolerance for turbine impulse pressure, at 100%, was ± 10 psi.

Results

The acceptance criteria were met;

- The final full power steam generator pressures are 985 ± 10 psig.
- The final full power T_{REF} does not exceed 588.5°F.

No adjustments at the 50% power level test plateau were required. Full power extrapolations of average S/G pressure and turbine impulse pressure were within required tolerances.

At the 75% power level test plateau, full power extrapolated values were:

- RCS Temperature (Average T_{AVG}) = 590°F
- S/G Pressure = 1035 psia
- Turbine Impulse Pressure = 730 psia

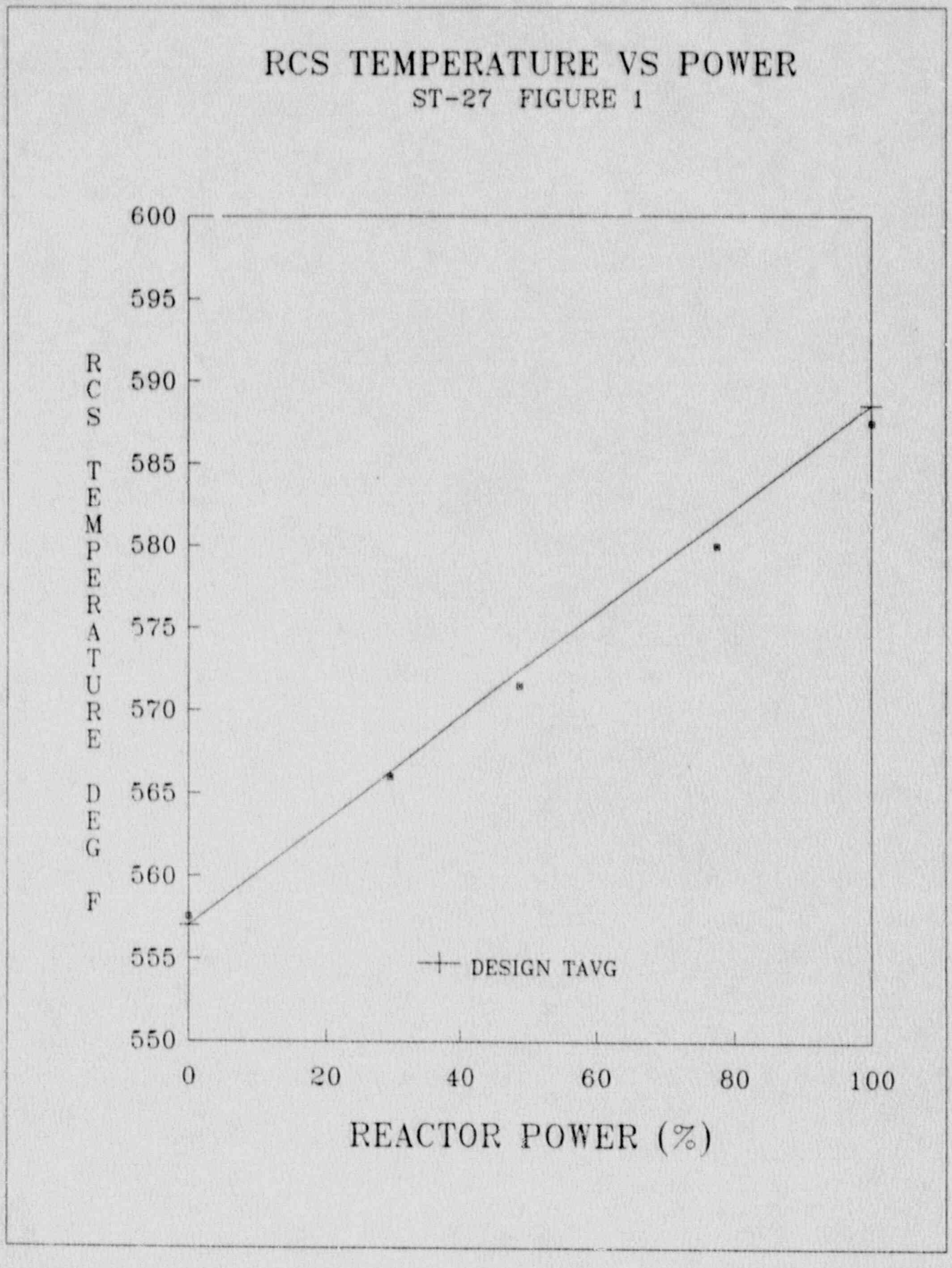
and rescaling of turbine impulse pressure and full power T_{AVG} program was necessary.

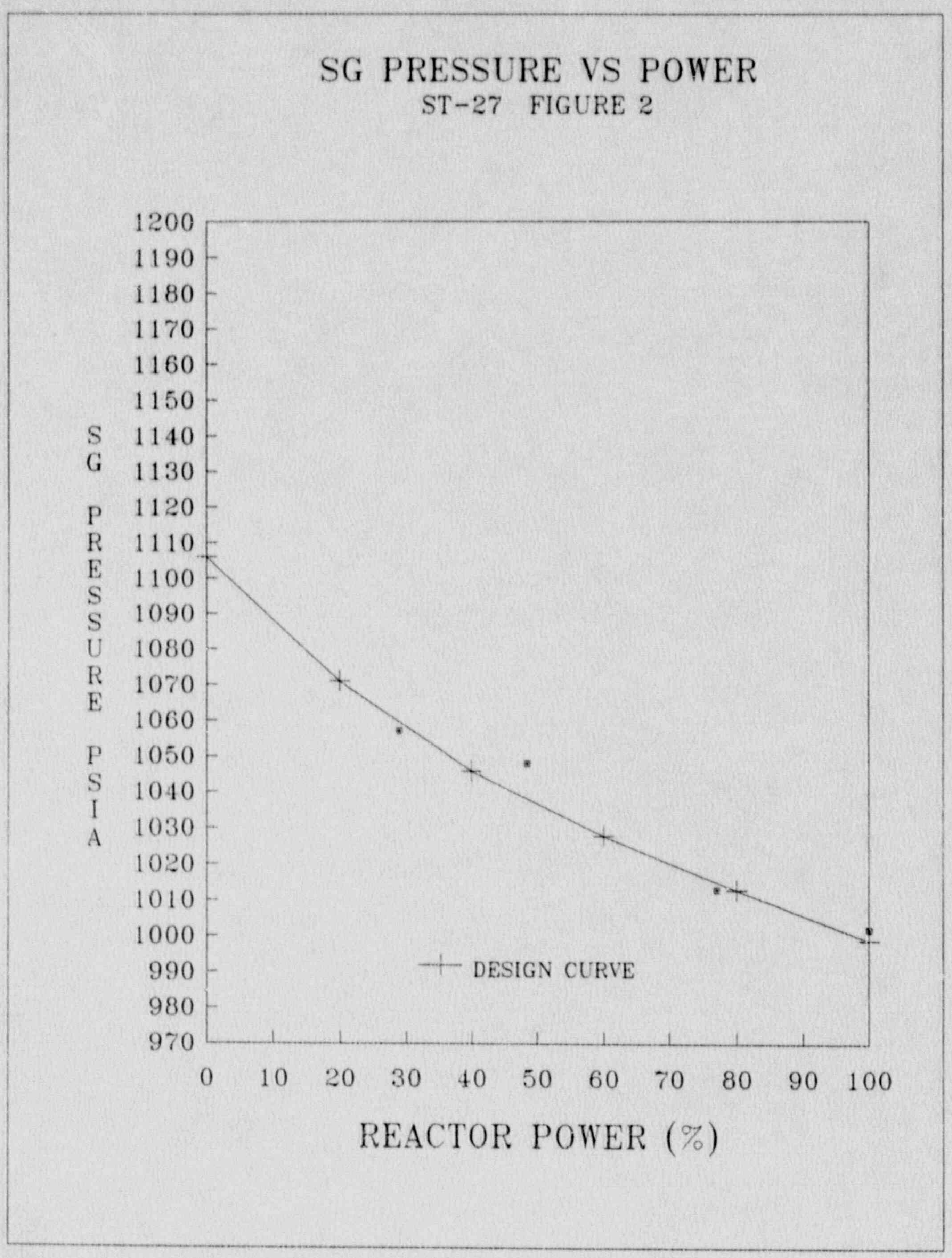
At 90% no adjustments were required.

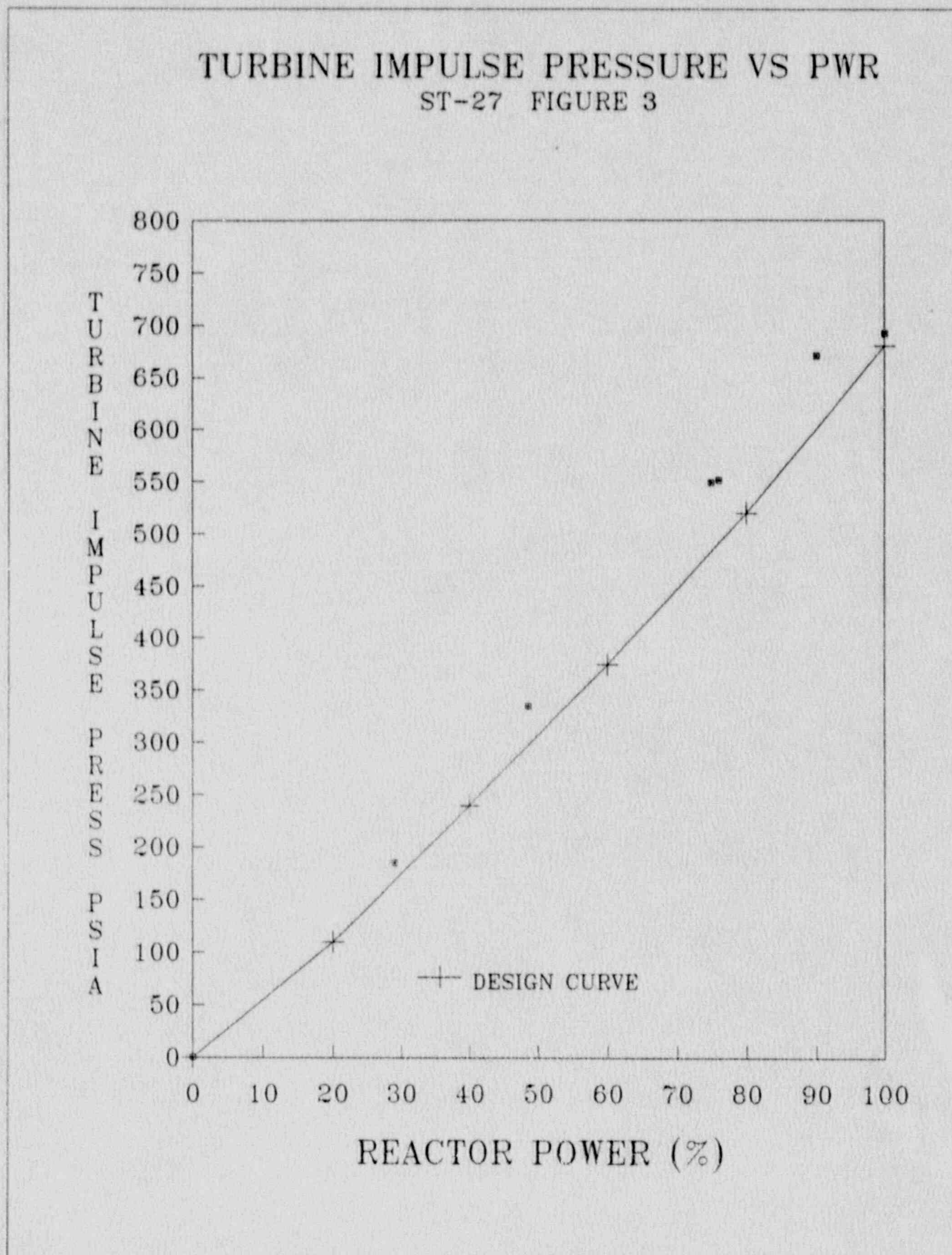
The moisture separator reheaters (MSR) were placed in service prior to testing at the 100% power level test plateau, which resulted in the turbine impulse pressure being reduced from the full power scaling of 730 psia (based on operation without MSRs). Rescaling of turbine impulse pressure and the full power T_{AVG} program resulted in final values of:

