PGandE Letter No.: DCL-86-141

ENCLOSURE

DIABLO CANYON UNIT 2

STARTUP REPORT

0835S/0044K

Dupe 8605290278

PREPARED BY: U. B. CHOPRA
N. S. SINGH
P. G. SARAFIAN
C. E. BARBEHENN
T. S. OHARA
R. A. SAVARD

REVIEWED BY: W. E. COLEY
C. G. RAO
C. E. BARBEHENN
L. F. WOMACK

APPROVED FOR ISSUE:

R. A. SAVARD
GENERAL CONSTRUCTION
LEAD STARTUP ENGINEER

L. F. WOMACK
DIABLO CANYON POWER PLANT
ENGINEERING MANAGER

DCC 33780

DIABLO CANYON POWER PLANT
UNIT 2 STARTUP REPORT

Contents

Page No.

Contents (Continued)

DIABLO CANYON POWER PLANT
UNIT 2 STARTUP REPORT

Contents (Continued)

DIABLO CANYON POWER PLANT
UNIT 2 STARTUP REPORT

List of Figures

List of Figures (Continued)

DIABLO CANYON POWER PLANT
UNIT 2 STARTUP REPORT

List of Tables

List of Tables (Continued)

SUMMARY

The Diablo Canyon Power Plant Unit 2 Startup Program activities included in this report are divided into the following sections:

1.0 Fuel Loading Program

2.0 Pre-Critical Test Program

3.0 Initial Criticality and Zero Power Physics Test Program

4.0 Turbine Driven Auxiliary Feedwater (AFW) Pump Endurance Test

5.0 Power Ascension Test Program

The Fuel Loading Program was performed during the period May 7-15, 1985. Fuel loading proceeded very smoothly except for two major delays (see Section 1.1).

Reactor assembly, reactor coolant system filling and venting and surveillance testing necessary to satisfy mode transitions to Hot Standby conditions were performed from May 16 to June 28, 1985.

The Pre-Critical Test Program was performed between June 28, 1985 and August 19, 1985. Cold System Tests included Rod Mechanism Timing and no flow and full flow Rod Drop Time tests. Hot System Tests included Rod Control System tests, Digital Rod Position Indication tests, Rod Mechanism Timing, no flow and full flow Rod Drop Time tests, Pressurizer Spray and Heater Capacity tests, RTD By-pass Loop Flow tests, Incore Thermocouple/RCS RTD Cross Calibrations, RCS Flow Measurement and RCS Flow Coastdown tests. Results were acceptable and no major equipment problems or delays were encountered. The primary reasons for the long duration of pre-critical testing were due to mode transition preparation and miscellaneous equipment problems.

Initial Criticality and Zero Power Physics testing were conducted from August 19 to August 26, 1985. All tests were completed satisfactorily, and no major problems were encountered. The all-rods-out zero power moderator temperature coefficient was slightly positive, requiring administrative limits to be placed on control rod withdrawal. These limits remained in place throughout the remainder of the Startup Program.

Following Zero Power Physics testing, a Turbine Driven Auxiliary Feedwater Pump endurance test with the reactor at low power was started on August 26, 1985. But, due to miscellaneous bearing temperature problems, a reactor trip and a subsequent reactor coolant pump motor failure, the test was postponed to October 9, 1985 and completed on October 12, 1985.

The Power Ascension Test Program commenced on October 12, 1985 with the performance of the Dynamic Steam Dump Test and was completed on March 13, 1986 with the unit being declared commercial. The major reasons for the delays during power ascension testing were equipment problems and the Strainer Outage having to be performed prior to the completion of the Power Ascension Test Program.

1.0 FUEL LOADING PROGRAM

1.1 Summary

The purpose of the Fuel Loading Program was to establish and maintain the prerequisite conditions for fuel loading and to perform fuel loading in a specified sequence.

The initial core loading for Diablo Canyon Power Plant Unit 2 was performed during the period of May 7-15, 1985. All but one of the 193 fuel assemblies were loaded per the original fuel loading sequence. One of the fuel assemblies was damaged during handling and had to be replaced at the end of the loading sequence. A replacement assembly was obtained through the NSSS vendor and loaded with less than a one day delay. The rest of the core loading proceeded relatively smoothly with only one other major delay to retrieve loose objects from the lower core plate. An improvement that was instituted on Unit 2 was the use of an IBM PC to accumulate and analyze count rate data for monitoring ICRR (Inverse Count Rate Ratio).

1.2 OP B-8D: INITIAL CORE LOADING (PREREQUISITES AND PERIODIC CHECKOUTS)

TEST OBJECTIVE

The purpose of Operating Procedure B-8D was to provide a checklist of prerequisites for Unit 2 fuel load operations.

TEST DESCRIPTION

Operating Procedure B-8D provided a checklist of prerequisites for Unit 2 fuel load along with their scheduling and frequency requirements, periodic tests to be completed during fuel loading, valve lineup checklists, and chemistry sampling requirements and data sheets.

TEST RESULTS

Preparations were begun several weeks ahead of the projected fuel load date and were signed off as each item was completed. Periodic tests were repeated as necessary and signed off.

1.3 INITIAL FUEL LOADING

OPERATIONS

Fuel loading operations commenced on May 7, 1985, with the first fuel assembly being placed in the core at 0715. Operations were performed in accordance with Operating Procedure B-8D, Supplement 2. The core loading map (Figure 1) and loading sequence that were used had been provided to Pacific Gas and Electric Company (PG&E) by Westinghouse Electric Corp., the NSSS vendor.

Prior to being loaded in the core, the fuel assemblies had been wrapped in polyethylene sheaths and dry stored in the Fuel Handling Building (FHB) spent fuel storage racks, arranged in the order of loading. Each fuel assembly consisted of a 17 X 17 square array of zircaloy-clad fuel rods with an active fuel length of twelve feet and one of three fuel enrichments (corresponding to assembly number prefixes L, M and N).

Fuel assemblies were carefully raised from the spent fuel racks as the sheath was stripped away, either by slitting the sheath with a knife or sliding the sheath off. The assemblies were placed into the fuel transfer mechanism and transferred along the partially flooded refueling canal into the Containment Building. They were then grappled by the Manipulator Crane and transferred to the partially filled reactor vessel. The assemblies were lowered at fast speed while offset into adjacent core vacancy positions and were then carefully positioned manually into the proper core location and lowered the final few inches in slow speed. Two observers at the vessel flange ensured that no interferences were encountered.

All physical operations were carried out by PG&E personnel with Westinghouse representatives on hand for technical advice. Two 10-hour shifts were used with a four hour early morning break each day. Personnel at major fuel handling workstations were rotated near the middle of each shift. Fuel handling operations included a dry-run training session at the start of many of the shifts in order to train the less experienced personnel. Fuel loading operations were completed at 1815 on May 15, 1985 with the insertion of the 193rd assembly into the core. This corresponds to an average of about one assembly per hour including all interruptions.

Prior to core loading, the two permanent plant Source Range Nuclear Instrument channels N31 and N32 read about 0.24 and 4.06 counts per second (cps). During breaks in the fuel load, work was performed on channel N32 to reduce noise levels. By the completion of core loading, the Source Range count rates had increased to about 11.23 and 11.39 cps, with occasional increases in count rate on N32 indicating that the channel was intermittently noisy. These count rates correspond to signal-to-noise ratios of about 47 and 3, above the required number of 2 for initial criticality. Inverse Count Rate Ratio (ICRR) plots from the fuel load for channels N31 and N32 are shown in Figures 2 and 3. These reflect the noisier nature of channel N32.

1.3 (Continued)

Three other temporary neutron detectors were obtained from the NSSS vendor and were used to continuously monitor neutron count rate. These were lowered into vacant core locations in the vessel. As the loading sequence progressed, the temporary detectors were moved around to strategic locations for core monitoring.

Count rate data were stored and analyzed on an IBM PC with graphics capability using software written by PG&E. Data were input manually from data sheets on which the counts were recorded by hand. The computer calculated count rates, calculated ICRR's and made criticality predictions. Additionally, the computer produced ICRR plots for all detectors upon request, either as displays on the monitor screen or as printouts (Figures 2 and 3).

Virtually all permanent data sheets were printed using the computer. The engineering workstation was located in Containment and required two engineers to assemble the data and operate the computer. By use of the computer, the speed of the data analysis was much improved over Unit 1. Fewer engineering personnel were required to maintain the flow of information required to support core loading.

Fuel loading operations were temporarily suspended approximately one day after they began when the observers at the vessel flange noted small objects on the lower core plate in the vicinity of assemblies being loaded. Previous to this time, the observers had noticed nothing in the vicinity of assemblies being set onto the core plate. TV monitor equipment was lowered into the core to examine the small objects. Three solid, loose objects were retrieved and included one small metal component from a pneumatic coupling that had come apart during previous work on upper internals. The other two were small pieces of tape and paint. Examination of areas below the lower core plate failed to reveal any other objects. Examination and retrieval were completed within one shift.

Fuel loading operations proceeded with only one other major interruption. While removing the 135th fuel assembly (M04) in the sequence from the spent fuel rack, the polyethylene sheath became lodged between the fuel assembly and the spent fuel rack cell. Cursory examination revealed that a sideplate on the lowermost spacer grid had a bent flow tab. Upon closer examination it was discovered that all four corner cells at the spacer grid were sufficiently distorted to disrupt proper dimple/fuel rod contact. Upon consultation with Westinghouse Fuels in Pittsburgh, PA, it was decided that the fuel assembly could not be used without being repaired at the fabrication facility. Fuel loading was resumed with the damaged assembly left out of the sequence temporarily. While fuel loading proceeded, a replacement assembly (B52) was flown to the site in sufficient time to cause a delay of only about one shift. This incident was the only occurrence of binding of the sheath sufficient to cause assembly damage concern on Unit 2. A similar occurrence on one assembly on Unit 1 had revealed no assembly damage.

Subsequent to the loading of the replacement fuel assembly (B52) into the core, core mapping was completed by visual inspection of fuel assembly serial numbers by two independent observers using binoculars. This concluded fuel loading operations on Unit 2.

1.3 (Continued)

PROBLEMS

Several minor equipment and related problems caused short delays during Unit 2 fuel loading and are summarized below.

1. Manipulator Crane - Manipulator crane operation caused severe electrical spikes on the temporary neutron detectors and their associated counting electronics. To alleviate the problem and allow fuel loading to continue, the manipulator crane and fuel transfer mechanism were stopped during the taking of count rate data. This slowed the overall operation considerably as it had on Unit 1.

2. Fuel Transfer Mechanism - The fuel transfer mechanism slowed and hesitated on occasion. Addition of oil to the air motor oil lines and cycling restored operation.

3. Temporary Neutron Detectors - Several NSSS vendor spare neutron detectors were brought to the site, but by the end of fuel load there were no spares left and one detector was acting somewhat erratically.

4. Source Range Nuclear Instruments - Noise was evident on both source range channels, with N32 having the most noise. Swapping to a spare cable and improvements in grounding were helpful in reducing noise. Fewer spurious containment evacuation alarms due to electrical interferences were received in Unit 2 than in Unit 1.

5. Underwater Lights - Several bulbs burned out again, as was the case in Unit 1. Fuel handling personnel complained of the poor lighting condition caused by narrow beam spotlights. Wider beam lights were installed.

6. Containment Personnel Hatch - The interlock mechanism failed on the personnel hatch, allowing both doors to open under the influence of the negative pressure inside containment. All fuel handling ceased while the doors were closed and the interlock was repaired. Personnel were stationed at both doors to operate them and the problem did not recur.

7. Underwater TV System - The underwater TV system used to map the core was unsuccessful in providing sufficient resolution to read fuel assembly serial numbers. Considerable improvement is needed in the TV, monitor, brackets and lighting in order to provide a sufficiently versatile, useful system.

DIABLO CANYON POWER PLANT
UNIT No. 2
CORE MAP

N-32

0° *(NORTH)

	A	B	C	D	E	F	G	H	J	K	L	M	N	P	R
15					N29	N58	N37	N10	N52	N14	N44				
14			N48	N57	N62	L19	N30	L34	N31	L09	N21	N23	N16		
13		N46	N33	M08	L54	M25	L45	M44	L57	M41	L08	M29	N59	N35	
12		N18	M55	M27	M21	L07	M60	L12	M30	L31	M51	M36	M54	N19	
11	N49	N64	L65	M40	L59	M52	L20	M58	L27	M42	L50	B52	L25	N08	N53
10	N39	L39	M05	L04	M26	L55	M01	L11	M23	L18	M35	L17	M56	L02	N09
9	N24	N11	L48	M64	L56	M02	L44	M17	L47	M33	L64	M50	L58	N61	N20
8	N13	L24	M57	L21	M32	L35	M53	L29	M11	L60	M38	L36	M18	L37	N22
7	N40	N55	L61	M06	L63	M13	L03	M07	L15	M15	L53	M12	L13	N03	N41
6	N34	L62	M14	L46	M63	L10	M31	L26	M10	L41	M62	L05	M37	L06	N38
5	N45	N25	L28	M22	L14	M09	L23	M61	L32	M20	L51	M16	L49	N42	N06
4		N05	M39	M59	M49	L01	M43	L52	M34	L43	M48	M24	M28	N47	
3		N17	N15	M47	L33	M19	L42	M45	L40	M03	L30	M46	N51	N56	
2			N63	N02	N28	L16	N04	L22	N01	L38	N54	N32	N36		
1					N26	N50	N07	N12	N60	N43	N27				

N-31

REMARKS Final Fuel Assembly Locations
Unit 2 Cycle 1

DATE ____________

TIME ____________

BY ____________

*Orientation corresponds to facing opposite direction from refueling machine control panel.

FIGURE 1

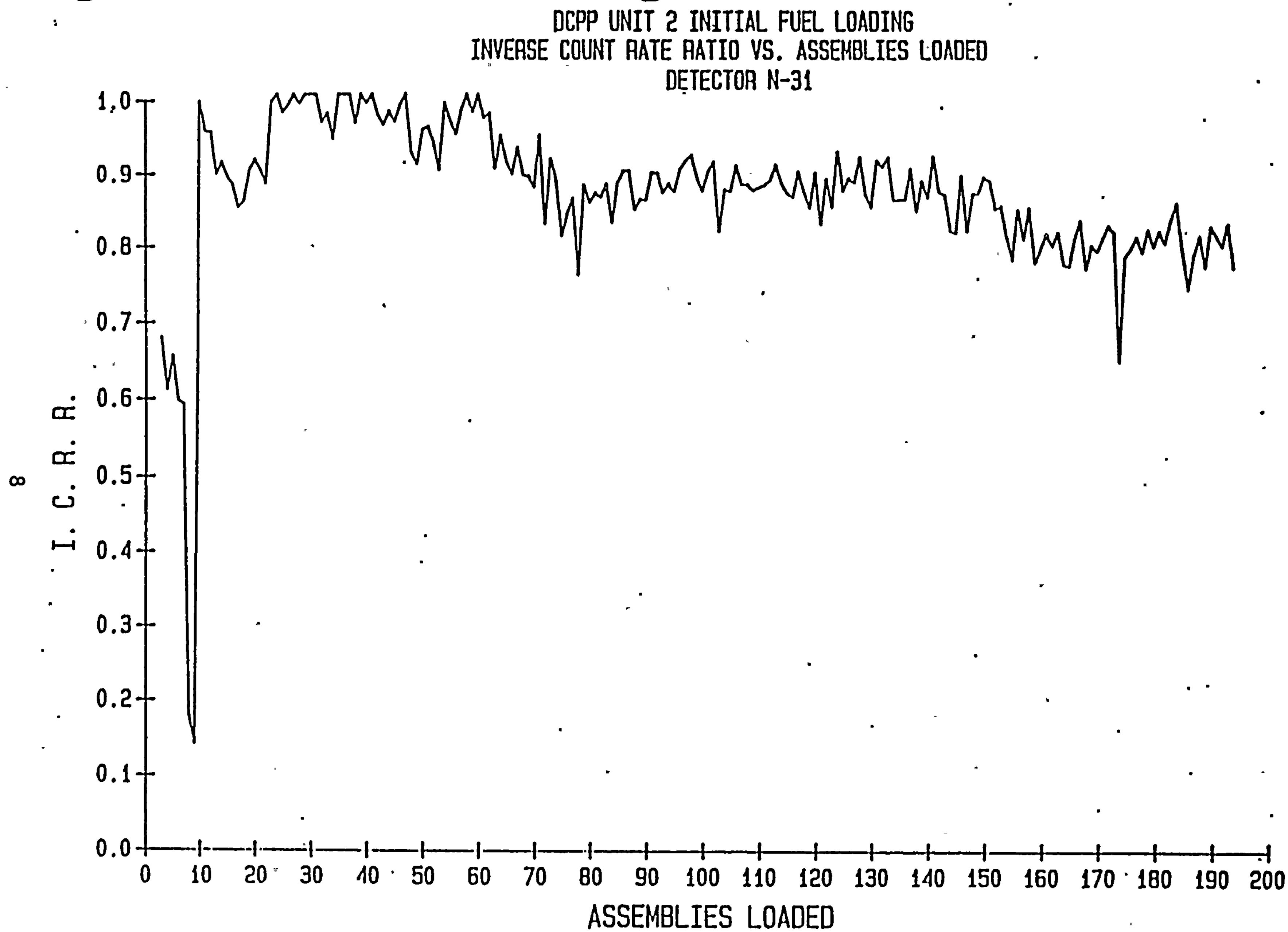

DCPP UNIT 2 INITIAL FUEL LOADING
INVERSE COUNT RATE RATIO VS. ASSEMBLIES LOADED
DETECTOR N-31
1.0
0.9
0.8
0.7
0.6
0.5
0.4
0.3
0.2
0.1
0.0
I. C. R. R.
0
10
20
30
40
50
60
70
80
90
100
110
120
130
140
150
160
170
180
190
200
ASSEMBLIES LOADED

FIGURE 2

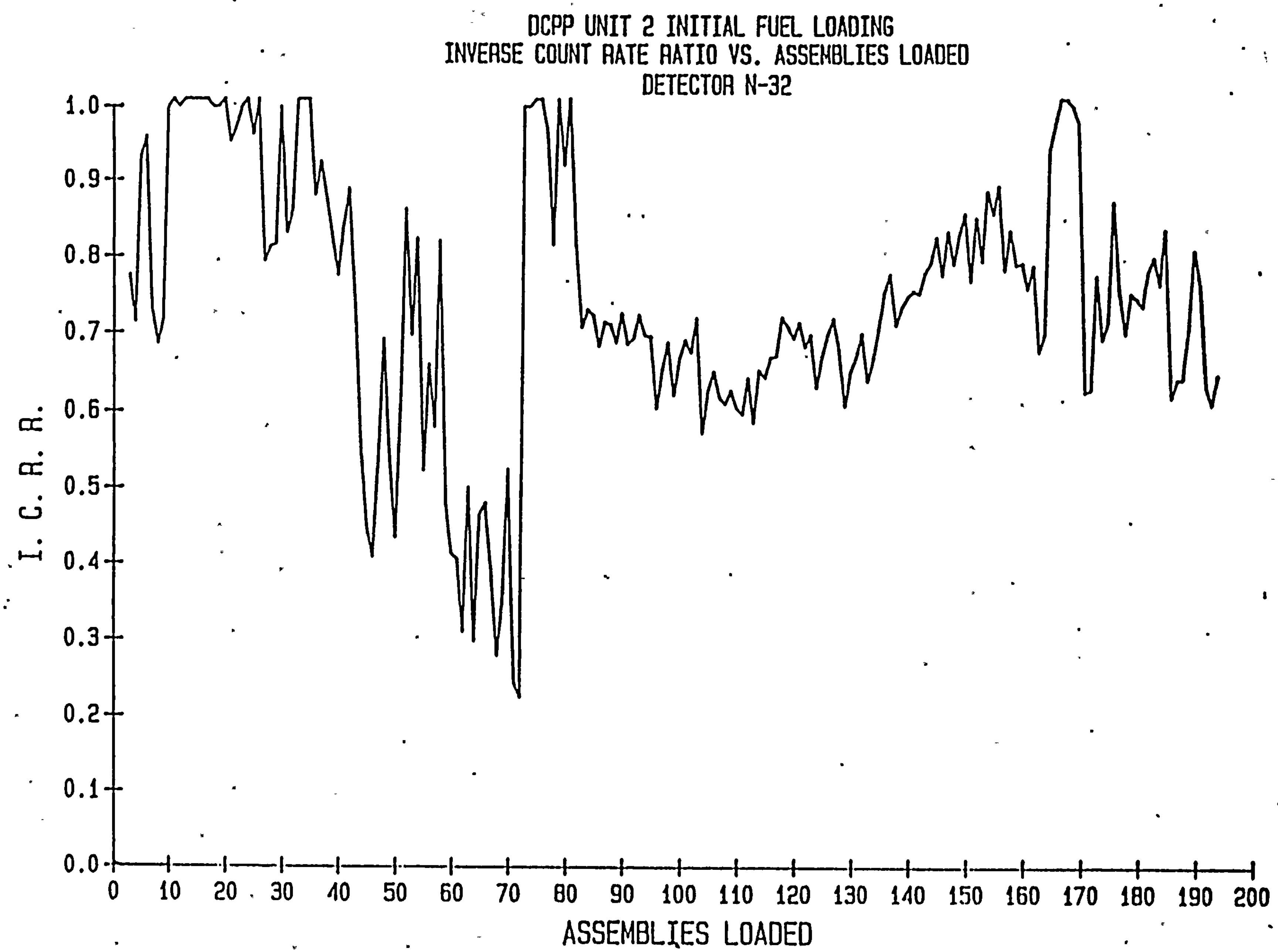

DCPP UNIT 2 INITIAL FUEL LOADING
INVERSE COUNT RATE RATIO VS. ASSEMBLIES LOADED
DETECTOR N-32
I. C. R. R.
1.0
0.9
0.8
0.7
0.6
0.5
0.4
0.3
0.2
0.1
0.0
0
10
20
30
40
50
60
70
80
90
100
110
120
130
140
150
160
170
180
190
200
ASSEMBLIES LOADED

FIGURE 3

2.0 PRE-CRITICAL TEST PROGRAM

2.1 Summary

Cold System Tests were performed after initial fuel load, reactor assembly and RCS filling and venting. The tests that were done during this phase were Incore Moveable Detector Checkout, Rod Drive Mechanism Timing and Rod Drop Time Measurements. These were performed during the period from June 28, 1985 to July 4, 1985.

Hot System Tests were performed with the RCS at rated temperature and pressure. The tests included Rod Drive Mechanism Timing, Rod Drop Time Measurements, Pressurizer Spray and Heater Effectiveness, RCS Flow Measurements, RCS Flow Coastdown, and RTD Bypass Loop Flow Measurements. These tests started on July 27, 1985 and were completed on August 12, 1985.

2.2 Test Procedure No. 38.5 - In-Core Moveable Detectors

TEST OBJECTIVE

The purpose of this test was to functionally check the operation of the In-Core Moveable Detector System.

TEST DESCRIPTION

This procedure was a comprehensive functional test of the In-Core Moveable Detector System. Using a dummy cable, operation of all five and ten path transfer devices was checked. The dummy cable was also used to verify path length measurements. In addition, all alarms and indicator lights were checked for proper actuation. The leak detection and gas purge systems related to the moveable detectors were tested. Finally, the actual detectors were installed and the corrected path lengths were determined.

TEST RESULTS

The high speed mode of transit did not meet the original acceptance criterion of 72 $\pm$ 1 feet per second. Because the high speed mode is used to transport the detectors to and from its thimble locations and no data is recorded during this maneuver, Westinghouse agreed to a change of 72 $\pm$ 2 feet per second which allowed all the original data to be accepted. All other acceptance criteria were met and the system was proven operable for standard flux mapping.

2.3 Test Procedure No. 36.1 - Rod Mechanism Timing

TEST OBJECTIVE

The purpose of this test was to operationally check the cycler timing for each control rod drive mechanism (CRDM) with a rod control cluster assembly (RCCA) attached under both cold and hot plant conditions.

TEST DESCRIPTION

Timing was checked by monitoring the lift coil, movable gripper coil and stationary gripper coil currents with an oscillograph. Microphones were placed on the top cap of each rod travel housing and their sound signals were monitored with their respective mechanism current traces. These traces were used to verify proper latch operation in conjunction with the lift, movable gripper and stationary gripper coil current traces.

Rod mechanism timing checks at cold system conditions were performed from June 28, 1985 to July 2, 1985 at approximately 370 psig and 136 deg. F. Because the Digital Rod Position Indication (DRPI) system had not been declared operable, Digital Rod Position Indication Functional Procedure, STP R-1C was performed in conjunction with T.P. 36.1. As each bank was being withdrawn, STP R-1C was performed at each 24 step increment. Then, with the bank 50 steps out, T.P. 36.1 was performed on each mechanism until all rods were tested. Finally, STP R-1C was resumed as the bank was withdrawn to its full 228 steps out position. This sequence was repeated until all of the banks were tested.

Rod mechanism timing checks at Hot System conditions were performed from July 31, to August 3, 1985 with the RCS at approximately 547 deg. F and 2235 psig. Because the DRPI was now declared operable it was possible to test the mechanisms using standard testing techniques (by pulling one bank up and testing one mechanism at a time).

TEST RESULTS

The traces for each mechanism were evaluated immediately following the test of that mechanism and were determined to be satisfactory.

Listed below are some of the problems encountered and their associated resolutions during the performance of the Rod Mechanism Timing Test:

1) DRPI indication problems/encoder cards were replaced

2) Blown stationary fuses/fuses were replaced

3) Rod N-9 would not move/loose connector pins at the bulkhead were cleaned and repaired

4) Miscellaneous DRPI indication problems/loose electrical connector pins from the DRPI coils at the head area were repaired

5) Data cabinet problems due to excessive environment temperature/cooling air to the Data Cabinets was supplied.

2.4 Test Procedure No. 36.3 - Rod Drop Time Measurements

TEST OBJECTIVE

The purpose of this test was to perform the following:

1) Measure the drop time of all control rods under four different conditions; cold no flow, cold full flow, hot no flow and hot full flow. Under each of the conditions, obtain a rod drop trace for a combined data coil signal ("A+B" trace) and an individual data coil signal ("A&B" trace).

2) Repeat the rod drop test ten times on the rods with the slowest and fastest drop times under all of the above mentioned conditions.

3) Demonstrate that the system meets the requirements of Technical Specification 3.1.3.4 which states that the individual full length (shutdown and control) rod drop time from the fully withdrawn position shall be $\leq$2.2 seconds from the beginning of decay of stationary gripper coil voltage to dashpot entry with Tavg $\geq$541 deg. F and all reactor coolant pumps running.

TEST DESCRIPTION

All measurements were made using a high speed visicorder to record the change in mechanism stationary gripper voltage, the output of the Digital Rod Position Indication (DRPI) data coils and the output of the microphones on the top cap of the mechanism housings. From the traces thus obtained, it was possible to measure the rod drop time from the loss of stationary gripper coil voltage to entry into the dashpot region as well as the time to reach the bottom of the dashpot. Figure 4 is an example of the traces obtained.

Listed below are the rod drop test plant conditions and their performance dates:

Cold No Flow	370 psig/136 deg. F	July 1, 1985
Cold Full Flow	380 psig/156 deg. F	July 3, 1985
Hot Full Flow	2235 psig/ 547 deg. F	August 4, 1985
Hot No Flow	2235 psig/530 deg. F	August 6, 1985

2.4 (Continued)

TEST RESULTS

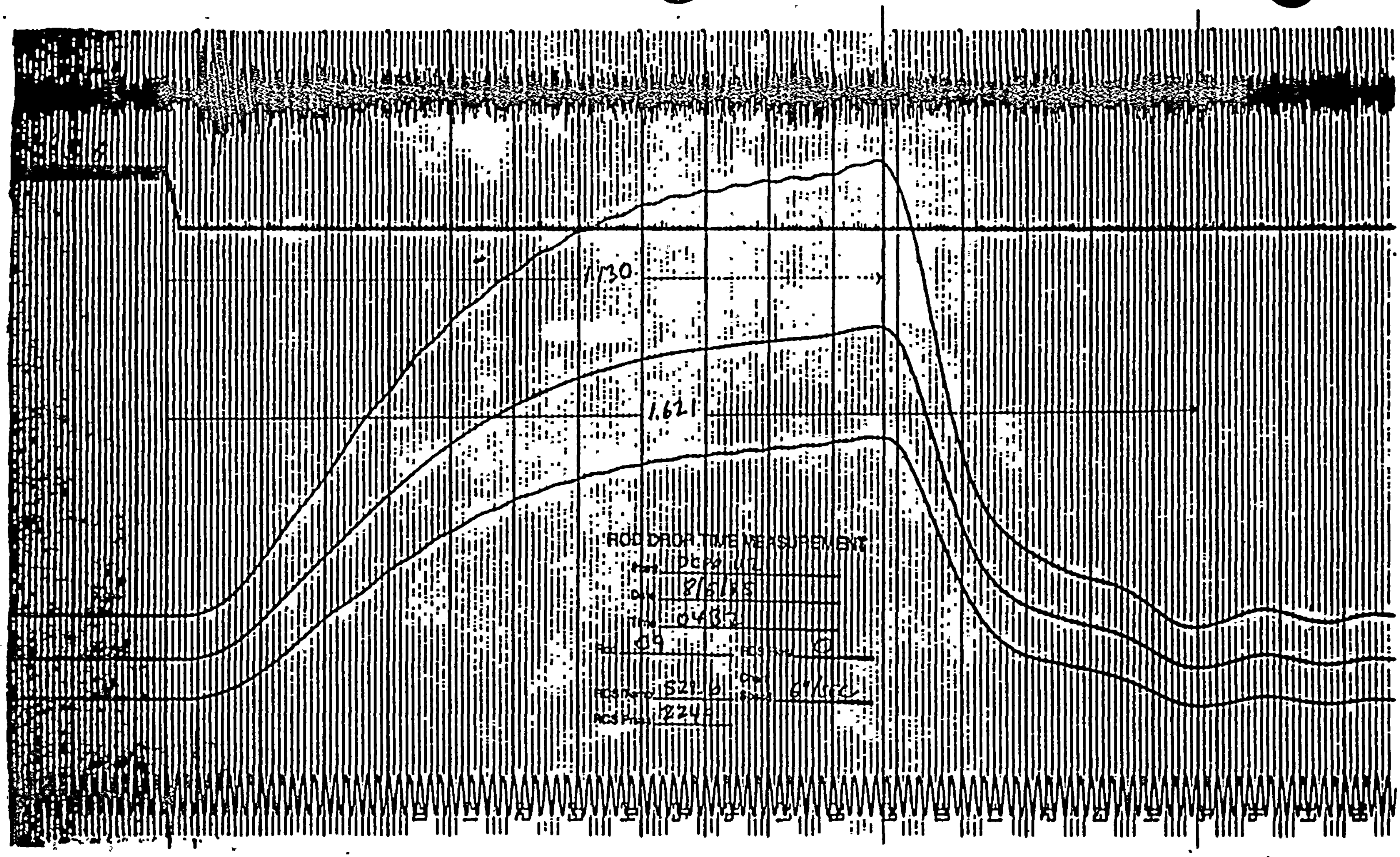
Figures 5 through 8 show the rod drop times for the four plant conditions and Table 1 lists the core average, slowest and fastest drop times. All rod drop times were well below the Technical Specification requirement of 2.2 seconds from initiation of event to dashpot entry. See section 2.3 (T.P. 36.1) for some of the typical problems encountered during the performance of this test.

Table 1

Rod Drop Times (Sec.) for Various Plant Conditions

Plant Conditions	Core Average	Slowest Rod	Fastest Rod	Standard Deviation
Cold Shutdown - No Flow	1.166/1.681	1.198/1.724	1.146/1.641	±0.011/0.018
Cold Shutdown - Full Flow	1.451/2.110	1.505/2.235	1.400/2.030	±0.023/0.034
Hot Standby - No Flow	1.129/1.616	1.158/1.654	1.106/1.598	±0.011/0.015
Hot Standby - Full Flow	1.317/1.863	1.368/1.965	1.277/1.808	±0.021/0.041

Times indicated represent: Initiation of event to dashpot entry/initiation of event to bottom of dashpot.



Combined A+B and A&B Traces

FIGURE 4

CONTROL ROD DROP TIME TABULATION

Temperature 136 **°F** **Pressure** 370 **psig** **% Flow** 0 **%**

X.XXX Breaker "Opening" to dashpot entry (seconds)

X.XXX Breaker "Opening" to bottom of dashpot (seconds)

	R	P	N	M	L	K	J	H	G	F	E	D	C	B	A
1															
2				1.172 1.708		1.198 1.724		1.161 1.667		1.161 1.635		1.151 1.661			
3					1.167 1.667		1.161 1.688		1.156 1.682		1.148 1.646				
4		1.167 1.703		1.161 1.682				1.193 1.714				1.167 1.698		1.151 1.688	
5			1.177 1.698										1.167 1.677		
6		1.188 1.708				1.172 1.682		1.156 1.677		1.161 1.667				1.151 1.677	
7			1.161 1.677										1.177 1.693		
8		1.167 1.698		1.161 1.677		1.177 1.672		1.167 1.672		1.172 1.677		1.167 1.677		1.167 .693	
9			1.151 1.672										1.177 1.688		
10		1.146 1.641				1.172 1.693		1.167 1.682		1.172 1.688				1.167 1.688	
11			1.167 1.677										1.156 1.646		
12		1.161 1.682		1.167 1.667				1.161 1.667				1.161 1.688		1.148 1.672	
13					1.177 1.693		1.172 1.688		1.156 1.677		1.177 1.688				
14				1.156 1.682		1.161 1.661		1.156 1.656		1.161 1.698		1.182 1.719			
15															

DIABLO CANYON POWER PLANT - UNIT 2

COLD SHUTDOWN - NO FLOW

FIGURE 5

CONTROL ROD DROP TIME TABULATION

Temperature 156 °F   Pressure 380 psig   % Flow 100 %

X.XXX Breaker "Opening" to dashpot entry (seconds)

X.XXX Breaker "Opening" to bottom of dashpot (seconds)

	R	P	N	M	L	K	J	H	G	F	E	D	C	B	A
1															
2				1.490 2.195		1.500 2.180		1.465 2.125		1.450 2.080		1.420 2.090			
3					1.470 2.120		1.440 2.120		1.440 2.105		1.440 2.100				
4		1.475 2.155		1.470 2.120				1.460 2.120				1.410 2.080		1.435 2.110	
5			1.465 2.110										1.435 2.100		
6		1.435 2.095				1.440 2.090		1.420 2.070		1.420 2.050				1.450 2.100	
7			1.471 2.110										1.465 2.100		
8		1.405 2.100		1.440 2.100		1.490 2.130		1.450 2.100		1.460 2.100		1.465 2.115		1.470 2.120	
9			1.435 2.090										1.440 2.090		
10		1.465 2.095				1.480 2.130		1.460 2.110		1.450 2.110				1.440 2.100	
11			1.455 2.095										1.445 2.065		
12		1.435 2.090		1.450 2.090				1.440 2.080				1.450 2.090		1.430 2.115	
13					1.475 2.125		1.445 2.115		1.447 2.140		1.480 2.170				
14				1.455 2.140		1.435 2.075		1.400 2.030		1.440 2.150		1.505 2.235			
15															

DIABLO CANYON POWER PLANT - UNIT 2

COLD SHUTDOWN FULL FLOW

FIGURE 6

CONTROL ROD DROP TIME TABULATION

Temperature 530 °F Pressure 2235 psig % Flow 0 %

X.XXX Breaker "Opening" to dashpot entry (seconds)

X.XXX Breaker "Opening" to bottom of dashpot (seconds)

	R	P	N	M	L	K	J	H	G	F	E	D	C	B	A
1															
2				1.140 1.635		1.158 1.654		1.130 1.640		1.140 1.604		1.123 1.600			
3					1.130 1.605		1.126 1.622		1.120 1.610		1.130 1.604				
4		1.134 1.630		1.140 1.620				1.111 1.603				1.130 1.620		1.116 1.618	
5			1.134 1.630										1.127 1.619		
6		1.132 1.627				1.120 1.615		1.106 1.598		1.120 1.593				1.130 1.620	
7			1.130 1.605										1.140 1.635		
8		1.140 1.633		1.118 1.593		1.138 1.622		1.142 1.632		1.107 1.602		1.133 1.620		1.133 1.610	
9			1.129 1.607										1.130 1.621		
10		1.137 1.614				1.140 1.630		1.117 1.603		1.118 1.613				1.120 1.615	
11			1.110 1.609										1.122 1.581		
12		1.132 1.622		1.133 1.620				1.106 1.591			1.140 1.625		1.122 1.629		
13					1.140 1.630		1.120 1.610		1.121 1.621		1.133 1.617				
14				1.126 1.621		1.126 1.603		1.133 1.595		1.129 1.628		1.142 1.647			
15															

DIABLO CANYON POWER PLANT - UNIT 2

HOT STANDBY - NO FLOW

FIGURE 7

CONTROL ROD DROP TIME TABULATION

Temperature 547 °F   Pressure 2235 psig   % Flow 100 %

X.XXX Breaker "Opening" to dashpot entry (seconds)

X.XXX Breaker "Opening" to bottom of dashpot (seconds)

	R	P	N	M	L	K	J	H	G	F	E	D	C	B	A
1															
2				1.368 1.965		1.335 1.902		1.314 1.856		1.332 1.853		1.352 1.902			
3					1.324 1.873		1.306 1.876		1.315 1.875		1.328 1.870				
4		1.350 1.920		1.318 1.860				1.308 1.859				1.282 1.853		1.330 1.890	
5			1.330 1.895										1.314 1.868		
6		1.310 1.870				1.300 1.852		1.378 1.830		1.284 1.819				1.299 1.855	
7			1.330 1.865										1.340 1.890		
8		1.330 1.879		1.300 1.841		1.319 1.864		1.310 1.870		1.294 1.837		1.316 1.859		1.312 1.861	
9			1.302 1.822										1.335 1.893		
10		1.320 1.850				1.312 1.872		1.302 1.855		1.306 1.856				1.298 1.858	
11			1.308 1.850										1.330 1.852		
12		1.311 1.866		1.281 1.845				1.280 1.824				1.299 1.838		1.350 1.900	
13					1.348 1.906		1.324 1.864		1.327 1.895		1.346 1.905				
14				1.340 1.900		1.305 1.840		1.277 1.808		1.297 1.883		1.360 1.953			
15															

DIABLO CANYON POWER PLANT - UNIT 2

HOT STANDBY - FULL FLOW

FIGURE 8

2.5 Test Procedure No. 36.5 - Digital Rod Position Indication System

TEST OBJECTIVE

The purpose of this test was to verify that the Digital Rod Position Indication (DRPI) System satisfactorily performed the required indication and alarm functions for each individual RCCA under Hot Standby conditions.