- Full Power T_{AVG} = 587.5°F
- S/G Pressure, A = 988.5 psig
  B = 984.5 psig
  C = 990.4 psig
  D = 986.0 psig
- Turbine Impulse Pressure = 691.4 psia
- S/G Saturation Temperature = 544.8°F

Graphs of RCS Temperature, S/G Pressure and Turbine Impulse Pressure versus Reactor Power are given in Figures 1, 2 and 3 respectively.


RCS TEMPERATURE VS POWER
ST-27 FIGURE 1
RCS TEMPERATURE DEG F
600
595
590
585
580
575
570
565
560
555
550
0
20
40
60
80
100
REACTOR POWER (%)
DESIGN TAVG


SG PRESSURE VS POWER
ST-27 FIGURE 2
SG PRESSURE PSIA
1200
1190
1180
1170
1160
1150
1140
1130
1120
1110
1100
1090
1080
1070
1060
1050
1040
1030
1020
1010
1000
990
980
970
DESIGN CURVE
0
10
20
30
40
50
60
70
80
90
100
REACTOR POWER (%)


TURBINE IMPULSE PRESSURE VS PWR
ST-27 FIGURE 3
TURBINE IMPULSE PRESS PSIA
800
750
700
650
600
550
500
450
400
350
300
250
200
150
100
50
0
0 10 20 30 40 50 60 70 80 90 100
DESIGN CURVE
REACTOR POWER (%)

6.6 ST-28, CALIBRATION OF STEAM AND FEEDWATER FLOW INSTRUMENTATION

Objective

The objective of this test was calibration of the main steam flow transmitters based on feedwater flow measurements.

This procedure is described in FSAR Section 14, Figure 14.2-5, Sheet 31.

Discussion

The procedure utilized the installed station feedwater flow instrumentation as the measured parameter of steam flow to empirically determine values used to calibrate the main steam flow transmitters. Data from ST-26, Thermal Power Measurement and Statepoint Data Collection, was used to determine steam flow and new calibration values. ST-28 was an ongoing startup test; data was collected at each test plateau:

At hot zero power - Main steam flow transmitter calibrations were performed to monitor any zero shift changes from cold conditions.

At each power level test plateau - Steam and feed flow calibration data were collected at the 30%, 50%, 75%, 90% and 100% power level test plateaus.

At the 100% power level test plateau - Steam flow and feed flow transmitter outputs were compared; steam flow output was corrected to agree with feed flow within the required accuracy.

Results

The acceptance criterion was met; the steam flow transmitter outputs for each channel have been matched to the associated feed flow transmitter outputs to within ± 100,000 pph (± 2%).

The original acceptance criterion, ± 25,000 pph (± 0.5%), could not be met, and the higher tolerance was introduced by procedure change after review of the basis for the initial value with Westinghouse. The review of the basis for the ± 0.5%, determined that the value was derived from a safety analysis performed for a Hi Steam Flow Rate Circuit protective function, utilized at some other Westinghouse plants. Seabrook does not utilize this protective function. The Seabrook system uses steam flow values along with feed flow to produce a mismatch error signal for S/G level control circuits. The ± 2% acceptance limit was recommended to replace the ± 0.5% value.

6.6 ST-28, CALIBRATION OF STEAM AND FEEDWATER FLOW INSTRUMENTATION (Continued)

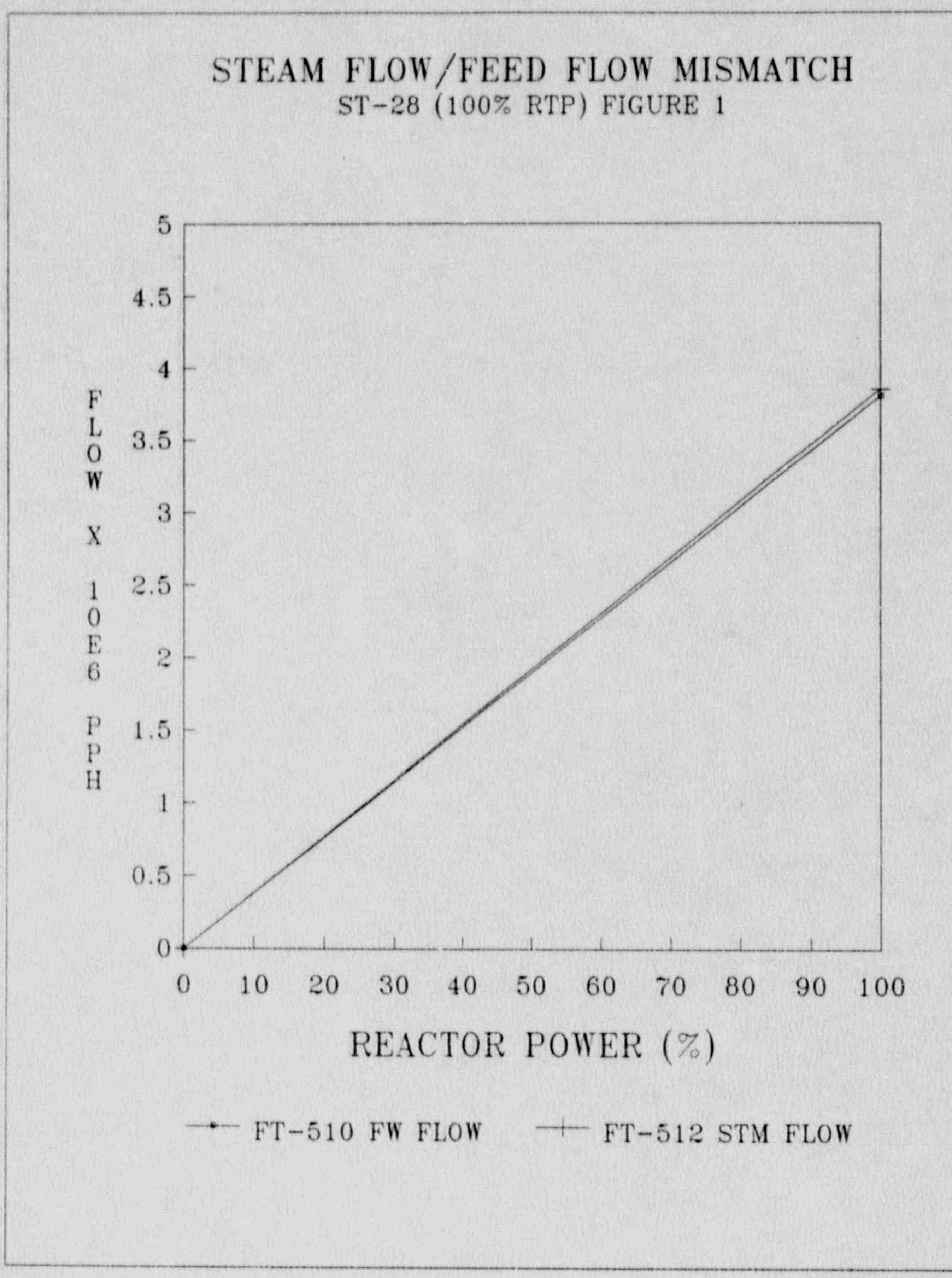
At each power level test plateau, for each loop, the steam flow transmitter data, for both transmitters in the same loop, was required to agree within ± 1% (± 50,000 pph). The steam flows did not always meet these limits:

Power Level	Loop	Flow Difference
30%	1	88,400 pph
	2	74,300 pph
50%	1	82,000 pph
	2	71,000 pph
	3	53,000 pph
75%	2	59,900 pph
	4	60,600 pph
90%	2	62,000 pph
100%	2	68,600 pph

The procedure required adjustments to steam flow transmitters if the difference limit could not be met. However, when steam flow and feed flow for a given loop were plotted versus reactor power, in all cases, the steam flow/feed flow mismatch at the extrapolated maximum flow rate was less than the allowable mismatch, 700,000 pph (for power level test plateaus < 100%). Four test exceptions were written, with Westinghouse concurrence, to continue to the next test plateau without adjustment, based on the allowable mismatch.

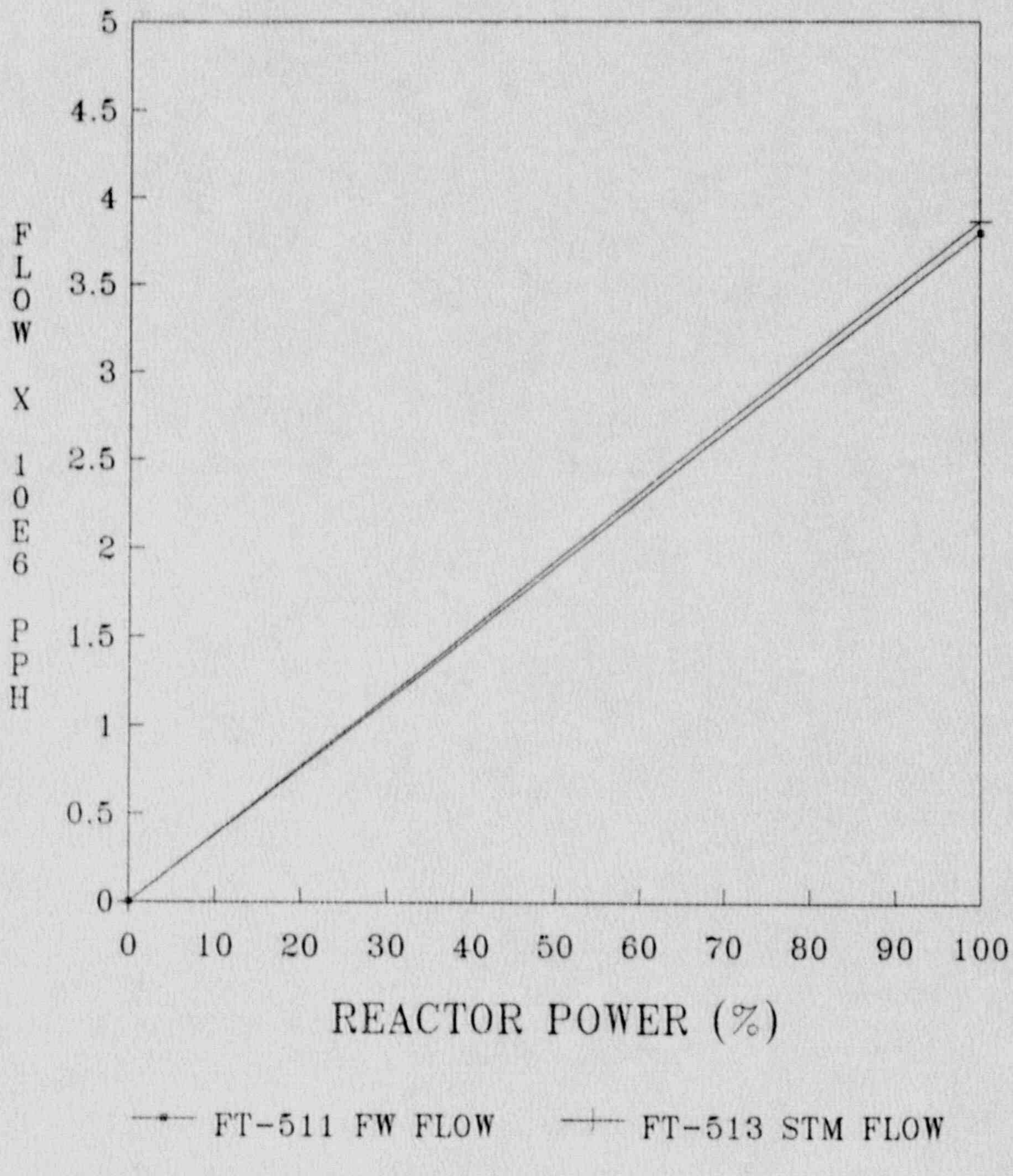
At the 100% power level test plateau, the eight steam flow transmitters were respanned to new full power delta-P values, and additional data taken. The maximum flow difference between steam flow transmitters in the same loop was then 36,700 pph.

Examples (Loop 1) of the final Steam/Feed Flow Mismatch graphical results are given in Figures 1 and 2.


STEAM FLOW/FEED FLOW MISMATCH
ST-28 (100% RTP) FIGURE 1
FLOW X 10E6 PPH
0
0.5
1
1.5
2
2.5
3
3.5
4
4.5
5
0
10
20
30
40
50
60
70
80
90
100
REACTOR POWER (%)
FT-510 FW FLOW
FT-512 STM FLOW

STEAM FLOW/FEED FLOW MISMATCH

ST-28 (100% RTP) FIGURE 2



6.7 ST-36, AXIAL FLUX DIFFERENCE INSTRUMENTATION CALIBRATION

Objective

Startup calibration of the excore power range detectors, and calibration data for the overtemperature delta-T f(Delta-I) summing amplifier gains are provided by this test. In addition, data for calibration of Main Control Board (MCB) axial flux difference (AFD) indicators and the AFD monitoring program on the Main Plant Computer System (MPCS) is obtained.

Discussion

The AFD test was performed at the 50%, 75% and 100% power level test plateaus. The three sections of the complete test are:

1. A preliminary incore-excore calibration, using a three point method for determining the relationship between incore and excore AFD, prior to escalation to and above 50% power.

2. Incore-excore calibration, using a multi-point method for determining the relationship between incore and excore AFD, at greater than 75% RTP.

3. A one-point calibration verification at 100% RTP.

The relationship between incore and excore axial flux difference was determined by plotting the incore flux difference versus excore detector currents. From this plot, slopes and intercepts permitted calculation of summing amplifier gains, MPCS AFD monitoring program constants and alignment of the MCB AFD instrumentation.

Station operating procedures for incore-excore calibration, incore-excore surveillance, flux mapping, and axial flux difference control were utilized in the test procedure.

Results

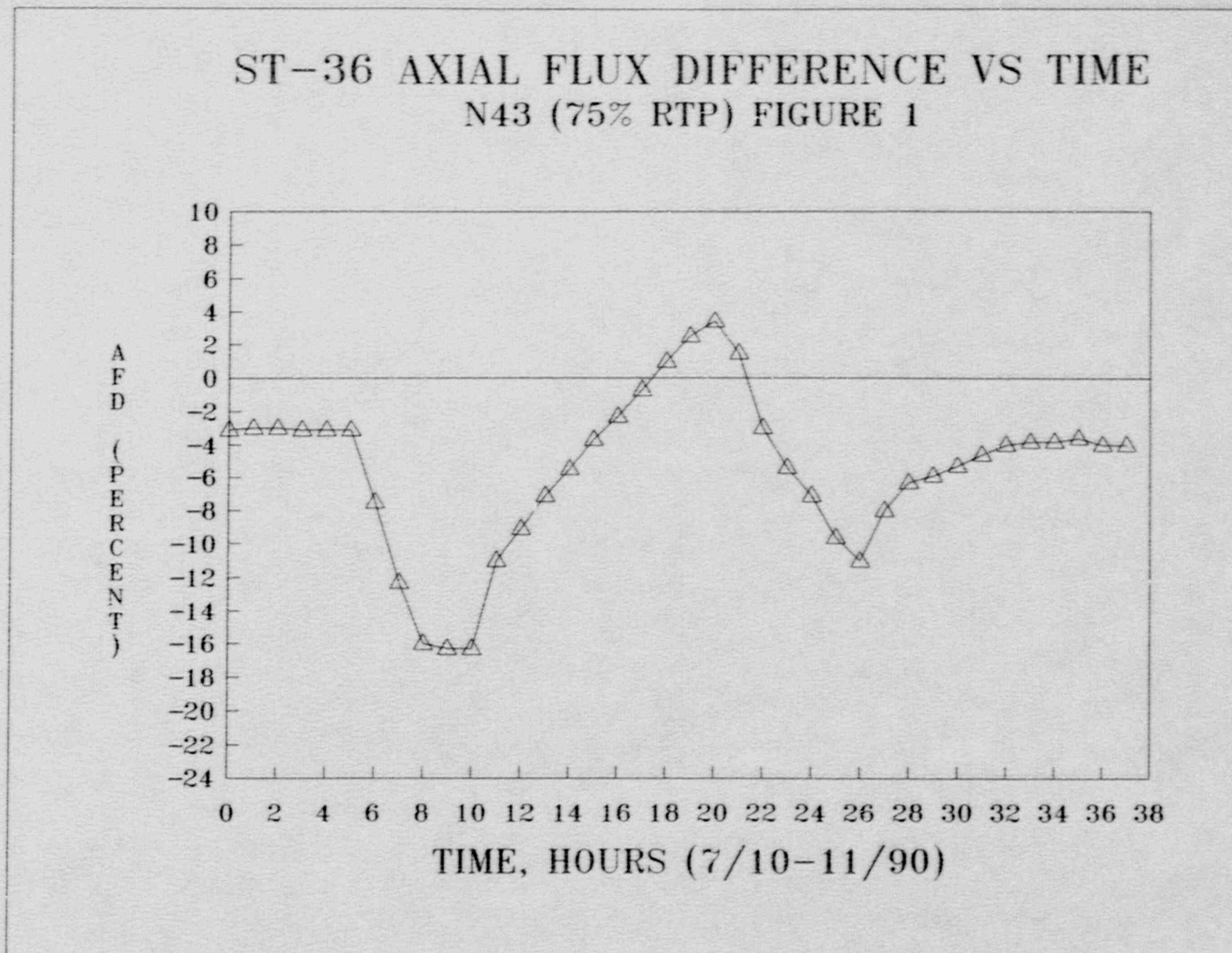
The acceptance criteria were met; determination of the relationship between incore axial offset and AFD, and calibration of the AFD instrumentation.

At the 50% power level test plateau, after an initial full core flux map (FCFM) was taken, a negative AFD was produced by dilution and compensating insertion of the controlling rod bank. An initial quarter core flux map (QCFM) was taken, followed by additional dilution. A second QCFM was then taken and the controlling bank was borated back to its initial position. From the data, extrapolated full power currents for NIS as a function of AFD, and the gain of the f(delta-I) portion of the over temperature delta-temperature (OTdeltaT) protection system were determined. The results were then used by I&C to calibrate power range (PR) and (OTdeltaT) inputs.

At the 75% power level test plateau, an FCFM was followed by dilution and control rod insertion until the indicated AFD was near the predetermined limit. The initial QCFM was taken, and dilution continued. After a 2-hour hold, a second QCFM was taken, boration was started and continued until the controlling bank (Bank D) was near its starting position. Five additional QCFMs were taken as the axial xenon oscillation progressed. The oscillation varied between -16% and +3% AFD. When data collection was completed, the xenon oscillation was terminated. Some scatter in the incore to excore data was noted, and some data points were discarded; however, the requirement of four useable points was always met. From the data, I&C carried out the necessary calibrations as noted above.