TEST DESCRIPTION

With the plant in Hot Standby conditions, the control rod system was operated and proper agreement of rod position between the step counters, DRPI, P-250 computer, and pulse-to-analog (control banks only) systems were verified.

TEST RESULTS

All DRPI, P-250 computer, pulse-to-analog and step counter readings agreed exceptionally well.

2.6 Test Procedure No. 36.6 - Rod Control System Operational Test

TEST OBJECTIVE

The purpose of this test was to verify the proper operation of the Rod Control System.

TEST DESCRIPTION

With the plant at Hot Standby conditions, the control rod system was operated to verify the proper functioning of the following:

1) Rod movement status lights.
2) Rod position indication systems.
3) Rod speed indicator.
4) DC hold supply cabinet.
5) Bank overlap.
6) "Rod Bottom" and "Rods at Bottom" alarms.

During the bank overlap test, rod control was in manual and the overlap settings were lowered from their normal values to preclude excessive rod withdrawal.

TEST RESULTS

All rod control system functions performed as expected. The only major delay was the inadvertent blowing of the DC hold cabinets' power supply diodes caused by improper switching and their subsequent replacement.

2.7 Test Procedure No. 7.10 - Pressurizer Spray and Heater Capacity and Continuous Flow Setting

TEST OBJECTIVE

This test had three objectives:

1) To establish the continuous pressurizer spray flow rate by adjusting the spray flow bypass valves.

2) To determine pressurizer spray effectiveness.

3) To determine pressurizer heater effectiveness.

TEST DESCRIPTION

For the continuous spray setting, the plant was initially stabilized at Hot Standby conditions with the spray flow bypass valves (valves 8050 and 8051) 3/4 turn and 1/4 turn open, respectively. Each spray valve was then adjusted to obtain the minimum possible continuous spray flowrate while maintaining a pressurizer to spray line temperature difference less than 200 deg. F and a spray line temperature above the low temperature set-point of 500 deg. F. The resulting valve positions represented the final settings.

To initiate the pressurizer spray effectiveness portion of this test, the plant was stabilized at Hot Standby conditions and all pressurizer heaters were deenergized. Next, both normal spray valves were fully opened to cause a rapid depressurization. The pressure transient response (i.e., pressure vs. time as measured on a strip chart recorder) was then compared to the acceptance criteria.

The final section of this test was intended to verify pressurizer heater effectiveness. With the plant at stable Hot Standby conditions and both normal spray valves closed, all pressurizer heaters were energized to their maximum capacity. The pressure transient response, as measured by a strip chart recorder, was then compared to the acceptance criteria.

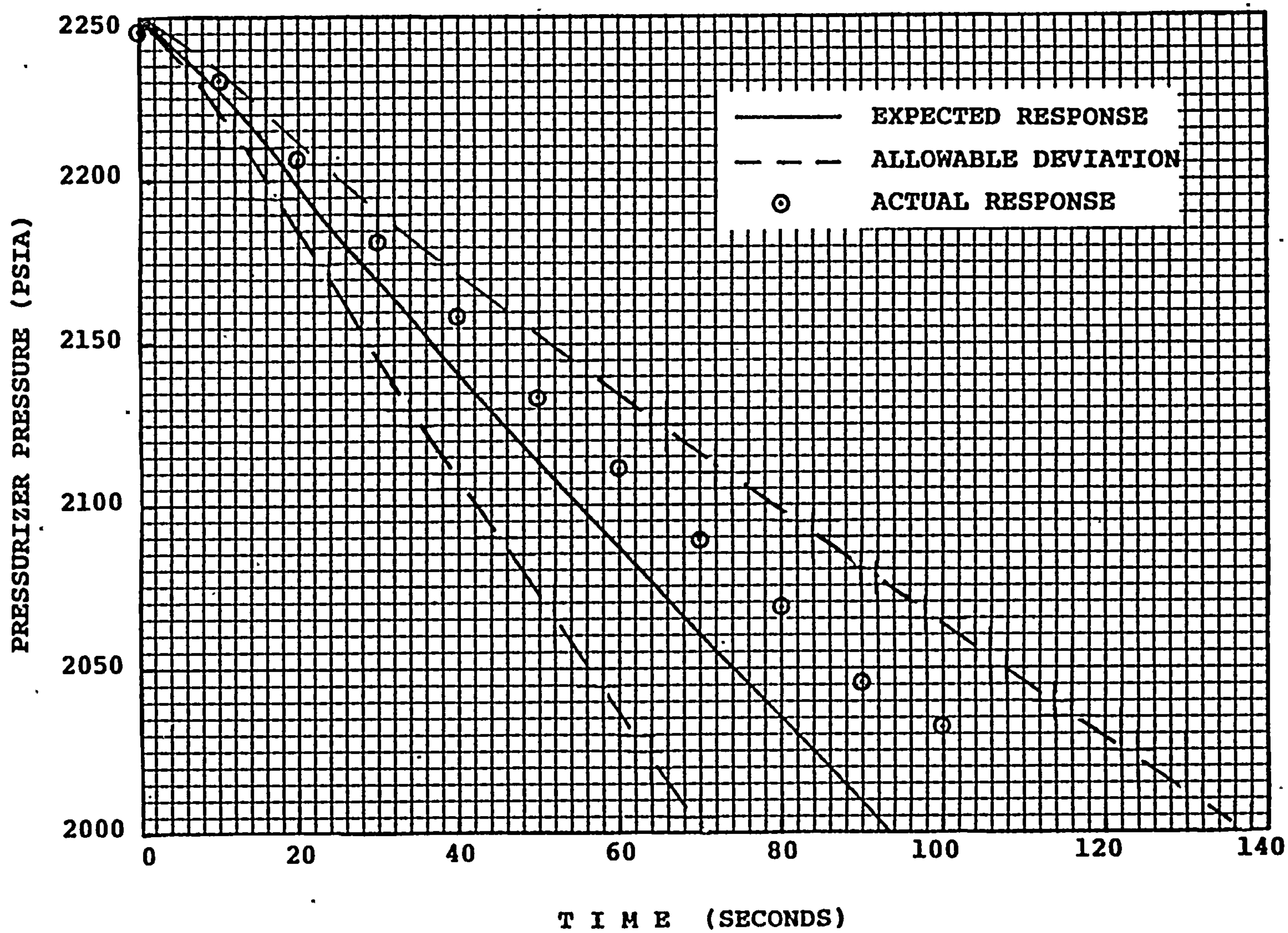
TEST RESULTS

The pressurizer continuous spray flow bypass valves were set as follows:

Loop 1: Valve 8050: 1/2 turn open,
Loop 2: Valve 8051: 3/4 turn open.

Pressurizer spray effectiveness was determined to be approximately -130 psi/minute. This rate was well within limits, as shown by Figure 9.

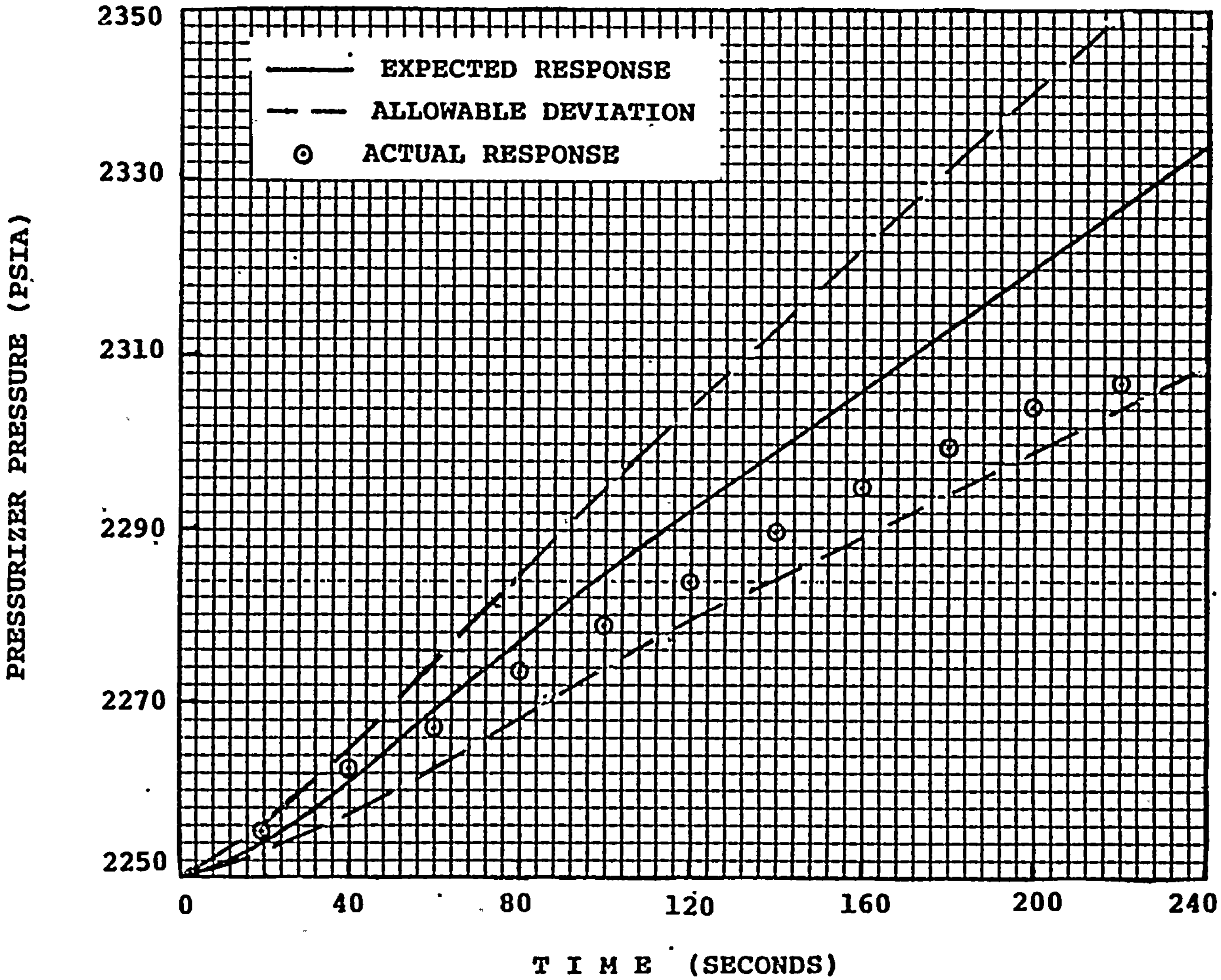
Pressurizer heater effectiveness was determined to be approximately 17 psi/minute. Again, the transient response was well within limits, as shown in Figure 10.



DIABLO CANYON POWER PLANT - UNIT 2

PRESSURIZER SPRAY EFFECTIVENESS

FIGURE 9



DIABLO CANYON POWER PLANT - UNIT 2

<u>PRESSURIZER HEATER EFFECTIVENESS</u>

FIGURE 10

2.8 Test Procedure No. 7.3 - Resistance Temperature Detector Bypass Loop Flow Measurements

TEST OBJECTIVE

The objective of this test was to verify transport times and alarms in the Resistance Temperature Detector (RTD) bypass loops for Hot Standby conditions after core loading.

TEST DESCRIPTION

RTD bypass loop total flow, hot leg flow, and cold leg flow were measured for each reactor coolant loop. These measured values were compared to calculated minimum flow rates necessary to achieve the design reactor coolant transport time (i.e. ≤1.0 second) for each loop RTD. In addition, the RTD bypass loop low flow alarms were set and verified.

TEST RESULTS

Prior to conducting the flow measurements, 0.73 inch restricting orifices were installed in each of the four cold leg bypass loops to balance the flows and to reduce the total bypass flows to within the flow indicator range.

RTD bypass loop low flow alarm setpoints were set and checked to trip within 90% of the total measured RTD loop bypass flow rate. RTD hot leg and cold leg bypass loop flows were significantly greater than the minimum required flows, thus ensuring acceptable reactor coolant transport times for each RTD. Final results are shown in Table 2.

Table 2

RTD Bypass Loop Flows

Reactor Coolant Loop	Cold Leg Flow (gpm)		Hot Leg Flow (gpm)		Total Flow (gpm)
	Minimum	Actual	Minimum	Actual	
2-1	50.7	108.2	64.4	151.8	260.0
2-2	67.7	95.3	61.2	148.7	244.0
2-3	56.7	102.5	61.9	144.5	247.0
2-4	64.2	93.3	61.9	146.7	240.0

DCC . 33780

2.9 STP R-27: Incore Thermocouple and RCS RTD Cross Calibration

TEST OBJECTIVE

Surveillance Test Procedure R-27 provided a means to calibrate the incore thermocouples using the RCS loop RTDs as a reference at 547 deg. F. The procedure also allowed a cross calibration to the RTDs themselves.

TEST DESCRIPTION

Surveillance Test Procedure R-27 consisted of establishing a stable, full-flow isothermal RCS temperature of 547 deg. F using a single condenser steam dump valve. Simultaneously the wide, narrow, and spare RTD resistance readings for each RCS loop and incore thermocouple temperatures at various locations were recorded. RTD resistance readings were obtained at the Hagan Racks. Thermocouple temperature readings were obtained from the output of the P-250 process computer, the Thermocouple Monitoring Systems (TMS), the Emergency Response Facility Data System (ERFDS), and the Subcooled Margin Monitor.

In order to read operating RTD resistances, those RTDs had to be taken out of service. Because of Technical Specification requirements, only the RTDs in a single loop were removed at any time and measured and repeated for each remaining RCS loop. Between loops, the previous loop RTDs were restored to service and isothermal temperature in the RCS was re-established by operating the steam dump system in the pressure control mode. The time required to re-establish isothermal conditions was minimized by feeding the steam generators to maintain a constant level between the data acquisition for each loop.

TEST RESULTS

All RTD readings were consistent. Most required small temperature corrections, all much less than ±1 degree at 547 deg. F. All wide and narrow range RTDs met the ±0.7 deg. F criterion to be declared OPERABLE. All but four met the ±0.3 deg. F accuracy specification. Thus, four RTD instrument loops needed recalibration.

Thermocouple readings at the TMS panels largely met the ±2 deg. F acceptance criterion. Greater than 50% of the thermocouple readings at the ERFDS, the Subcooled Margin Monitor, and from the P-250 were outside the acceptance criteria and thus required recalibration.

Plant I&C Engineering have evaluated the data and have determined that the offsets necessary to bring the readings to the specified accuracies were of such small magnitude that recalibration at this time would not be productive.

2.10 Test Procedure No. 7.5 - Reactor Coolant System (RCS) Flow Measurement

TEST OBJECTIVES

The primary objective of this test was to calculate steady state Reactor Coolant System (RCS) flow at pre-critical conditions. Additional data, to serve as baseline information for an undamaged core, were also collected.

TEST DESCRIPTION

Loop flow instrumentation consisted of three elbow tap differential pressure transmitters on each of the four reactor coolant loops. In order to dampen flow oscillations, snubbers were temporarily installed on these loop flow transmitters.

Initial conditions for the RCS flow measurement required steady state Hot Standby conditions with all four reactor coolant pumps operating. With the RCS stable, flow transmitter output and RCS temperatures were recorded for a ten minute period. The voltage readings from each elbow tap flow transmitter were averaged and converted to a differential pressure based on calibration data.

Reactor coolant loop flow was determined as a function of the loop flow transmitter differential pressure and temperature through the use of a Westinghouse supplied curve.

RCS baseline data were collected for various operating pump configurations to serve as a reference to which future data could be compared, if required.

TEST RESULTS

The total RCS flow rate was 388,217 gpm. The individual loop flow rates were all within ±3% of the average and all acceptance criteria were met. Table 3 provides the details of the results.

Table 3

Reactor Coolant Loop Flows

Reactor Coolant Loop	Loop Flow (gpm)	% Difference * From Average
2-1	96,183	-0.9
2-2	95,817	-1.3
2-3	99,417	2.4
2-4	96,800	-0.3
Total Flow	388,217	
Loop Average	97,054	

$$* \frac{\text{Loop Flow} - 97{,}054}{97{,}054} \times 100$$

ACCEPTANCE CRITERIA

1. Flow rate for each loop within 5% of average.

2. Individual loop flow rates $\geq$ 88,500 gpm.

3. At Hot Standby, total RCS flowrate $\geq$ 90% of 366,000 gpm.

2.11 Test Procedure No. 7.6 - Reactor Coolant System Flow Coastdown

TEST OBJECTIVE

The main objective of this test was to measure changes in the reactor coolant flow rate resulting from trips of various reactor coolant pump (RCP) breakers. Delay times associated with these trips were also determined.

TEST DESCRIPTION

Two coastdowns were analyzed:

1) Four pumps operating initially, two pumps coasting down (2/4),

2) Four pumps operating initially, four pumps coasting down (4/4),

In each case, the pumps coasting down were tripped within 100 msec of one-another under Hot Standby conditions. The resulting coastdowns, i.e., flow as a function of time, were compared to coastdowns in the FSAR.

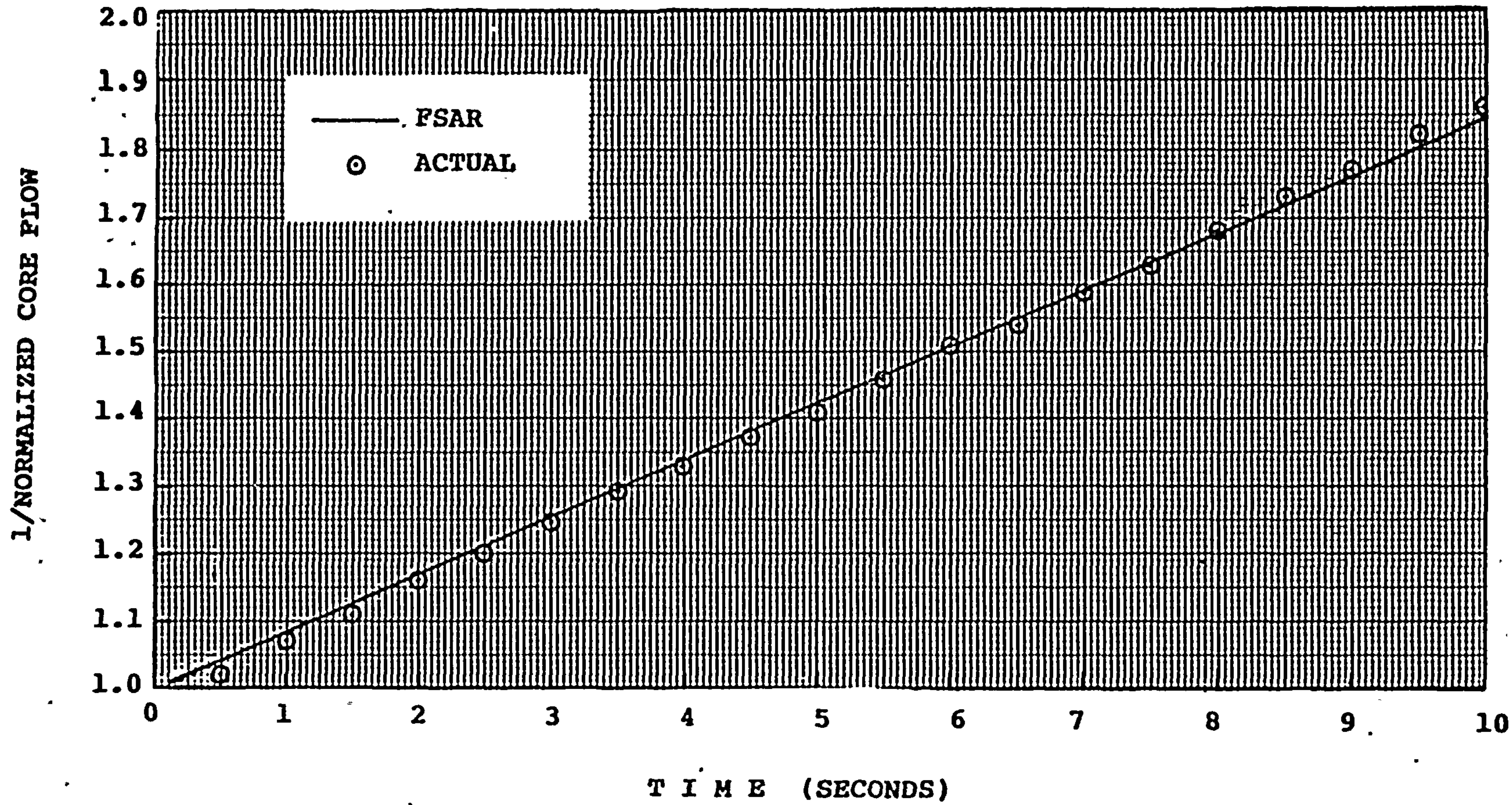
TEST RESULTS

For the 4/4 coastdown, the rate at which actual flow changed was evaluated through the slope of the inverse core flow curve, as shown in Figure 11. This curve was compared to the FSAR inverse core flow curve in the time range of 3 to 10 seconds. Although the slope of the actual curve was greater than the slope of the FSAR curve, the results were evaluated by Westinghouse and determined to be acceptable. The actual inverse flow curve was also used to determine flow sensor delay. (Flow sensor delay is defined as the time at which the best straight line approximation to the inverse flow curve drawn in the 4/4 coastdown, between three and ten seconds, intersects the inverse flow value of 1.0).

For both coastdowns, the actual flow, corrected for flow sensor delay, was compared to the flow in the FSAR. Results are shown in Figure 12. To be conservative, the FSAR curve must lie below (i.e., show a more rapid reduction in coolant flow) the actual curve. However, due to conservative testing methodology, actual flow curves typically lie slightly below the FSAR curves for Westinghouse plants. The results were evaluated and declared acceptable by Westinghouse.

Data from the 2/4 coastdown was used to calculate the low flow time delay, the undervoltage trip delay time, and the under frequency trip delay time. All three parameters met their respective Acceptance Criteria (A.C.). The low flow time delay, defined as the time from beginning of coastdown until rod motion, was calculated to be 1.63 seconds (A.C. of $\leq$3.06 seconds). The undervoltage trip delay time, defined as the difference between the time undervoltage trip conditions are reached and the time the rods are free to fall, was calculated to be 0.118 second (A.C. of $\leq$1.2 seconds). The underfrequency trip delay time, defined as the difference between the time underfrequency trip conditions are reached and the time the rods are free to fall, was calculated to be 0.127 second (A.C. of $\leq$0.6 second).

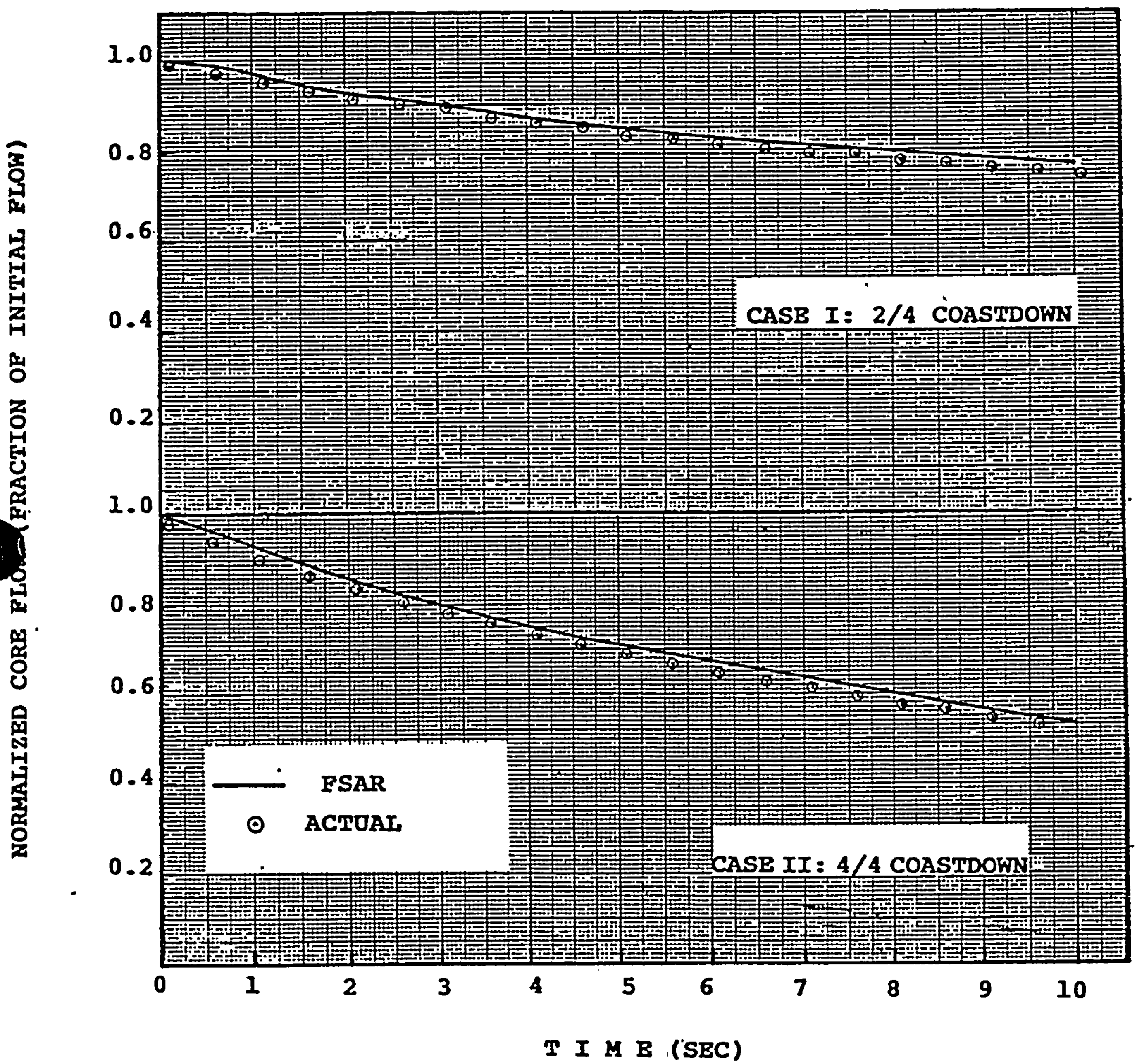
INVERSE CORE FLOW CURVE



DIABLO CANYON POWER PLANT - UNIT 2

FIGURE 11

RCS FLOW COASTDOWN CURVES



DIABLO CANYON POWER PLANT - UNIT 2

FIGURE 12

3.0 INITIAL CRITICALITY AND ZERO POWER PHYSICS TEST PROGRAM

3.1 Summary

This portion of the Startup Program consisted of Initial Criticality and Zero Power Physics Testing. The approach to criticality started on August 19, 1985 and the low power physics measurements were completed on August 25, 1985. No major problems were encountered during the conduct of these tests.

Initial criticality was achieved on August 20, 1985 at 0216 hours. Next, nuclear design checks were performed by measuring parameters including:

- Critical boron concentrations
- Isothermal temperature coefficients
- Control rod bank reactivity worths
- Zero power neutron flux distributions
- Boron reactivity worths

These parameters were determined at nominal all-rods-out conditions as well as for various control bank configurations.

Additional physics testing included a pseudo rod ejection and a minimum shutdown margin verification. For the pseudo ejection, an individual control rod was withdrawn in order to obtain the flux distribution and ejected reactivity worth. Finally, adequate shutdown margin was verified by measuring the reactivity worth of the shutdown banks and the worth of the most reactive stuck rod.

The physics testing was completed in a timely manner and verified that the zero-power physics characteristics of the reactor core are consistent with design.

3.2 Test Procedure No. 41.2 - Initial Criticality

TEST OBJECTIVE

The objectives of this procedure were to 1) achieve criticality, 2) increase reactor power to the point of adding heat, 3) establish the zero power test range, and 4) verify proper operation of the reactivity computer.

TEST DESCRIPTION

Initial conditions were established with the shutdown banks fully withdrawn, control banks fully inserted, boron concentration at 1819 ppm, RCS temperature at 547 deg. F, and RCS pressure at 2252 psig.

The control banks were withdrawn in 50 step intervals until Control Bank D reached 170 steps. An inverse count rate ratio (ICRR) was taken at each interval. During the control rod withdrawal, the ICRR dropped from 1.0 to approximately 0.65.

Normal mode dilution to criticality was then commenced at approximately 1000 pcm/hr. Again, ICRR was tracked and plotted. When the ICRR reached 0.2, the dilution was stopped to allow RCS mixing. Control rods were driven in to offset the dilution as criticality was achieved during mixing at approximately 0216 hours on August 20, 1985.

Rods were pulled to obtain a positive startup rate and power increased to 1×10^{-8} amp on the intermediate range. Power was then stabilized and reference initial criticality data taken: 113.5 steps for Bank D, 1313 ppm RCS boron concentration.

During the approach to criticality and the subsequent increase to 1×10^{-8} amps, the reactivity computer was not operable due to the erratic behavior of the unit's recorder. The problem was traced to a noise signal from the power range's lower detector. The signal was not serious enough to affect the operability of the power range channel but it was of a large enough amplitude to affect the operation of the reactivity computer. Therefore, it was decided to hook the reactivity computer to another power range channel (NI-44) and return the other power range channel (NI-43) to service.

Following the resolution of noise problems related to the reactivity computer set-up, power was increased toward the point of adding heat (POAH). Approach to POAH was repeated three times to ensure data repeatability. From the POAH data, (1×10^{-6} amp as indicated on the reactivity computer), the zero power test range (ZPTR) was established as 1×10^{-8} to 1×10^{-7} amp on the reactivity computer.

Reactor power was then reduced to the lower end of the ZPTR in preparation for the reactivity computer checkout. Twenty-five, forty and sixty pcm positive reactivity additions were made and the neutron doubling times were measured. The results were checked against Westinghouse design criteria and found to be satisfactory.

3.2. (Contined)

TEST RESULTS

All parameters measured during this testing were within the Acceptance Criteria provided by Westinghouse. Critical boron concentration was measured at 1313 ppm with Bank D at 113.5 steps. The estimated critical condition was 1313 ppm with Bank D at 170 steps. The difference was well within the design margin allowance.

The POAH was measured at 5×10^{-7} amp on the intermediate range detectors. Recording the same data for each of the three approaches to the POAH verified the value was correct and repeatable.

The last test to verify proper operation of the reactivity computer indicated proper response for reactivity changes. All test cases were within the $\pm 4\%$ Acceptance Criteria. This test was repeated several times during the Low Power Physics Test Program to ensure continued proper operation of the reactivity computer throughout testing.

3.3 Test Procedure No. 41.3 - Nuclear Design Checks

TEST OBJECTIVE

The objective of this test was to measure the Boron Endpoints, Isothermal Temperature Coefficients and the Zero Power Neutron Flux Distributions and compare the results with design predictions.

TEST DESCRIPTION

At various control rod configurations, measurements were made to determine the Boron Endpoint, the Isothermal Temperature Coefficient, and the Zero Power Neutron Flux Distribution.

Boron End Point Measurements

These measurements were performed to determine the boron concentrations at which the reactor would be just critical for several control rod configurations.

The control rod configurations at which this measurement was performed were:

1) All rods out (ARO)
2) Control bank D fully inserted
3) Control banks D and C fully inserted
4) Control banks D, C, and B fully inserted
5) Control banks D, C, B, and A fully inserted
6) Shutdown bank D, and all control banks fully inserted
7) Shutdown banks D and C, and all control banks fully inserted
8) All control banks fully inserted less the most reactive rod control cluster assembly.

These measurements were performed with the reactor just critical and within 60 pcm of the endpoint configurations. The critical RCS boron concentrations were based on RCS sampling. The controlling banks were then withdrawn/inserted to the endpoint configuration and the reactivity changes were measured. The corresponding critical boron endpoint concentrations were then determined as follows:

$$(C_B)_{end} = (C_B)_{j.c.} - [\Delta\rho/(\text{Boron Worth})]$$

Where:

$(C_B)_{end}$ = Critical boron endpoint concentration.

$(C_B)_{j.c.}$ = Measured just critical boron concentration at beginning of measurement

$\Delta\rho$ = The reactivity change by bank insertion/withdrawal to endpoint configuration.

Boron Worth = The reactivity change per unit boron concentration change as specified by the Nuclear Design Report.

3.3 (Continued)

Isothermal Temperature Coefficient Measurements

These measurements determined the reactivity changes due to the overall temperature changes of the core. These measurements were performed at the following control rod configurations:

1) All rods out.
2) Control bank D fully inserted.
3) Control bank D and C fully inserted.

With the output from the reactivity computer and an average RCS Tavg signal connected to an x-y recorder, the RCS was gradually cooled approximately 5 deg. F using the steam dump system and then reheated to the no-load Tavg. The slope generated on the x-y recorder was then taken to be the isothermal temperature coefficient (ITC).

Another parameter of interest, the moderator temperature coefficient (MTC), was then determined from the relationship:

ITC = MTC + FTC

where:

ITC = Isothermal Temperature Coefficient

MTC = Moderator Temperature Coefficient

FTC = Doppler (Fuel) Temperature Coefficient (from Nuclear Design Report)

Zero Power Flux Distributions

In order to verify the correct fuel loading pattern and to verify design calculations, low power testing included two flux distribution measurements: the first with all rods out and the second with Control Bank D almost fully inserted. The core average temperature was maintained at approximately 547 deg. F and reactor power was maintained just above the nominal zero power physics test range and just below the point of adding nuclear heat. The core average radial power distributions are shown in Figures 13 and 14 for the two cases.

The Movable Detector Flux Mapping System was used to collect data from the 58 fuel assemblies with instrument paths. Due to small detector currents during zero power testing, the movable detector system required a special setup for each detector consisting of a high quality power supply and a Keithly Picoammeter for signal input to the flux trace recorders and the P-250 computer.

The collected data (i.e., the P-250 output) were then input to the INCORE computer code, which expands the measured information to a detailed three-dimensional full-core power distribution.

3.3 (Continued)

TEST RESULTS

Boron Endpoint Measurements

The results of the Boron Endpoint measurements are shown in Table 4. The measured values agreed very well with predicted values and all acceptance criteria were met.

Isothermal Temperature Coefficient Measurements

The results of the Isothermal Temperature Coefficient measurements are summarized in Table 5. All acceptance criteria were met. It was determined that the moderator temperature coefficient was positive at the ARO endpoint configuration. Rod withdrawal limits were established using an interpolation technique on the isothermal temperature coefficient data of the ARO and Control Bank D fully inserted endpoints. The rod withdrawal limits are a function of boron concentration and power level as shown in Figure 15. They will remain in effect until sufficient core burnup has occurred such that the critical boron concentration is reduced to the point where the moderator temperature coefficient is always negative. (The Technical Specifications require only that the moderator temperature coefficient be negative).

Zero Power Flux Distributions

Both flux distribution measurements yielded results close to expectations and well within the acceptance criteria. The core average axial power distribution was close to a cosine shape while the unrodded radial distribution was reasonably flat with the peak assemblies closer to the core periphery than the center (see Figure 13 for relative assembly powers). Insertion of Control Bank D caused a slight increase in flux peaking, as shown in Figure 14. The radial distribution was also characterized by a small, but acceptable, flux tilt. Peaking factors are summarized in Table 6.

Table 4

Measured Versus Predicted Boron Endpoint Concentrations

Endpoint Configuration	Critical Boron Concentration	
	Actual (ppm)	Predicted (ppm)
ARO	1352	1322 ± 50
CD in	1217	1217 ± 14
CD,CC in	1102	1102 ± 12
CD,CC,CB in	978	981 ± 12
CD,CC,CB,CA in	927	929 ± 5
CD,CC,CB,CA,SDD in	857	*
CD,CC,CB,CA,SDD,SDC, in	762	*
ARI, N-1	741	719 ± 63

* no predicted concentration

Table 5

Measured Versus Predicted Isothermal Temperature Coefficient and Derived Moderator Temperature Coefficient

Endpoint Configuration	Isothermal Temperature Coefficient (ITC)		Derived * Moderator Temperature Coefficient (MTC) (pcm/deg. F)
	Measured (pcm/deg. F)	Predicted (pcm/deg. F)	
ARO	-0.26	-0.84 ±3.0	+1.64
CD in	-4.07	-4.75 ±3.0	-2.17
CD,CC in	-7.85	-8.75 ±3.0	-5.95

* From Design Predictions, FTC = -1.9 pcm/deg.F

MTC = ITC - FTC
= ITC + 1.9 pcm/deg. F

TABLE 6

Power Distribution Results

ITEM		UNRODDED FLUX MAP (All Rods Out)	RODDED FLUX MAP (Control Bank D In)
CONDITIONS	- temperature	547 deg. F	547 deg. F
	- boron conc.	1356 ppm	1223 ppm
	- power	0%	0%
	- burnup	0 MWD/MTU	0 MWD/MTU
DATE		August 24, 1985	August 24, 1985
ROD CONFIGURATION		Bank D @ 228 steps Bank C @ 228 steps	Bank D @ 18 steps Bank C @ 228 steps
$F^N_{\Delta H}$	- measured value	1.422	1.577
	- location*	M04-IJ	J02-AQ
	- acceptance criteria	1.42 $\pm$10%	1.53 $\pm$10%
F^T_Q	- measured value	2.315	2.509
	- location*	M12-IH @ 77"	J02-AQ @ 77"
QUADRANT TILT	- measured value	1.008	1.010
	- acceptance criteria	$\leq$1.020	$\leq$1.020
	- by quadrant	1.005 | 1.002 0.991 | 1.002	1.004 | 1.006 0.992 | 0.998

* Assembly locations (i.e., M12) are shown in Figure 1

Pin location within assembly (i.e., IH) are based on 17 x 17 matrix ranging from AA to QQ.