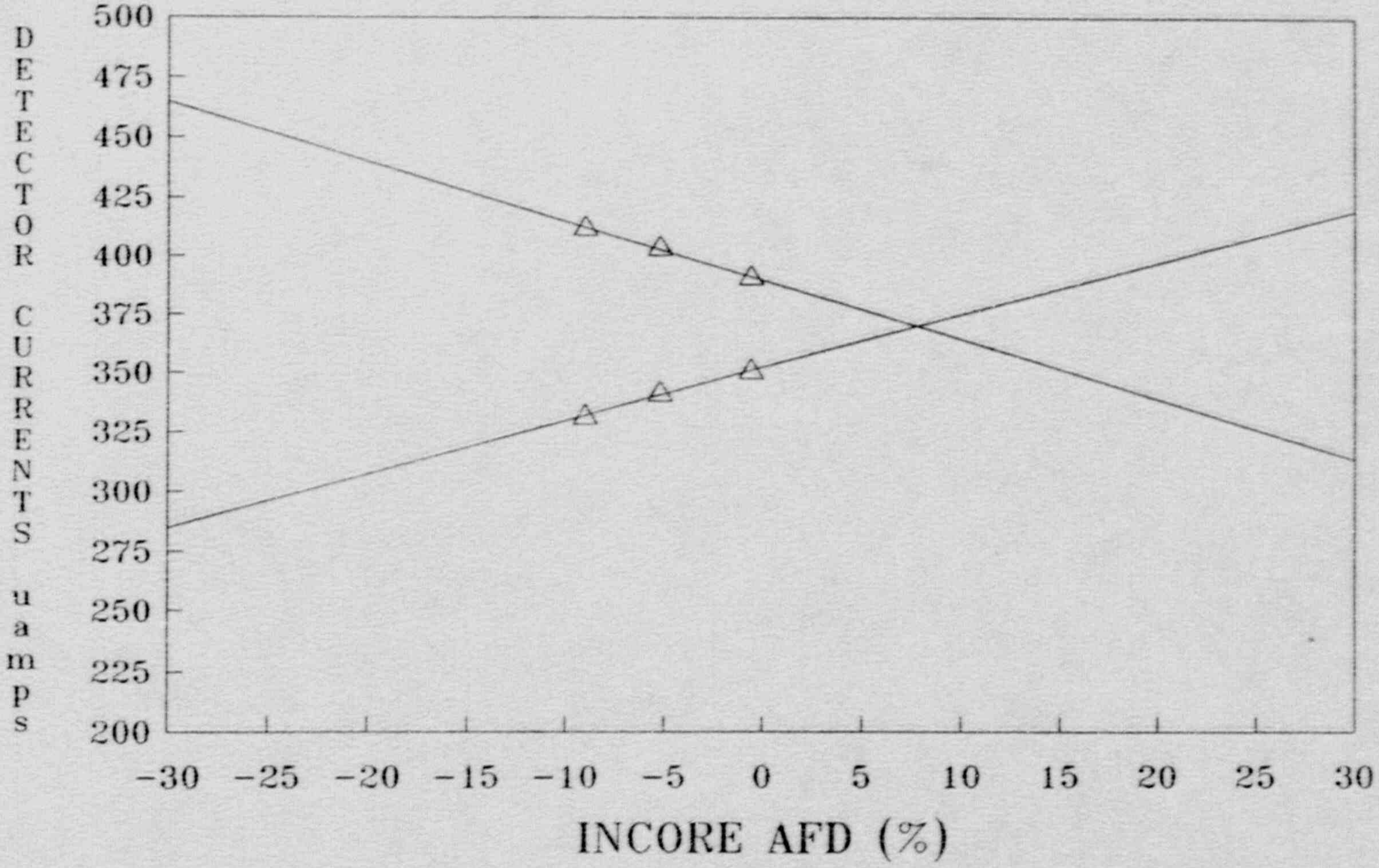
Results of the calibrations at the 75% power level test plateau are given in Figures 1-9.

At the 100% power level test plateau, a FCFM was taken to verify the previous calibrations. The calibrations are acceptable if the incore-excore comparison differs by < 3%; the maximum actual difference was 0.55%.

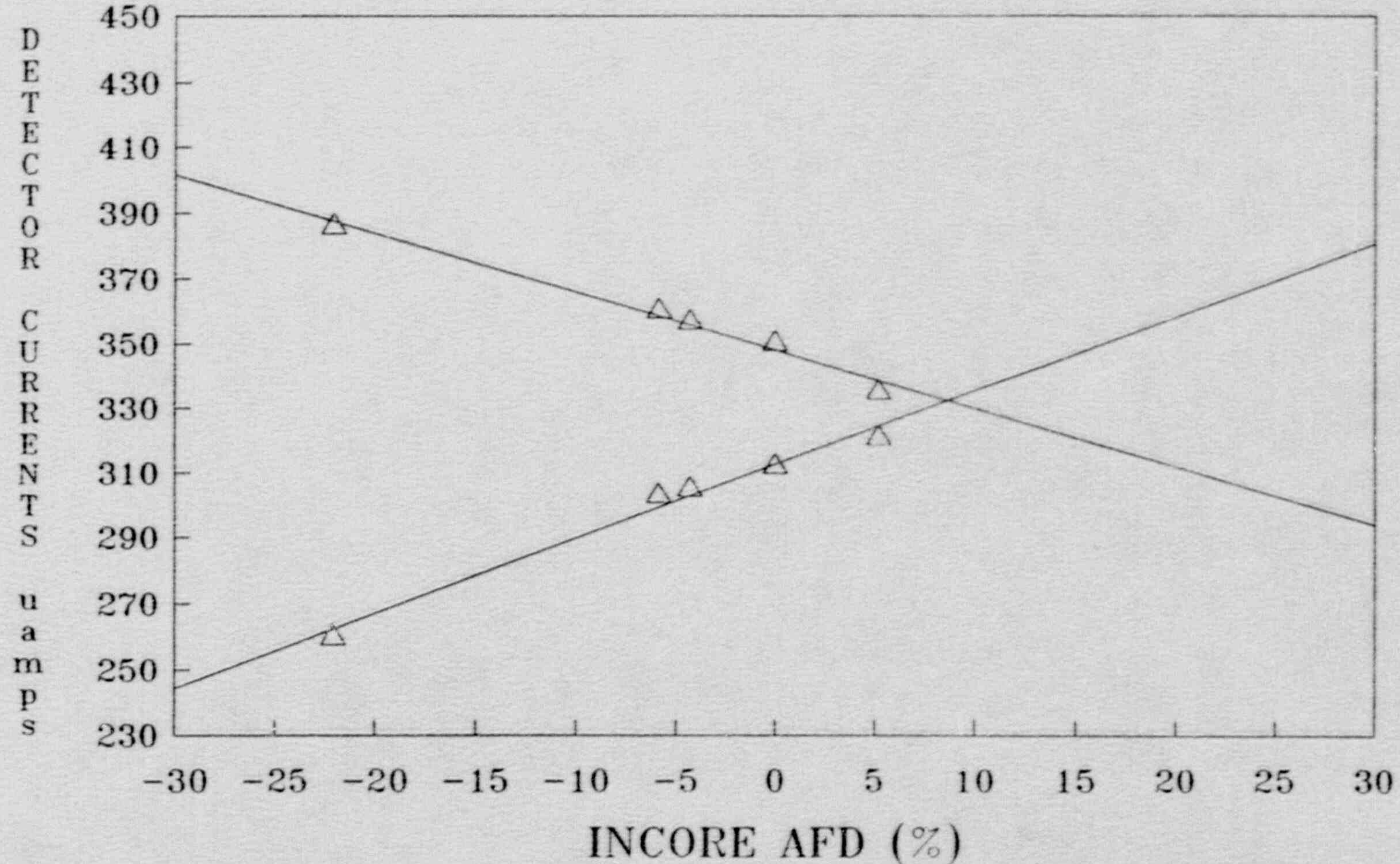


ST-36 AXIAL FLUX DIFFERENCE VS TIME
N43 (75% RTP) FIGURE 1
AFD (PERCENT)
10
8
6
4
2
0
-2
-4
-6
-8
-10
-12
-14
-16
-18
-20
-22
-24
0 2 4 6 8 10 12 14 16 18 20 22 24 26 28 30 32 34 36 38
TIME, HOURS (7/10-11/90)

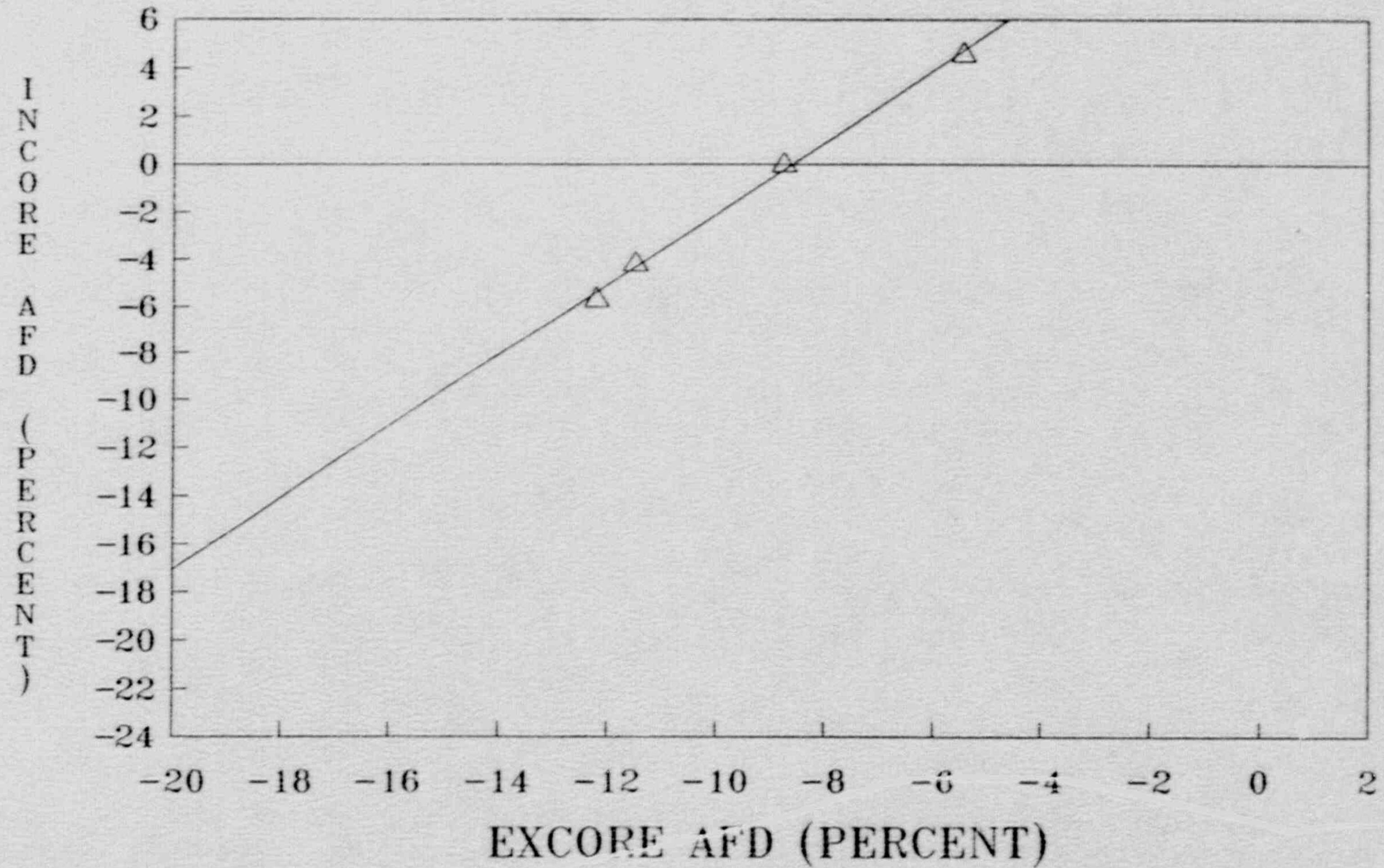
ST-36 DETECTOR CURRENTS VS INCORE AFD
N41 (75% RTP) FIGURE 2



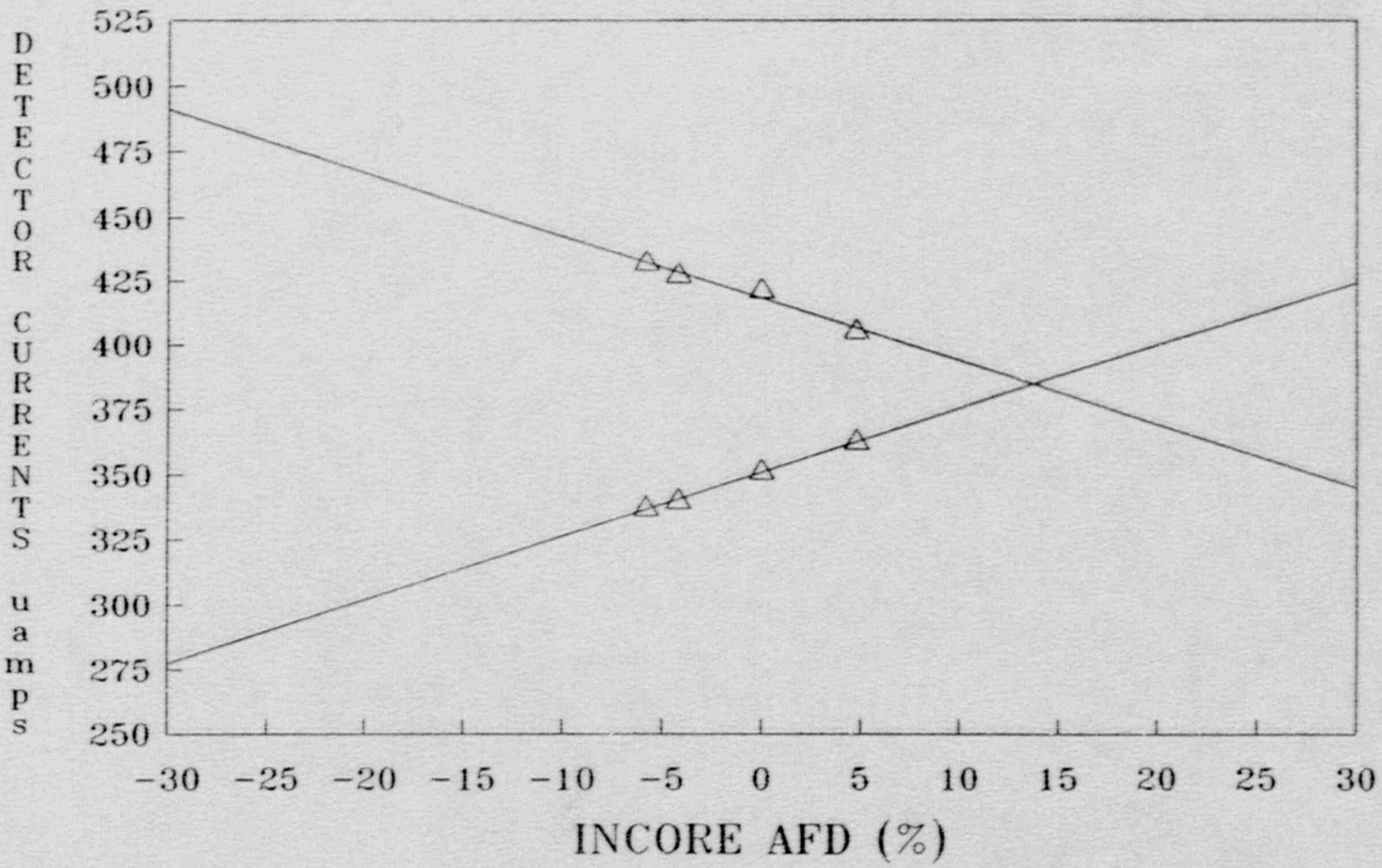
ST-36 DETECTOR CURRENTS VS INCORE AFD
N41 (75% RTP) FIGURE 3



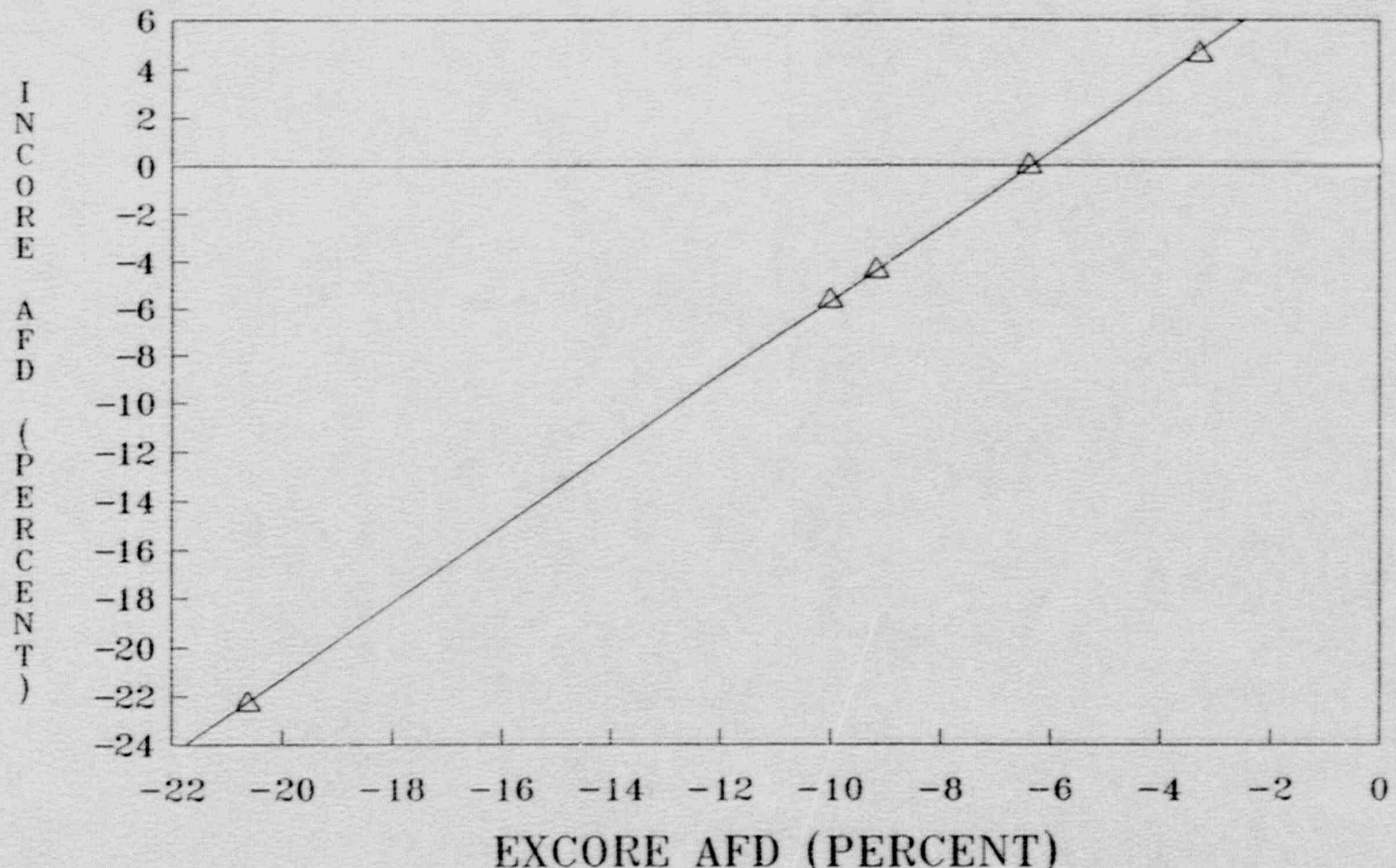
ST-36 INCORE AFD VS EXCORE AFD
N42 (75% RTP) FIGURE 4