CORE AVERAGE RADIAL POWER DISTRIBUTION -ALL RODS OUT

Relative Assembly Power (Pi)

$\frac{\text{Measured Pi - Expected Pi}}{\text{Expected Pi}} \times 100$

NORTH

	R	P	N	M	L	K	J	H	G	F	E	D	C	B	A
1					.576 2.7	.683 3.2	.816 3.5	.744 3.9	.809 2.7	.671 1.3	.568 1.3				
2			.513 4.1	.905 4.1	1.043 2.7	1.033 3.1	1.066 2.0	1.050 1.8	1.047 .2	.997 -.5	1.007 -.8	.857 -1.4	.494 .2		
3		.498 1.2	1.071 1.2	.980 1.2	1.144 2.7	1.134 2.7	1.181 1.8	1.141 1.3	1.167 .5	1.079 -2.3	1.093 -1.8	.955 -1.4	1.061 .2	.501 1.8	
4		.868 -.1	.965 -.4	1.316 .1	1.103 2.0	1.203 2.0	1.160 .3	1.210 .6	1.154 -.1	1.180 .0	1.074 -.7	1.314 -.1	.969 .1	.876 .8	
5	.563 .4	1.005 -1.0	1.099 -1.3	1.071 -.9	1.194 1.2	1.152 1.1	1.187 1.6	1.148 1.8	1.189 1.8	1.153 1.2	1.197 1.5	1.088 .6	1.121 .6	1.023 .7	.571 1.8
6	.660 -.4	.983 -1.9	1.070 -3.1	1.162 -1.5	1.138 -.1	1.140 .8	1.021 .7	1.093 1.0	1.009 -.6	1.128 -.7	1.124 -1.3	1.188 .7	1.112 .8	1.010 .8	.674 1.8
7	.792 .5	1.035 -.9	1.132 -2.5	1.128 -2.4	1.150 -1.6	1.009 -.6	1.040 .3	.952 .3	1.033 -.4	1.000 -1.4	1.157 -.9	1.157 .1	1.170 .8	1.056 1.1	.804 2.0
8	.711 -.7	1.020 -1.1	1.099 -2.4	1.177 -2.1	1.100 -2.5	1.073 -.9	.947 -.3	.999 -.3	.941 -.9	1.077 -.5	1.121 -.5	1.204 .1	1.132 .5	1.039 .7	.724 1.1
9	.781 -.9	1.032 -1.2	1.135 -2.3	1.127 -2.5	1.132 -3.1	.988 -2.6	1.015 -2.1	.935 -1.5	1.043 .6	1.020 .6	1.174 .5	1.155 -.1	1.165 .4	1.051 .6	.796 1.0
10	.660 -.4	.996 -.5	1.068 -3.2	1.153 -2.3	1.113 -2.3	1.106 -2.6	.984 -3.0	1.059 -2.2	.999 -1.5	1.140 .4	1.146 .6	1.186 .6	1.104 .0	1.012 1.0	.669 1.0
11	.574 2.3	1.039 2.3	1.140 2.3	1.058 -2.1	1.167 -1.1	1.121 -1.6	1.149 -1.7	1.093 -3.1	1.133 -3.0	1.129 -.9	1.174 -.5	1.074 -.6	1.095 -1.7	1.035 2.0	.572 2.0
12		.879 1.2	.968 .0	1.316 .0	1.086 .4	1.185 .5	1.135 -1.8	1.162 -3.4	1.116 -3.4	1.167 -1.0	1.071 -1.0	1.299 -1.2	.968 -.0	.893 2.7	
13		.496 .7	1.065 .6	.974 .6	1.122 .8	1.103 -.1	1.141 -1.7	1.089 -3.2	1.147 -1.2	1.111 .6	1.124 .9	.960 -.9	1.065 .6	.510 3.5	
14			.496 .7	.877 .9	1.027 1.2	1.002 .1	1.032 -1.2	1.019 -1.1	1.056 1.0	1.045 4.4	1.046 3.0	.883 1.6	.505 2.5		
15					.571 1.9	.663 .0	.791 .4	.720 .6	.816 3.5	.691 4.4	.578 3.0				

DIABLO CANYON POWER PLANT - UNIT 2

FIGURE 13

DCC 33780

CORE AVERAGE RADIAL POWER DISTRIBUTION - CONTROL BANK D INSERTED

Relative Assembly Power (Pi)

$$\frac{\text{Measured Pi} - \text{Expected Pi}}{\text{Expected Pi}} \times 100$$

NORTH

	R	P	N	M	L	K	J	H	G	F	E	D	C	B	A
1					.698 4.3	.857 4.2	1.048 5.1	.957 5.5	1.055 5.7	.860 4.5	.700 4.5				
2			.493 3.9	.920 4.0	1.179 4.2	1.236 4.2	1.286 3.8	1.265 3.7	1.276 2.9	1.216 2.5	1.148 1.6	.883 -.2	.482 1.7		
3		.480 1.3	.949 1.1	.826 1.2	1.151 4.1	1.245 4.2	1.229 1.0	1.082 -.1	1.217 .1	1.199 .4	1.106 .0	.812 -.4	.953 1.6	.491 3.6	
4		.885 .1	.813 -.4	.654 -1.6	.947 .2	1.203 .4	1.009 -3.7	.633 -3.0	1.032 -1.5	1.206 .6	.945 -.0	.659 -.8	.826 1.2	.905 2.4	
5	.675 .8	1.126 -.4	1.096 -.8	.934 -1.2	1.141 .7	1.169 .0	1.117 -.9	.974 -1.0	1.118 -.7	1.169 .8	1.136 .3	.950 .5	1.120 1.3	1.148 1.5	.683 2.0
6	.826 .4	1.176 -.9	1.161 -2.8	1.176 -1.9	1.153 -.5	1.192 .8	1.036 1.4	1.079 1.4	1.025 .4	1.178 -.4	1.145 -1.2	1.204 .5	1.207 1.0	1.200 1.2	.839 2.0
7	.999 .2	1.231 -.7	1.187 -2.3	1.015 -3.1	1.101 -2.3	1.010 -1.1	.997 .4	.828 .1	.994 .1	1.011 -1.0	1.089 -3.3	1.008 -3.8	1.202 -1.1	1.248 .7	1.023 2.6
8	.897 -1.0	1.207 -1.1	1.057 -2.4	.625 -4.2	.943 -4.2	1.042 -2.1	.818 -1.2	.512 -1.7	.821 -.7	1.059 -.5	.957 -2.8	.626 -4.1	1.061 -2.1	1.226 .4	.924 1.9
9	.986 -1.1	1.225 -1.2	1.194 -1.8	1.012 -3.4	1.078 -4.3	.988 -3.3	.968 -2.5	.813 -1.9	.998 .5	1.024 .2	1.108 -1.7	1.019 -2.8	1.195 -1.7	1.245 .5	1.015 1.8
10	.829 .7	1.192 .5	1.166 -2.4	1.174 -2.0	1.134 -2.1	1.156 -2.3	.995 -2.7	1.045 -1.8	1.010 -1.1	1.183 .1	1.162 .3	1.202 .3	1.205 .9	1.214 2.3	.842 2.4
11	.692 3.4	1.168 3.3	1.140 3.1	.925 -2.1	1.116 -1.4	1.139 -1.7	1.105 -2.0	.956 -2.8	1.098 -2.5	1.148 -1.0	1.096 -3.2	.911 -3.7	1.046 -5.4	1.175 3.9	.696 4.0
12		.894 1.0	.804 -1.5	.655 -1.5	.945 .0	1.201 .2	1.026 -2.0	.628 -3.8	1.018 -2.8	1.186 -1.1	.890 -5.8	.614 -7.6	.781 -4.2	.920 4.0	
13		.475 .2	.938 -.1	.817 .1	1.109 .3	1.196 -.4	1.194 -1.8	1.055 -2.6	1.211 -.4	1.208 1.1	1.088 -1.5	.777 -4.7	.923 -1.7	.494 4.1	
14			.476 .3	.895 1.2	1.156 2.3	1.196 .8	1.234 -.4	1.217 -.3	1.263 1.9	1.239 4.4	1.171 3.6	.907 2.6	.490 3.3		
15					.694 3.6	.830 .9	1.007 .9	.918 1.2	1.039 4.2	.860 4.5	.694 3.6				

DIABLO CANYON POWER PLANT - UNIT 2

FIGURE 14

UNIT 2 CYCLE 1

ROD WITHDRAWAL LIMITS

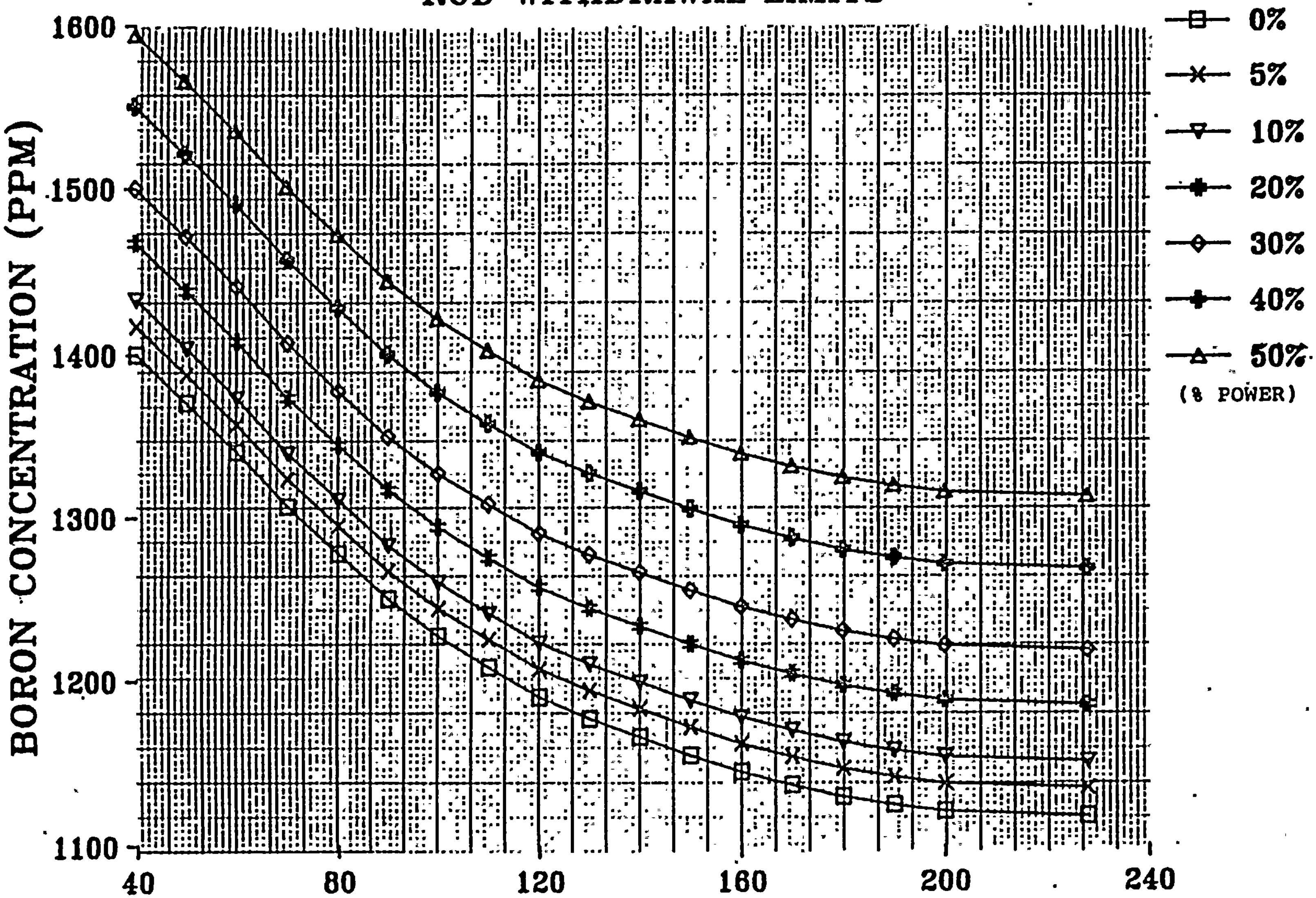


FIGURE 15

3.4 Test Procedure No. 41.4 & 41.5 - Rod and Boron Worth Measurements

TEST OBJECTIVE

The objective of these tests was to determine the reactivity worth of each control bank, total reactivity worth of control banks with normal 100 step overlap and the average boron reactivity worth.

TEST DESCRIPTION

Individual Control Bank Worth

With all control rods withdrawn, a reactor coolant system boron dilution was established. The control banks were inserted to compensate for the resulting reactivity gain. The sequence of individual control bank insertion was Control Bank D, Control Bank C, Control Bank B, then Control Bank A.

The reactivity changes were recorded using the reactivity computer. The data obtained were used to develop integral and differential bank worth curves. Figures 16 through 19 show these curves.

Control Bank Worth With Normal Bank Overlap

With all shutdown banks withdrawn and all control banks inserted, a reactor coolant system boration was established. The control banks were withdrawn to compensate for the resulting reactivity insertion. The withdrawal was done in normal sequence with normal bank overlap.

The integral and differential worth curves are shown in Figure 20.

Boron Worth

The average boron reactivity worth was based on data obtained during the individual control bank worth measurements. A typical boron worth consisted of the ratio of the reactivity worth of the individual control bank to the change in critical boron concentrations associated with the insertion of the control bank.

TEST RESULTS

The measured values agreed well with predictions as can be seen in Tables 7 and 8. All acceptance criteria were met.

Table 7

Measured Versus Predicted Control Bank Reactivity Worth

Control Bank	Measured Worth (pcm)	Predicted (pcm)
CD	1398	1365 ± 137
CC	1186	1162 ± 116
CB	1301	1242 ± 124
CA	534	533 ± 53
Total	4419	

Control Banks in Overlap	Measured Worth	Within ±4% of the Total Measured Worth of Individual Banks
	4397 pcm	4419 ± 177 pcm

INDEXED RMS

Table 8

Measured Versus Predicted Average Boron Worth

Control Bank	Bank Worth (pcm)	C_B (ppm)	Boron Worth (pcm/ppm)
CD	-1398	135	-10.4
CC	-1186	115	-10.3
CB	-1300	124	-10.5
CA	-534	51	-10.5

Average	-10.4
Predicted	-10.2 ±1.0

Unit: 2
T.P.: 41.4
Date: 8-22-85

CONTROL BANK D

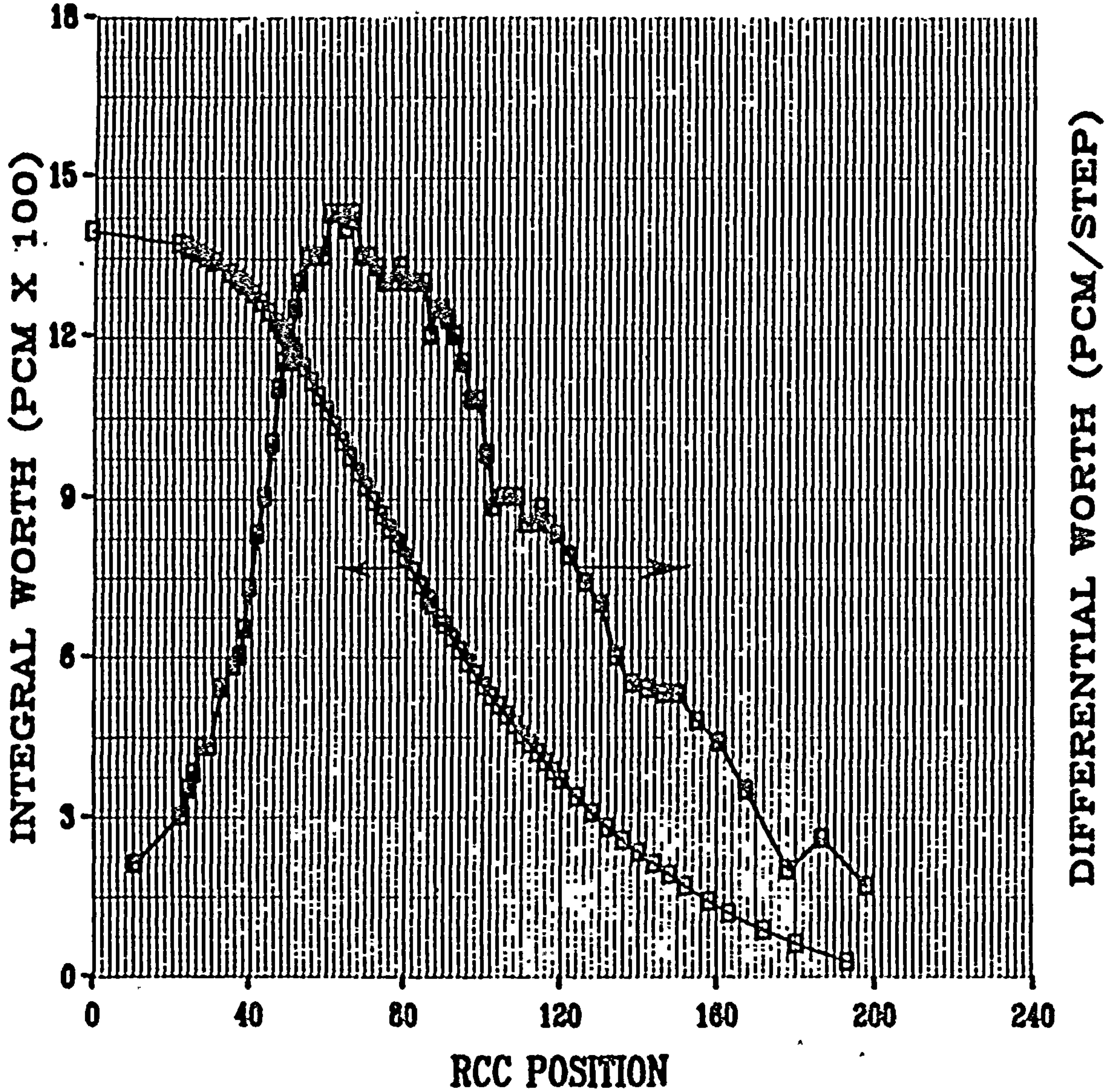


FIGURE 16

Test Conditions:

1. RCC Bank Positions:

SDA	228
SDB	228
SDC	228
SDD	228
CA	228
CB	228
CC	228
CD	moving
RCCA	-

2. Power Level:
   0 %RTP
3. RCS Temperature:
   Initial 547 F
   Final 547 F
4. RCS Pressure:
   Initial 2245 psig
   Final 2245 psig
5. Avg. Core Burnup:
   0 MWD/MTU

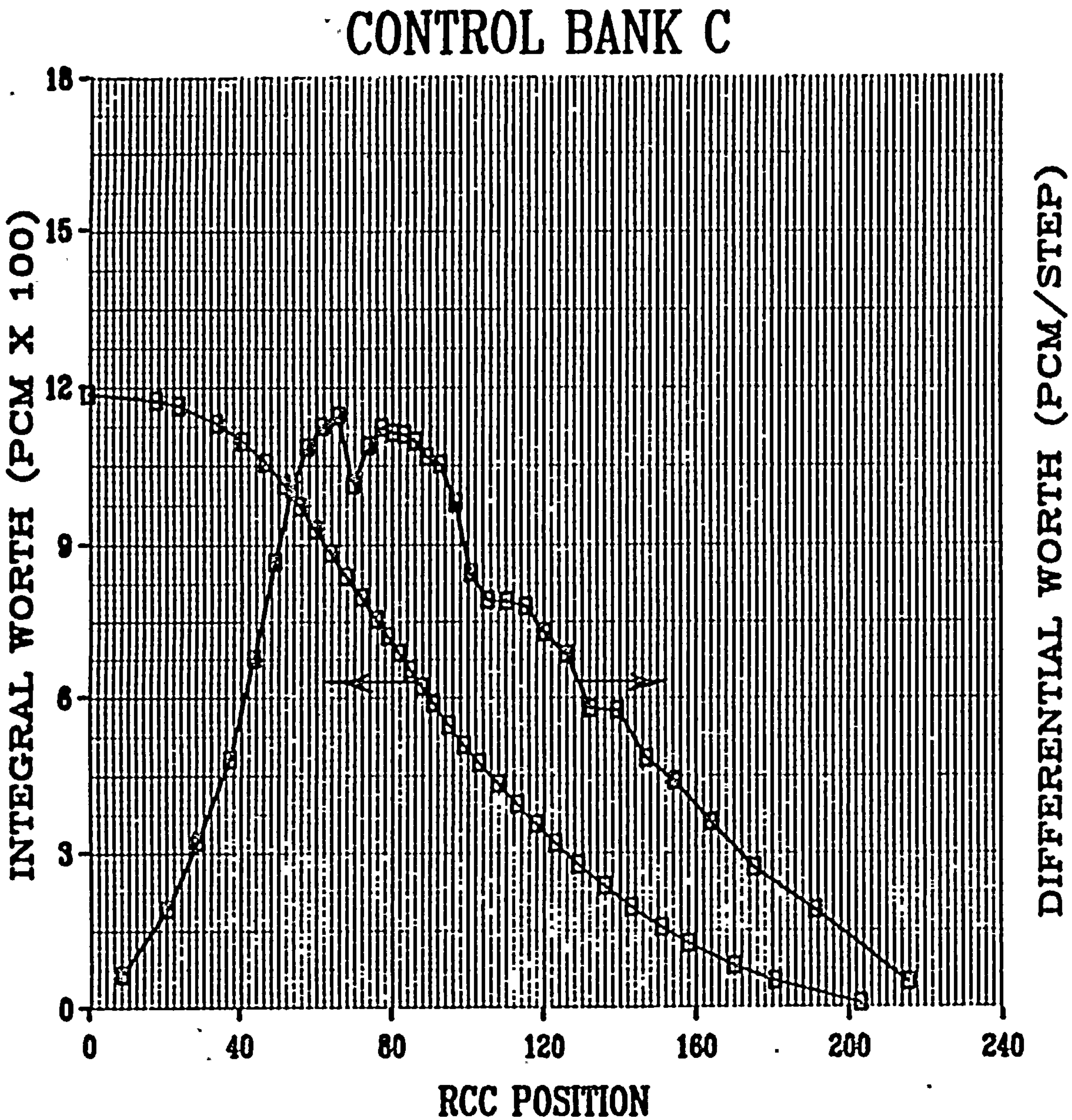


FIGURE 17

Unit: 2

T.P.: 41.4

Date: 8-22-85

Test Conditions:

1. RCC Bank Positions:

SDA	228
SDB	228
SDC	228
SDD	228
CA	228
CB	228
CC	moving
CD	0
RCCA	-

2. Power Level:
   0 %RTP
3. RCS Temperature:
   Initial 546.7 F
   Final 547.2 F
4. RCS Pressure:
   Initial 2250 psig
   Final 2250 psig
5. Avg. Core Burnup:
   0 MWD/MTU

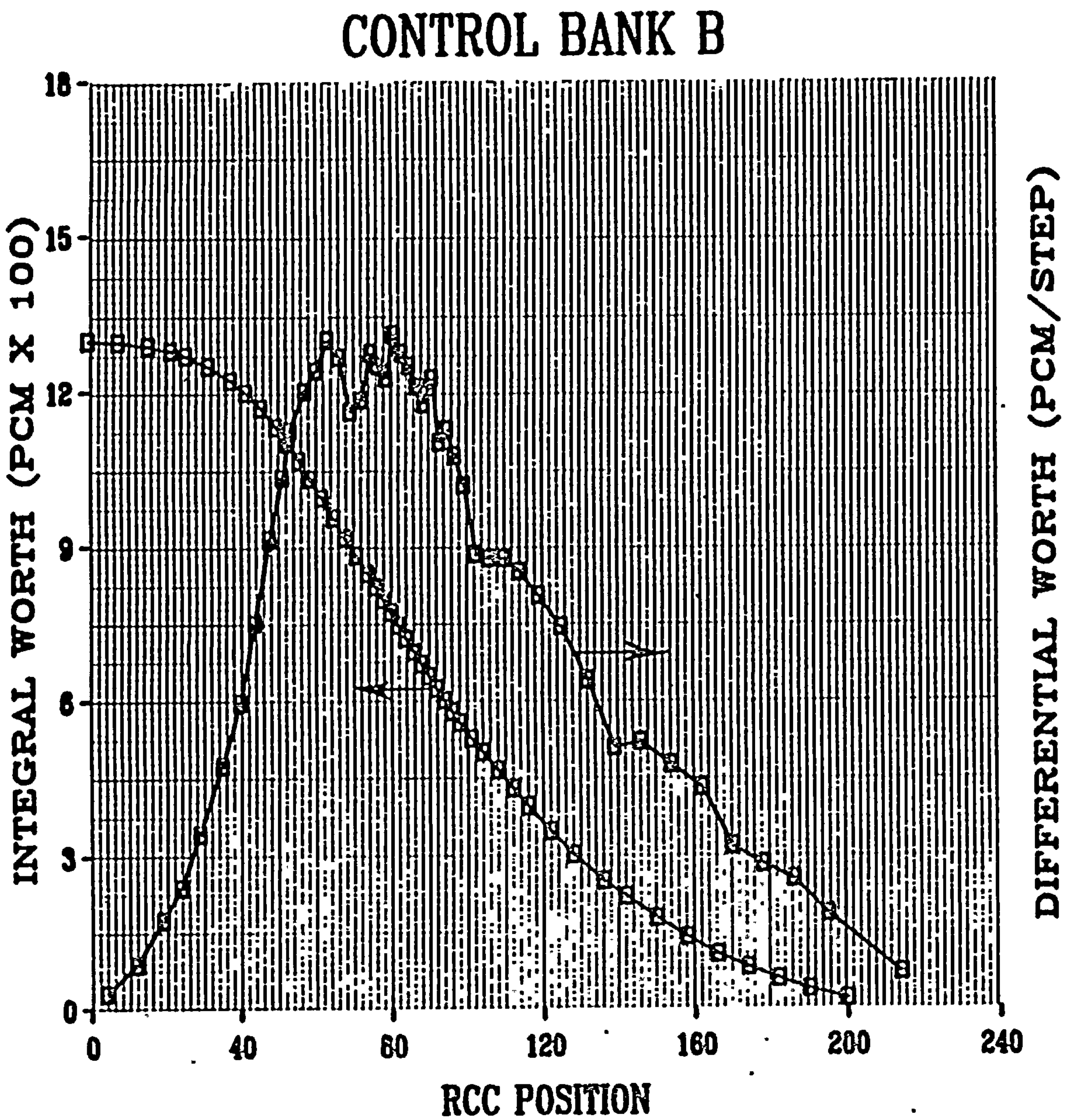


FIGURE 18

Unit: 2

T.P.: 41.4

Date: 8-23-85

Test Conditions:

1. RCC Bank Positions:

SDA	228
SDB	228
SDC	228
SDD	228
CA	228
CB	moving
CC	0
CD	0
RCCA	-

2. Power Level:
   0 %RTP
3. RCS Temperature:
   Initial 548 F
   Final 548 F
4. RCS Pressure:
   Initial 2235 psig
   Final 2235 psig
5. Avg. Core Burnup:
   0 MWD/MTU

CONTROL BANK A

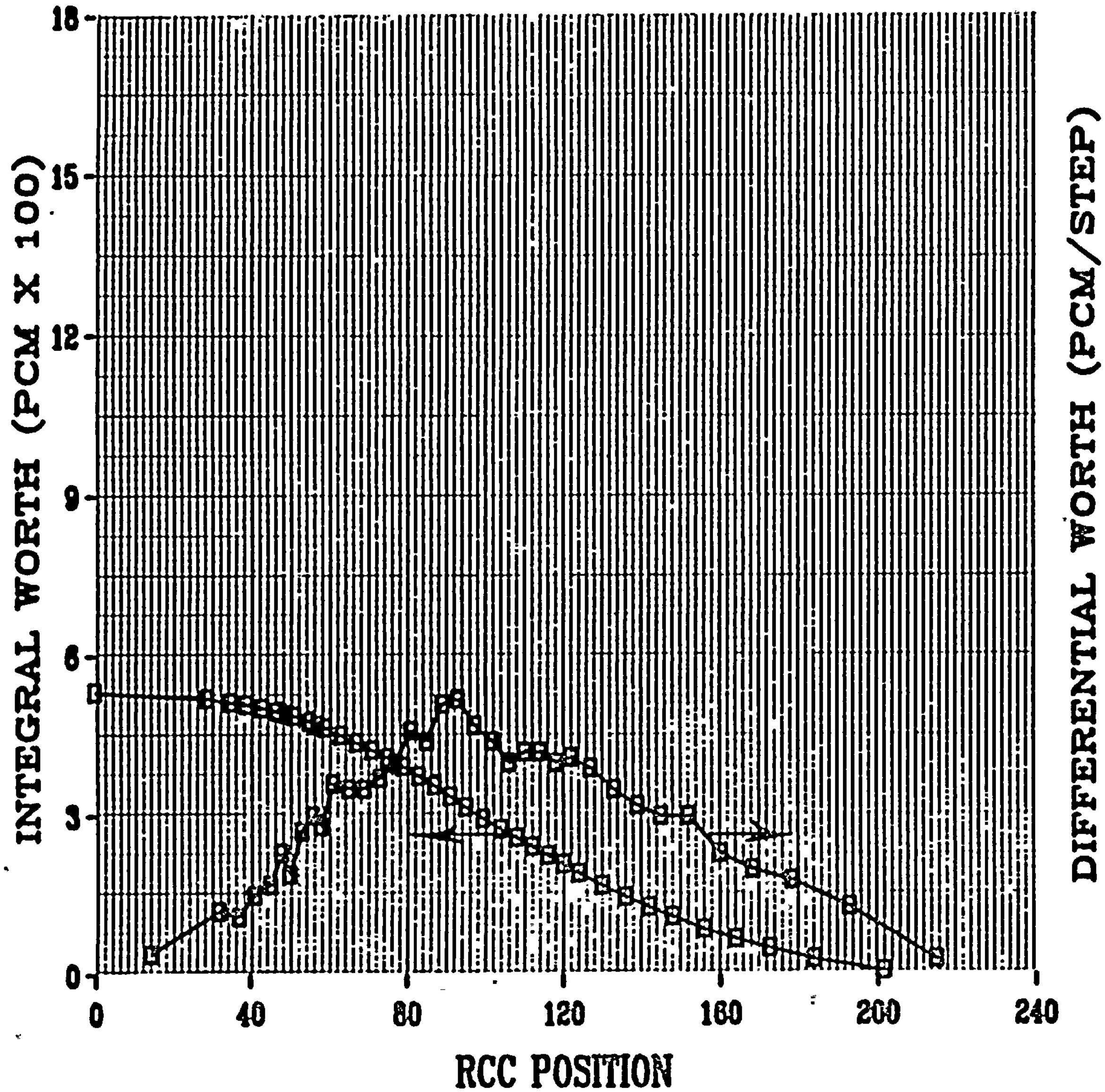


FIGURE 19

Unit: 2
T.P.: 41.4
Date: 8-23-85

Test Conditions:

1. RCC Bank Positions:

SDA	228
SDB	228
SDC	228
SDD	228
CA	moving
CB	0
CC	0
CD	0
RCCA	-

2. Power Level:
   0 %RTP
3. RCS Temperature:
   Initial 547.5 F
   Final 547 F
4. RCS Pressure:
   Initial 2237 psig
   Final 2235 psig
5. Avg. Core Burnup:
   0 MWD/MTU

CONTROL BANK WORTH CURVES

NORMAL BANK OVERLAP

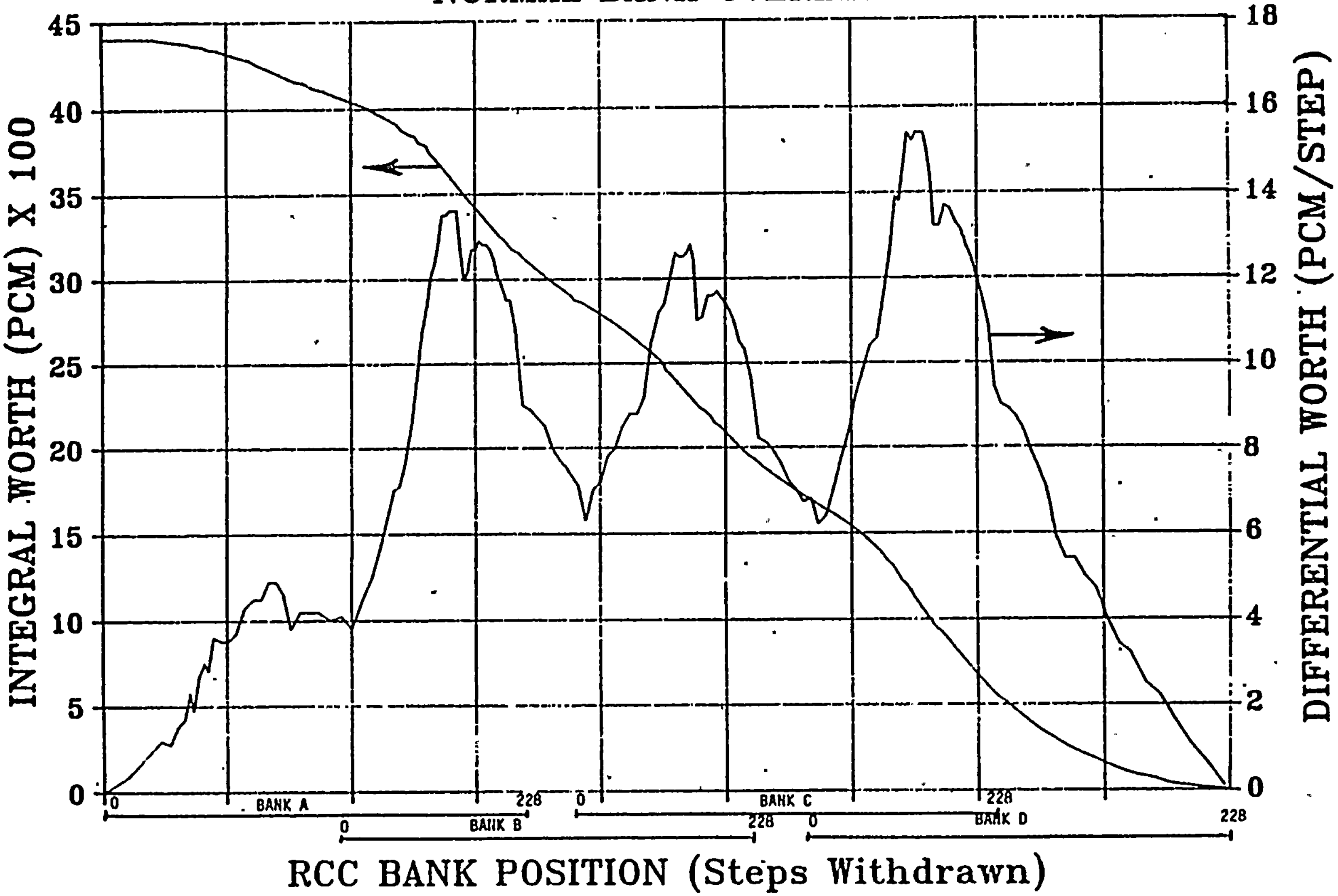


FIGURE 20

3.5 Test Procedure No. 41.6 - Rod Cluster Control Assembly (RCCA) Pseudo Ejection At Zero Power

TEST OBJECTIVE

The objective of this test was to simulate ejection of the most reactive control rod (D12) with control banks at the zero power insertion limit. The integral worth of Rod D12 was measured and compared with design values. Core power distribution was determined to verify that hot channel factors were within predicted values and FSAR limits.

TEST DESCRIPTION

Reference conditions were established with the reactor critical in the zero power test range. Control rods were positioned at approximately the zero power rod insertion limit and Control Bank D adjusted to reference conditions as shown below:

Shutdown Banks A, B, C, D	228 steps
Control Bank A	228 steps
Control Bank B	227 steps
Control Bank C	99 steps
Control Bank D	5 steps

A baseline incore flux map was obtained and analyzed.

Lift coil disconnect switches for all rods in Control Bank D except rod D12 were opened and a 300 pcm/hr continuous boron addition commenced. Criticality was maintained by withdrawal of the "ejected rod", D12. During withdrawal, integral rod worth for D12 was measured. With D12 near the full out position, boron addition was stopped. With D12 fully withdrawn, the ejected rod incore flux map was taken. Following the flux map, D12 was inserted with reactivity compensated by the withdrawal of Control Bank C. In order to allow a flux map measurement required by Test Procedure 41.3 (ref. Section 3.3), the final configuration consisted of Bank C fully withdrawn and Bank D realigned at a position of 5 steps.

TEST RESULTS

The reactivity worth measured during the withdrawal of control rod D12 was 505 pcm. When this value was increased by 10% for conservatism, the worth was 555 pcm. This value is well below the design upper-limit acceptance criteria of 737 pcm and the safety analysis upper-limit acceptance criteria of 785 pcm.

The power distributions met acceptance criteria and were consistent with expectations as the post-ejection flux peaking factors were located near rod D12.

3.5 (Continued)

The post-ejection value of F_Q^T (i.e., heat flux hot channel factor) was 8.0, well under the design limit acceptance criteria of 10.45 and the safety analysis limit acceptance criteria of 13.0. The power distribution results are summarized in Table 9.

TABLE 9

Power Distribution Results (Pre and Post Pseudo RCCA Ejection)

ITEM	PRE-EJECTED FLUX MAP	POST-EJECTED FLUX MAP
CONDITIONS	547 deg. F	547 deg. F
- temperature	1174 ppm	1222 ppm
- boron conc.	0%	0%
- power	0 MWD/MTU	0 MWD/MTU
- burnup	HZP insertion limit	HZP insertion limit with rod D12 withdrawn
- rod configuration		
DATE	August 25, 1985	August 25, 1985
$F^N_{\Delta H}$ - measured value - location*	1.64 E14-AA	4.68 C13-CC
F^T_Q - measured value - location*	3.15 J02-AQ @ 38"	8.00 C13-CC @ 55"
QUADRANT TILT - by quadrant	1.000 | 1.009 0.988 | 1.003	0.359 | 0.703 0.713 | 2.225

* Assembly locations (i.e., E14) are shown in Figure 1.