ST-36 DETECTOR CURRENTS VS INCORE AFD
N42 (75% RTP) FIGURE 5

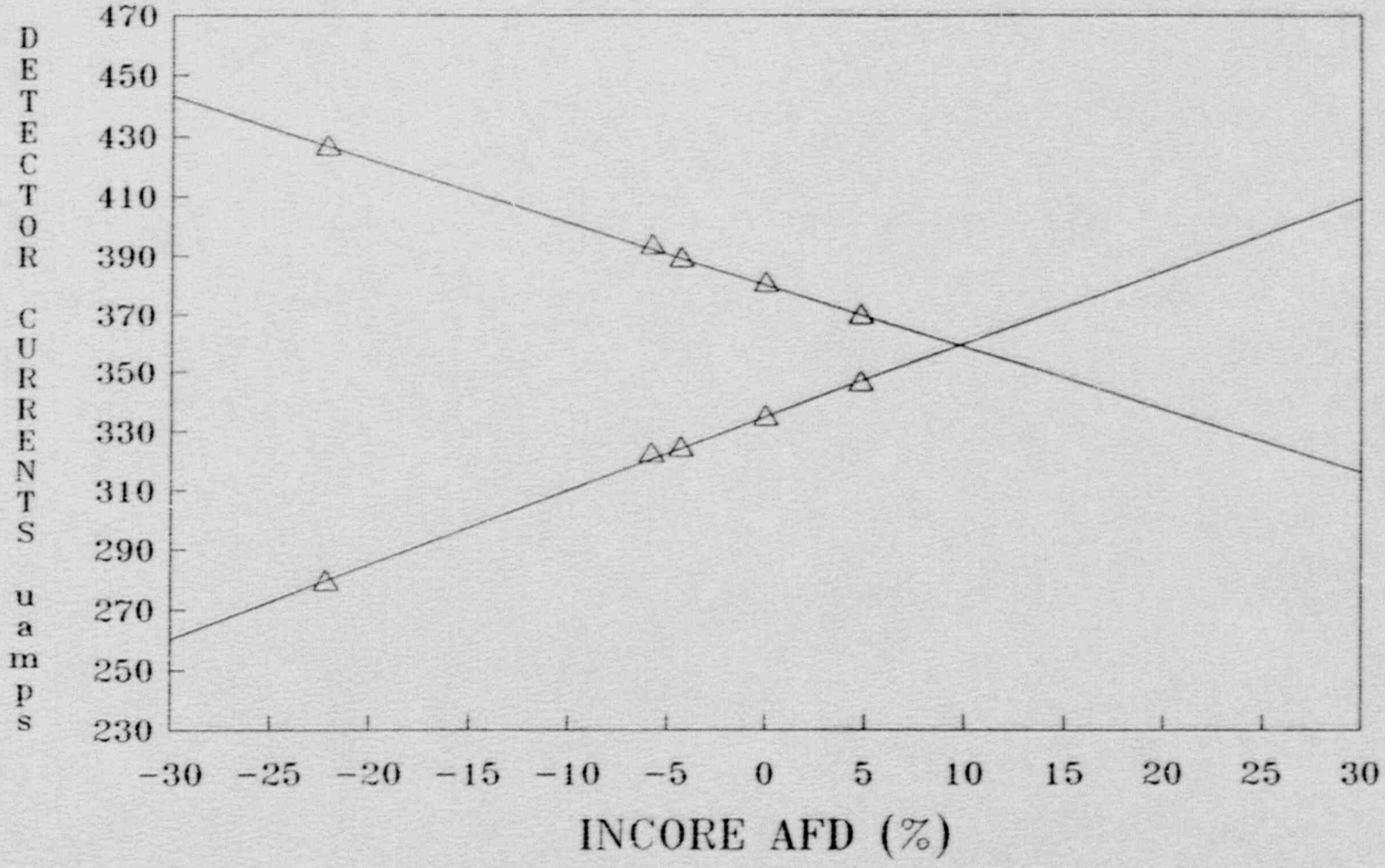


ST-36 INCORE AFD VS EXCORE AFD
N43 (75% RTP) FIGURE 6



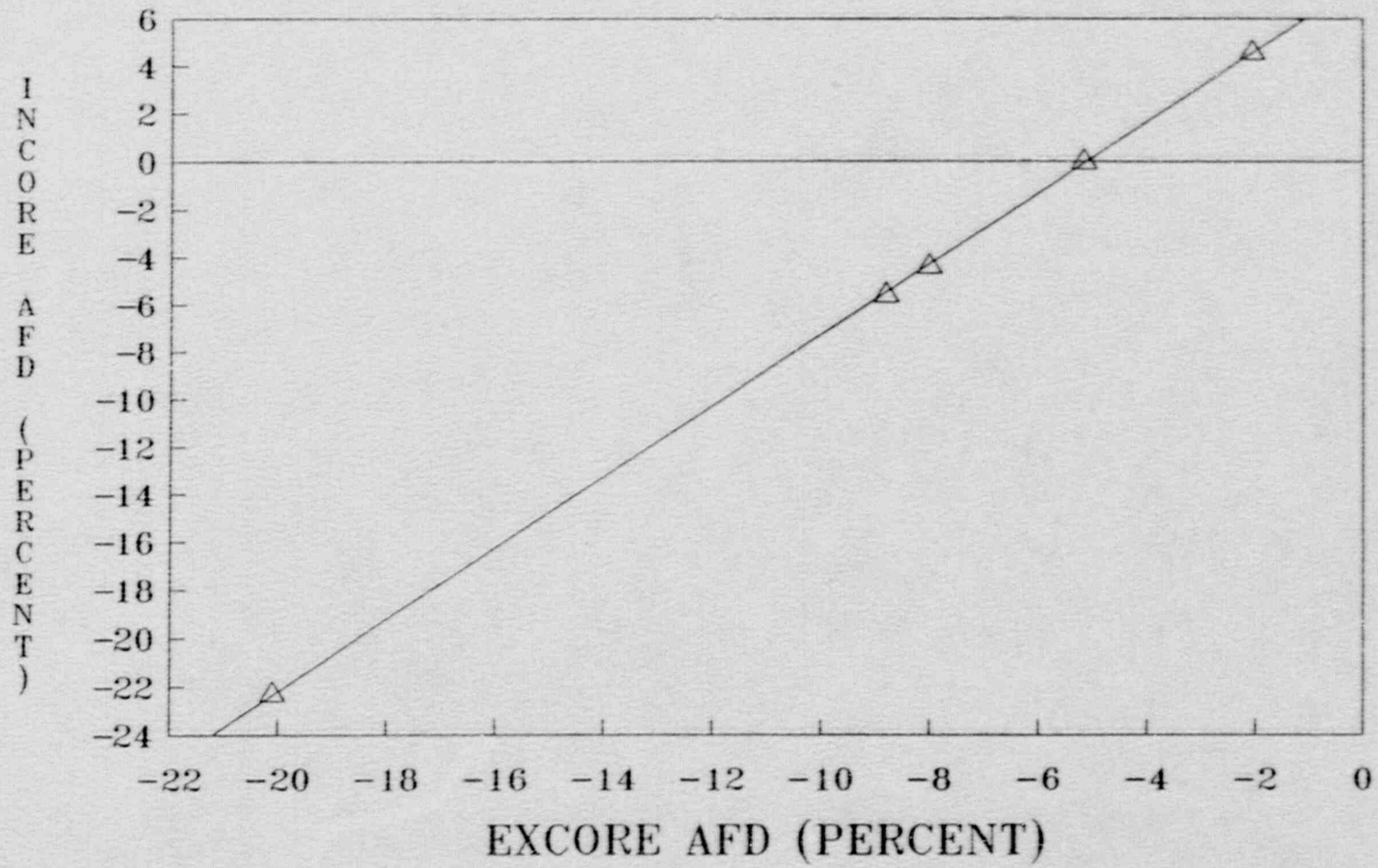
ST-36 DETECTOR CURRENTS VS INCORE AFD

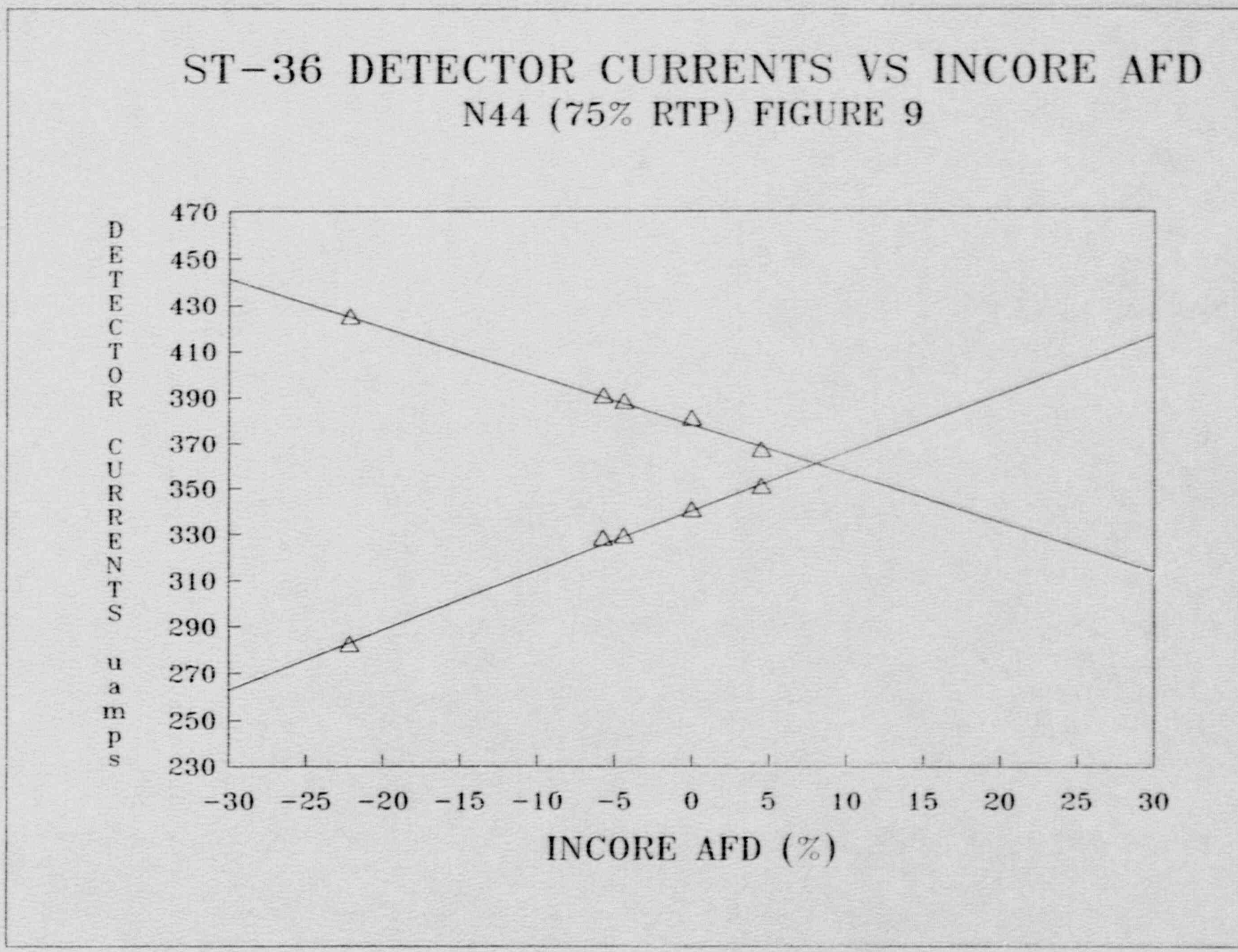
N43 (75% RTP) FIGURE 7



ST-36 INCORE AFD VS EXCORE AFD

N44 (75% RTP) FIGURE 8




ST-36 DETECTOR CURRENTS VS INCORE AFD
N44 (75% RTP) FIGURE 9
DETECTOR CURRENTS uamps
470
450
430
410
390
370
350
330
310
290
270
250
230
-30 -25 -20 -15 -10 -5 0 5 10 15 20 25 30
INCORE AFD (%)

7.0 GENERAL PLANT TESTING

Contents:

7.1 ST-41, RADIATION SURVEY

Objective

The objective of this test was determination of neutron and gamma dose rate levels and verification of operation of selected radiation monitors by comparison of monitor response to survey readings.

Discussion

Radiation surveys were conducted to verify that the radiation protection design features of the facility, as described in the FSAR, have been met. Data was taken at the 50% and 100% power level test plateaus.

Health Physics personnel, using survey instruments and the normal Health Physics shield survey procedure, conducted the tests. Survey maps of the areas to be evaluated were utilized, and any discrepancies noted and evaluated. During the surveys, the incore detection system was caution tagged to prevent operation of the system with survey personnel in the area.

Results

All acceptance criteria were met;

1. Neutron and gamma radiation dose rates have been measured at the required locations, and high radiation areas have been properly identified.

2. Except for documented discrepancies, all measured dose rates are within zone criteria.

3. The response of the radiation monitors agrees with the survey results within ± 20% or are dispositioned per the survey procedure.

Seven shield survey discrepancies were found in the final (100%) survey. All were dispositioned in the survey procedure. Three were due to streaming at doorways, not due to defects in shielding. One, a higher than expected reading, is located where the shield thickness was decreased to permit maintenance on pressurizer heaters. The problem is very localized and would not affect the general area dose rate. The remaining three discrepancies, survey readings which exceeded the greater than 20% criteria, resulted when the survey instrument was reading below a minimum sensitivity of the monitor.

7.2 ST-42, WATER CHEMISTRY CONTROL

Objective

The procedure demonstrated that chemical and radiochemical control and analysis systems function to maintain primary and secondary water chemistry within the requirements of the Station Chemistry Control Program.

Water chemistry control is described in FSAR, Section 14, Table 14.2-5, Sheet 45.

Discussion

With the plant operating under steady-state conditions at the 30%, 50%, 75% and 100% power level test plateaus, samples of reactor coolant, steam generator water, and feedwater were obtained and analyzed using the Primary Chemistry Control and the Secondary Chemistry Control portions of the Seabrook Station Chemistry Program. A minimum of four hours stable power conditions at the specified power level test plateau was required.

Readings of selected secondary system on-line analyzers were compared to respective sample analyses to determine in-plant analyzer reading accuracy.

Results

All acceptance criteria were met; as discussed below, chemistry in several systems was out of specification, but was acceptable by Westinghouse for a plant at this operating stage.

Agreement between selected secondary system on-line analyzers and the respective sample analyses was very good. Table 1 lists values obtained and acceptable tolerance.