Pin location within assembly (i.e., AA) are based on 17 x 17 matrix ranging from AA to QQ.

RELATIVE ASSEMBLY POWER DISTRIBUTIONS - PSEUDO RCCA EJECTION

Pre Ejection Relative Assembly Power
Post Ejection Relative Assembly Power

NORTH ↓

	R	P	N	M	L	K	J	H	G	F	E	D	C	B	A
1					.756 .222	.887 .278	.989 .325	.843 .305	.985 .395	.888 .403	.757 .360				
2			.560 .179	1.026 .326	1.270 .379	1.285 .390	1.204 .403	1.005 .360	1.191 .492	1.261 .552	1.253 .593	1.001 .477	.551 .315		
3		.551 .166	1.079 .316	.914 .288	1.229 .364	1.279 .412	1.195 .398	.996 .379	1.185 .489	1.238 .580	1.198 .570	.906 .480	1.081 .648	.557 .395	
4		1.001 .292	.906 .275	.683 .217	.995 .320	1.239 .405	1.009 .377	.600 .264	1.026 .487	1.237 .607	.986 .552	.688 .439	.915 .644	1.015 .746	
5	.731 .238	1.227 .389	1.188 .364	.974 .321	1.116 .368	1.089 .402	1.065 .428	.949 .452	1.087 .557	1.121 .650	1.141 .720	.994 .729	1.216 .913	1.257 1.010	.747 .621
6	.848 .300	1.217 .398	1.207 .400	1.200 .404	1.079 .404	.956 .387	.935 .436	1.029 .526	.941 .585	.958 .658	1.098 .850	1.236 1.000	1.264 1.114	1.263 1.103	.878 .818
7	.960 .353	1.168 .421	1.163 .398	1.002 .379	1.039 .426	.915 .436	.915 .471	.777 .488	.925 .686	.931 .823	1.061 .981	1.015 1.025	1.191 1.215	1.192 1.283	.974 1.053
8	.809 .319	.966 .374	.975 .384	.588 .263	.908 .449	.994 .520	.768 .485	.462 .386	.770 .765	1.007 1.080	.931 1.147	.597 .837	.995 1.362	.994 1.305	.822 1.138
9	.948 .397	1.162 .496	1.162 .492	.998 .480	1.030 .551	.898 .579	.893 .672	.757 .754	.926 1.056	.938 1.291	1.074 1.672	1.023 1.827	1.184 2.055	1.179 2.070	.961 1.643
10	.849 .367	1.229 .529	1.204 .575	1.192 .598	1.068 .645	.927 .656	.906 .824	.989 1.075	.922 1.286	.959 1.545	1.112 2.230	1.240 2.638	1.269 2.809	1.261 2.652	.870 1.823
11	.739 .335	1.252 .594	1.211 .596	.960 .543	1.107 .717	1.078 .854	1.060 .988	.919 1.154	1.053 1.683	1.093 2.244	1.106 2.805	.964 3.080	1.159 3.281	1.276 3.335	.753 1.810
12		.993 .509	.887 .516	.680 .452	.987 .734	1.228 1.007	1.019 1.023	.591 .839	1.010 1.827	1.215 2.659	.947 3.060	.653 4.052	.887 3.474	1.039 3.189	
13		.540 .325	1.059 .668	.913 .661	1.208 .916	1.244 1.129	1.174 1.232	.977 1.393	1.187 2.118	1.255 2.889	1.188 3.407	.875 3.365	1.065 3.861	.573 2.020	
14			.541 .383	.999 .746	1.253 1.043	1.245 1.140	1.167 1.351	.972 1.370	1.194 2.206	1.288 2.715	1.275 3.341	1.020 3.082	.564 1.955		
15					.739 .654	.854 .849	.959 1.147	.819 1.239	.989 1.809	.888 1.867	.752 1.814				

DIABLO CANYON POWER PLANT - UNIT 2

FIGURE 21

3.6 Test Procedure No. 41.7 - Minimum Shutdown Margin Verification and Stuck Rod Worth Measurement

TEST OBJECTIVE

The objectives of this test were to 1) measure the reactivity worth of the Shutdown Banks, 2) measure the critical boron concentration with all control rods inserted and the most reactive rod (F10) fully withdrawn, and 3) measure the reactivity worth of the most reactive rod.

TEST DESCRIPTION

The test began with the reactor critical in the zero power test range with all control banks inserted and all shutdown banks withdrawn. Prior to the reactivity worth measurement, for Shutdown Bank D, preparations were made to enter the Technical Specification Special Test Exception for minimum shutdown margin. This required demonstration of the ability to trip from at least 50% withdrawn each control rod not fully inserted within 24 hours prior to reducing the shutdown margin to less than 1.6% $\Delta k/k$. Therefore, with all Shutdown Banks fully withdrawn, the reactor was tripped. Control Bank C (i.e., the bank with the most reactive rod, F10) was then withdrawn to greater than 114 steps and tripped. These actions met the requirements for entry into Test Exception 3.10.1. The reactor was then returned to criticality with all control banks inserted and all shutdown banks fully withdrawn.

An RCS boron dilution of 500 pcm/hr was then commenced in order to measure the individual bank worths of Shutdown Banks D and C. The dilution was stopped when Shutdown Banks D and C were inserted and Shutdown Banks A and B were still withdrawn.

Control Bank C (the bank containing the most reactive rod, F10) was then pulled to 5 steps. Lift coil disconnect switches were opened for all rods on Control Bank C with the exception of F10. While maintaining criticality, Shutdown Bank A was exchanged with Rod F10. As Shutdown Bank A reached the fully inserted position, the exchange with F10 was continued using Shutdown Bank B. When Rod F10 reached its fully withdrawn condition a dilution was commenced to allow the insertion of the remainder of Shutdown Bank B. At this point the reactor was critical with all rods inserted with the exception of the most reactive rod, F10, and Shutdown Bank B 33 steps from the bottom. This was considered to be the design All-Rods-In N-1 configuration. The boron endpoint was obtained for this condition.

Once the boron endpoint was obtained, the reactivity computer was rescaled in order to observe the reactivity insertion associated with dropping F10. The stationary gripper coil fuses for Rod F10 were pulled, dropping F10 into the core.

The reactor trip breakers were then opened and the RCS borated until conditions were reached to achieve criticality with all shutdown banks withdrawn.

TEST RESULTS

All acceptance criteria were met, with the exception of the reactivity worth of Shutdown Bank D. The measured worth was 756 pcm, just outside the design acceptance criteria of 675 ± 68 pcm. A subsequent Westinghouse review deemed the results acceptable. It should be noted that the individual worth of a shutdown bank has little significance; the combined worth of shutdown and control banks is much more important.

The measured worth of Shutdown Bank C was 1021 pcm, well within the acceptance criteria of 975 ± 98 pcm.

The total reactivity worth of all control and shutdown banks less rod F10 was 6421 pcm, very close to the design acceptance criteria of 6432 ± 643 pcm and well above the safety analysis lower limit acceptance criteria of 5789 pcm. The worth of rod F10, based on the drop from the N-1 configuration, was 858 pcm.

The critical boron concentration for the all-rods-in less F10 (i.e., N-1) configuration was 741 ppm, which meets the acceptance criteria of 719 ± 63 ppm.

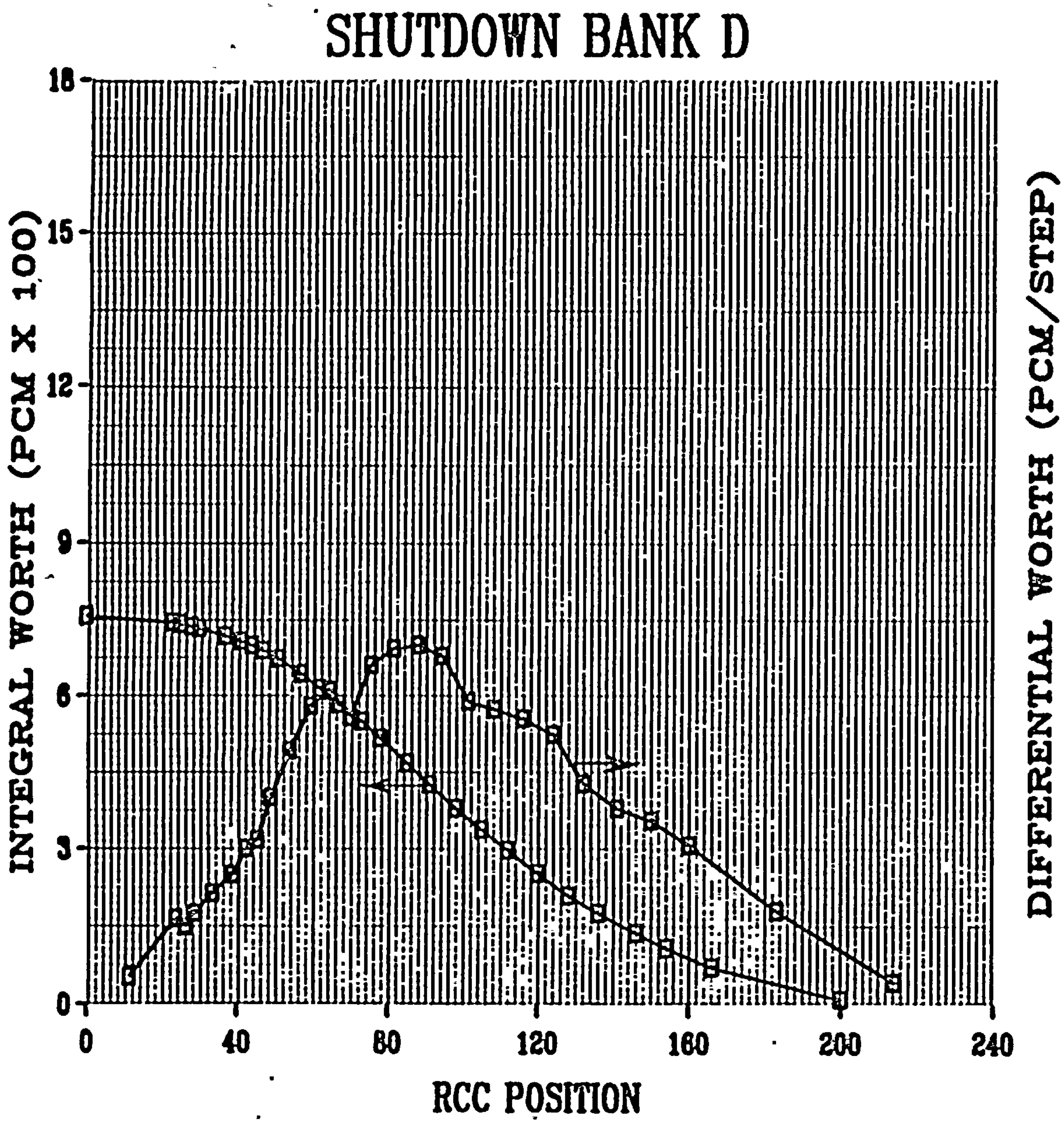


FIGURE 22

Unit: 2

T.P.: 41.4

Date: 8-23-85

Test Conditions:

1. RCC Bank Positions:

SDA	228
SDB	228
SDC	228
SDD	moving
CA	0
CB	0
CC	0
CD	0
RCCA	-

2. Power Level:
   0 %RTP
3. RCS Temperature:
   Initial 547 F
   Final 547 F
4. RCS Pressure:
   Initial 2245 psig
   Final 2245 psig
5. Avg. Core Burnup:
   0 MWD/MTU

SHUTDOWN BANK C

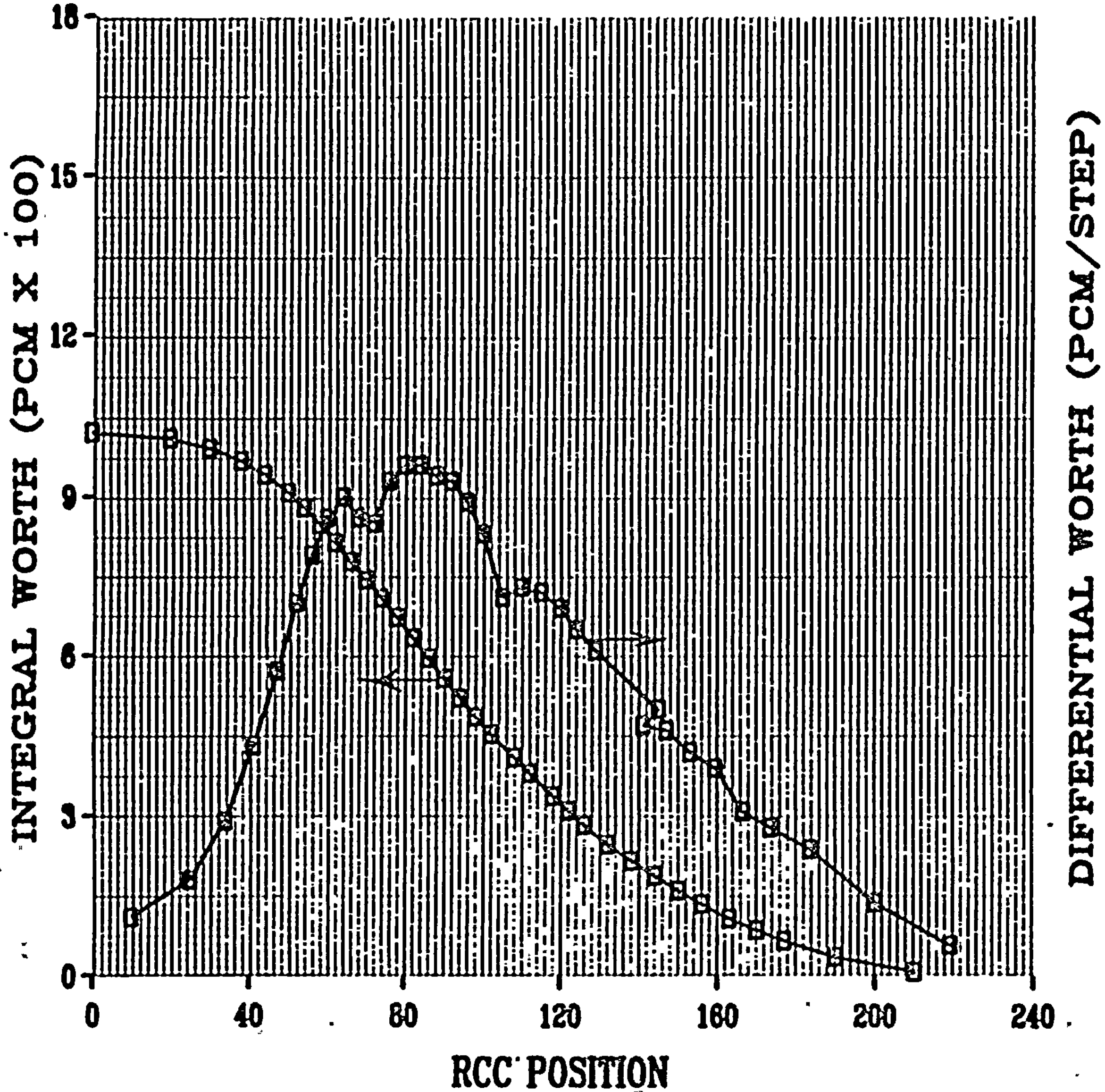


FIGURE 23

Unit: 2

T.P.: 41.4

Date: 8-23-85

Test Conditions:

1. RCC Bank Positions:

SDA	228
SDB	228
SDC	moving
SDD	0
CA	0
CB	0
CC	0
CD	0
RCCA	-

2. Power Level: 0 %RTP
3. RCS Temperature:
   Initial 547.1 F
   Final 547 F
4. RCS Pressure:
   Initial 2245 psig
   Final 2245 psig
5. Avg. Core Burnup: 0 MWD/MTU

4.0 Test Procedure No. 3.7 Addendum 3 - Turbine Driven Auxiliary Feedwater Pump Endurance Test

TEST OBJECTIVE

The objective of this test was to demonstrate the reliability of the Turbine Driven Auxiliary Feedwater Pump (AFW Pump 2-1) by operating the pump for an extended period of time to comply with the requirements of NUREG-0737.

TEST DESCRIPTION

With the reactor at approximately 3.7% power, sufficient to support rated flow from the Turbine Driven Auxiliary Feedwater Pump (2-1), AFW Pump 2-1 was run in the minimum recirculation mode. After an initial inspection of all components, the pump was aligned to supply water to the Steam Generators and return it to the Condensate Storage Tank via the Condensate Reject Loop from the hotwell. Feedwater flow to the Steam Generators was established and the system was run at rated flow for 48 hours. During this endurance run, selected data such as pump head, pump flow, turbine bearing temperatures, pump bearing temperatures, vibration readings, pump room temperature and humidity were monitored. Pump flow was maintained above a rated flow of 880 gpm during the test. At the end of the 48 hour run, flow was returned to rated flow and all the parameters recorded to ensure pump performance had not degraded. After 48 hours, the pump was shutdown and pump temperatures were allowed to cool to within 20 deg. F of their initial values. Following the cooldown, the pump was restarted and run for one hour at rated flow.

TEST RESULTS

Auxiliary Feedwater Pump 2-1 was started on August 26, 1985 and hourly readings were taken to determine rated flow and bearing temperature stability. Listed below is a sequence of events explaining the problems experienced during this test and their resolutions:

August 27: Secured AFW Pump 2-1 due to Turbine inboard and outboard bearing temperatures exceeding 175 deg. F.

August 28: After a ground at the Thermocouple Cold Junction Box was repaired, AFW Pump 2-1 was restarted and the endurance test officially begun at 1100 hrs.

August 29: 0540 hrs, a Reactor Trip occurred due to Low Steam Generators Level/ Steam and Feed Flow mismatch; consequently, a safety injection occurred on a Low-Low Tave signal.

August 31: Reactor Coolant Pump (RCP) 2-4 tripped on overcurrent. Upon further investigation, it was determined that it had some damaged windings. Testing was put on hold until RCP 2-4 was rewound and reinstalled.

September 27: RCP 2-4 was rewound and was onsite.

4.0 (Continued)

October 8: Plant was at 2.2 % power and AFW Pump 2-1 was running on recirculation flow in preparation for aligning it to the Steam Generators. Valves LCV 106 and 107, Auxiliary Feedwater Flow to Steam Generators 2-1 and 2-2, would not close. AFW Pump 2-1 was secured and the motor-driven Auxiliary Feedwater Pumps were aligned to feed the Steam Generators. Upon further investigation, it was determined that the valves had broken shear pins which were eventually repaired and LCVs 106 and 107 were placed back in service.

October 9: AFW Pump 2-1 was supplying rated flow to the Steam Generators. Forty-eight hour endurance test had begun.

October 11: Secured AFW Pump 2-1. Waiting for AFW Pump 2-1 to cooldown to within 20 deg F. of its initial temperatures prior to restart.

October 12: Restarted AFW Pump 2-1 after its cooldown period and ran it for at least one hour at full rated flow. This completed the AFW Pump endurance test.

During the 48 hour run, Auxiliary Feedwater Pump 2-1 operated satisfactorily and all acceptance criteria were met. Pump flow remained above the 880 gpm minimum required flow, pump suction pressure varied between 15.6 and 21.8 psig ($\geq$10.14 psig required) and pump differential pressure remained very close to 1400 psid ($\geq$1253.4 psid required). Bearing temperatures and vibration readings were within limits for both the AFW pump and turbine.

5.0 POWER ASCENSION TEST PROGRAM

5.1 Summary

Preparations for power ascension began on October 12, 1985, after the successful completion of the Turbine Driven Auxiliary Feedwater Pump Endurance Test. Mode 1 was entered for the first time on October 12, 1985 for test procedure 41.8, Dynamic Automatic Steam Dump Control Test. The generator was synchronized and on line on October 20, 1985 and testing at the 15% power plateau was completed on October 21, 1985.

The power level was increased to 30% on October 22, 1985, and testing at this plateau was completed on October 31, 1985.

The 50% power level testing plateau was established on November 1, 1985 and testing at this plateau was completed on November 15, 1985. The major reasons for the delay at this plateau were due to two unscheduled reactor trips (see trips no. 5 and 6 in Section 5.29, Unscheduled Reactor Trips) and a forced ramp down to approximately 1% power to repair the steam leaks at the high pressure stop valves for Main Feedwater Pumps 2-1 and 2-2.

The power level was increased to 75% power on November 15, 1985, and testing at this plateau was completed on December 8, 1985. The major reasons for the delay at this power plateau were due to:

1) Steam Generator chemistry problems caused by inefficient resin beds in the Condensate Polishing System (CPS). This problem could only be resolved by an on going program to change out the resin beds in each CPS vessel.

2) The 12kv power supply cable to Circulating Water Pump 2-1 was accidentally cut during core drilling, requiring power to be reduced.

3) Four unscheduled reactor trips (see trips no. 7, 8, 9 and 10 in Section 5.29, Unscheduled Reactor Trips).

4) The steam dump program modifications after the test data obtained during the transient testing indicated that the steam dump valves' performance had to be improved.

5) Severe storms caused an excessive accumulation of kelp at the Intake Structure which caused damage and forced a ramp down in power until the kelp was removed and the damage was repaired.

6) Repeating of T.P. 43.7, Net Load Trip from 50% power, after the steam dump program was modified.

The power level was subsequently increased on December 9, 1985 with the unit reaching 90% power on December 11, 1985. Testing at this plateau was completed on December 20, 1985. The major reasons for the delay at this power plateau were due to:

1) Steam Generator chemistry being out of specified limits and an excessive pressure drop across the CPS.

5.1 (Continued)

2) Plant power being reduced to approximately 20% power to repair damaged air supply line to Main Feedwater Regulating Valve, FCV-530.

3) Several salt water leaks in the main condenser which caused several forced power reductions to locate the faulty condenser tubes.

On December 20, 1985 the power level was increased to 100% and testing at this plateau was completed on March 13, 1986.

The main reasons for the delay at this power plateau were due to:

1) The plant being taken off line after discovery of a loose and arcing neutral to ground strap on the B phase main transformer.

2) Six unscheduled reactor trips (see trips no. 11, 12, 13, 14, 15, and 16 in Section 5.29, Unscheduled Reactor Trips).

3) Power being reduced several times to repair leaking condenser tubes in the main condenser. On January 13, 1986, condenser tube leakage had increased to the point that it was no longer possible for the CPS to maintain the steam generator chemistry within specified limits. Therefore, it was decided to perform T.P. 43.4 (Plant Trip from 100% Power), begin the Strainer Outage, find/repair the leaking condenser tubes, perform the required Westinghouse repairs to the main generator and then continue with the remainder of the testing required at the 100% power plateau.

On February 20, 1986 the Strainer Outage was declared complete with the plant being synchronized to the grid and the plant ramping up at approximately 3% per hour with holds at selected power plateaus for turbine performance testing. On February 22 the plant experienced a reactor trip during the power escalation (see trip no. 15 in Section 5.29, Unscheduled Reactor Trips). The plant was on line again on February 25 and ramping up to 100% power at 3% per hour with selected hold points. Startup testing was subsequently resumed on March 1, 1986 and Diablo Canyon Power Plant Unit 2 was declared commercial at 0300 hours on March 13, 1986.

5.2 Test Procedure 41.8 - Dynamic Automatic Steam Dump Control

TEST OBJECTIVE

The objective of this test was to verify proper operation of the Turbine Trip, Load Rejection, and Steam Pressure controllers in the Steam Dump Control System and to adjust controller setpoints, if needed, to obtain satisfactory response.

TEST DESCRIPTION

With the Main Turbine tripped, reactor power at 1% and steam dump being controlled in the steam pressure control mode, the turbine trip controller was tested by raising Tavg to 550 deg. F and then transferring into the Tavg mode. The Steam Dump System and Tavg were monitored for proper response. Reactor power was then increased to 6% at a fast rate while monitoring the Steam Dump System and Tavg for proper response.

Testing the load rejection controller required the Main Turbine latched, reactor power at 3% and the Steam Dump System in the steam pressure control mode. Two additional special requirements were to have:

(1) The sudden load loss interlock actuated to place the load rejection controller in the Tavg control circuit and to unblock the Steam Dump Valves, and

(2) A simulated Tref signal of 543 deg. F into the load rejection controller to create a temperature mismatch.

The Steam Dump System was then transferred to the Tavg mode while monitoring the Steam Dump System and Tavg response.

Testing the Steam Header Pressure Controller required the reactor to be at 1% power and the Steam Dump System in the steam pressure control mode. With the steam pressure controller in automatic, reactor power was increased to 5% while monitoring the Steam Dump System and steam pressure response.

TEST RESULTS

The testing of the Turbine Trip Controller was performed without any problems. During the power increase transient from 1% to 6%, the Steam Dump System responded satisfactorily and Tavg stabilized at 550 deg. F (within the acceptance criteria of 549.4 deg. F to 554.6 deg. F).

The testing of the Load Rejection Controller was performed without any problems. During the transient, the Steam Dump System responded satisfactorily and Tavg stabilized at 549.7 deg. F (within the acceptance criteria of 545.4 deg. F to 550.6 deg. F).

5.2 (Continued)

The Steam Dump Control System responded satisfactorily during the Steam Pressure Controller Test. The steam pressure stabilized at 990 psig (within the acceptance criteria of 986.2 to 1023.8 psig).

Since the Steam Dump Control System responded satisfactorily, there was no need to adjust any of the controller setpoints.

5.3 Test Procedure No. 22.9 - Main Turbine Overspeed Trip Test

TEST OBJECTIVE

The objective of this test was to verify operability of the Main Turbine Overspeed Protection System.

TEST DESCRIPTION

The turbine was run between 80-90 MW for a ten hour "Soak" period and then unloaded. The overspeed setpoints (103%, 111% and 111.5% of normal speed) were verified by Surveillance Test Procedure (STP) M-21B.

TEST RESULTS

STP M-21B was performed satisfactorily and the results are shown in Table 10.

Table 10

Turbine Overspeed Setpoints

	Trip Setting	Actual	Acceptance Criteria.
DEH	103%	1857 rpm	1850 - 1858 rpm
Mechanical	111%	1969 rpm	1963 - 1999 rpm
DEH*	111.5%	1918 rpm	1862 - 1960 rpm

*NOTE: The setpoints are automatically reduced by 4.5.% while testing the 111.5% trip setting from the digital electrohydraulic (DEH) unit.

5.4 Test Procedure No. 42.9 - Operational Alignment of Nuclear Instrumentation System

TEST OBJECTIVE

The objective of this test was to align and monitor the Nuclear Instrumentation System (NIS) prior to and during core loading and through power ascension.

TEST DESCRIPTION

Prior to core loading, the pulse amplifier attenuator and discriminator voltage settings, the high voltage power supply plateau and the operating voltage settings for the source range channels were determined.

Prior to Startup, the initial trip setpoint for all the nuclear instrumentation channels was determined. During Startup, the overlap between source range and intermediate range and between the intermediate range and power range channels were determined. During power ascension, the power range detector currents vs. core power were determined and the flux deviation alarm settings were monitored. At the 50% power test plateau, the intermediate and power range operating detector voltages were checked. While at 100% power, the power range operating detector currents were obtained.

After shutdown from power operations at the 100% power test plateau, the intermediate range detectors' compensation voltages were set.

TEST RESULTS

Required adjustments, calibrations, and setpoint determinations were accomplished without significant problems using standard I&C procedures. The source range instrumentation data prior to core loading is listed in Table 11. The Nuclear Instrumentation data prior to Startup is shown in Table 12. Results of the Nuclear Instrumentation overlap data taken prior to criticality and at various power levels are shown in Table 13. Power range detector high level trip setpoints which were reset prior to each power increase to the next power plateau are listed in Table 14.

Intermediate range and power range detector characteristics were determined at the 50% power plateau prior to the power range Incore-Excore detector calibration. Detector plateaus were also determined at this power level.

While at 100% power, indicated power range detector currents were taken. Results are shown in Table 15.

Shortly after shutdown from power operation at the 100% power test plateau and with a core burnup of at least 1200 MWD/MTU, the intermediate range detectors' compensating voltages were set to provide an overlap of about three decades with the source range detectors (31.506 Vdc for N35 and 34.450 Vdc for N36).

Table 11

Source Range Instrumentation Data Prior to Core Loading

Parameter	Units	Detector	
		N31	N32
Attenuator Setting	db.	10	10
Discriminator Voltage	Vdc	-0.450	-0.631
Detector Voltage	Vdc	2200	2200
Detector Voltage bistable trip	Vdc	2100	2100
High Flux alarm	cps	35.51	35
High Flux trip	cps	1.0×10^5	9.9×10^4

Table 12

Nuclear Instrumentation Data Prior to Startup

Intermediate Range Channels

Parameter	Units	Detector	
		N35	N36
1. High Voltage Setting	Vdc	800	800
2. Compensating Voltage	Vdc	40.03	40.00
3. Compensating Voltage Bistable Trip	Vdc	20.06	20.02
4. Loss of Detector Trip	Vdc	701	698
5. P-6 Bistable Trip	amp	1.05×10^{-10}	1.2×10^{-10}
6. Rod Stop Bistable Trip	amp	8.0×10^{-5}	6.0×10^{-5}
7. Reactor Trip Bistable	amp	8.0×10^{-5}	7.6×10^{-5}

Power Range Channels

Parameter	Units	Detector			
		N41	N42	N43	N44
1. High Voltage Setting	Vdc	800	800	800	800
2. High Voltage Bistable Trip	Vdc	700	698	699.3	700
3. P10 Bistable Trip	%	9.97	9.99	9.99	9.98
4. P8 Bistable Trip	%	34.92	34.96	34.97	34.99
5. Overpower Rod Stop Bistable Trip	%	102.97	103.02	103.03	103.03
6. High Neutron Flux Rate Trip	%	4.78	4.72	4.80	4.75
7. Flux Rate Time Constant	sec	2.15	2.15	2.15	2.16

Table 13

Nuclear Instrumentation Overlap Data

DETECTOR	PRECRITICAL READINGS	0% POWER	~15% POWER	~30% POWER	~50% POWER	~75% POWER	~90% POWER	~100% POWER
SOURCE RANGE (cps)								
N31 - Control Board	50	$3x10^{4}$	Blocked	Blocked	Blocked	Blocked	Blocked	Blocked
N31 - NI Drawer	60	$3x10^{4}$	Blocked	Blocked	Blocked	Blocked	Blocked	Blocked
N32 - Control Board	45	$2x10^{4}$	Blocked	Blocked	Blocked	Blocked	Blocked	Blocked
N32 - NI Drawer	55	$2x10^{4}$	Blocked	Blocked	Blocked	Blocked	Blocked	Blocked
INTERMEDIATE RANGE (amps)								
N35 - Control Board	$1.0x10^{-11}$	$9x10^{-11}$	$7x10^{-5}$	$1.5x10^{-4}$	$2.0x10^{-4}$	$3x10^{-4}$	$3.6x10^{-4}$	$4.0x10^{-4}$
N35 - NI Drawer	$1.0x10^{-11}$	$9x10^{-11}$	$8x10^{-5}$	$1.3x10^{-4}$	$2.0x10^{-4}$	$3x10^{-4}$	$3.3x10^{-4}$	$3.6x10^{-4}$
N36 - Control Board	$1.0x10^{-11}$	$1x10^{-10}$	$7x10^{-5}$	$1.4x10^{-4}$	$2.0x10^{-4}$	$3x10^{-4}$	$3.5x10^{-4}$	$3.8x10^{-4}$
N36 - NI Drawer	$1.0x10^{-11}$	$1x10^{-10}$	$8x10^{-5}$	$1.4x10^{-4}$	$2.1x10^{-4}$	$3x10^{-4}$	$3.5x10^{-4}$	$3.9x10^{-4}$
POWER RANGE (%)								
N41 - Control Board	0	0	15.0	30.2	49.5	76.2	91.0	99.6
N41 - NI Drawer	0	0	15.5	30.8	49.0	75.5	90.5	99.0
N42 - Control Board	0	0	16.0	30.5	50.0	76.9	91.0	100.0
N42 - NI Drawer	0	0	16.0	30.8	49.5	76.1	91.0	100.0
N43 - Control Board	0	0	16.0	30.4	50.0	77.1	91.0	100.0
N43 - NI Drawer	0	0	16.0	30.8	49.0	75.8	90.5	99.5
N44 - Control Board	0	0	16.0	30.5	49.0	76.2	91.0	99.9
N44 - NI Drawer	0	0	16.0	30.9	49.5	76.3	91.0	99.8

Table 14

Power Range High Level Trip Set Points

Power Plateaus (% RTP)	Desired Setpoint (%)	Actual Set Point (%)			
		N41	N42	N43	N44
0 to 5	24.5 $\pm$ 0.5	24.4	24.2	24.2	24.1
15	24.5 $\pm$ 0.5	24.4	24.2	24.2	24.1
30	40 $\pm$ 0.5	40.0	40.0	40.0	40.0
50	60 $\pm$ 0.5	60.0	59.9	60.0	60.2
75	90 $\pm$ 0.5	90.0	90.2	90.0	90.0
90	109 $\pm$ 0.5	109.0	109.0	109.2	109.0
100	109 $\pm$ 0.5	109.0	109.0	109.2	109.0

Table 15

Power Range Detector Currents at 100% Power

Detector	Upper Detector Current (μa)	Lower Detector Current (μa)
N41	350	397
N42	312	352
N43	359	410
N44	384	413

5.5 Test Procedure No. 42.8 - Operational Alignment of Reactor Coolant System Temperature Instrumentation

TEST OBJECTIVE

The purpose of this test procedure was to align the ΔT and Tave instrumentation channels during power ascension.

TEST DESCRIPTION

At isothermal conditions, ΔT and Tave values were determined from Thot and Tcold readings. At each power ascension test plateau, ΔT and Tave data were collected and transcribed to this test. At the 75% power plateau, linear regression analysis was used to determine extrapolated Tave and ΔT for each loop at 100% power. These extrapolated full power values were averaged, and the average ΔT and Tave were used to make the necessary adjustments to ΔT, Overtemperature ΔT, and Overpressure ΔT instrumentation.

At 100% power, the calibrations were to be refined by adjusting the instrumentation based on the actual ΔT values of each reactor coolant loop. A final verification for Tave consisted of comparing loop Tave values to the average Tave. A final verification for ΔT involved comparing the power level inferred from each loop's ΔT to core-average power based on a secondary side heat balance (i.e., Surveillance Test Procedure R-2B).

TEST RESULTS

At isothermal conditions, ΔT and Tave values agreed with the values calculated from Thot and Tcold readings within the specified tolerance as shown in Table 16. Instrumentation adjustments were not needed.

At 75% power, a linear regression was performed and the extrapolated Tave and ΔT values for each loop at 100% power were 569.12 deg. F and 62.53 deg. F respectively. Both the extrapolated ΔT and Tave values were below the respective upper limits of 64.4 deg. F and 577.6 deg. F. These values are consistent with the fact that the measured RCS flow rate is slightly greater than design.

While at 90% power, it was determined by analyzing the test data and the plant's indicators that 100% power could not be achieved with the existing ΔT and Tave scaling. It was decided at this time that the adjustment of the instrumentation would be based on each loop's extrapolated 100% power ΔT and Tave values. Following this adjustment, power was increased to 100% power.

Based on data collected at 100% RTP, the ΔT and Tave scaling was further refined. Following these adjustments, verification data was collected and all acceptance criteria were met. The final results are summarized in Table 17.

Table 16

ΔT and Tave at Isothermal Conditions

Parameter (deg. F)	Loop 1	Loop 2	Loop 3	Loop 4
Thot	546.94	546.99	547.00	546.98
Tcold	547.42	547.35	547.03	547.02
ΔT (calculated)	-0.48	-0.36	-0.03	-0.04
ΔT (measured)	-0.39	-0.40	-0.45	-0.14
Tave (calculated)	547.18	547.17	547.02	547.00
Tave (measured)	547.38	547.17	546.82	547.04

Acceptance Criteria

ΔT (measured) = ΔT (calculated) ± 0.5 deg. F
Tave (measured) = Tave (calculated) ± 1.0 deg. F

Table 17

ΔT and Tave at Full Power
(data recorded at 99.76% of RTP)

Parameter	Loop 1	Loop 2	Loop 3	Loop 4
Tave (deg. F)	571.34	572.37	570.11	572.23
ΔT (deg. F)	60.78	63.28	59.61	61.70
Power-based on ΔT (%)	99.57	100.03	99.52	99.79
Power-based on R-2B(%)	99.76	99.76	99.76	99.76
Loop power deviation (%)	-0.19	+0.27	-0.24	+0.03
Tave - Avg Tave (deg. F)	-0.17	+0.86	-1.40	+0.72

Acceptance Criteria

Upper Limits: Tave <577.6, ΔT <64.4 deg. F

Tave: Loop Tave within 2 deg. F of average Tave

ΔT: Power based on ΔT within 1% of power based on R-2B
(i.e., Loop power deviation $\leq$1.0%)

5.6 Test Procedure No. 4.1 - Calibration of Steam and Feedwater Flow Instrumentation at Power.

TEST OBJECTIVE

The objective of the test was to calibrate the steam flow instrumentation to feedwater flow and to perform a cross-check verification of all signals indicating feedwater and steam flow with the reference feedwater flow determined by high accuracy differential pressure (d/p) gauges across the feedwater system venturis.

TEST DESCRIPTION

The feedwater and steam flow instrumentation output signals were checked against the reference feedwater flow (Barton gauges) at steady state power levels of 15%, 30%, 50%, 75%, 90% and 100% RTP. Test data collected were analyzed to determine the deviation of steam and feedwater flow compared to the reference feedwater flow. Any transmitted signal data found to be outside the allowable tolerance was submitted to the Instrumentation Department for evaluation and recalibration as required. If any adjustments were made, verification data were collected and analyzed prior to ascending to the next power plateau.

TEST RESULTS

Several feedwater flow transmitters were just outside the allowed 1.5% deviation at low power levels but indicated values within tolerance at higher test plateaus. This discrepancy was attributed to noise in the data, as the transmitters were well within tolerance at the 100% test plateau.

The eight steam flow transmitters required rescaling as follows:

50% power - Transmitters 532, 533, 542, 543
75% power - Transmitters 512, 513, 522, 523, 542, 543
90% power - Transmitters 532, 533
100% Power outage - Transmitters 512, 513, 522, 523, 532, 533, 542, 543

Rescaling a transmitter generally brought it within tolerance at that power plateau. The rescaling process presumed that the transmitter's output voltage was proportional to the square of the steam flow. Observed behavior was slightly different than a square function, so the transmitter was usually out of tolerance at a higher power level.

At the completion of power ascension testing, all feedwater transmitters were within their tolerance of 1.0% of reference flow while the steam flow transmitters were within their tolerance of 2.0% of reference flow.

5.7 Surveillance Test Procedure R-3A - Incore Power Distribution

TEST OBJECTIVE

The purpose of this procedure was to obtain flux maps using the Movable Incore Detector System (MIDS). The detector outputs were used to determine such core parameters as axial flux distributions, peaking factors, and core tilts for several startup tests during the power ascension power plateaus. The flux maps were also used to fulfill the routine surveillance requirements.

TEST DESCRIPTION

Various full core flux maps and quarter core flux maps were performed during power ascension testing. Full core maps nominally involved 12 passes through the core by the six incore detectors. Quarter core maps involved three passes of the detectors in selected locations and were used only for determining axial flux distribution. Digitized detector output from the flux maps served as input to the INCORE computer code which calculated relative assembly powers, peaking factors, and quadrant power tilts for the full core cases.

Below is a chronological summary of the flux maps taken during the power ascension test program:

- 30% power, all rods out (ARO), equilibrium xenon; provided base line data for T.P. 42.5 - Statepoint Data Collection.

- 30% power, Control Bank D at approximately 177 steps (100% RTP Rod Insertion Limit), equilibrium xenon; provided reference data for T.P. 42.2 - RCCA Pseudo Ejection and RCCA above bank position measurements.

- 30% power, Control Bank D (except RCCA D-12) at approximately 177 steps, RCCA D-12 fully withdrawn to simulate a rod ejection; provided post-ejection data for T.P. 42.2 - RCCA Pseudo Ejection and RCCA above bank position measurement.

- 50% power, ARO, equilibrium xenon; provided baseline data for T.P. 42.5 - State Point Data Collection; and provided reference data for STP R-13 - Nuclear Power Range Incore-Excore Detector Calibration.

- 6 quarter-core maps at 50% power provided data for STP R-13, Nuclear Power Range Incore-Excore Detector Calibration.

- 75% power, all rods out (ARO), equilibrium xenon; provided base line data for T.P. 42.5 - Statepoint Data Collection.

- 75% power, ARO, equilibrium xenon; provided reference data for STP R-13 - Nuclear Power Range Incore-Excore Detector Calibration.

- 6 quarter-core maps at 75% power provided data for STP R-13, Nuclear Power Range Incore-Excore Detector Calibration.

5.7 (Continued)

-90% power, ARO, equilibrium xenon; provided base line data for Test Procedure 42.5 - Statepoint Data Collection.

-100% power, ARO, equilibrium xenon; provided base line data for Test Procedure 42.5 - Statepoint Data Collection.

TEST RESULTS

Each of the flux maps listed above was analyzed and determined to be satisfactory. Results are discussed in more detail in Section 5.8 (Test Procedure 42.5) and Section 5.18 (Surveillance Test Procedure R-13).

5.8 Test Procedure No. 42.5 - Statepoint Data Collection

TEST PROCEDURE

The objective of this test was to collect statepoint data and verify neutron flux distribution at various power ascension test plateaus. Core power level was determined by secondary system heat balance calculations.

TEST DESCRIPTION

This test was performed at nominal power levels of 15%, 30%, 50%, 75%, 90%, and 100% rated thermal power. Initial conditions at each plateau consisted of stable plant parameters, equilibrium xenon, and control rods at or near fully withdrawn positions. Upon establishing these conditions, data were collected as concurrently as possible. Recorded information included:

- Full core flux maps through the use of the Incore Movable Detector System, (except at 15% power)

- Steady state plant process data

- Secondary plant parameters such as steam line pressure, steam generator pressure, turbine inlet pressure and turbine impulse pressure.

The collected information served as a data base for steady state conditions at each of the power plateaus during the power ascension test program.

RESULTS

Results specific to this test procedure included the calculated power levels and the core power distributions.

Measured steady state, equilibrium power distributions (i.e., relative assembly power, radial power shape, axial power shape, quadrant power tilt, peaking factors) were within Acceptance Criteria and very close to design predictions. Peaking factors were well below limits specified by the Technical Specifications. Results of the flux maps are summarized in Table 18 and Figures 24 through 28. At each power plateau, $F^N_{\Delta H}$ and F^T_Q obtained were compared to the limiting values at the next power plateau and found acceptable.

At each test plateau, test equipment was used to measure steam generator pressure. These pressures were compared to steam line pressure readings taken at the pressure taps at which permanent plant equipment is connected. The steam line pressure drop from the steam generator to the PT taps just outside containment increased with power level. Each loop had a full power pressure drop of just under 17 psi, as shown in Figure 29.

Table 18

Power Distribution Results at Power

ITEM	30% POWER TEST PLATEAU	50% POWER TEST PLATEAU	75% POWER TEST PLATEAU	90% POWER TEST PLATEAU	100% POWER TEST PLATEAU
CONDITIONS* - temperature - boron concentration - burnup	~555 deg. F 1086 ppm 68 MWD/MTU	~560 deg. F 1015 ppm 189 MWD/MTU	~564 deg. F 947 ppm 360 MWD/MTU	~566 deg. F 931 ppm 735 MWD/MTU	~569 deg. F 923 ppm 1218 MWD/MTU
DATE	10-27-85	11-04-85	11-18-85	12-13-85	1-11-86
$F^N_{\Delta H}$ - Measured value - location**	1.363 B07-AQ	1.346 M12-LE	1.341 B07-AQ	1.335 J04-AQ	1.339 J05-LE
F^T_Q - Measured value - location**	2.036 M04-IJ @79"	1.996 M12-LE @74"	2.049 M12-LE @58"	2.014 M04-LM @60"	1.987 M04-LM @58"
F_Z - Measured value	1.360	1.339	1.390	1.366	1.363
Quadrant Tilt - Measured value	1.006	1.005	1.007	1.006	1.009

* Common conditions include stable plant parameters, equilibrium xenon, control rods at or near fully withdrawn positions.

** Assembly locations (i.e., D12) as shown in Figure 1. Pin location within assembly (i.e., IH) based on 17x17 matrix ranging from AA to QQ.

CORE AVERAGE RADIAL POWER DISTRIBUTION 30% TEST PLATEAU

ASSEMBLY AVERAGE POWERS FROM UNRODDED FLUX MAP

Relative Assembly Power (Pi)
$\frac{\text{Measured Pi} - \text{Expected Pi}}{\text{Expected Pi}} \times 100$

NORTH

	R	P	N	M	L	K	J	H	G	F	E	D	C	B	A
1					.590 2.2	.689 1.0	.821 1.6	.747 1.5	.818 1.2	.678 -.7	.574 -.6				
2			.498 .2	.870 .1	1.042 2.3	1.027 1.2	1.061 .7	1.048 .3	1.047 -.6	.999 -1.6	.999 -1.8	.848 -2.3	.496 -.2		
3		.496 -.3	1.038 -.2	.946 -.3	1.134 2.5	1.129 2.4	1.180 1.6	1.133 1.3	1.165 .4	1.076 -2.5	1.081 -2.3	.929 -2.1	1.039 -.2	.506 1.8	
4		.864 -.5	.945 -.4	1.258 .3	1.071 .9	1.184 .8	1.153 .8	1.183 1.0	1.143 -.1	1.169 -.6	1.049 -1.2	1.258 .3	.953 .4	.881 1.4	
5	.573 -.7	1.006 -1.2	1.094 -1.2	1.054 -.7	1.185 1.4	1.151 1.4	1.202 3.0	1.148 2.5	1.194 2.2	1.140 .4	1.175 .6	1.072 1.0	1.120 1.2	1.029 1.1	.584 1.2
6	.674 -1.3	.996 1.9	1.077 -2.4	1.160 -1.3	1.138 .3	1.161 1.7	1.048 2.4	1.116 1.9	1.028 .4	1.136 -.5	1.126 -.8	1.184 .8	1.114 1.0	1.025 .9	.690 1.1
7	.799 -1.2	1.037 -1.6	1.137 -2.0	1.122 -1.9	1.159 -.8	1.030 .7	1.073 2.0	.980 2.1	1.068 1.4	1.028 .5	1.184 1.4	1.153 .8	1.179 1.6	1.067 1.3	.833 3.1
8	.721 -2.0	1.025 -1.9	1.095 -2.1	1.153 -1.5	1.097 -2.0	1.097 .2	.973 1.4	1.016 1.6	.969 1.0	1.103 .8	1.134 1.2	1.182 1.0	1.137 1.7	1.055 1.0	.752 2.2
9	.791 -2.1	1.033 -2.0	1.135 -2.2	1.121 -2.0	1.139 -2.4	1.007 -1.6	1.045 -.7	.960 .0	1.076 2.2	1.039 1.5	1.189 1.8	1.152 .7	1.181 1.8	1.064 .9	.823 1.9
10	.678 -.6	1.008 -.7	1.070 -3.0	1.152 -2.0	1.112 -2.0	1.119 -2.0	1.005 -1.7	1.086 -.8	1.022 -.1	1.145 .3	1.138 .3	1.179 .3	1.114 .9	1.025 .9	.688 .9
11	.588 1.9	1.038 2.0	1.131 2.1	1.043 -1.8	1.154 -1.2	1.119 -1.4	1.155 -1.1	1.104 -1.4	1.153 -1.2	1.127 -.7	1.162 -.6	1.057 -.5	1.098 -.8	1.044 2.5	.592 2.5
12		.873 .5	.940 -.9	1.258 .2	1.065 .3	1.177 .2	1.127 -1.5	1.151 -1.7	1.122 -1.9	1.167 -.8	1.048 -1.3	1.240 -1.2	.937 -1.3	.879 1.2	
13		.495 -.6	1.035 -.5	.951 .2	1.117 .9	1.098 -.5	1.135 -2.2	1.089 -2.6	1.148 -1.1	1.106 .3	1.108 .1	.943 -.6	1.036 -.4	.497 -.1	
14			.495 -.5	.872 .4	1.031 1.3	1.013 -.3	1.029 -2.4	1.020 -2.3	1.046 -.7	1.031 1.6	1.032 1.4	.879 1.2	.500 .6		
15					.584 1.2	.676 -1.0	.795 -1.6	.725 -1.5	.814 .7	.692 1.5	.585 1.3				

DIABLO CANYON POWER PLANT - UNIT 2

FIGURE 24

CORE AVERAGE RADIAL POWER DISTRIBUTION 50% TEST PLATEAU

ASSEMBLY AVERAGE POWERS FROM UNRODDED FLUX MAP

Relative Assembly Power (Pi)
$\frac{\text{Measured Pi} - \text{Expected Pi}}{\text{Expected Pi}} \times 100$

NORTH

	R	P	N	M	L	K	J	H	G	F	E	D	C	B	A
1					.575 1.6	.679 1.4	.806 2.2	.742 2.6	.804 1.9	.670 .1	.567 .1				
2			.498 3.7	.864 3.6	1.002 1.7	1.015 1.5	1.045 1.2	1.045 1.1	1.034 .1	.993 -.7	.982 -.3	.837 .4	.488 1.6		
3		.493 2.6	1.014 2.7	.941 2.6	1.110 1.8	1.116 1.7	1.177 1.3	1.134 1.2	1.168 .5	1.082 -1.4	1.085 -.4	.923 .7	1.003 1.6	.493 2.6	
4		.851 2.1	.938 2.2	1.232 1.0	1.058 -.1	1.185 -.2	1.162 .0	1.201 .5	1.158 -.2	1.183 -.4	1.059 -.0	1.236 1.4	.934 1.8	.852 2.2	
5	.572 1.1	.989 .4	1.097 .7	1.057 -.2	1.179 -.3	1.155 -.4	1.208 .5	1.164 .8	1.210 .7	1.160 .1	1.186 .3	1.072 1.2	1.110 1.8	1.002 1.7	.574 1.5
6	.665 -.6	.986 -1.4	1.068 -2.6	1.165 -1.9	1.144 -1.3	1.172 -.8	1.059 -.8	1.140 -.5	1.053 -1.3	1.165 -1.3	1.138 -1.8	1.199 1.0	1.113 1.4	1.013 1.3	.679 1.4
7	.789 -.1	1.027 -.7	1.140 -1.9	1.138 -2.0	1.177 -2.1	1.046 -2.0	1.094 -1.4	.999 -1.5	1.095 -1.2	1.046 -2.0	1.193 -.7	1.170 .8	1.176 1.3	1.045 1.1	.805 2.0
8	.719 -.5	1.028 -.5	1.104 -1.5	1.177 -1.5	1.124 -2.7	1.117 -2.5	.990 -2.4	1.044 -1.8	.998 -1.6	1.127 -1.6	1.147 -.7	1.204 .8	1.135 1.2	1.044 1.0	.735 1.7
9	.784 -.7	1.027 -.6	1.151 -.9	1.136 -2.2	1.163 -3.2	1.029 -3.6	1.066 -3.9	.984 -3.0	1.100 -.8	1.057 -.9	1.205 .3	1.164 .3	1.175 1.1	1.044 1.0	.802 1.6
10	.668 -.3	.997 -.3	1.080 -1.6	1.164 -1.9	1.135 -2.1	1.148 -2.8	1.030 -3.5	1.113 -2.9	1.045 -2.1	1.170 -1.0	1.158 -.1	1.187 -.1	1.104 .6	1.015 1.5	.679 1.4
11	.573 1.1	.997 1.2	1.104 1.3	1.042 -1.6	1.169 -1.1	1.136 -2.0	1.175 -2.2	1.120 -3.0	1.172 -2.5	1.145 -1.2	1.185 .2	1.062 .3	1.091 .1	1.006 2.1	.578 2.1
12		.851 2.1	.947 3.2	1.249 2.4	1.070 1.0	1.197 .8	1.142 -1.7	1.167 -2.3	1.130 -2.6	1.172 -1.3	1.059 .0	1.224 .4	.929 1.2	.854 2.4	
13		.493 2.6	1.014 2.7	.941 2.6	1.103 1.2	1.098 .1	1.142 -1.7	1.092 -2.5	1.049 -1.1	1.098 .1	1.110 1.8	.930 1.4	1.005 1.8	.493 2.7	
14			.493 2.6	.852 2.2	1.002 1.7	1.004 .4	1.019 -1.3	1.020 -1.3	1.037 .3	1.027 2.8	1.013 2.8	.858 2.9	.494 2.8		
15					.574 1.4	.668 -.3	.787 -.3	.722 -.1	.805 2.0	.688 2.7	.582 2.7				

DIABLO CANYON POWER PLANT - UNIT 2

FIGURE 25

CORE AVERAGE RADIAL POWER DISTRIBUTION 75% . TEST PLATEAU

ASSEMBLY AVERAGE POWERS FROM UNRODDED FLUX MAP

Relative Assembly Power (Pi)

$$\frac{\text{Measured Pi} - \text{Expected Pi}}{\text{Expected Pi}} \times 100$$

NORTH

	R	P	N	M	L	K	J	H	G	F	E	D	C	B	A
1					.570 .5	.678 .9	.806 1.8	.745 2.6	.805 1.7	.669 -.4	.565 -.3				
2			.490 2.6	.851 2.6	.991 .5	1.012 .9	1.048 1.0	1.049 1.3	1.039 .1	.994 -.9	.977 -.9	.822 -.9	.478 .1		
3		.487 2.0	.997 2.0	.923 1.4	1.094 .6	1.108 .6	1.176 1.2	1.134 1.3	1.170 .7	1.086 -1.4	1.076 -1.1	.904 -.7	.979 .2	.483 1.0	
4		.844 1.8	.925 1.6	1.184 -.4	1.046 -1.0	1.177 -1.0	1.164 .4	1.181 1.0	1.161 .1	1.185 -.4	1.050 -.7	1.191 .2	.913 .3	.835 .7	
5	.570 .6	.985 -.1	1.092 .5	1.048 -.9	1.185 .1	1.168 .2	1.222 1.4	1.169 1.3	1.218 1.1	1.168 .1	1.180 -.3	1.058 .1	1.093 .5	.990 .5	.571 .7
6	.660 -1.8	.984 -1.8	1.076 -2.3	1.167 -1.9	1.157 -.8	1.192 .3	1.084 .7	1.157 .5	1.072 -.4	1.180 -.8	1.150 -1.4	1.195 .5	1.110 .8	1.010 .7	.676 .7
7	.790 -.2	1.031 -.6	1.143 -1.6	1.143 -1.5	1.194 -.9	1.074 -.2	1.117 .2	1.017 -.1	1.117 .2	1.076 -.1	1.217 1.0	1.178 1.5	1.178 1.4	1.046 .9	.802 1.3
8	.722 -.5	1.030 -.5	1.100 -1.7	1.157 -1.0	1.134 -1.8	1.142 -.8	1.013 -.5	1.042 -.4	1.018 -.0	1.147 -.4	1.161 .5	1.183 1.1	1.137 1.6	1.043 .7	.732 .8
9	.786 -.7	1.030 -.7	1.145 -1.5	1.143 -1.5	1.178 -2.2	1.054 -2.1	1.093 -2.0	1.003 -1.5	1.124 .8	1.077 .0	1.219 1.2	1.168 .7	1.180 1.5	1.042 .4	.797 .7
10	.665 -1.0	.992 -1.0	1.071 -2.8	1.173 -1.4	1.149 -1.5	1.167 -1.8	1.054 -2.1	1.133 -1.6	1.064 -1.1	1.184 -.4	1.171 .4	1.195 .5	1.117 1.4	1.009 .7	.676 .6
11	.572 .9	.995 .9	1.099 1.1	1.045 -1.2	1.178 -.5	1.154 -1.0	1.192 -1.0	1.133 -1.9	1.183 -1.8	1.162 -.4	1.186 .2	1.059 .2	1.082 -.5	1.001 1.5	.575 1.5
12		.835 .6	.915 .5	1.204 1.3	1.069 1.2	1.202 1.0	1.148 -1.0	1.148 -1.9	1.135 -2.1	1.184 -.5	1.052 -.5	1.184 -.4	.912 .1	.848 2.2	
13		.480 .6	.984 .6	.920 1.0	1.101 1.3	1.103 .1	1.145 -1.4	1.092 -2.4	1.150 -1.0	1.103 .1	1.092 .4	.906 -.5	.983 .6	.491 2.8	
14			.480 .6	.836 .8	.995 .9	1.001 -.2	1.027 -1.0	1.027 -.9	1.044 .6	1.025 2.2	1.003 1.7	.840 1.3	.487 2.0		
15					.569 .4	.666 -.9	.793 .2	.729 .5	.811 2.4	.686 2.1	.576 1.7				

DIABLO CANYON POWER PLANT - UNIT 2

FIGURE 26

CORE AVERAGE RADIAL POWER DISTRIBUTION 90% TEST PLATEAU

ASSEMBLY AVERAGE POWERS FROM UNRODDED FLUX MAP

Relative Assembly Power (Pi)
$\dfrac{\text{Measured Pi} - \text{Expected Pi}}{\text{Expected Pi}} \times 100$

NORTH

	R	P	N	M	L	K	J	H	G	F	E	D	C	B	A
1					.563 1.5	.667 1.4	.789 1.9	.727 2.2	.780 .7	.649 -1.4	.547 -1.4				
2			.480 1.8	.832 1.8	.987 1.6	1.002 1.5	1.036 1.0	1.032 .9	1.021 -.6	.972 -1.7	.956 -1.6	.805 -1.5	.469 -.5		
3		.474 .7	.974 .8	.919 .8	1.100 1.7	1.117 1.6	1.176 1.5	1.140 1.4	1.165 .6	1.079 -1.9	1.065 -1.6	.900 -1.3	.962 -.4	.473 .4	
4		.818 .1	.914 .2	1.219 1.0	1.087 1.9	1.213 1.8	1.183 .9	1.208 1.2	1.174 .2	1.180 -1.0	1.052 -1.4	1.197 -.8	.908 -.4	.816 -.1	
5	.550 -.8	.962 -1.0	1.075 -.6	1.066 -.1	1.212 1.9	1.199 1.8	1.251 3.0	1.204 2.6	1.242 2.3	1.170 -.6	1.173 -1.4	1.058 -.9	1.086 .4	.975 .4	.562 1.4
6	.644 -2.1	.967 -2.2	1.076 -2.1	1.182 -.8	1.184 .6	1.213 1.3	1.114 1.9	1.183 1.5	1.100 .6	1.184 -1.1	1.156 -1.8	1.190 -.1	1.109 .9	.996 .8	.667 1.3
7	.773 -.1	1.019 -.7	1.142 -1.4	1.157 -1.3	1.209 -.4	1.094 .1	1.146 1.3	1.056 1.3	1.148 1.5	1.096 .3	1.220 .5	1.184 1.0	1.172 1.2	1.037 1.1	.785 1.4
8	.709 -.4	1.018 -.5	1.107 -1.5	1.185 -.7	1.157 -1.4	1.159 -.5	1.043 -.0	1.082 .1	1.044 .1	1.170 .5	1.185 1.0	1.212 1.5	1.137 1.1	1.030 .6	.715 .4
9	.769 -.7	1.019 -.7	1.139 -1.6	1.158 -1.2	1.192 -1.8	1.076 -1.6	1.118 -1.2	1.032 -1.1	1.141 .8	1.106 1.1	1.237 1.9	1.183 .9	1.168 .8	1.028 .2	.777 .3
10	.650 -1.2	.976 -1.3	1.066 -3.0	1.175 -1.4	1.160 -1.4	1.178 -1.6	1.075 -1.7	1.148 -1.4	1.081 -1.1	1.201 .3	1.189 1.0	1.204 1.1	1.111 1.0	.989 .1	.658 .0
11	.558 .7	.979 .7	1.092 .9	1.054 -1.2	1.184 -.5	1.167 -.9	1.206 -.7	1.155 -1.6	1.196 -1.5	1.171 -.6	1.194 .4	1.072 .4	1.083 .1	.979 .8	.558 .7
12		.819 .2	.910 -.2	1.208 .1	1.072 .4	1.196 .4	1.161 -1.0	1.173 -1.8	1.149 -2.0	1.184 -.6	1.068 .0	1.210 .3	.921 1.0	.831 1.7	
13		.470 -.3	.965 -.2	.912 -.0	1.088 .6	1.096 -.3	1.141 -1.5	1.097 -2.4	1.145 -1.2	1.099 -.0	1.086 .4	.910 -.2	.973 .7	.483 2.6	
14			.470 -.2	.815 -.2	.970 -.2	.979 -.9	1.006 -1.9	1.004 -1.9	1.020 -.6	1.005 1.7	.980 .9	.817 .1	.477 1.3		
15					.552 -.4	.649 -1.4	.764 -1.2	.704 -1.1	.779 .7	.669 1.6	.559 .8				

DIABLO CANYON POWER PLANT - UNIT 2

FIGURE 27

CORE AVERAGE RADIAL POWER DISTRIBUTION 100% TEST PLATEAU

ASSEMBLY AVERAGE POWERS FROM UNRODDED FLUX MAP

Relative Assembly Power(Pi)
$\frac{\text{Measured Pi} - \text{Expected Pi}}{\text{Expected Pi}} \times 100$

NORTH

	R	P	N	M	L	K	J	H	G	F	E	D	C	B	A
1					.548 .3	.648 -.3	.765 .3	.707 .5	.763 .0	.639 -1.7	.537 -1.6				
2			.476 1.9	.822 1.8	.966 .3	.977 -.2	1.018 -.2	1.012 -.3	1.009 -1.1	.958 -2.0	.943 -2.1	.789 -2.3	.459 -1.7		
3		.469 .4	.962 .5	.919 .6	1.082 .5	1.104 .4	1.162 .7	1.136 .8	1.154 .0	1.075 -2.3	1.053 -2.2	.895 -2.0	.942 -1.6	.462 -1.1	
4		.805 -.3	.911 -.2	1.218 .7	1.092 1.5	1.207 1.4	1.186 .5	1.214 1.1	1.180 .0	1.190 -.1	1.065 -.9	1.215 .4	.907 -.7	.805 -.3	
5	.542 -.8	.951 -1.2	1.066 -1.0	1.070 -.5	1.217 2.1	1.212 2.1	1.258 3.2	1.219 2.7	1.249 2.5	1.199 1.0	1.205 1.1	1.087 1.1	1.087 .9	.971 .8	.551 1.0
6	.638 -1.8	.958 -2.1	1.077 -2.1	1.180 -.9	1.197 .9	1.227 2.0	1.140 2.8	1.202 2.3	1.123 1.2	1.206 .2	1.186 -.1	1.203 1.0	1.111 1.0	.987 .9	.656 .9
7	.753 -1.2	1.007 -1.3	1.137 -1.4	1.168 -1.0	1.222 .3	1.123 1.2	1.170 2.2	1.087 2.2	1.170 2.3	1.123 1.3	1.236 1.5	1.190 .8	1.165 .9	1.028 .8	.770 1.0
8	.691 -1.7	1.001 -1.3	1.112 -1.4	1.198 -.2	1.177 -.8	1.182 .6	1.077 1.2	1.124 2.0	1.085 2.0	1.196 1.8	1.206 1.7	1.219 1.5	1.138 .9	1.018 .4	.704 .1
9	.748 -1.9	1.006 -1.5	1.135 -1.6	1.172 -.7	1.205 -1.2	1.100 -.8	1.140 -.4	1.071 .6	1.180 3.1	1.138 2.6	1.251 2.7	1.190 .8	1.160 .6	1.021 .0	.762 .0
10	.638 -1.8	.960 -1.8	1.068 -3.0	1.180 -.9	1.175 -1.0	1.190 -1.1	1.100 -.8	1.177 .2	1.121 1.1	1.223 1.6	1.203 1.4	1.207 1.4	1.097 -.3	.975 -.3	.647 -.4
11	.543 -.6	.958 -.5	1.073 -.4	1.068 -.7	1.189 -.2	1.180 -.6	1.216 -.2	1.175 -.9	1.206 -1.0	1.183 -.3	1.193 .1	1.076 .1	1.061 -1.5	.965 .2	.547 .1
12		.801 -.8	.905 -1.0	1.214 .4	1.081 .6	1.196 .4	1.173 -.6	1.185 -1.3	1.161 -1.7	1.186 -.4	1.076 .1	1.214 .4	.911 -.3	.811 .4	
13		.464 -.6	.952 -.5	.912 -.2	1.082 .4	1.091 -.9	1.131 -2.0	1.096 -2.8	1.136 -1.6	1.093 -.7	1.072 -.4	.905 -.9	.953 -.4	.469 .6	
14			.464 -.5	.803 -.5	.959 -.4	.963 -1.6	.993 -2.6	.988 -2.6	1.009 -1.2	.983 .5	.962 -.1	.802 -.6	.467 -.0		
15					.541 -.9	.635 -2.2	.745 -2.3	.689 -2.1	.761 -.2	.653 .5	.545 -.1				

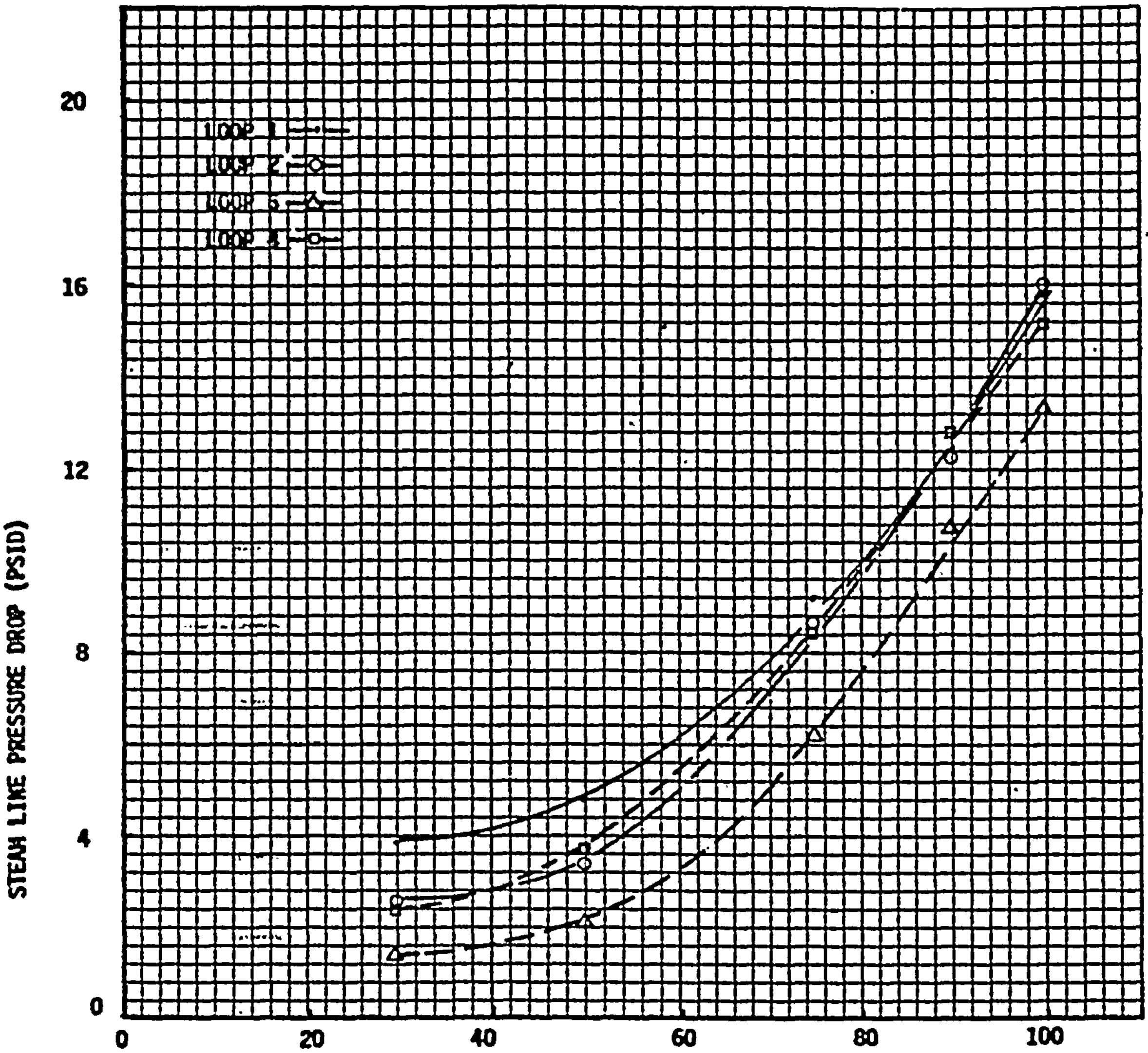
DIABLO CANYON POWER PLANT - UNIT 2

FIGURE 28

DCC 33780

STEAM LINE PRESSURE DROP

(Steam Generator Pressure Minus Steam Line Pressure at PT Tap Outside Containment)



POWER* (%)	PRESSURE DROP (psid) LOOP 1	LOOP 2	LOOP 3	LOOP 4
30	3.9	2.6	1.4	2.4
50	4.8	3.4	2.2	3.7
75	9.2	8.7	6.2	8.5
90	12.5	12.3	10.7	12.8
100	15.8	16.0	13.3	15.2

*DATA UNRELIABLE BELOW 30% POWER

FIGURE 29

5.9 Test Procedure No. 38.6 - Startup Adjustments of Reactor Control System

TEST OBJECTIVE

The main objective of this test procedure was to determine the reactor coolant average temperature program required to maintain the design full load Steam Generator pressure without exceeding steam generator pressure or Tave limitations.

TEST DESCRIPTION

Reactor Coolant Tave, Steam Generator pressure and Turbine Impulse Chamber pressure were recorded at 0%, 30% and 50% RTP. Each of these parameters was extrapolated to 100% RTP. A temperature program correction was then computed from the difference between the saturation temperature of the extrapolated Steam Generator pressure and the saturation temperature of the design full load steam generator outlet pressure of 805 psia. This correction was applied to the design temperature program generated by the Reactor Control System, Steam Dump Control System and plant computer. With Tave controlled at the new Tref, Turbine Impulse Chamber pressure was compared to the 50% load design value and agreement was verified. This entire process was repeated at 75% RTP to obtain a further refinement in the temperature program. Upon reaching 100% RTP, the temperature program was adjusted to obtain the design value of Steam Generator pressure. Throughout this procedure, changes in the temperature program were verified to maintain the 100% RTP value for Tref below 577.6 deg. F.

TEST RESULTS

Test data were taken at 0%, 30% and 50% RTP and the results were plotted and extrapolated to 100% RTP. A Tref correction of -2.53 deg. F was subtracted from the most recently determined Tref (100%) of 573.24 deg. F to yield the new projected 100% RTP Tref of 570.71 deg. F. This value correlated closely with the extrapolated 100% Tave of 569.12 deg. F. Tref as a linear function of percent load was used to determine the desired voltage as a linear function of power for recalibration of the Turbine Impulse Chamber Pressure Controllers TC-505 and TC-505A. This calibration was done at 50% RTP and retest results of the data taken at 50% RTP after calibration were acceptable.

Test data taken at 75% RTP indicated no need for any additional refinement in the Tref program.

At 100% RTP, steam generator pressures were 782.55, 782.85, 781.05 and 786.15 psia for loops 1, 2, 3 and 4, respectively and were outside the Acceptance Criteria range of 805 $\pm$ 10 psia. Tave values were 568.36, 569.65, 567.28 and 567.5 deg. F for loops 1, 2, 3 and 4, respectively, all were below the limit of 577.6 deg. F. Turbine impulse chamber pressure was 548.42 psia, 14.58 psia below the full load design value of 563.0 psia.

5.9 (Continued)

In order to maintain steam generator pressure within the Acceptance Criteria range, a recalculation of the Tref program was performed taking into consideration any Tave/Tref mismatch that existed while taking the last set of measurements. This technique resulted in a new 100% Tref program of 572.8 deg. F. A recalibration of the Tref program was performed and test data taken again at 100% RTP. The results of the data are listed below in Table 19.

Table 19

Tave Conditions at 100% Power

Parameter	Loop 1	Loop 2	Loop 3	Loop 4
Tave (deg. F)	571.5	572.8	570.1	572.7
Steam Generator Pressure (psia)	810.1	811.02	809.75	811.42
Turbine Impulse Pressure (psia)	552.65	----	----	----
Measured Reactor Power (%)	100.23%	----	----	----

Acceptance Criteria

Tave $\leq$ 577.6 deg. F

Steam Generator Pressure: 805 psia $\pm$10

A calculation of the change in Moderator Temperature Coefficient (MTC) due to the reduced Tref at 100% RTP resulted in the equivalent of approximately 1 ppm boron. This change in Tref will have no appreciable effect on rod withdrawal limits and will not result in a positive Moderator Temperature Coefficient for the current rod withdrawal limits.

5.10 Test Procedure No. 1.15 - Radiation Surveys and Shielding Effectiveness

TEST OBJECTIVE

The objective of this procedure was to verify the adequacy of the radiation surveys and shielding effectiveness program as prescribed by Nuclear Plant Operations (NPO) Procedure TC 8410. The main objective of the test program was to measure radiation levels in accessible areas of Unit 2 at various power levels and identify any locations where shielding may be deficient.

TEST DESCRIPTION

Radiation measurement points or radiation base points (RBPs) were located throughout the DCPP site. The purpose of the RBPs was to provide fixed points outside radiation shields from which the neutron and gamma radiation levels could be measured. The radiation levels at shield wall pipe penetrations and the area close to the Steam Generators were also measured. The measured radiation levels were then compared to the FSAR design criterion to determine the adequacy of the shield. Most RBPs were located outside the secondary shield wall of Unit 2. Secondary shielding is defined as the shielding in the reactor building designed to attenuate the gamma radiation emanating from the primary coolant system external to the reactor vessel. Labyrinth entrances and shielding penetrations were closely monitored to determine the adequacy of the shielding. Most RBPs were selected to verify that the radiation levels at labyrinth entrances and penetrations met design radiation levels. Radiation base points were also located to test shielding thickness adequacy by measuring the radiation levels on the shield side farthest away from the radiation source.

Neutron and gamma radiation measurements were taken at each fixed radiation base point unless dose rates precluded measurement. The radiation dose rates measured during the early stages of the testing program were linearly extrapolated to the 100% power level. Background radiation measurements were taken prior to the start-up of the Unit 2 reactor. Background radiation measurements were taken so values against which measurements made during the start-up phase of Unit 2 could be compared. Radiation measurements during the start-up phase of DCPP Unit 2 were taken at 0%, 18%, 50%, and 100% power levels.

Penetrations less than 2 meters above the floor were measured for neutron and gamma radiation. Those penetrations located greater than 2 meters above the floor were surveyed only for gamma radiation. Measurements were taken with the radiation detector as close to the penetration as possible.

TEST RESULTS

The adequacy of the "as-built" DCPP Unit 2 radiation shielding was verified by comparing the startup bioshield survey results with the Final Safety Analysis Report Update radiation zone requirements and the Shielding Design Review for Diablo Canyon Units 1 and 2. All radiation zone requirements were met. Radiation dose equivalent rates in the Unit 2 containment were found to be much lower than at similar plants. As expected, most RBPs exhibited a high degree of positive linear correlation with reactor power level.

5.11 Test Procedure No. 1.16 - Effluents and Effluents Monitoring

TEST OBJECTIVE

The objective of this procedure was to document the existence of an adequate program to verify the level of liquid and gaseous radwaste releases. Specifically, this test verifies the calibration of the effluent monitors by comparing with laboratory sample analysis results.