At the 30% power level test plateau, cation conductivity and sulfates in the steam generators, and the specific cation conductivity in the main steam, feedwater and condensate systems were above the limits in the Westinghouse Secondary Water Chemistry Manual. The source of these contaminates was identified as original system preservatives. With Westinghouse and Chemistry Department approval, a test exception was prepared. The mechanism for removal is continued operation.

Again, data taken at the 50%, 75% and 100% power levels indicated a continued problem with contamination due to system preservatives. Westinghouse, after a review of the results, noted that these were typical of similar plants at this operating stage. The Chemistry Department, with Westinghouse concurrence has approved the results.

Plant chemistry out-of-specification results at the 100% power level test plateau are given in Table 2.

TABLE 1
ST-42 PROCESS INSTRUMENT/GRAB SAMPLE COMPARISONS

Power Level: 100%

Sample Point	Parameter	Analyzer Reading	Sample Analysis	Tolerance *
Blowdown S/G A	Na +	14.0	11.2	≤ 20 ppb
	pH	8.9	8.96	± 0.5 pH
	Cation Cond	3.4	3.43	± 0.5 umho
S/G B	Na +	9.5	10.8	≤ 20 ppb
	pH	8.8	9.04	± 0.5 pH
	Cation Cond	3.5	3.46	± 0.5 umho
S/G C	Na +	12.0	11.0	≤ 20 ppb
	pH	8.8	9.04	± 0.5 pH
	Cation Cond	3.4	3.4	± 0.5 umho
S/G D	Na +	10.0	9.3	≤ 20 ppb
	pH	8.9	9.07	± 0.5 pH
	Cation Cond	3.1	2.96	± 0.5 umho
Cond Pump Discharge	Oxygen	2.0	<5.0	± 5 ppb
	pH	9.0	9.2	± 0.5 pH
	Na+	0.75	<1.0	± 3 ppb
	Cation Cond	0.24	0.243	± 0.1 umho
Condensate Htr 22 Outlet	Hydrazine	26.0	26.0	± 20%
Feedwater Outlet 26 Htr	Silica	9.0	<10.0	± 10%
	Cation Cond	0.42	0.44	± 10%
	pH	9.0	9.14	± 0.5 pH
	Oxygen	<1.0	<5.0	± 5 ppb
	Hydrazine	20.0	20.0	± 20%

* All tolerances given in % are of the full scale reading for that instrument.

7.2 ST-42, WATER CHEMISTRY CONTROL (Continued)

TABLE 2
ST-42 OUT-OF-SPECIFICATION RESULTS, 100% RTP

Cation Conductivity (uS/cm)	Measured Values	SSCP* Limit
Main Steam	0.45	0.2
Feedwater	0.37-0.42	0.3
Steam Gen.	3.0-3.7	0.8
Sulfate (ppb)		
Steam Gen.	23.6-19.3	20
Silica (ppb)		
Steam Gen.	570-520	300

* Seabrook Station Chemistry Program Manual

7.3 ST-43, PROCESS COMPUTER

Objective

Procedure ST-43 verified that the Main Plant Computer System (MPCS) receives correct inputs from process variables and performs related calculations correctly. In conjunction with selected transient tests, the procedure evaluated the response of the Safety Parameter Display System (SPDS) during transient conditions.

The procedure is described in FSAR, Section 14, Table 14.2-5, Sheet 46.

Discussion

At the 30%, 50%, 75% and 100% power level test plateaus, computer values of plant process parameters, indicated on MPCS, were compared to other indications to validate the computer. In general, MPCS values were compared to Main Control Board (MCB) hard-wired indications.

Where the process instruments are RTDs, the only indicators are on MPCS; in the case of containment air temperature RTDs, testing and calibration was completed (prior to ST-43), using Integrated Leak Rate Test (ILRT) procedures; feedwater heater RTDs were verified during the performance of ST-26, Thermal Power Measurement and Statepoint Data Collection, also prior to performance of ST-43.

The MPCS software verification required comparison of the program output to the appropriate station procedures which are utilized when the plant computer is out of service.

The verification schedule is given below:

Test Plateau -	30%	50%	75%	100%
• RCS Leakage Monitor	X	X	X	X
• SPDS	X	X	X	X
• Secondary Heat Balance	X	X	X	X
• Xenon/Samarium Monitor	X	X	X	X
• Core Burnup Monitor		X		X
• Containment Average Air Temperature Monitor		X		X
• RCS Delta-T and T_{AVG} Monitor		X		X
• Rod Deviation Monitor and Report		X		X
• Condenser Performance Report		X		X
• Feedwater Heater Performance Report		X		X
• Turbine Performance Report		X		X
• AFD Monitor			X	X
• QPT Ratio Monitor and Report			X	X

Response time of SPDS during transient conditions was evaluated by accumulating data indicative of MPCS system load during periods of expected heavy alarm activity. Four transient startup tests were utilized; 10% load swing from 100% power (ST-34), large load reduction from 100% power (ST-35), unit trip from 100% power (ST-38) and loss of offsite power (ST-39).

Results

All acceptance criteria were met, except for those parameters noted below, where test exceptions were taken.

The following criteria apply to the 30%, 50%, 75% and 100% power level test plateaus:

- All MPCS indications and Main Control Board (MCB) indications agree within the tolerance specified.

- All MPCS calculations are being performed correctly as demonstrated by the program verification attachments.

Additional acceptance criteria apply to the transient tests (above) used to evaluate the response time of SPDS during transient conditions:

- The average SPDS response time on the STA work station is less than 10 seconds.

- No MPCS failovers occurred due to lack of CPU availability.

A number of test exceptions were taken at each test plateau, several addressed the same problems at successive power levels. Most exceptions were related to data acquisition and instrument problems; all of this nature were easily resolved.

Two exceptions remain open. One, requires software changes to the turbine and condenser performance reports; the second was a computer failover which occurred shortly after the start of ST-39, Loss of Offsite Power. Both exceptions are now under evaluation.

7.4 ST-44, LOOSE PARTS MONITORING

Objective

The procedure obtained RCS baseline noise level and signature data from the TEC Loose Parts Monitoring System (LPMS) during steady state conditions at the 50% and 100% power level test plateaus. Additional data was obtained in conjunction with the performance of three transient tests.

ST-44 is described in FSAR, Section 14, Table 14.2-5, Sheet 47.

Discussion

The LPMS provides a means for detection of loose metallic parts in the RCS. Twelve sensors (accelerometers) and associated circuitry continuously monitor noise levels at the reactor vessel and steam generators. If the noise level in a channel increases beyond a predetermined alert setpoint, an MPCS alarm alerts control room personnel.

With the plant operating in steady state, at the 50% and 100% power level test plateaus, the procedure utilized two station procedures, an operational test and a quarterly surveillance test to obtain the necessary noise level and signature data. The measurements taken yielded alert setpoints for the LPMS channels, which were subsequently entered into ST-15, Reactor Plant System Setpoint Verification.

The LPMS sensor response was further evaluatod during transient tests; the load swing test (ST-34), at 50% and 100% RTP for decreasing and increasing transients, and the large load reduction (ST-35) and unit trip at 100% RTP (ST-38).

Results

All acceptance criteria were met except for one channel;

- Steady state and transient baseline data were obtained at the specified test plateaus.

- Background values measured during normal plant operation were less than the maximum allowable background values for each individual LPDS channel as defined in the quarterly surveillance.

- Final alert setpoints have been determined.*

* LPMS alert level information submitted to meet the requirements of Reg. Guide 1.133, Part C.3.a, is given in Table 1.

At the 50% power level test plateau, with the plant in steady state, the required operational test and quarterly surveillance was performed. LPMS was found to be operating within its dynamic range, and no setpoint adjustments were required.

Also at this power level, steady state and transient baseline noise level and signature data were recorded during the performance of ST-34, Load Swing Test. All LPMS auto functions operated as required during the transients.

Similar steady state results were obtained at the 100% power level test plateau. At 100%, steady state and transient baseline noise level and signature data were recorded during the performance of three transient tests:

- ST-34, Load Swing Test
- ST-35, Large Load Reduction
- ST-38, Unit Trip from 100% Power

All LPMS auto functions again operated as required during the transients.

The channel which failed acceptance, did not meet the background value requirement at both the 50% and 100% power level test plateaus. The quarterly surveillance acceptance criterion for LPMS is identical to the above background value requirement. The surveillance specified that in this circumstance, a new alert setpoint should be calculated per the "LPMS Technical Manual". The calculation was made, and per vendor recommendation, the value was left unchanged (1.0), since no alarms were occurring during normal operation. A test exception was written to address the problem.

TABLE 1
ST-44 LPMS ALERT SETPOINTS

CHANNEL ID	SENSOR LOCATION	ALERT SETPOINTS FOR POWER OPERATION INITIAL	FINAL
VB-YM-6824-1	Reactor Vessel Head	1	1
VB-YM-6824-2	Reactor Vessel Head	1	1
VB-YM-6825-1	Reactor Vessel Bottom	1	1
VB-YM-6825-2	Reactor Vessel Bottom	1	1
VB-YM-6826-1	SG A Below Tube Sheet	1	1
VB-YM-6826-2	SG A Above Tube Sheet	1	1
VB-YM-6827-1	SG B Below Tube Sheet	1	1
VB-YM-6827-2	SG B Above Tube Sheet	1	1
VB-YM-6828-1	SG C Below Tube Sheet	1	1
VB-YM-6828-2	SG C Above Tube Sheet	1	1
VB-YM-6829-1	SG D Below Tube Sheet	1	1
VB-YM-6829-2	SG D Above Tube Sheet	1	1

NOTE: Alert setpoint values were initially set to a value of (1) for all LPMS channels as recommended by the vendor. Changes to these values are based on the results of RN 1714, Loose-Part Detection System Quarterly Surveillance, performed during this procedure and an evaluation of tabulated alarm data.

7.5 ST-45, PROCESS AND EFFLUENT RADIATION MONITORING SYSTEM TEST

Objective

The procedure verified that the process and effluent radiation monitors respond correctly to actual sample activity determined by radiochemical analysis.

A description of the procedure is found in FSAR, Chapter 14, Table 14.2-5, Sheet 48.

Discussion

The Radiation Data Management System (RDMS), through the RM-11 console, provides readings from the process and effluent radiation monitors. RDMS is designed to continuously monitor selected process and effluent steams wherever the potential for a significant release of radioactivity exists during normal operation, including anticipated operational occurrences, and during postulated accidents.

At the 50% and 100% power level test plateaus, Chemistry Department personnel collected samples, coordinated with measurements from RDMS. After radiochemical sample analysis, the results were compared to the RDMS values to verify performance of the RDMS system.

Results

The acceptance criterion, monitor results agree with laboratory sample anslysis to within a factor of 2, was not expected to be met under all conditions. Sample analysis could not always be directly compared to RDMS readings, particularly at beginning of core life. Results were acceptable even though RDMS reported values and sample analysis differed, provided there was a reasonable explanation for the difference.