TEST DESCRIPTION

Effluent monitoring is an ongoing program by the DCPP staff which involves following plant procedures. The test collects data to verify the effluent monitoring program and from this data verifies the calibration of the effluent monitors. The intent was to perform these verifications at the 30, 50, 75, and 100% power test plateaus.

TEST RESULTS

A minimum activity level is required to adequately judge the calibration of each monitor. However, throughout the power ascension program, the activity levels at each monitor had not been large enough to verify the calibration of the monitors.

Only rad monitor RE-18 (i.e., the rad monitor for the Liquid Batch Tanks to Outfall) had a sufficiently high activity level. The data collected using Unit 1 procedures (RE-18 is a common monitor for both units) have provided an empirical and reasonably consistent relationship between rad monitor counts and sample activity. However, as more data become available, this correlation will be further refined. With respect to all other rad monitors, the activity had been too low for meaningful analysis. With continued operation of the plant and a corresponding increase in effluent inventories, rad monitor readings will be correlated and verified against sample activities through the use of DCPP procedures.

5.12 Test Procedure No. 1.17 - Chemical and Radiochemical Analysis

TEST OBJECTIVE

The objective of this test was to document the ability to control water chemistry and perform reactor plant chemical and radiochemical analysis.

TEST DESCRIPTION

The results of the on-going DCPP Systems Sampling, Analytical and Chemistry Control Program using approved plant procedures were reviewed to verify chemical control was being maintained. Random samples were taken and analyzed during the Startup Power Ascension Test Program at various steady state power levels. The results were checked against the plant's ongoing program to verify the sampling and analytical procedures utilized in the plant manual. Effectiveness of selected plant filters and demineralizers was verified.

TEST RESULTS

A review of the performance of DCPP Chemistry and Radiation Protection Department analyses found them to be in accordance with the approved plant procedures. Samples taken and analyzed during power escalation were checked against the on-going program results and found to be comparable. Plant chemistry was being maintained within the limits established and specified in DCPP Operating Procedure F-5, or corrective action was taken to bring the system back within specifications.

The effectiveness of the plant filters and demineralizers was verified by the ability to maintain the water chemistry limits required in Operating Procedure F-5.

5.13 Surveillance Test Procedure R-26 - RCS Primary Coolant Flow Measurement

TEST OBJECTIVE

The objective of this test was to verify the calibration of RCS flow instruments at 30%, 50%, 75%, 90% and 100% of full power and to confirm that the total flow of all four loops is greater than the Technical Specification requirement of 366,000 gpm and that each RCS loop flow is at least 88,500 gpm.

TEST DESCRIPTION

Prior to obtaining data, the plant load was stabilized at a constant value and plant parameters were checked or adjusted to be within normal operating limits. Data was then obtained during a nominal 30 minute period with plant conditions stabilized and plant load constant.

Reactor coolant flow was determined by performing a heat balance on the RCS. This was done by using the gross steam generator thermal output calculated in the high accuracy heat balance test (STP R-2A) and narrow range hot-leg and cold-leg temperature measurements.

The heat balance across the secondary side of the steam generators (STP R-2A) produced an accurate determination of primary system heat rate. The heat rate results were then refined by compensating for RCS peripheral and convective heat loads to determine actual core heat generation. Actual RCS flow was then calculated.

TEST RESULTS

30% Power Test

Total RCS flow was measured as 382,000 gpm which is approximately 4% more than required by Technical Specifications. Lowest RCS loop flow was 93,000 gpm.

50% Power Test

Total RCS flow was measured as 387,000 gpm which is approximately 5.8% more than required by Technical Specifications. Lowest RCS loop flow was 95,000 gpm.

5.13 (Continued)

75% Power Test

Total RCS flow was measured as 378,000 gpm which is approximately 3.3% more than required by Technical Specifications. Lowest RCS loop flow was 93,000 gpm.

90% Power Test

Total RCS flow was measured at 383,000 gpm which is approximately 4.6% more than that required by Technical Specifications. Lowest RCS loop flow was 93,000 gpm.

100% Power Test

Total RCS flow was measured as 381,000 gpm which is approximately 4.2% more than required by Technical Specifications. Lowest RCS loop flow was 93,000 gpm.

At each test plateau no recalibration of the loop flow meters was required and no changes to the loop flow constants (specified in Surveillance Test Procedure I-1A and done each shift by the operators) were required.

5.14 Test Procedure No. 38.2 - Automatic Steam Generator Level Control

TEST OBJECTIVE

The objective of this test was to verify proper operation and stability of the Automatic Steam Generator Level Control System and Automatic Feedwater Pump Speed Controller.

TEST DESCRIPTION

This test was performed at a nominal reactor power of 30%. The programmed level setpoint signal was disconnected from the level controller and a constant test signal of equal magnitude substituted. The test signal was then raised and lowered with the controller in AUTOMATIC while system response was recorded. The steam flow signal input to the flow balancing controller was next substituted with a test signal of equal value. This test signal was then increased and decreased 5% with the controller in AUTOMATIC while system response was recorded. The controllers were restored to their operational configuration and integrated system response was checked by manually increasing Steam Generator level 5%, switching the controller to AUTOMATIC and monitoring system response. The entire procedure was completed on one Steam Generator Level Control System before proceeding to the next.

The Feedwater Pump Speed Controllers were tested by varying the master controller ±5% of feedpump operating speed with the control stations in AUTOMATIC. Feedwater pump speed response was monitored and the procedure was repeated for the second feedwater pump.

TEST RESULTS

The Automatic Steam Generator Level Control System test was successfully completed without any modifications to the initial Westinghouse settings.

The Feedwater Pump 2-1 speed controller worked satisfactorily without any adjustment to controller settings.

The Feedwater Pump 2-2 required adjustments to the zero point of H.P. Governor Valve by Westinghouse after the initial testing resulted in pump speed oscillations. After that adjustment, retest of Feedwater Pump 2-2 speed controller was satisfactory.

5.15 Test Procedure No. 38.1 - Automatic Reactor Control

TEST OBJECTIVES

The objective of this test was to verify the performance of the Automatic Reactor Control System in maintaining reactor coolant average temperature within acceptable steady state limits.

TEST DESCRIPTION

The Rod Control System was switched from manual to automatic control with the reactor at equilibrium conditions at 30% power and system response monitored.

With the Rod Control System in manual, Tavg was increased approximately 6 deg. F above Tref by withdrawing Control Bank D. The Rod Control System was then transferred from manual to automatic and plant response recorded.

After the plant stabilized, Tavg was decreased approximately 6 deg. F below Tref by insertion of Control Bank D with the Rod Control System in manual. The system was transferred from manual to automatic and plant response was recorded.

TEST RESULTS

The Automatic Reactor Control System responded properly to a ±6 deg. F temperature mismatch between Tavg and Tref. The Acceptance Criteria were met and the plant stabilized properly, within acceptable limits, after the Rod Control System automatically compensated for the temperature mismatch and brought Tavg back to Tref. Table 20 compares the actual data obtained with acceptable data limits.

Table 20

Automatic Reactor Control System Response

Initial Condition	Description	Acceptable Data Limits	Actual Data
Tavg > Tref	Maximum - Initial Pressurizer Pressure	≤65 psig	10.7 psig
	Initial - Minimum Pressurizer Pressure	≤65 psig	55.6 psig
	Peak-to-Peak Amplitude of Tavg Oscillation	≤5 deg. F	4 deg.F
	Minimum Period of Tavg Oscillation	≥60 sec.	288 sec.
	|Tref - Tavg| (After Transient)	±1.5 deg. F	0.375 deg.F
Tavg < Tref	Maximum - Initial Pressurizer Pressure	≤65 psig	6.2 psig
	Initial - Minimum Pressurizer Pressure	≤65 psig	6.3 psig
	Peak-to-Peak Amplitude of Tavg Oscillation	≤5 deg. F	4 deg. F
	Minimum Period of Tavg Oscillation	≥60 sec	208 sec.
	|Tref - Tavg| (After Transient)	±1.5 deg. F	1,1 deg.F

5.16 Test Procedure No. 42.1 - Power Coefficient Measurement

TEST OBJECTIVE

The objective of this test was to verify nuclear design predictions of the Doppler-only power coefficient.

TEST DESCRIPTION

After establishing stable plant conditions at the 30%, 50%, 75% and 90% test plateaus with equilibrium xenon and axial flux difference at or near its target value, the turbine load was decreased approximately 22 MWe at a rate of 2200 MWe/min. This action caused about 2% drop in reactor power level. Subsequent 44 MWe load swings were performed in order to vary reactor power by about 4% in each case, and a final load swing of 22 MWe returned power to its initial value. This sequence is shown on Figure 30. The period between each load swing was long enough to allow stabilization of Tave and ΔT. Boron concentration and control rod position were maintained constant throughout the test.

For each increase in turbine load, ΔT increases and Tave decreases. This increase in ΔT (i.e. fuel temperature) causes a negative reactivity effect due to the fuel's negative Doppler coefficient. This is offset by the positive reactivity due to the negative isothermal temperature coefficient (ITC) and the decreased Tave. In a similar manner, load decreases involve a decrease in ΔT and an associated increase in Tave.

The load swings done in this test directly measured the change in core average coolant temperature required to offset a change in ΔT. By relating the changes in ΔT to changes in reactor power level, ratios of Doppler coefficient to ITC were calculated by dividing the change in Tave by the change in power for each load swing. The acceptance criterion was that this measured ratio (Doppler coefficient/ITC) must be within 0.5 deg. F/% of the design value.

TEST RESULTS

As shown in Table 21, the measured ratios of Doppler coefficient to ITC for each test plateau were within the acceptance criterion.

Table 21

Power Coefficient Measurements

TEST PLATEAU (% RTP) (NOMINAL)	MEASURED RATIO* (deg.F/% power)	DESIGN RATIO* (deg.F/% power)	MEASURED - DESIGN RATIO * (deg. F/% power)	DESIGN DOPPLER COEFFICIENT (pcm/% power)
30	2.80	3.21	-0.41	-13.5
50	2.17	2.35	-0.18	-12.9
75	1.56	1.47	+0.09	-11.8
90	1.52	1.21	+0.31	-11.3

$$\text{*RATIO} = \frac{\text{DOPPLER COEFFICIENT}}{\text{ISOTHERMAL TEMPERATURE COEFFICIENT}}$$

Acceptance criterion: $|\text{measured ratio} - \text{design ratio}| \leq 0.5$ deg.F/% power

LOAD CYCLING PATTERN FOR POWER COEFFICIENT VERIFICATION

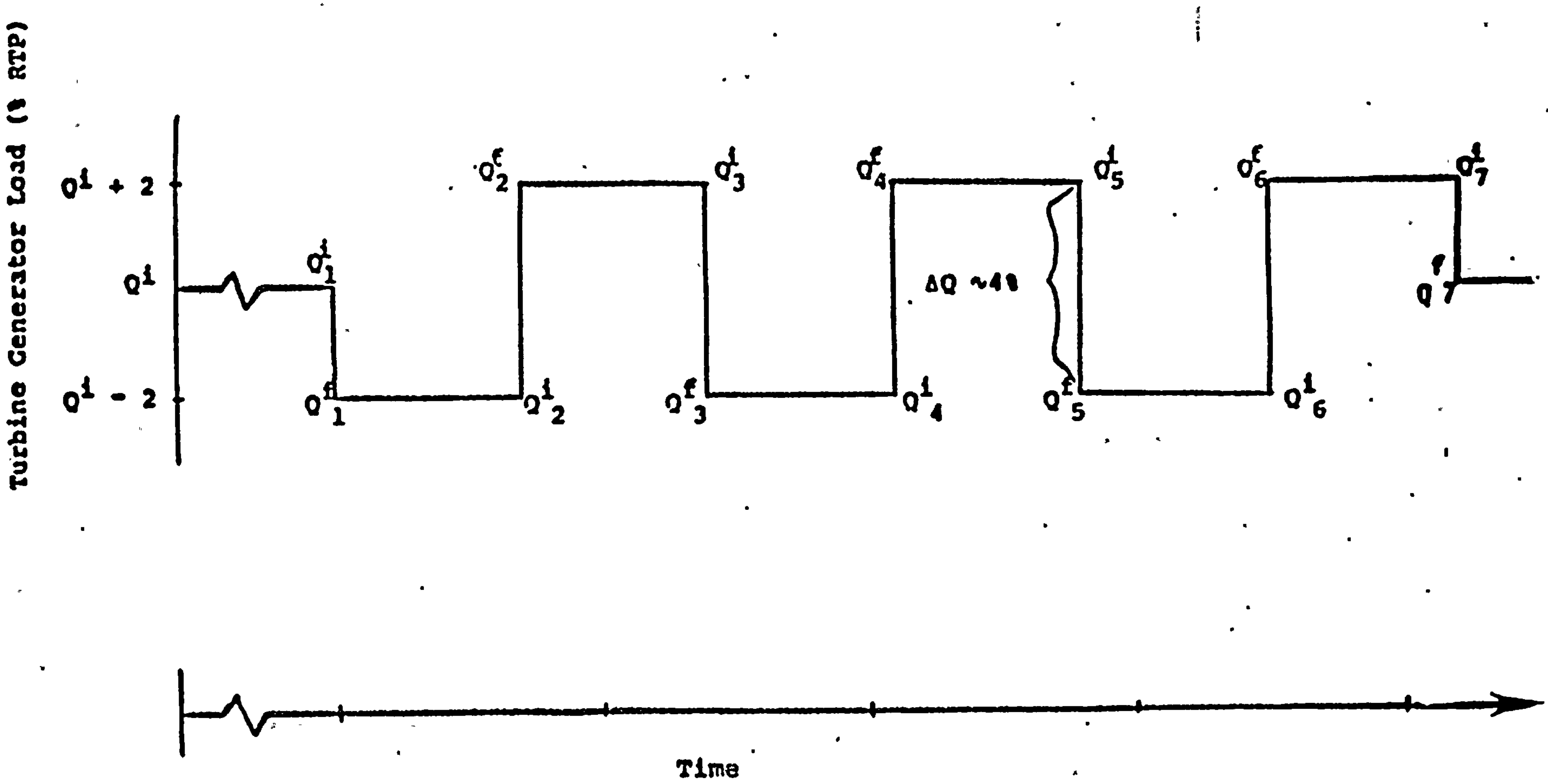


FIGURE 30

5.17 Test Procedure No. 42.2 - RCCA Pseudo Ejection and RCCA Above Bank Position Measurements

TEST OBJECTIVE

The objective of this test was to determine the power distribution and rod worth associated with an ejected RCCA.

TEST DESCRIPTION

The test was performed at the 30% power plateau with the plant stable and Control Bank D at the hot full power rod insertion limit of 188 steps.

After verifying these conditions, the movable incore detectors were used to perform a flux map to determine the "pre-ejection" power distribution. While maintaining constant turbine power and boron concentration, RCCA D-12 was withdrawn from 188 to 200 steps. At 200 steps a partial flux map (i.e., data from 6 of the 58 flux thimble locations) was taken, after which the rod was fully withdrawn. With RCCA D-12 withdrawn, a full core flux map was taken for the post-ejection power distribution. Finally, RCCA D-12 was returned to its initial position.

TEST RESULTS

Power distribution results are summarized in Table 22 and core average radial power distributions are shown in Figure 31. All Acceptance Criteria were met and no significant problems were encountered during the performance of this test.

The post-ejection value of F_Q^T (i.e., heat flux hot channel factor) was 2.275, well below the Acceptable Criteria (FSAR) limit of 7.07. Ejected rod worth was approximately 9 pcm, well below the Acceptance Criteria limit of 200 pcm.

Table 22

Power Distribution Results - Pre and Post Pseudo Rod Ejection

ITEM	PRE-EJECTED FLUX MAP	POST-EJECTED FLUX MAP
Conditions - power - temperature (RCS) - boron concentration - burnup	30% ~551 deg. F 1078 ppm 90 MWD/MTU	30% ~553 deg. F 1082 ppm 90 MWD/MTU
Rod Configuration	Bank D @ 188 steps	RCCA D-12 @ 228 steps
$F^N_{\Delta H}$ - measured value	1.360	1.501
- location*	J02-QQ	P09-QA
F^T_Q - measured value	2.116	2.275
- location*	M12-IH @ 58 in	P09-QA @ 60 in
F_Z - measured value	1.402	1.390
QUADRANT TILT - measured value	1.006	1.037

* Assembly locations (i.e., M12, etc.) are shown on Figure 1.
Pin locations within assembly (i.e., IH) are based on 17x17 matrix ranging from AA to QQ.

POWER DISTRIBUTIONS FOR PSEUDO EJECTION (T.P. 42.2)

PRE-EJECTION ASSEMBLY POWER
POST-EJECTION ASSEMBLY POWER

NOTE: a) Box entries are relative assembly average powers.

b) Ejected location: D-12

NORTH

	R	P	N	M	L	K	J	H	G	F	E	D	C	B	A
1					.591 .572	.695 .684	.828 .812	.755 .750	.822 .879	.680 .742	.575 .625				
2			.504 .534	.880 .922	1.039 1.024	1.033 .997	1.068 1.082	1.057 1.037	1.056 1.171	1.009 1.083	1.007 1.057	.853 .822	.494 .478		
3		.497 .507	1.038 1.036	.945 .992	1.126 1.091	1.125 1.128	1.176 1.121	1.128 1.083	1.167 1.091	1.092 1.232	1.089 1.060	.928 .933	1.031 .978	.500 .480	
4		.864 .853	.939 .963	1.240 1.185	1.079 1.040	1.199 1.117	1.144 1.117	1.165 1.110	1.143 1.122	1.184 1.120	1.055 1.055	1.242 1.193	.946 .982	.878 .834	
5	.575 .603	1.006 1.055	1.091 1.097	1.051 1.056	1.186 1.111	1.154 1.120	1.193 1.118	1.143 1.109	1.193 1.124	1.156 1.128	1.192 1.121	1.077 1.066	1.123 1.080	1.036 1.016	.589 .569
6	.674 .732	.996 1.068	1.076 1.072	1.162 1.118	1.139 1.121	1.161 1.096	1.045 1.023	1.119 1.059	1.033 1.022	1.148 1.097	1.135 1.116	1.192 1.144	1.119 1.111	1.032 .981	.696 .680
7	.806 .873	1.044 1.163	1.138 1.098	1.121 1.110	1.158 1.111	1.029 1.011	1.067 1.013	.972 .961	1.065 1.014	1.025 1.016	1.176 1.114	1.150 1.129	1.174 1.123	1.068 1.057	.828 .808
8	.728 .806	1.033 1.115	1.092 1.091	1.139 1.097	1.099 1.096	1.097 1.041	.963 .955	.995 .955	.965 .961	1.105 1.059	1.128 1.109	1.166 1.118	1.128 1.116	1.053 1.009	.745 .744
9	.798 .879	1.041 1.172	1.136 1.086	1.124 1.106	1.144 1.106	1.008 1.012	1.040 1.006	.955 .962	1.075 1.031	1.044 1.042	1.187 1.138	1.151 1.142	1.174 1.131	1.060 1.071	.817 .819
10	.675 .704	1.001 1.028	1.064 1.064	1.164 1.112	1.123 1.120	1.128 1.094	1.009 1.018	1.090 1.060	1.029 1.041	1.156 1.115	1.145 1.143	1.185 1.147	1.114 1.107	1.023 1.056	.688 .722
11	.582 .593	1.024 1.063	1.110 1.134	1.050 1.038	1.163 1.115	1.127 1.122	1.157 1.121	1.098 1.112	1.148 1.141	1.125 1.124	1.162 1.147	1.053 1.089	1.096 1.086	1.035 1.104	.588 .613
12		.867 .875	.938 .989	1.241 1.231	1.067 1.056	1.185 1.138	1.126 1.129	1.128 1.113	1.115 1.148	1.163 1.128	1.046 1.083	1.220 1.257	.934 1.049	.877 .957	
13		.494 .518	1.031 1.060	.943 .990	1.113 1.057	1.100 1.106	1.138 1.138	1.083 1.160	1.146 1.184	1.106 1.174	1.107 1.141	.935 1.014	1.032 1.102	.499 .578	
14			.494 .522	.869 .877	1.027 1.005	1.014 .987	1.036 1.130	1.028 1.092	1.055 1.157	1.042 1.032	1.036 1.092	.875 .923	.500 .558		
15					.583 .562	.678 .683	.803 .854	.733 .795	.823 .871	.701 .706	.589 .606				

DIABLO CANYON POWER PLANT - UNIT 2

FIGURE 31

5.18 Surveillance Test Procedure No. R-13 - Incore-Excore Detector Calibration

TEST OBJECTIVE

The objective of this test was to determine the relationship between the axial power distribution in the core (as established by use of movable incore flux detectors) and power range excore upper/lower detector signals. The scaling factors that were calculated were used to calibrate the excore nuclear instruments. The test was performed initially at the 50% power plateau and repeated at 75% power.

TEST DESCRIPTION

Each of the four (4) power range nuclear instrumentation channels consists of a pair of uncompensated ion chambers stacked vertically. Each detector in a given channel is located symmetrically above and below the core axial midplane. The calibration test was performed to provide the data necessary to calibrate the pairs of detectors and provide upper/lower signals that are proportional to the power split between the upper/lower halves of the core over a wide range of axial power distributions.

To provide such data, the control rod position and soluble boron content of the RCS were varied initially in such a way to provide power distributions ~~axially~~ skewed toward the bottom of the core. Flux maps (initially full core, quarter-core thereafter) were recorded using the movable incore detector system to determine the amount of asymmetry in the axial power distribution (expressed as axial flux difference, AFD*, or axial offset, AO**). The excore detector signals also were measured during the flux maps so that a direct comparison could be made of incore detector vs. excore detector AO. Control rods then were returned to their initial position and the asymmetric buildup of xenon in the core was allowed to produce a xenon-induced axial power oscillation, shifting power toward the top of the core. Periodically, flux maps and excore signals were recorded. At a prescribed point in the xenon/power oscillation (end of the test), a control rod maneuver was performed to dampen out the axial oscillation and return core conditions to normal.

The data from the test were used to plot incore AO vs. excore AO for each power range excore channel and also incore AO vs. normalized (full power) detector currents for each power range excore upper and lower detector. These plots provided best-fit straight lines from which excore detector gains (slopes) and offsets (intercepts) were obtained. A subsequent I&C surveillance test procedure STP I-2D used these gains and offsets to calibrate the power range excore nuclear instruments.

*AFD (%) = AO (%) x % core power / 100
**AO (%) = (Upper detector current or core power - Lower detector current or core power) x 100/ (Upper + Lower)

5.18 (Continued)

TEST RESULTS

(A) 50% Plateau:

Control Bank D rods were inserted in two (2) increments of approximately seventeen (17) steps each. Incore AO shifted from about -3.6% initially (Bank D at 207 steps) to about -10% (189 steps) and then to about -19.8% (173 steps). Upon holding the latter configuration for approximately two hours, rods were returned to their original position. During the subsequent axial xenon oscillation, four more quarter-core flux maps were produced at incore AOs ranging from about -6.9% to +4.9%. Thus, the total span of minimum to maximum AO was from about -19.8% to about +4.9%

The entire test required approximately 22 hours to complete.

Slopes (gains) of the incore vs. excore AO plots were approximately 1.6 for all channels. Offsets constants ranged from about +1.5% for channel N44 to +6.1% for channel N43.

A full calibration was performed on all 4 power range channels per STP I-2D using these gains and offset constants.

(B) 75% Plateau:

Control Bank D rods were inserted in 2 increments of approximately 10 steps each. Incore AO shifted from about -9.1% initially (Bank D at 186 steps) to about -14.9% (175 steps) and then to about -21.6% (166 steps). Upon holding the latter configuration for approximately 2 hours, rods were returned to their original position. During the subsequent axial xenon oscillation, four more quarter-core flux maps were produced at incore AOs ranging from about -13.5% to -4.1%. Thus, the total span of minimum to maximum AO was from about -21.6 to about -4.1%. The entire test required approximately 18 hours to complete.

Slopes (gains) of the incore vs. excore AO plots were approximately 1.6 for all channels. Offset constants ranged from about +1.5% for channel N44 to +6.1% for channel N43. A sample of each type of plot is enclosed for channel N41 (Figures 32 and 33).

Because of agreement with results of the test at 50% power level, recalibration of the power range channels was deemed not to be required. This test confirmed the adequacy of performing Incore/Excore calibrations at 50% power.

STP R-13 SAMPLE PLOT: N-41 INCORE vs. EXCORE AXIAL OFFSET
75% RTP, STP R-13, 11-18-85

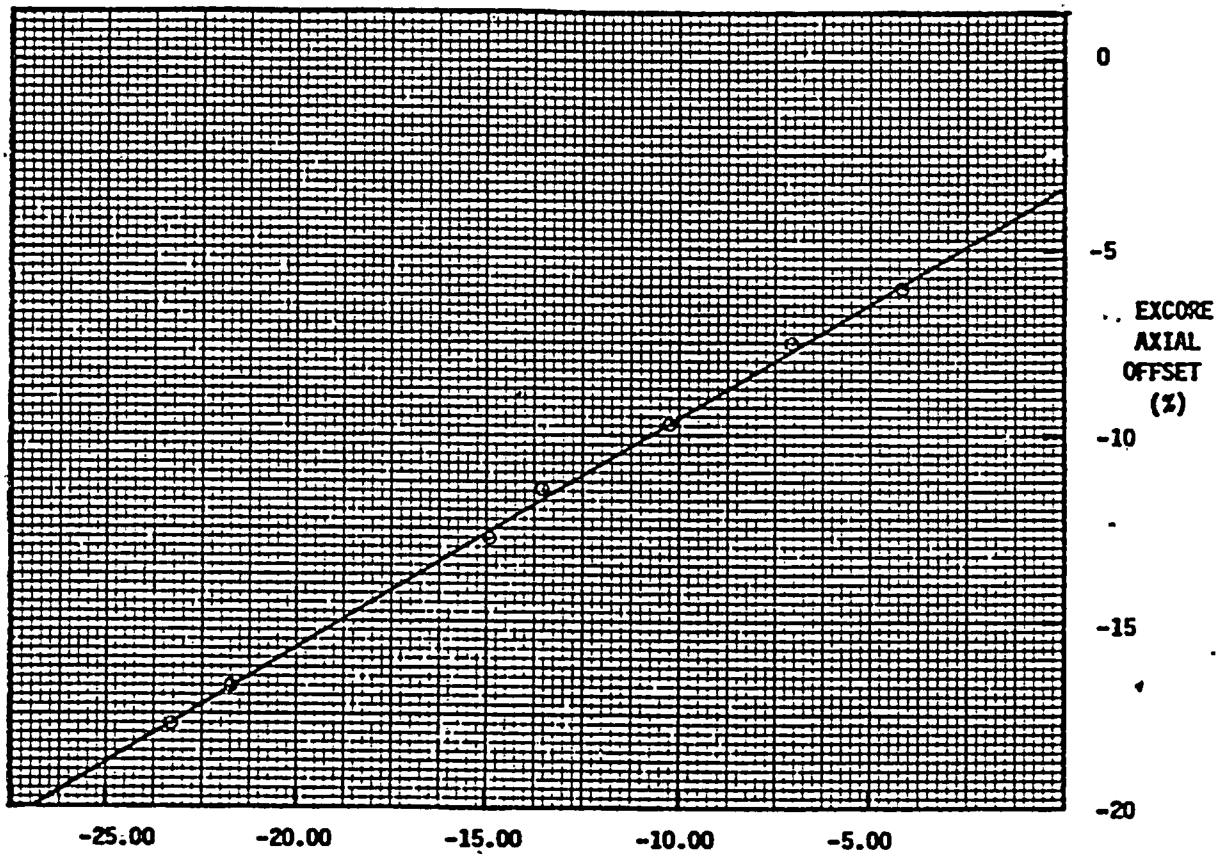


FIGURE 32

DCC No. 33780

STP R-13 SAMPLE PLOT: N-41 INCORE AXIAL OFFSET vs. NORMALIZED DETECTOR CURRENT
75% RTP, STP R-13, 11-18-85

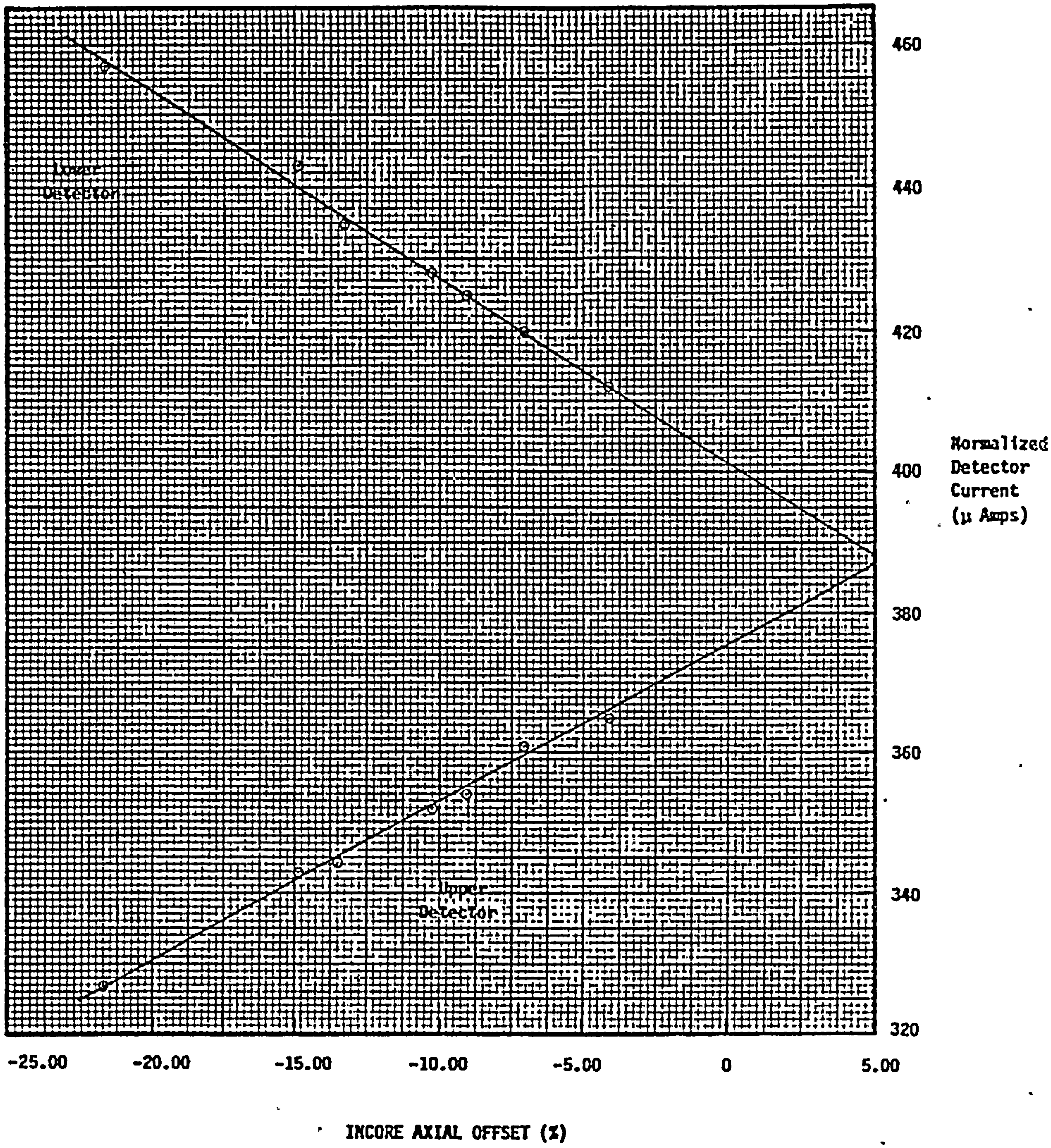


FIGURE 33

5.19 Test Procedure No. 43.7 - Net Load Trip From 50% Power

TEST OBJECTIVE

The objective of this test was to demonstrate the ability of the unit to sustain a net load rejection from nominal 50% power.

TEST DESCRIPTION

The plant was stable at nominal 50% RTP and on automatic control. The load rejection was initiated by opening the main transformer high side breakers. The plant was then stabilized using DCPP Emergency Operating Procedure OP AP-2, Full Load Rejection.

The results were evaluated for acceptable dynamic response. Also, the interaction between the control systems was studied for possible setpoint changes to improve transient response.

TEST RESULTS

The following is a listing of the sequence of events explaining the problems experienced during this test and their resolutions:

November 6, 1985: With the plant stable at 50% power and in automatic mode, 50% load rejection transient was initiated at 1915 hours. Approximately 60 seconds into the transient, control room operators took manual control of the Main Feedwater Pumps Master Controller in an attempt to restore steam generator levels. Approximately 10 seconds later, the reactor tripped on Steam Generator 2-4 low level. Analysis of the transient data suggested that the trip was caused by the slow response time of the Main Turbine intercept valves. After obtaining concurrence from Westinghouse, an orifice changeout was implemented on all Main Turbine intercept valves to improve their response time.

November 13, 1985: During the second attempt, approximately 90 seconds after initiation of the transient, manual control of Main Feedwater Pump 2-1 was taken by the control room operators. An additional 90 seconds later, control rods were switched to manual mode and steam dump control system was switched from Tavg mode to pressure mode. Transient data analysis revealed that after manual control of the Main Feedwater Pump 2-1 was taken, feedwater header pressure increased to greater than 1500 psi, but the feedwater pump did not trip. Investigation revealed that 2 out of 3 feedwater high pressure trip switches were isolated. This event was analyzed by the plant staff and it was determined that this anomaly did not impact the overall response of the plant to the transient.

5.19 (Continued)

December 8, 1985: Due to inadequate control system response observed during the first two attempts at 50% load reduction from 75% power level per Test Procedure 43.3 (described in Section 5.24), a design was implemented to modify the steam dump program. Since this design change modified key control system parameters tested previously on November 13, 1985, it was decided to retest the ability of the unit to sustain net load rejection from 50% power. During this retest, all Acceptance Criteria listed below were met successfully:

1. Reactor and Turbine did not trip.

2. Safety Injection was not initiated.

3. Main Steam safety valves did not lift.

4. Pressurizer safety valves did not lift.

5. Minimal manual intervention until nuclear power level decreased to approximately 20%.

Operators took manual control of the control rods at approximately 2 minutes and 17 seconds after the transient was initiated. Main feedwater pumps were put in manual mode an additional 44 seconds later. Also, the plant response was within expectations except for steam generator 2-1 level undershoot which was 24.7% versus an expected value of $\leq$15.0% and control rod maximum speed time which was 42 seconds versus the expected value of 30 seconds. Westinghouse and PG&E Engineering considered these deviations to be acceptable. The response of key plant parameters are tabulated in Table 23 and illustrated in Figures 34A through 34E.

Table 23

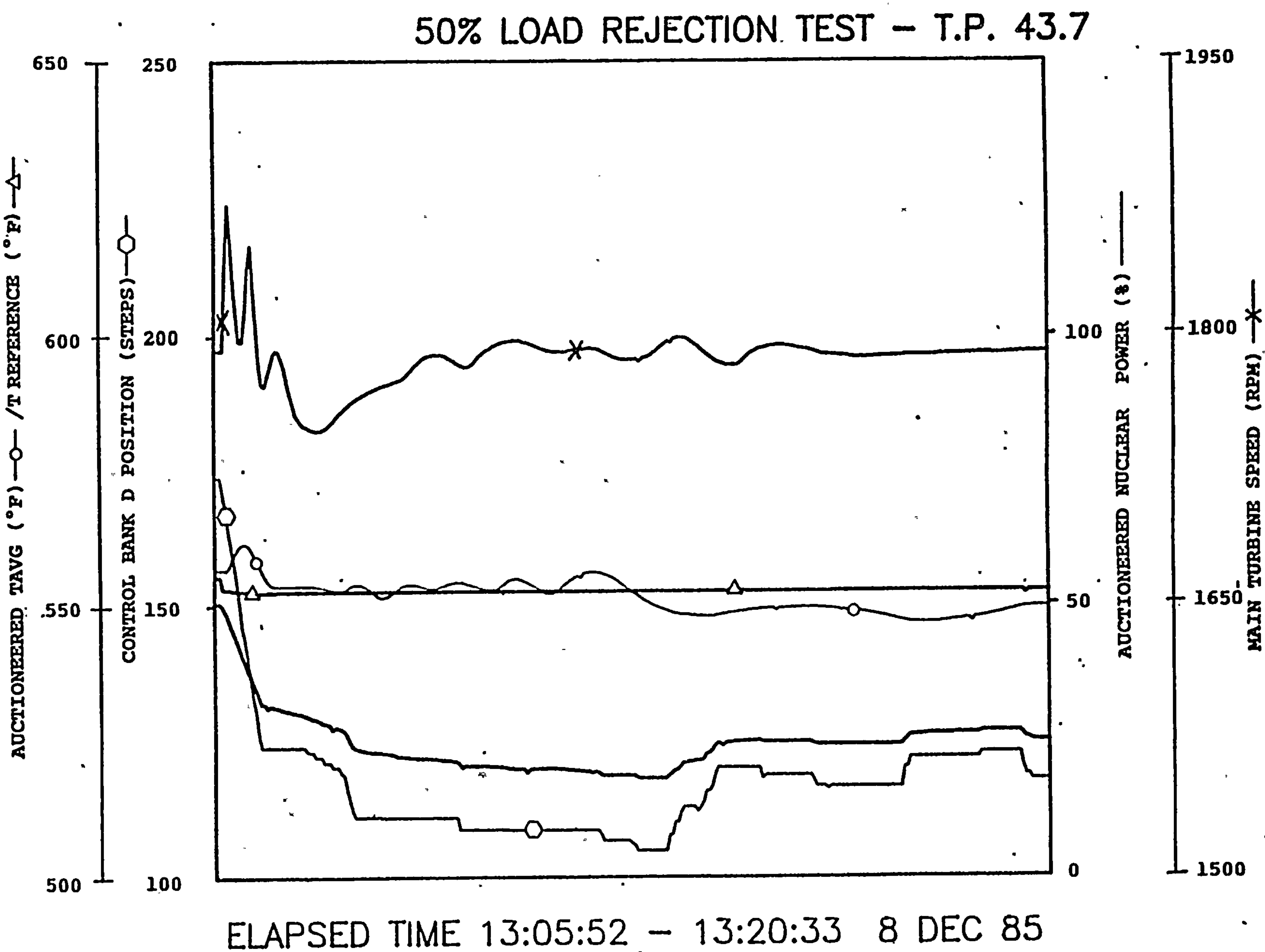
Key Plant Parameters During Net Load Trip From 50% Power

Tave:	
Initial Tave (deg. F)	557.8
Peak Tave (deg. F)	562.7
Final Tave (deg. F)	551.2
Tave Oscillations (deg. F)	≤ 2
Pressurizer Pressure:	
Initial Pressure (psig)	2229.6
Maximum Pressure (psig)	2278.8
Minimum Pressure (psig)	2163
Maximum - Initial (psid)	49.2
Initial - Minimum (psid)	66.6
Steam Generator Level:	
Initial S/G 2-1 Level (%)	42.9
Maximum S/G 2-1 Level (%)	56.5
Minimum S/G 2-1 Level (%)	18.2
Maximum - Initial (%)	13.6
Initial - Minimum (%)	24.7 *
Control Rod Speed:	
Time of Maximum Speed (sec)	42 *
Steam Dumps:	
Actuation and Modulation	yes
Cycling	no

Expected Responses:

1. The Tave peak should be less than 5 deg. F above the initial value, while Tave oscillations should be ≤ 2 deg. F peak to valley.
2. Pressurizer pressure should not vary more than +100 psi and -150 psi from the initial pressure.
3. Steam Generator levels should not vary more than $\pm 15\%$ from the initial value.
4. Maximum control rod speed should exist for approximately 30 seconds.
5. Steam dumps should actuate and modulate and shut off with no cycling.

* As described in Section 5.19, these deviations from the expected response were judged to be acceptable.



50% LOAD REJECTION TEST - T.P. 43.7
650
600
550
500
250
200
150
100
AUCTIONEERED TAVG (°F) —o— /T REFERENCE (°F) —△—
CONTROL BANK D POSITION (STEPS) —⬡—
100
50
0
AUCTIONEERED NUCLEAR POWER (%) ——
1950
1800
1650
1500
MAIN TURBINE SPEED (RPM) —✕—
ELAPSED TIME 13:05:52 - 13:20:33 8 DEC 85

FIGURE 34A

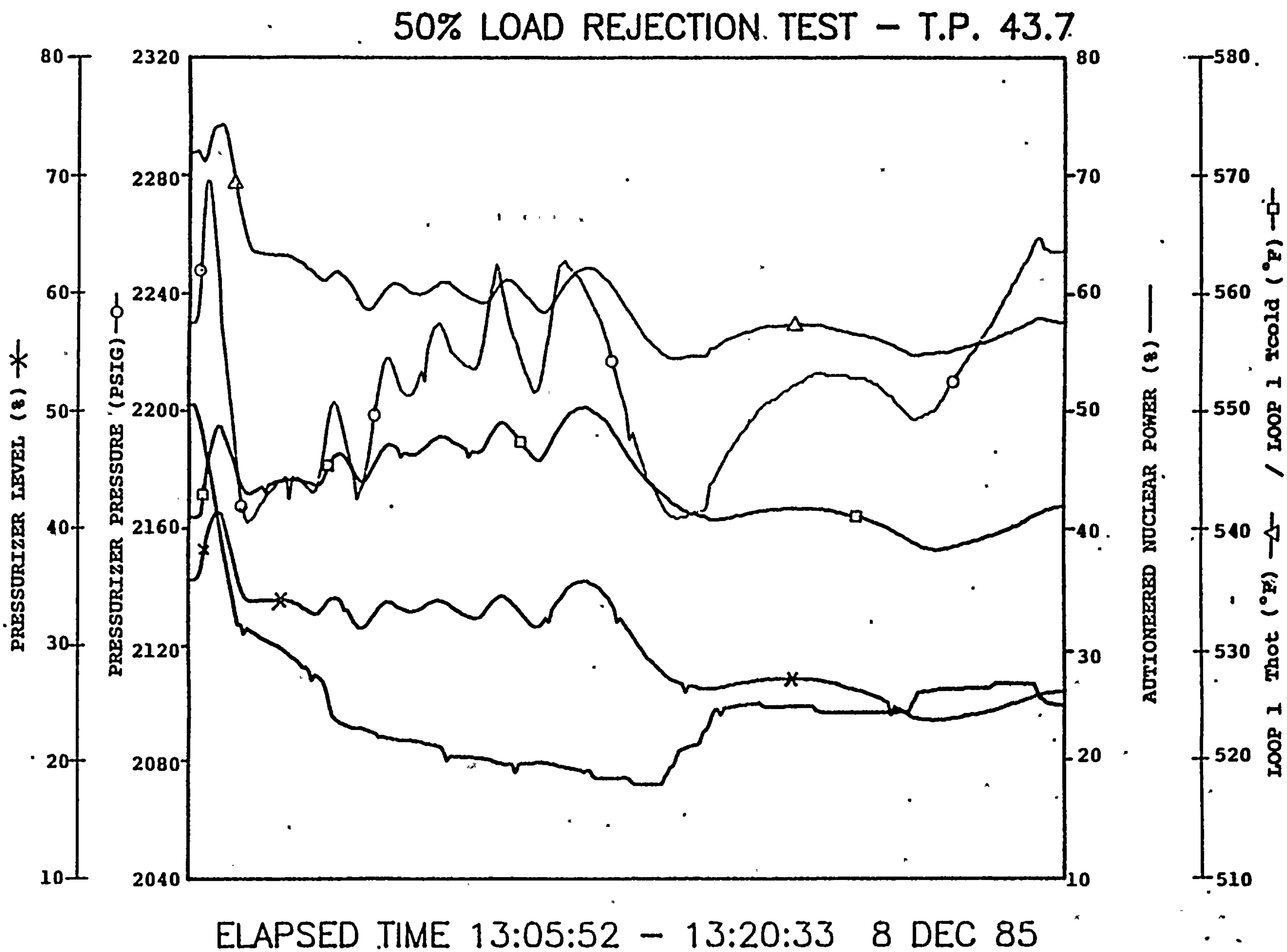

50% LOAD REJECTION TEST - T.P. 43.7
PRESSURIZER LEVEL (%)
80
70
60
50
40
30
20
10
PRESSURIZER PRESSURE (PSIG)
2320
2280
2240
2200
2160
2120
2080
2040
AUTIONEERED NUCLEAR POWER (%)
80
70
60
50
40
30
20
10
LOOP 1 Thot (°F) / LOOP 1 Tcold (°F)
580
570
560
550
540
530
520
510
ELAPSED TIME 13:05:52 - 13:20:33 8 DEC 85

FIGURE 34B

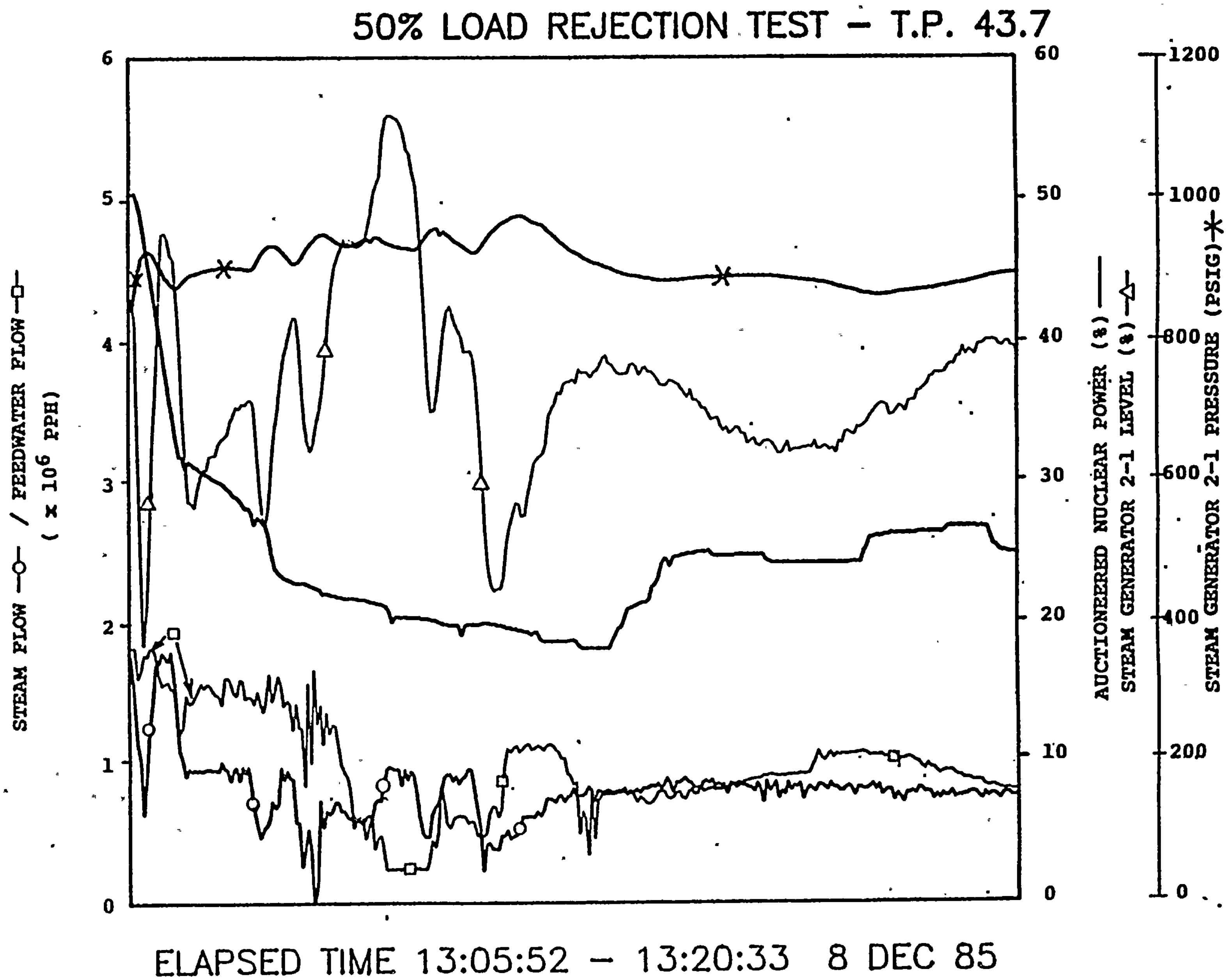
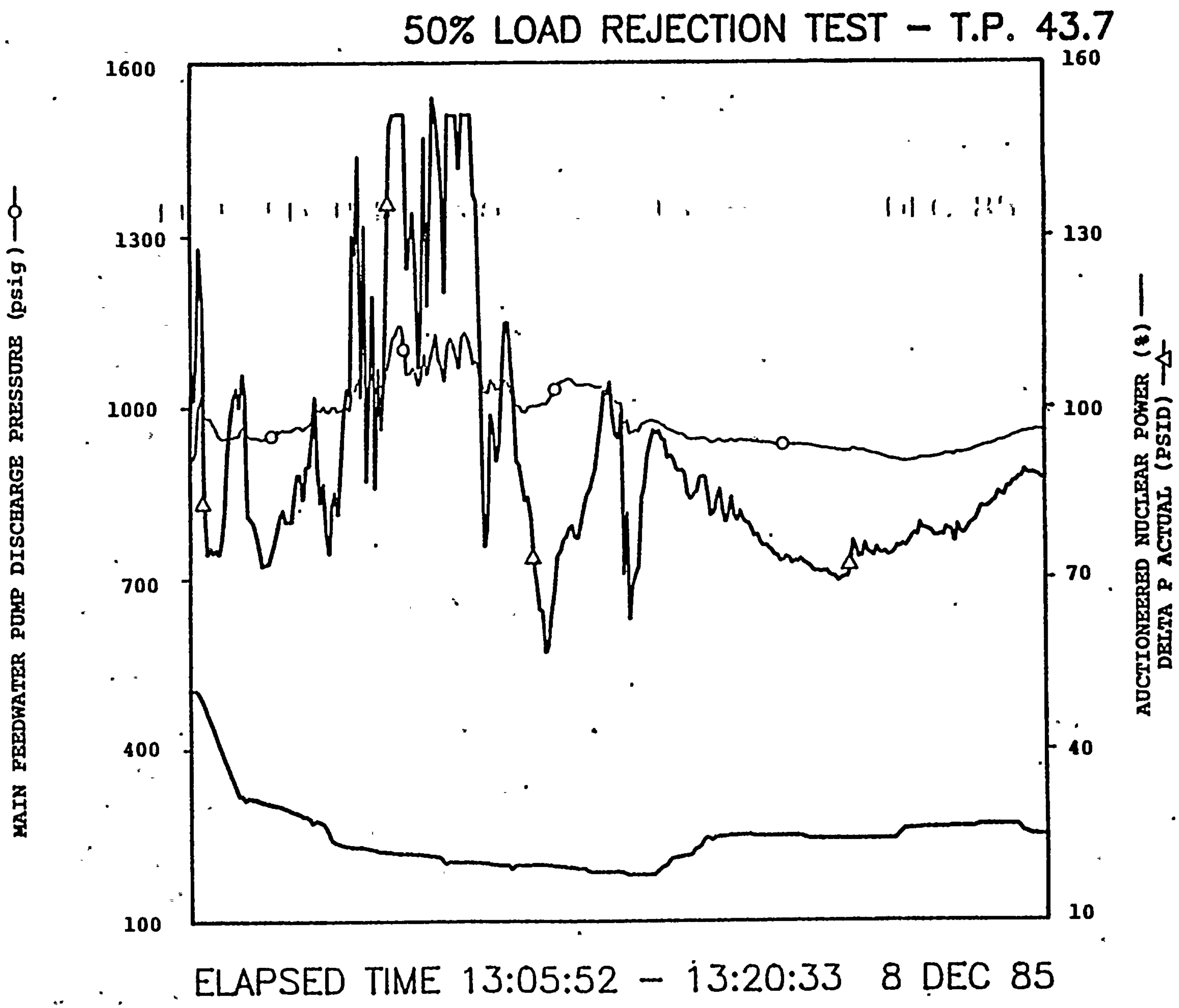
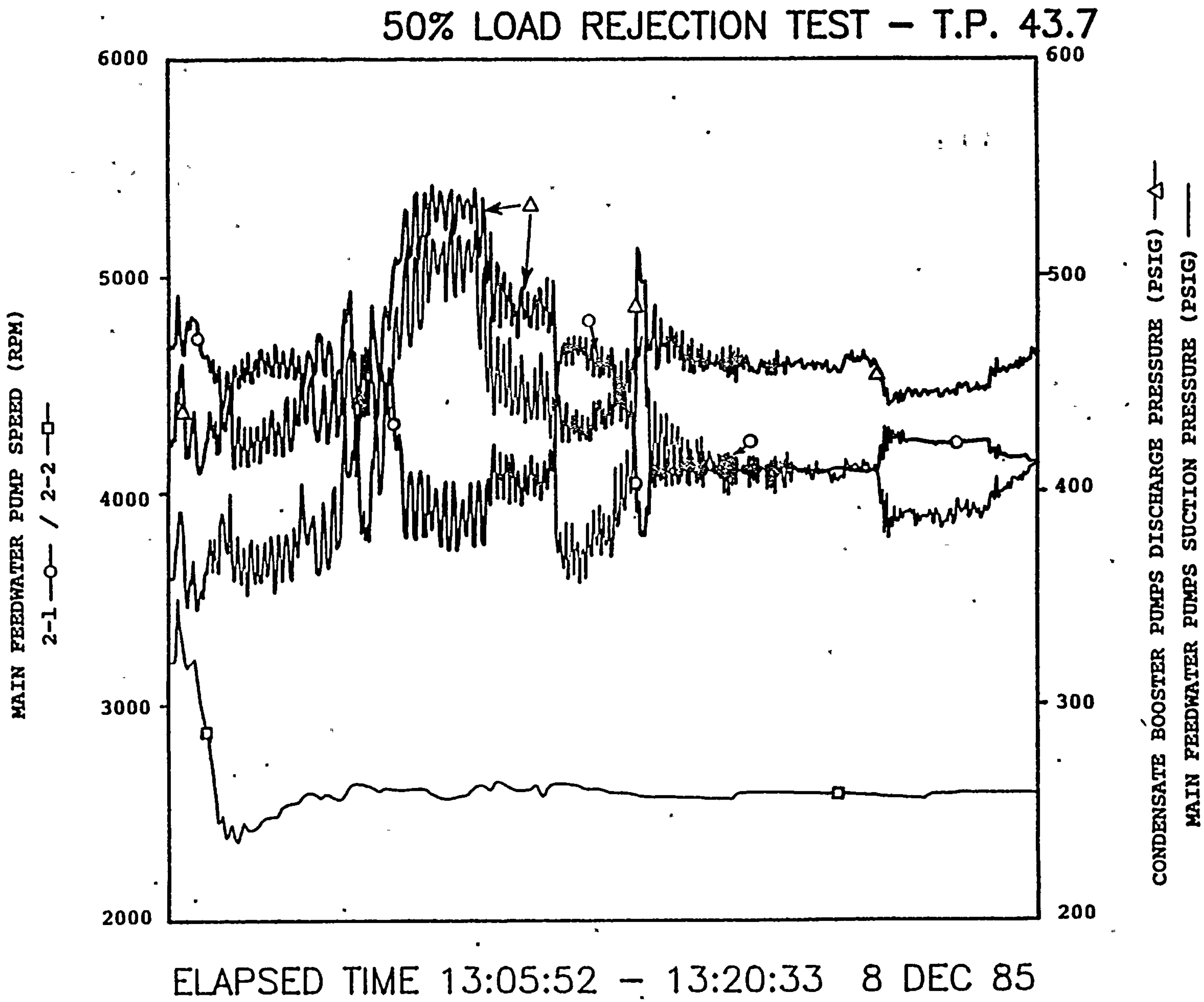

50% LOAD REJECTION TEST - T.P. 43.7
STEAM FLOW —○— / FEEDWATER FLOW —□—
(x 10^6 PPH)
AUCTIONEERED NUCLEAR POWER (%) ——
STEAM GENERATOR 2-1 LEVEL (%) —△—
STEAM GENERATOR 2-1 PRESSURE (PSIG) —✳
0
1
2
3
4
5
6
10
20
30
40
50
60
200
400
600
800
1000
1200
ELAPSED TIME 13:05:52 - 13:20:33 8 DEC 85

FIGURE 34C



50% LOAD REJECTION TEST - T.P. 43.7
MAIN FEEDWATER PUMP DISCHARGE PRESSURE (psig)
AUCTIONEERED NUCLEAR POWER (%)
DELTA P ACTUAL (PSID)
1600
1300
1000
700
400
100
160
130
100
70
40
10
ELAPSED TIME 13:05:52 - 13:20:33 8 DEC 85

FIGURE 34D



50% LOAD REJECTION TEST – T.P. 43.7
MAIN FEEDWATER PUMP SPEED (RPM)
2-1 / 2-2
CONDENSATE BOOSTER PUMPS DISCHARGE PRESSURE (PSIG)
MAIN FEEDWATER PUMPS SUCTION PRESSURE (PSIG)
6000
5000
4000
3000
2000
600
500
400
300
200
ELAPSED TIME 13:05:52 – 13:20:33 8 DEC 85

FIGURE 34E

5.20 Test Procedure No. 43.5 - Rod Group Drop and Plant Trip

TEST OBJECTIVE

The objectives of this test was to demonstrate the ability of the Excore Detector System Negative Rate Circuitry to detect a two rod drop and subsequently cause a reactor trip and to review plant response and control system behavior to the resulting plant trip.

TEST DESCRIPTION

With the plant stable at a nominal power level of 50% and on automatic control, two rods (L-13 and E-3) were simultaneously dropped. The rod motion caused an Excore Detector System negative flux-rate trip which tripped the Reactor and the Turbine. The plant response was monitored during the transient.

TEST RESULTS

The two rods dropping simultaneously caused a negative rate trip which caused a Reactor trip and a Turbine trip. The acceptance criteria for the test was met as the transient did not cause (i) a safety injection, (ii) reactor coolant pump tripping (iii) steam line safety valve lifting or (iv) pressurizer safety valve lifting. The plant responded as expected to the plant trip with the exception of pressurzer level and Tavg both of which went below expected values due to the Auxiliary Steam demand. Westinghouse and PG&E Engineering judged these deviations to be acceptable. A summary of selected parameter response to the transient is shown in Table 24 and illustrated in Figures 35A through 35E.

Table 24

Rod Group Drop and Plant Trip

Parameter	Units	Initial	TRANSIENT Max.	TRANSIENT Min.	Final
Reactor Power	%	51	-	-	0
Electrical Output (Gross)	MW	490	-	-	0
Tref	deg. F	557	-	-	547
* Tavg	deg. F	557.2	-	-	533.5
Pressurizer Pressure	psig	2241	2241	2099	2148
* Pressurizer Level	%	36.4	36.4	18	21
Steam Header Pressure	psig	835	920	<600	<600
(Loop1) Steam Generator Level	%	45.4	45.4	1	26
Core Exit Thermocouple (F5)	deg. F	578	-	-	530

* Pressurizer level and Tavg did not remain above the expected minimum values of 20% and 547 deg. F respectively. This was caused by the auxiliary feed-water system operating at maximum flow rates until steam generator levels returned to their setpoint value. This quantity of cold water being injected into the steam generators tended to lower Tave initially, thereby dropping pressurizer level excessively. Pressurizer level dropped with Tavg and went below 20% about 2 min. 50 sec. after trip: Tavg dropped below 547 deg. F about 19 secs. after trip and settled at around 533 deg. F.

Feedwater isolation occurred about 8 secs. after trip.

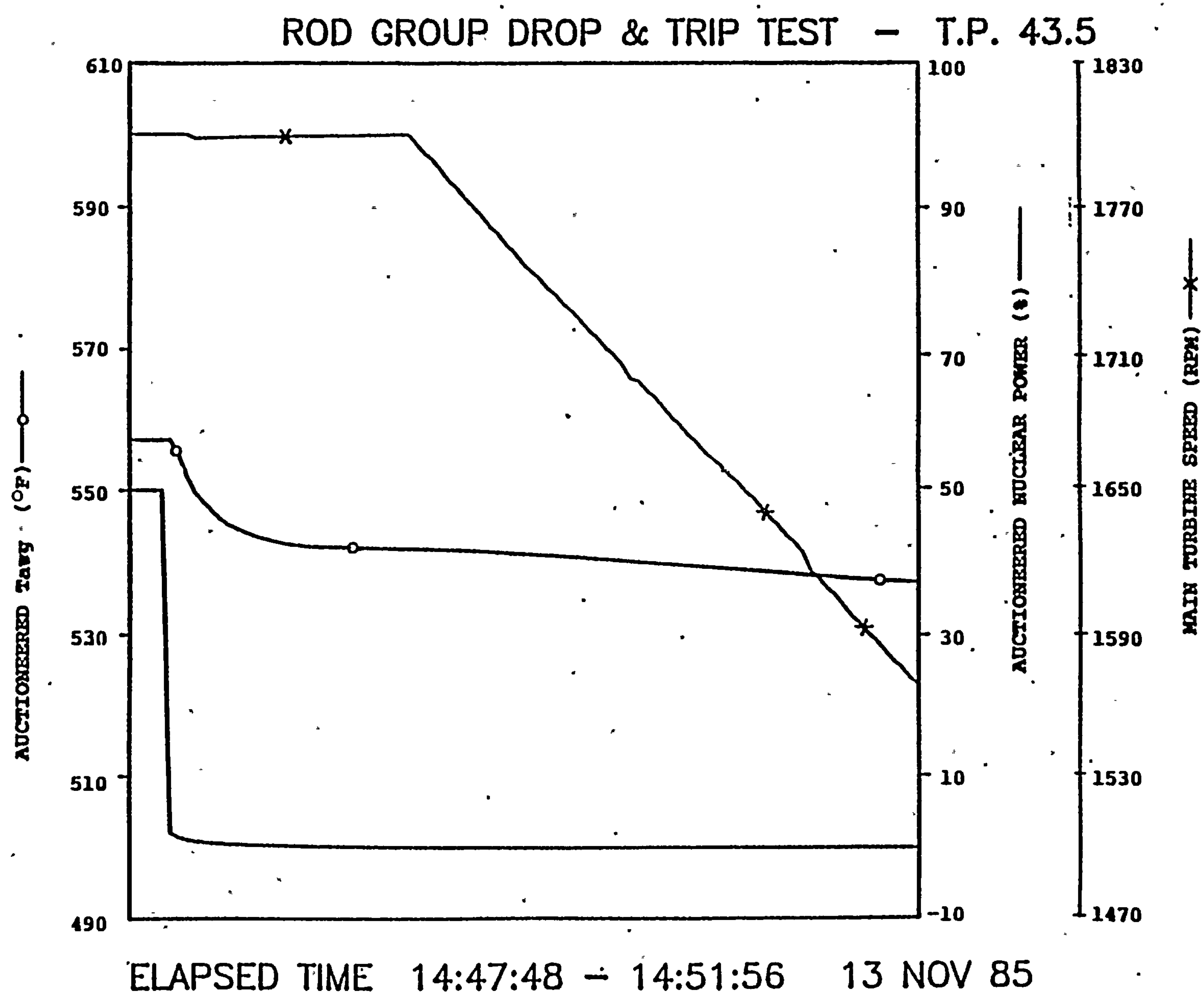

ROD GROUP DROP & TRIP TEST - T.P. 43.5
610
590
570
550
530
510
490
100
90
70
50
30
10
-10
1830
1770
1710
1650
1590
1530
1470
AUCTIONEERED Tavg (°F)
AUCTIONEERED NUCLEAR POWER (%)
MAIN TURBINE SPEED (RPM)
ELAPSED TIME 14:47:48 - 14:51:56 13 NOV 85

FIGURE 35A

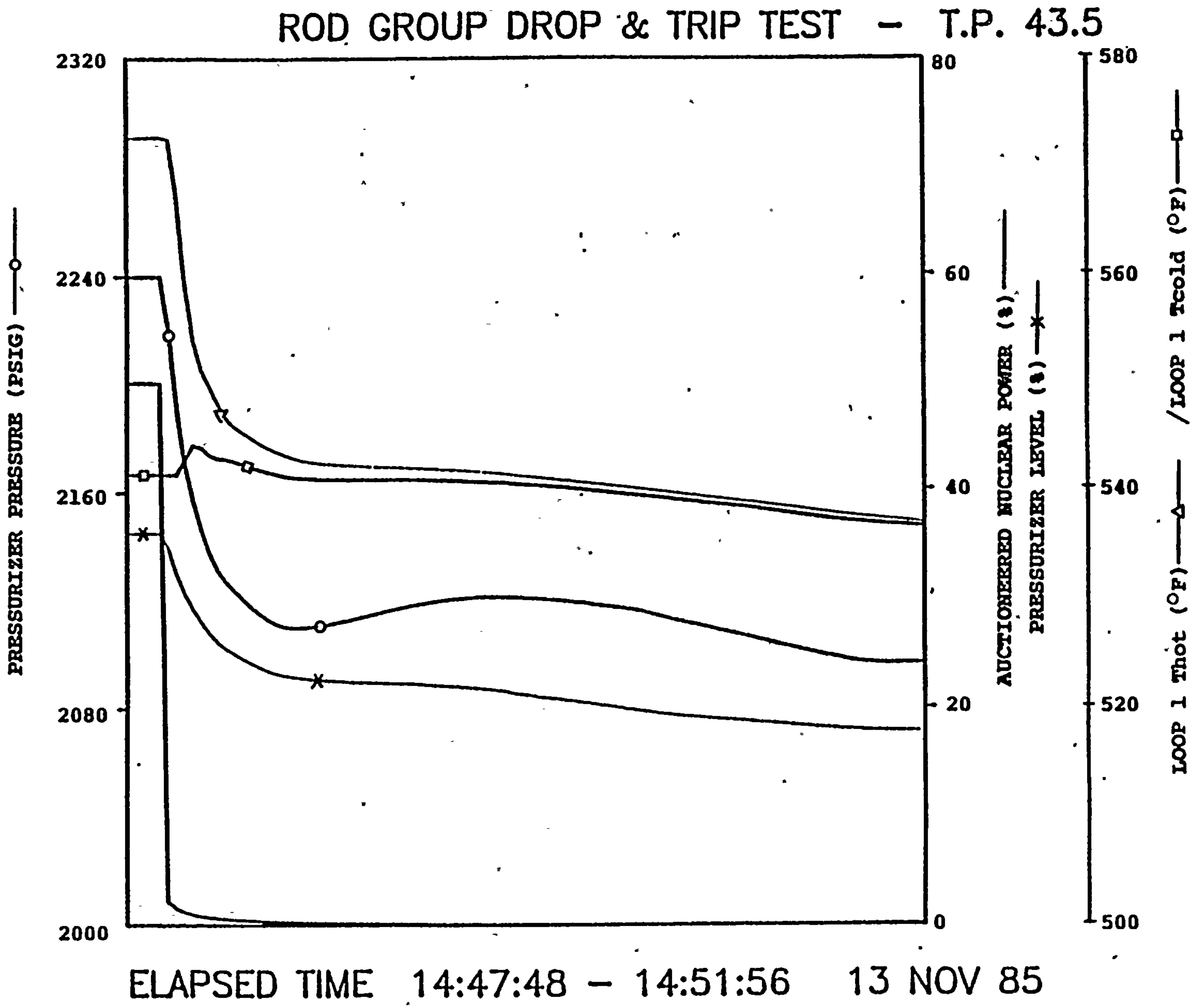

ROD GROUP DROP & TRIP TEST - T.P. 43.5
2320
2240
2160
2080
2000
80
60
40
20
0
580
560
540
520
500
PRESSURIZER PRESSURE (PSIG)
AUCTIONEERED NUCLEAR POWER (%)
PRESSURIZER LEVEL (%)
LOOP 1 Thot (°F) / LOOP 1 Tcold (°F)
ELAPSED TIME 14:47:48 - 14:51:56 13 NOV 85

FIGURE 35B

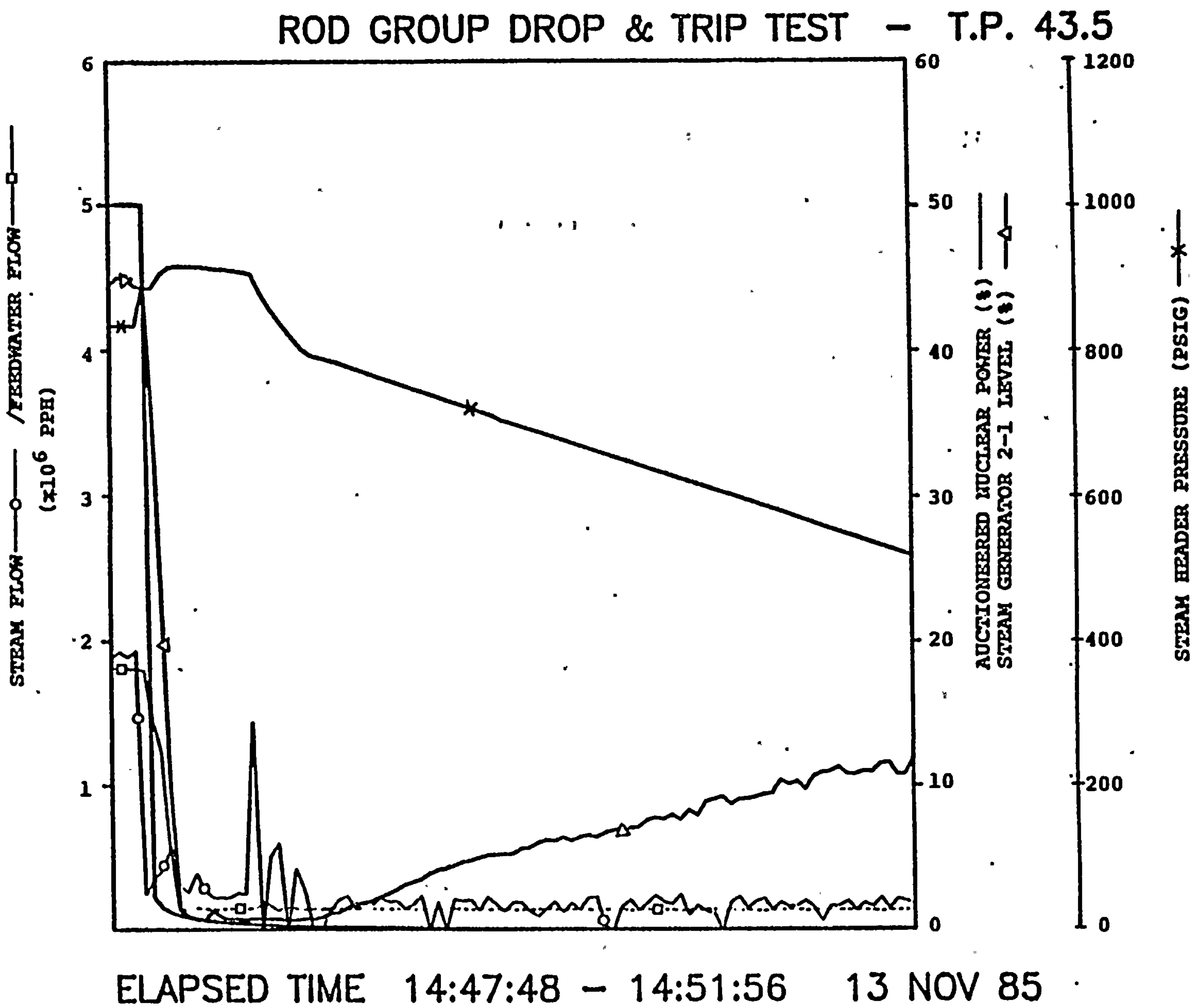
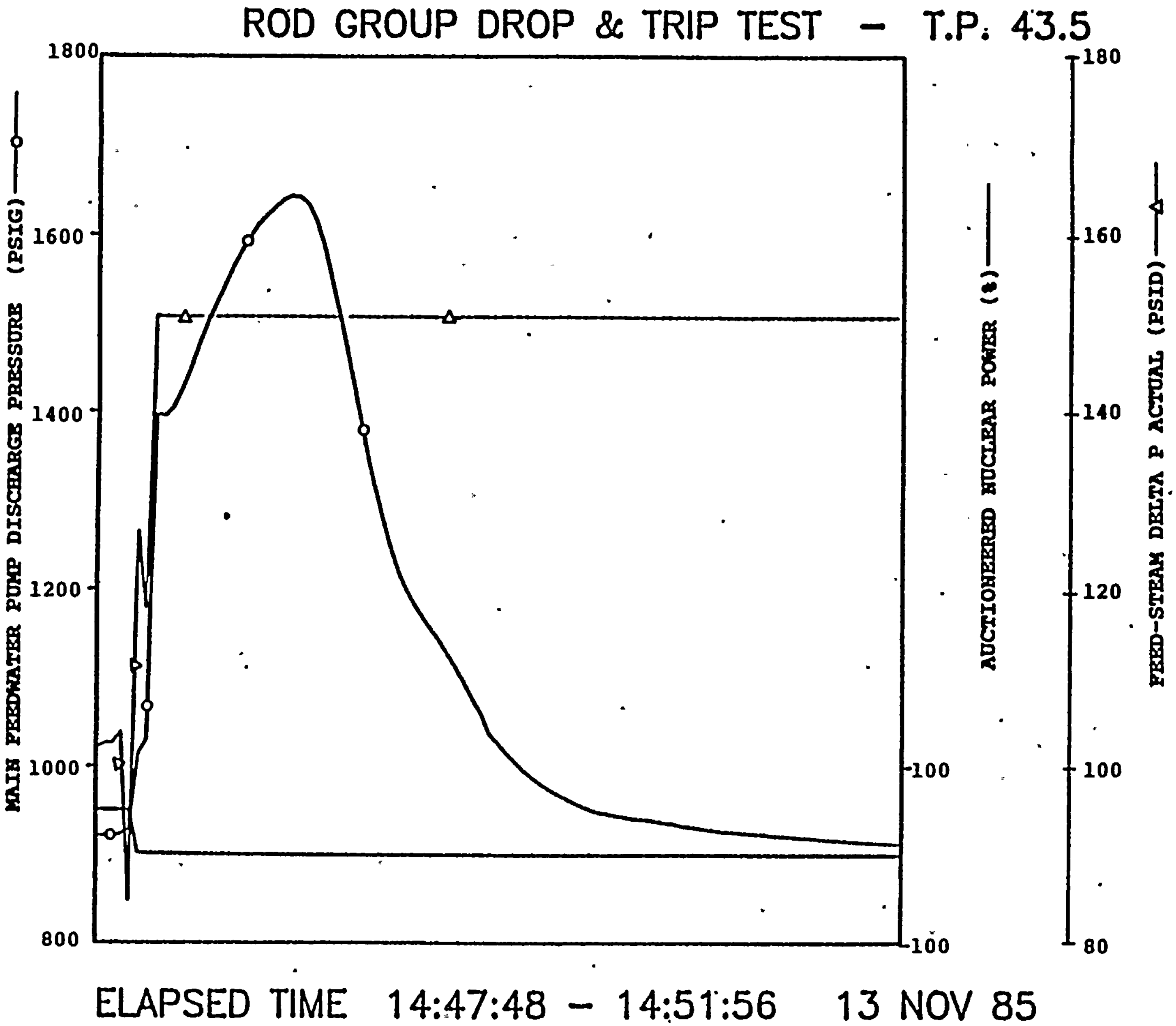
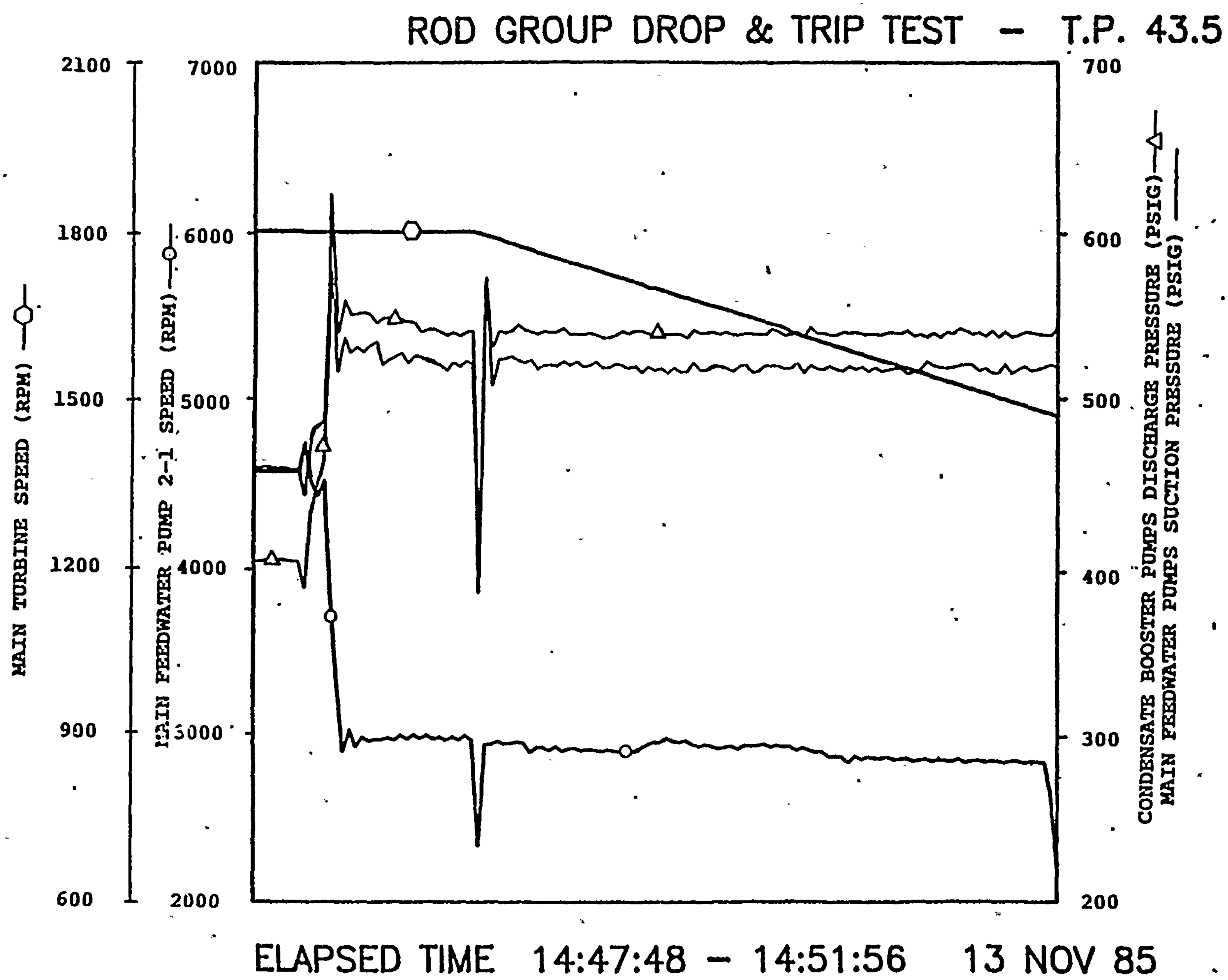

ROD GROUP DROP & TRIP TEST - T.P. 43.5
STEAM FLOW —o— /FEEDWATER FLOW —□—
(x10^6 PPH)
AUCTIONEERED NUCLEAR POWER (%) ——
STEAM GENERATOR 2-1 LEVEL (%) —△—
STEAM HEADER PRESSURE (PSIG) —*—
6
5
4
3
2
1
60
50
40
30
20
10
0
1200
1000
800
600
400
200
0
ELAPSED TIME 14:47:48 - 14:51:56 13 NOV 85

FIGURE 35C



ROD GROUP DROP & TRIP TEST - T.P. 43.5
1800
1600
1400
1200
1000
800
MAIN FEEDWATER PUMP DISCHARGE PRESSURE (PSIG)
AUCTIONEERED NUCLEAR POWER (%)
100
100
FEED-STEAM DELTA P ACTUAL (PSID)
180
160
140
120
100
80
ELAPSED TIME 14:47:48 - 14:51:56 13 NOV 85

FIGURE 35D



ROD GROUP DROP & TRIP TEST - T.P. 43.5
MAIN TURBINE SPEED (RPM)
MAIN FEEDWATER PUMP 2-1 SPEED (RPM)
CONDENSATE BOOSTER PUMPS DISCHARGE PRESSURE (PSIG)
MAIN FEEDWATER PUMPS SUCTION PRESSURE (PSIG)
2100
1800
1500
1200
900
600
7000
6000
5000
4000
3000
2000
700
600
500
400
300
200
ELAPSED TIME 14:47:48 - 14:51:56 13 NOV 85

FIGURE 35E

5.21 Test Procedure No. 41.1 - Plant Shutdown from Outside the Control Room

TEST OBJECTIVE

The purpose of this test was to demonstrate that normal Hot Standby conditions can be established and maintained from outside the Main Control Room (MCR).