The acceptance criterion was met for all monitors except those listed in Table 1. The monitor location (system) and an explanation for the difference for each exception is included.

Three test exceptions were written to address these failures to meet acceptable limits.

TABLE 1
ST-45 PROCESS AND EFFLUENT RADIATION MONITORING
TEST EXCEPTIONS

Monitor	System	Explanation
RM-6500	Boron Waste Storage Tank Inlet	Poor sampling point location; dilution has occurred after monitor, but before sampling point. Evaluation determined that dilution occurred due to a mispositioned valve, the result of a drawing error. A DCR is in preparation to correct the drawing.
RM-6502	Inlet to Carbon Delay Bed Room	Defective instrument; work request issued to repair.
RM-6504	Waste Gas Compressor Discharge	Discrepancy because short-lived isotopes decay before sample analysis can be completed.
RM-6509	Liquid Waste Test Tank Discharge to Circulating Water System (CWS)	Level of activity measured was below monitor sensitivity limit.
RM-6520-2	RC Letdown Gross Activity Monitor	Same as RM-6504.
RM-6514	Liquid Waste from Evaporators	A Temporary Modification which is not scheduled for removal until the first refueling outage prevents sample flow to the monitor.

NOTE: Additional monitors, RM-6490, RM-6501, RM-6502, RM-6515, RM-6516, RM-6519, and RM-6528 were excepted at the 50% power level plateau, because levels of activity measured were below monitor sensitivity limits.

7.6 ST-46, VENTILATION SYSTEM OPERABILITY TEST

Objective

Heating, ventilation and air conditioning systems were monitored to demonstrate that the systems maintain their service environment areas within design limits under normal plant operating conditions. A comparison of permanent room temperature indicator readings with survey instruments was made.

The procedure is described in FSAR, Section 15, Table 14.2-5, Sheet 49.

Discussion

With the reactor in steady state, at the 50% and 100% power level test plateaus, a test team monitored, using temperature and humidity measuring devices, ventilation systems operating in their normal operating mode.

The areas monitored were included in the following structures:

- Containment Building
- Containment Enclosure
- Diesel Generator Building
- Waste Process Building (Tank Farm)
- Service Water Pumphouse and Cooling Tower
- Control Building
- Emergency Feedwater Pumphouse
- Fuel Storage Building
- Main Steam and Feedwater Pipe Chases
- Primary Auxiliary Building
- Equipment Vaults

For the identified environmental zones in the above areas, six representative ambient area temperature and/or humidity readings were taken at 50% power; based on the results, only one set of readings was taken at 100% power.

Results

The acceptance criteria:

- Test data has determined that the ventilation systems are capable of maintaining equipment space environmental conditions, based on temperature and humidity, within FSAR specified design values.

- A comparison of permanent plant temperature indicator readings to measured readings verifies that plant equipment used to satisfy Technical Specifications is sensing a representative area temperature.

were met with three exceptions as noted below.

7.6 ST-46, VENTILATION SYSTEM OPERABILITY TEST (Continued)

Measurements were made in the indicated areas except for the east and west main steam and feedwater pipe chases. Temporary fans were installed in these areas to maintain acceptable temperatures. A modification to change design air flows for the pipe chase cooling fans is in process, and the pipe chase area will be monitored after it is approved and installed. The modification will be made when the plant is in a refueling or maintenance outage. Test exceptions were taken at the 50% and 100% plateaus for the areas where temporary fans were in use.

Certain environmental zones failed to meet design requirements:

Containment	2 Areas
Control Building	10 Areas
East Pipe Chase	10 Areas
West Pipe Chase	5 Areas

A test exception was written for the above areas, and a Request for Engineering Services (RES) submitted for engineering evaluation.

7.7 ST-49, CIRCULATING WATER SYSTEM THERMAL-HYDRAULIC TEST (DEFERRED)

Objective

The procedure was prepared to demonstrate that the thermal-hydraulic characteristics of the Circulating Water System are such that the intake tunnel can be treated with 110°F to 120°F water for 1 to 2 hours with the plant at 75% RTP. The procedure was also to demonstrate that heat treatment can be performed within the constraints of the National Pollutant Discharge Elimination System (NPDES) permit, and determine the effects on the power plant.

The circulating water system thermal-hydraulic test is described in FSAR, Chapter 14, Table 14.2-5, Sheet 52.

Discussion

The major source of fouling in the circulating water system tunnels is expected to be the common mussel. Experiments on mussels typical of the area demonstrated that a temperature of 110°F for 2 hours would produce 100% mortality. The NPDES permit for the test limited the warm water discharge to 120°F for 2 hours, and is the basis for the acceptance criteria.

Heat treatment of the intake tunnel requires establishment of flow reversal and warm water recirculation to reach the required temperature condition, followed by the necessary treatment period.

The chlorination system installed after the procedure had been planned appears to be effectively eliminating the buildup of marine organisms in the intake tunnel; a decision was made to defer the procedure.

ST-49 will be performed prior to any heat treatment operation. It should be noted that Technical Specifications do not contemplate operation in the heat treatment mode. NHY plans to request a technical specification change which will explicitly address heat treatment.

Heat treatment of the intake tunnel is a design feature of the Seabrook plant, not assumed in safety analysis. Therefore, the decision to defer ST-49 will not affect a successful completion of PATP or the overall Startup Test Program.

7.8 ST-51, POWER ASCENSION DYNAMIC TEST

Objective

The objective of this procedure was measurement of the dynamic response of certain main steam, feedwater and pressurizer relief systems under transient conditions.

Dynamic testing is described in FSAR, Section 3.9(B).2, and Table 3.9(B)-1.

Discussion

Displacement transducers, installed on the piping and components to be monitored, measured the behavior of the system during specified transients. The data was recorded on GETARS for later analysis.

During the precritical test program, dynamic testing of the pressurizer relief system, individual operation of the condenser steam dump and atmospheric steam dump valves, and trip of the emergency feedwater pump Terry Turbine was conducted under ST-51.

The feedwater pump dynamic testing was performed in conjunction with ST-53, Turbine Driven Emergency Feedwater Pump Start Verification. The steam dump portion was performed in conjunction with ST-55, Steam Dump System Test.

In the Power Ascension Test Program, additional dynamic response testing, coordinated with ST-38, Unit Trip from 100% Power, was conducted. The transients monitored were:

1. Main Steam System - Turbine trip, and simultaneous operation of the condenser steam dump valves.

2. Feedwater System - Closure of the feedwater containment isolation valve and trip of the steam generator main feedwater pump.

Results

The acceptance criteria was a review and verification by New Hampshire Yankee Engineering that the measured stresses do not exceed code limits. All measurements met this criteria.

7.9 ST-52, POWER ASCENSION THERMAL EXPANSION TEST

Objective

The objective of this test was a demonstration that piping systems were free to thermally expand consistent with design. These measurements confirmed that associated restraints and supports allow the required thermal movement.

The test is described in FSAR, Section 3.6.3.4.d, and Section 14, Table 14.2-3, Sheet 6.

Discussion

Thermal expansion data was obtained from displacement measuring transducers and by visual observation of spring hangers snubbers, and pipe whip restraints. Walkdowns were performed to identify areas of potential restraint to free movement.

The following systems were monitored for baseline, no-load (557°F), 30%, 50%, 75% and 100% conditions:

1. Snubbers:
   Auxiliary Steam Condensate
   Primary Component Cooling
   Chemical and Volume Control
   Diesel Generator
   Feedwater
   Main Steam
   Main Steam Drains
   Nitrogen Gas
   Reactor Coolant
   Residual Heat Removal
   Steam Generator Blowdown
   Spent Fuel Pool Cooling
   Safety Injection
   Service Water
   Waste Processing Liquid Drains

2. Spring Hangers*:
   Condensate
   Extraction Steam
   Feedwater
   Heater Drains
   Main Steam
   Main Steam Drains
   Moisture Separator & Reheater Drains/Sampling System
   *Adjustments were made, during the test sequence, to spring hangers which were not within their hot and cold settings.

3. Pipe Whip Restraints**:
   Feedwater
   Main Steam
   **Baseline and 100% test conditions only.

The following system was monitored during a turbine driven EFW pump run:

1. Snubbers:
   Main Steam (associated with EFW pump)

Results

Acceptance criteria for thermal movements were specified for Westinghouse (NSSS) scope, for UE&C scope, and for completion of NHY Engineering review:

- Piping and components are free to expand without restriction other than by design during heatup and operation of the reactor coolant and associated systems.

- The measured thermal movements shall be within ± 50% of the analytical value or ± 0.25 inches, whichever is greater for movements up to 1 inch. For analytical movements greater than 1 inch, the measured thermal movements shall be within ± 25% of the analytical value.

- Problem log discrepancies have been resolved.

- NHY Engineering evaluation is complete for data obtained outside the acceptance criteria.

All acceptance criteria were met.

Eighty-six problem log sheets were developed during performance of ST-52. The procedure required preparation of a problem log sheet, and NHY Engineering resolution for each problem identified at each power level test plateau. Thus, in some cases, four problem log sheets were prepared and resolved for a single monitored system. A summary of actual problems is given in Table 1.

No thermal expansion problems were identified during the EFW pump measurements.

A test exception was written for test equipment which is not accessible for removal under present plant operating conditions.

TABLE 1
SUMMARY OF THERMAL EXPANSION PROBLEMS

Problem Reported	Number of Cases
Predicted growth does not match actual growth within acceptable tolerance.	13*
Bolting on instrument loose	4
Instrument broken	2
Instrument displaced slightly	2
Counterweight striking structure	2
Spring can still has shipping lugs	1
Spring can no longer installed	1
Clamp holding instrument moved	1
Valve body in contact with grating	1
Unable to move as designed	1
Total	28

*Thirteen instrumented locations responsible for 71 problem sheets.

7.10 ST-56, PIPING VIBRATION TESTING

Objective

The objective of this test was verification that the vibration level of selected portions of the condensate, feedwater and main steam systems within the containment, MS/FW chases, and turbine building are within design limits under steady state conditions at 100% RTP.

The procedure is required by FSAR, Section 3.9(B).2.1a and Reg. Guide 1.68, Revision 2, Appendix A, Sec. 5.0.0.

Discussion

Piping vibration data was obtained with hand-held vibration meters at potential high vibration areas on condensate, feedwater and main steam systems. Points where high vibration was expected, such as pumps, control valves, heat exchangers, etc. received particular attention.

The data was recorded and determined for acceptability by comparing measured values with previously determined acceptability limits. Unacceptable results were evaluated for resolution by NHY Engineering.

Results

The acceptance criterion was met; acceptability of the data by NHY Engineering when compared to analytically predicted limits.

All vibration amplitudes measured were small compared to the limiting values specified in the procedure.

The procedure specified an initial condition of two condensate pumps operating; normal plant operating conditions at the 100% power level test plateau was with three condensate pumps operating.

Twenty-five data points on condensate systems were remeasured when it was found that one vibration meter failed a post-measurement calibration test. All remeasured data points were acceptable. A test exception was written to permit remeasurements.