TEST DESCRIPTION

Following a reactor trip, essential primary and secondary system conditions such as RCS Temperature and Pressure, RCS Boron Concentration and Steam Generator levels and pressures were controlled from outside the Main Control Room - primarily from the Hot Shutdown Panel (HSDP) - as required to establish and maintain stable shutdown conditions.

TEST RESULTS

The test was performed on November 13, 1985, with the unit operating at approximately 50% power. Following a reactor trip from outside the Main Control Room (see Section 5.20), a minimum Operations test crew consisting of six members evacuated the Main Control Room, assumed control of the plant from the HSDP and manned other stations to monitor and control plant parameters in accordance with Operating Procedure OP AP-8, "Control Room Inaccessibility."

The on-shift Shift Foreman and his crew remained on watch during this test to monitor the plant and note any problems encountered. Control was maintained from the HSDP for approximately three hours. During this period, additional actions were required from various locations within the plant by the test crew.

Once the test crew had established Hot Standby conditions (RCS temperature approximately 547 deg. F, Pressurizer pressure approximately 2235 psig and Pressurizer level approximately 22%), these conditions were maintained for approximately 30 minutes. The test was then terminated by transferring control back to the Control Room.

Some Control Room Operator actions were performed during this test. These actions and their justifications are listed below:

1. Close MSIVs and Bypasses after trip.
   No adverse effect on results. Closure of valves is part of immediate action prior to leaving MCR per OP AP-8.

2. Manual verification of closure of Main Feed Reg. Valves.
   No adverse effect. Checking the valves to ensure that they are closed is part of immediate action in EP E-0.1; valves were already closed. Check valves also exist in FW lines if Main Feed Regulator Valves are not closed.

5.21 (Continued)

3. Manual adjustment of PCV-135 twice during test to re-establish letdown after letdown isolation.
No adverse effect. Operating experience shows that establishing letdown without PCV-135 adjustment causes lifting of the letdown line relief. PCV-135 adjusted in attempt to prevent unnecessary cycling of relief valve. In actual emergency, relief valve would be allowed to lift.

4. Circ. Water Pump 2-1 restarted.
No adverse effect. Circ. water pump restarted to maintain condenser vacuum and prevent condensate system perturbations. Does not affect primary plant with MSIVs and MFW system isolated.

5. MSIV bypass valves opened assuming downstream isol. valves shut; MSIV bypass valves reshut when improper valve lineup identified.
No adverse effect. Opening MSIV bypasses with downstream isolation valves shut is essentially the same as MSIV bypasses shut (one valve in bypass line shut). Improper lineup caused further primary system cooldown and may have contributed to second letdown isolation, but did not affect ability to recover and maintain RCS temperature from Hot Standby Panel.

6. Opening and closing of MFW Pump Recirc. valve.
No effect. MFW Pump essentially isolated from Primary Plant.

7. Manual adjustment to letdown temperature (TCV-130).
No effect due to possible alternatives. TCV-130 was in manual for several days due to erratic operation in AUTO. Valve worked satisfactorily during T.P. 37.20, Control Room Inaccessability. In actual emergency with control in manual, either (a) temperature would be allowed to stay at higher value or (b) a test signal could be injected into TCV-130 control circuit and temperature monitored locally. It should be noted that adjustment was made only while letdown flow was increased to decrease Pressurizer level from approximately 50% after letdown isolation; if strictly following OP AP-8, Pressurizer level is increased to 50% in preparation for cooldown and excess letdown flow would not be required. Temperature was maintained satisfactorily with normal (75 gpm) letdown flow.

8. Draining Pressurizer Relief Tank (PRT) level to normal operating band.
No effect. Operator convenience only; PRT level could have been left as is throughout the transient. (Level increase apparently due to Letdown Relief valve lifting even though PCV-135 was adjusted to prevent/minimize lifting.)

Because these Control Room operators' actions could be resolved administratively and all Acceptance Criteria were met, this test demonstrates satisfactorily the capability to remotely maintain the plant in Hot Standby conditions.

5.22. Test Procedure No. 43.1 - Load Swing Tests

TEST OBJECTIVE

The main objective of this test was to verify plant dynamic response, including automatic control system performance, to 10% step load changes introduced at the turbine generator control panel.

TEST DESCRIPTION

This test was performed at 30%, 50%, 75%, and 100% power. At each test plateau, the plant was verified to be stable and on automatic control. By using the turbine load control system, plant output was then reduced by 10% (i.e., approximately 115 MWe) at the maximum rate of 2200 MWe per minute.

After stability was achieved and all necessary data were collected, plant output was increased to the previous power level at a rate of 2200 MWe per minute, again monitoring the above parameters. This increase was not done at the 100% test plateau.

TEST RESULTS

For all power plateaus at which this test was performed, Acceptance Criteria were satisfied as follows:

1. Reactor and Turbine did not trip.
2. Safety injection did not initiate.
3. Main steam safety valves did not lift.
4. Pressurizer relief valves and safety valves did not lift.
5. Nuclear power undershoot and overshoot was less than 3%.

In addition a final acceptance criteria, no manual intervention to bring the plant to steady state conditions, was satisfied.

At the 30% plateau, the 10% load decrease was initiated and all systems responded correctly. The plant stabilized within 8 minutes. Then the 10% load increase was initiated and all systems responded correctly. The plant stabilized within 15 minutes. During the load increase, the expected Steam Generator 2-1 level overshoot value of less than 10% was exceeded (actual was 11.6%). Westinghouse and PG&E Engineering judged this deviation to be acceptable.

At the 50% plateau, the plant response to the load decrease and increase was satisfactory. During the load decrease and increase, the expected steam pressure overshoot value of 25.0 psi was exceeded (31.0 psi during load decrease and 32.5 psi during load increase).

At the 75% plateau, all expected values were met except that, during the load decrease, steam pressure overshoot was 28 psi, exceeding the expected value of 25 psi.

5.22 (Continued)

At the 100% plateau, the Load Transient Bypass system was actuated during the load decrease. Plant response was satisfactory and equilibrium conditions were reached in 4 1/2 minutes. The steam pressure overshoot was 53.0 psi and exceeded the expected overshoot value of 25.0 psi.

Deviations from the expected values for the pressures at the 50%, 75% and 100% plateau were reviewed and judged to be acceptable by Westinghouse and PG&E Engineering prior to continuing with any further transient testing at that test plateau. A summary of selected parameter response to the transient is tabulated in Table 25 and illustrated in Figures 36A through 40E.

Table 25

Key Plant Parameters During 10% Load Swings

	30% Pwr to 20% Pwr	20% Pwr to 30% Pwr	50% Pwr to 40% Pwr	40% Pwr to 50% Pwr	75% Pwr to 65% Pwr	65% Pwr to 75% Pwr	100% Pwr to 90% Pwr
Pressurizer Pressure:							
Initial Pressure (psig)	2230	2232	2230.8	2232	2227.2	2232	2236.8
Maximum Pressure (psig)	2232.5	2235	2236.8	2232.2	2232	2235	2242.8
Minimum Pressure (psig)	2202.5	2213	2203.8	2220	2202	2227.2	2191.8
Maximum - Initial (psid)	2.5	3	6.0	1.2	4.8	3	6.0
Initial - Minimum (psid)	27.5	19	27.0	12.0	25.2	4.8	45.0
Steam Generator Level:							
Initial S/G 2-1 Level (%)	44	42.4	45	45	44.1	43.5	44.5
Maximum S/G 2-1 Level (%)	45.9	54	50.4	51.8	50.5	50.8	51.0
Minimum S/G 2-1 Level (%)	36.5	37	38.0	39.8	37.8	37.0	39.0
Maximum - Initial (%)	1.9	11.6 *	5.4	6.8	6.4	7.3	6.5
Initial - Minimum (%)	8.5	5.4	7.0	5.2	6.3	6.5	5.5
Steam Pressure:							
Initial Pressure (psig)	893	912	846	867.5	781.5	795	715
Final Pressure (psig)	911.9	888	865	860	795	780	712
Maximum Pressure (psig)	935	911	896	862.0	823	783	765
Minimum Pressure (psig)	893	864.2	846	827.5	800	756	708.5
Final - Minimum (psid)	N/A	23.8	N/A	32.5 *	N/A	24	N/A
Maximum - Final (psid)	23.1	N/A	31.0 *	N/A	28 *	N/A	53*

Expected Response

1. Pressurizer pressure swings of less than 50 psi.
2. Steam Generator levels not varying by more than 10% from initial level.
3. Steam pressure not having an overshoot or undershoot from the final value of more than ±25 psi. For a load decrease, steam pressure should increase and show a slight overshoot before stabilizing to a final value. For a load increase, steam pressure should show a decrease with a slight undershoot.

* As described in Section 5.22, these deviations from the expected response were judged to be acceptable.

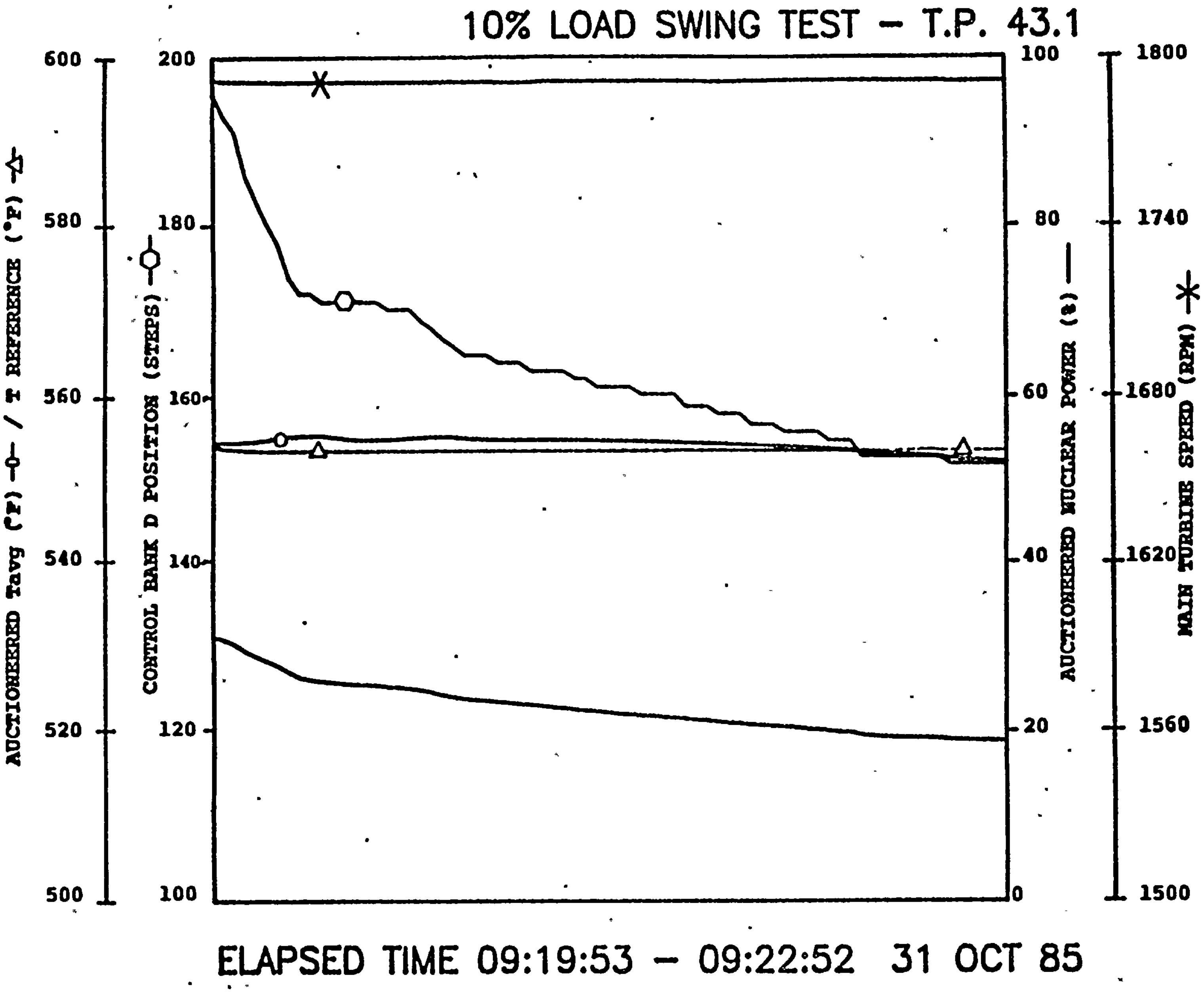


FIGURE 36A

126

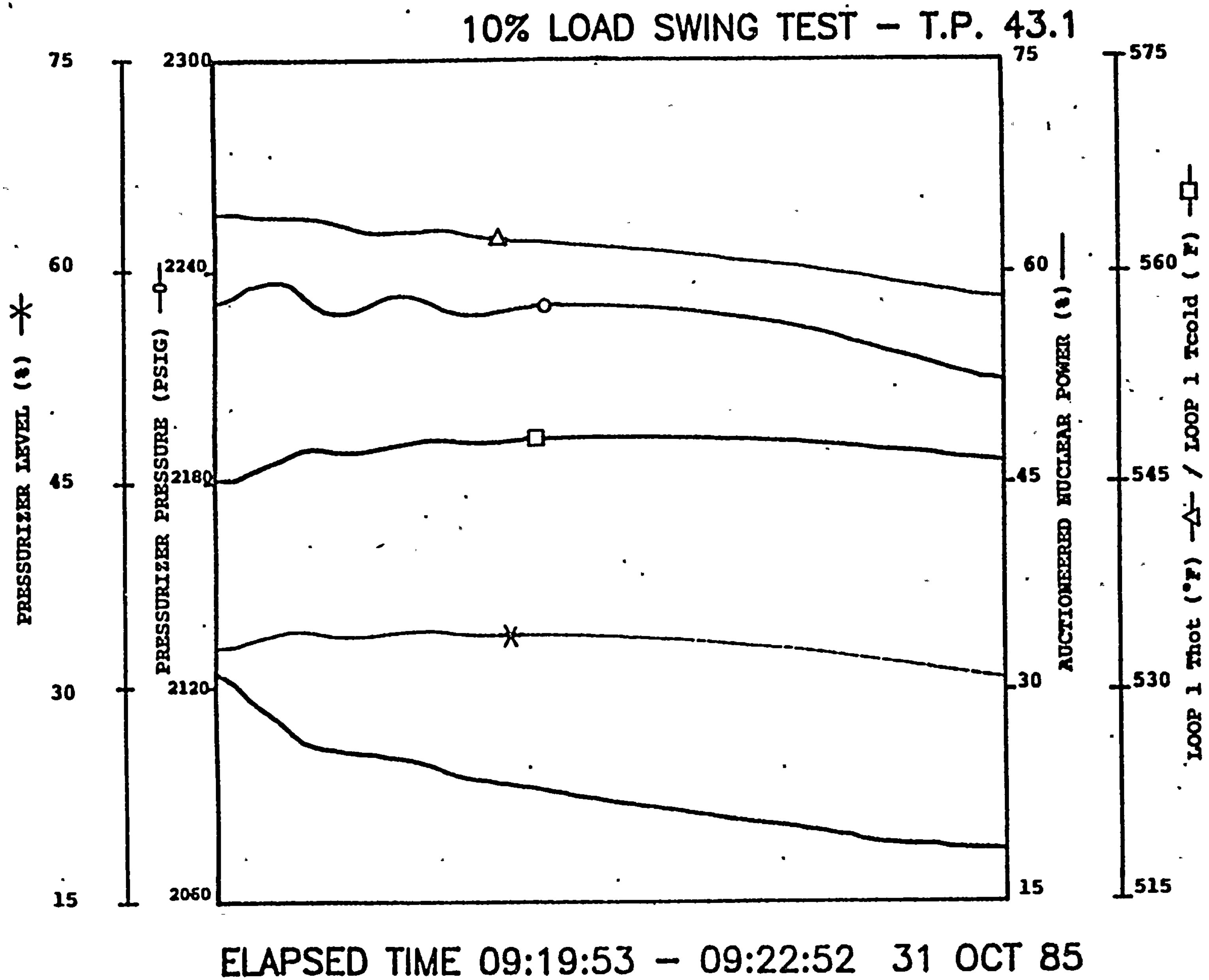
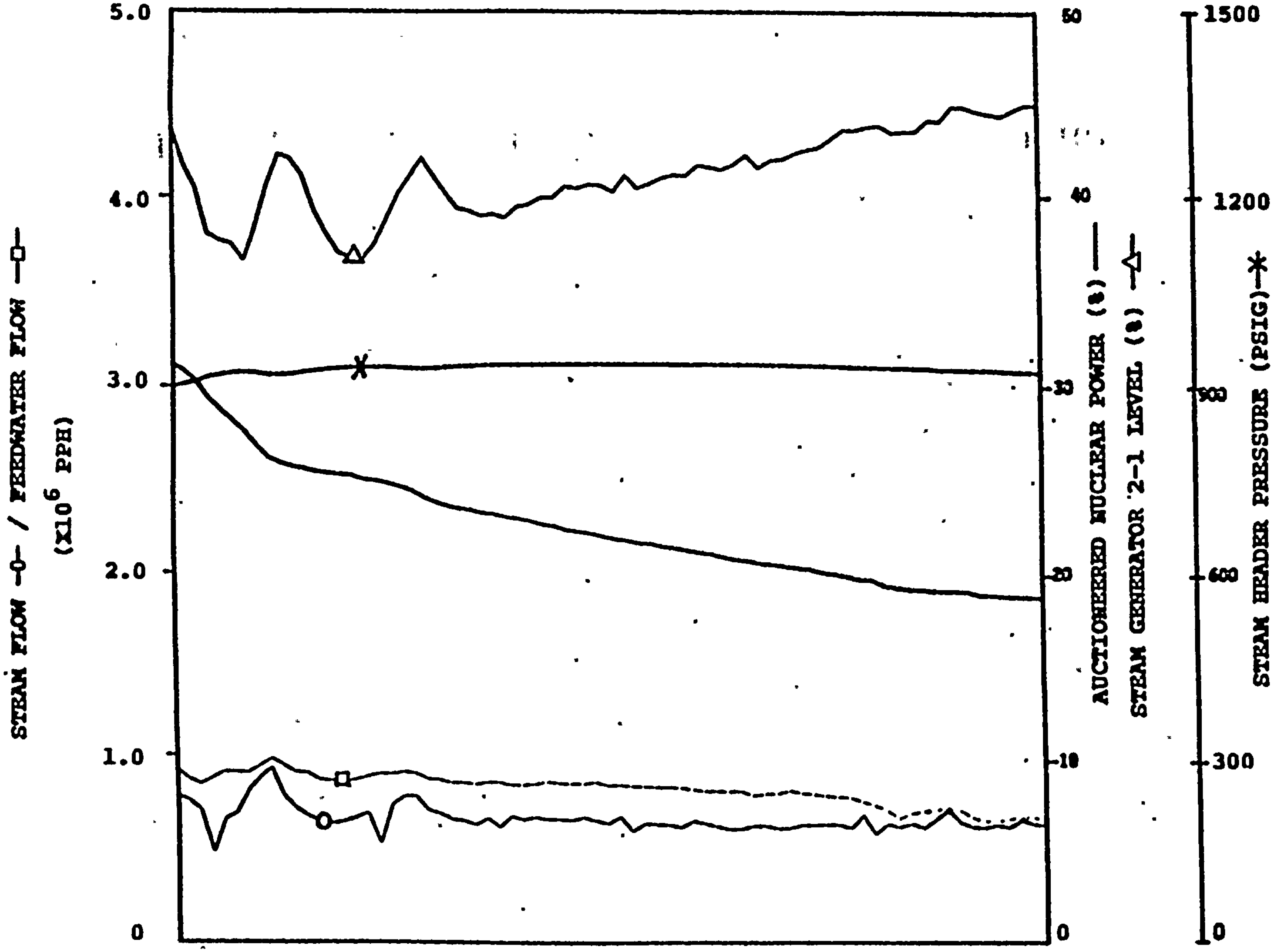

10% LOAD SWING TEST - T.P. 43.1
75
60
45
30
15
PRESSURIZER LEVEL (%)
2300
2240
2180
2120
2060
PRESSURIZER PRESSURE (PSIG)
75
60
45
30
15
AUCTIONEERED NUCLEAR POWER (%)
575
560
545
530
515
LOOP 1 Thot (°F) / LOOP 1 Tcold (°F)
ELAPSED TIME 09:19:53 - 09:22:52 31 OCT 85

FIGURE 36B

10% LOAD SWING TEST - T.P. 43.1



ELAPSED TIME 09:19:53 - 09:22:52 31 OCT 85

FIGURE 36C

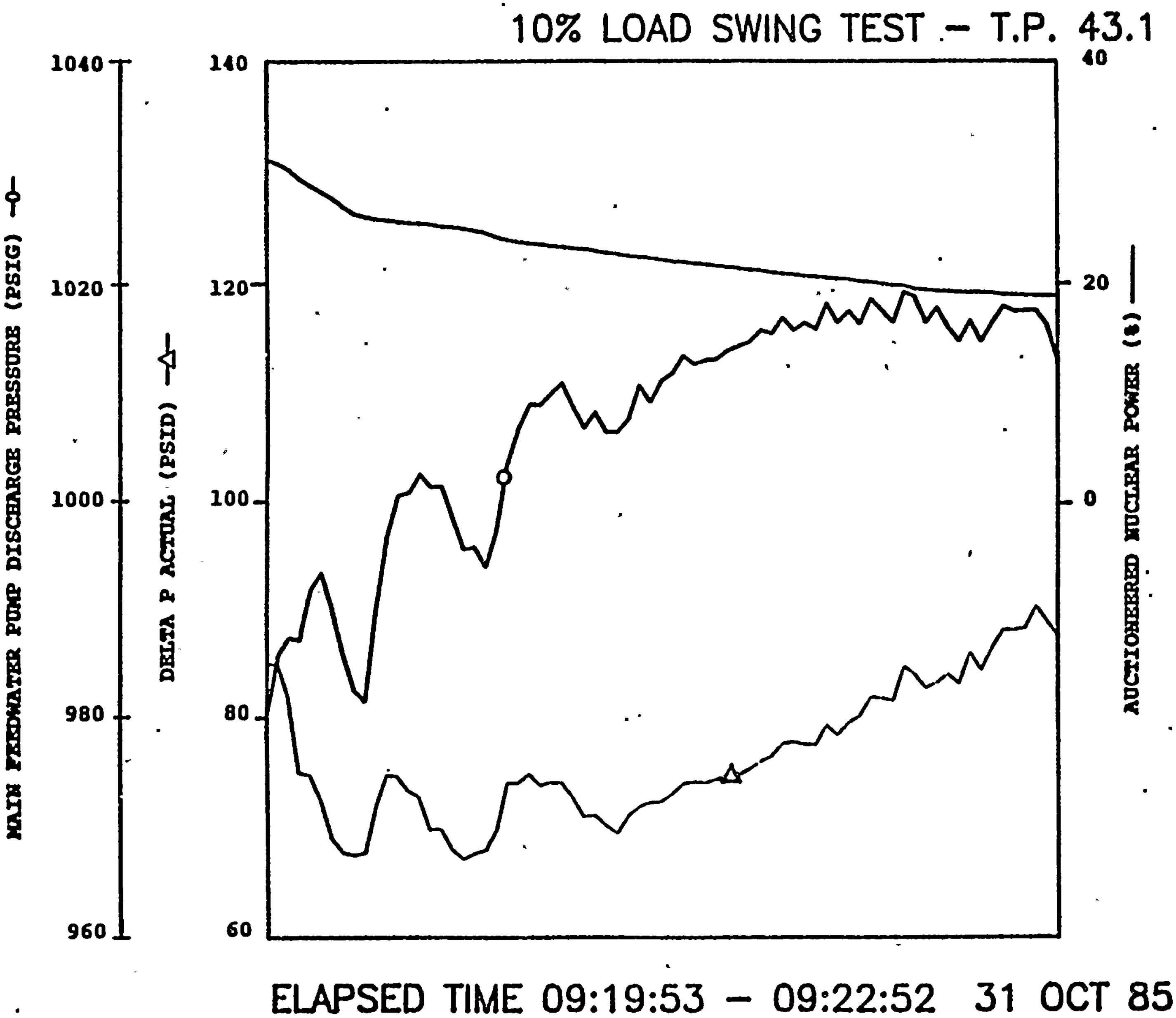
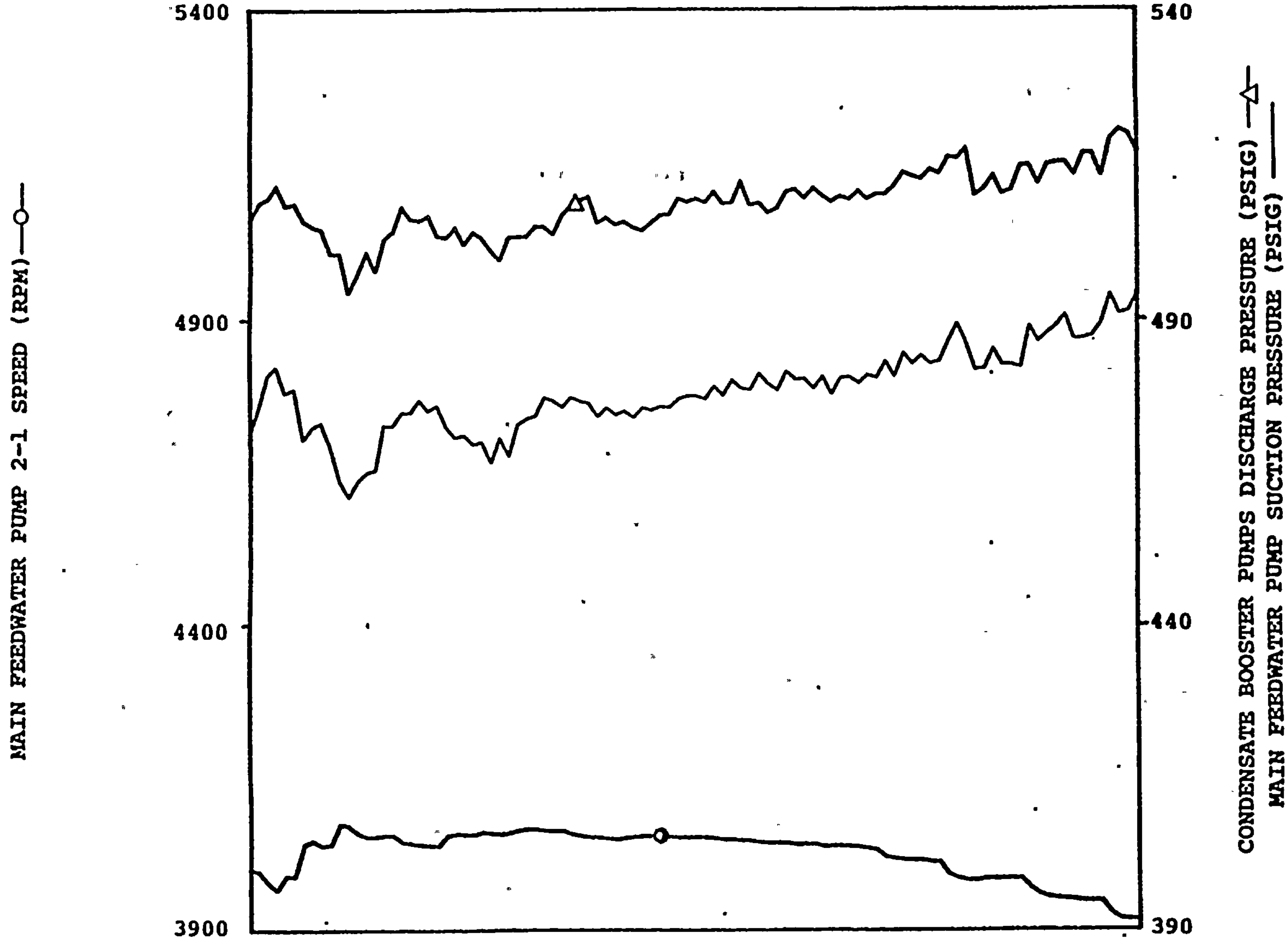

10% LOAD SWING TEST - T.P. 43.1
MAIN FEEDWATER PUMP DISCHARGE PRESSURE (PSIG)
1040
1020
1000
980
960
DELTA P ACTUAL (PSID)
140
120
100
80
60
AUCTIONEERED NUCLEAR POWER (%)
40
20
0
ELAPSED TIME 09:19:53 - 09:22:52 31 OCT 85

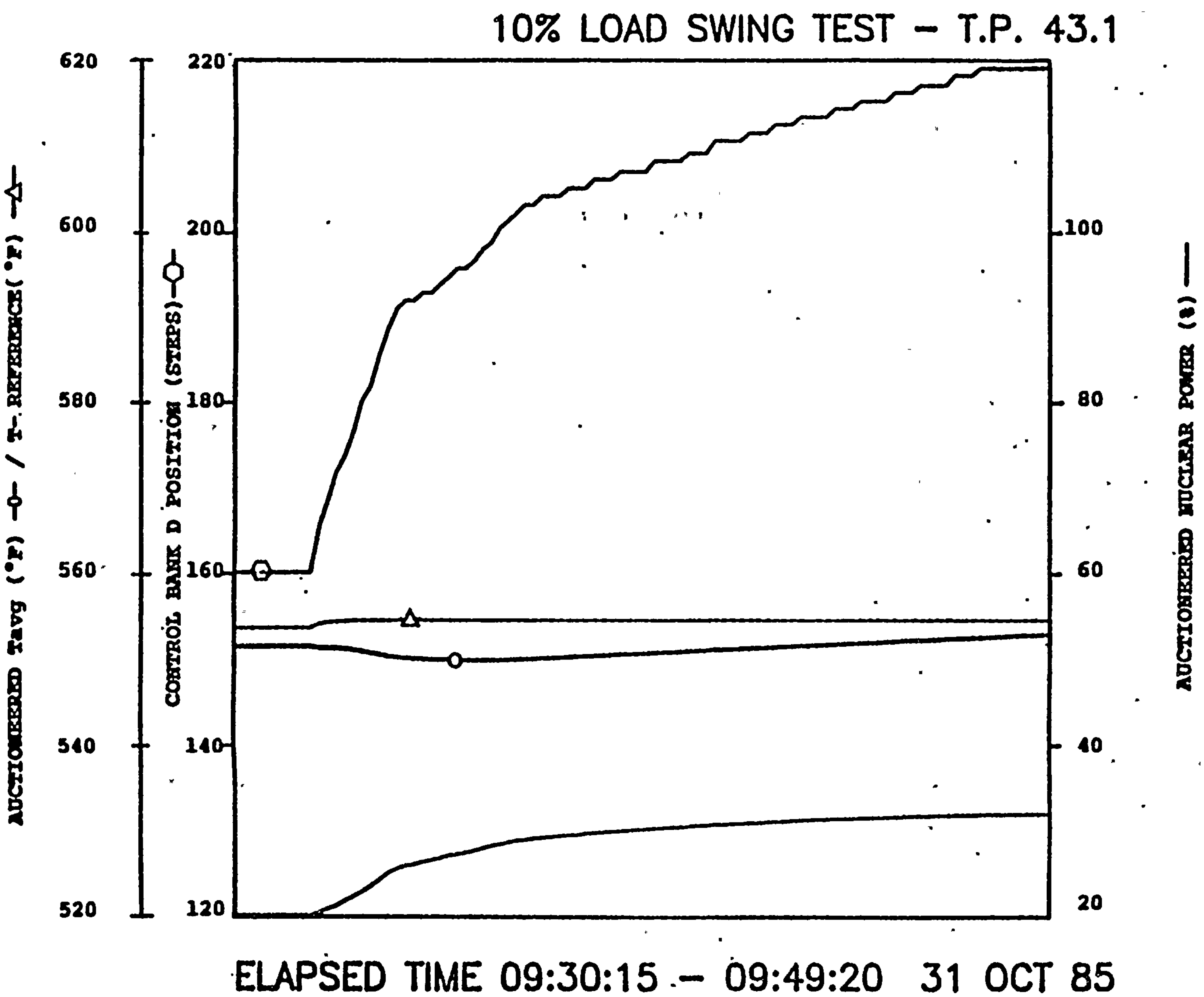
FIGURE 36D

10% LOAD SWING TEST – T.P. 43.1



ELAPSED TIME 09:19:53 – 09:22:52 31 OCT 85

FIGURE 36E



10% LOAD SWING TEST - T.P. 43.1
AUCTIONEERED Tavg (°F) / T-REFERENCE(°F)
CONTROL BANK D POSITION (STEPS)
AUCTIONEERED NUCLEAR POWER (%)
620
600
580
560
540
520
220
200
180
160
140
120
100
80
60
40
20
ELAPSED TIME 09:30:15 - 09:49:20 31 OCT 85

FIGURE 37A

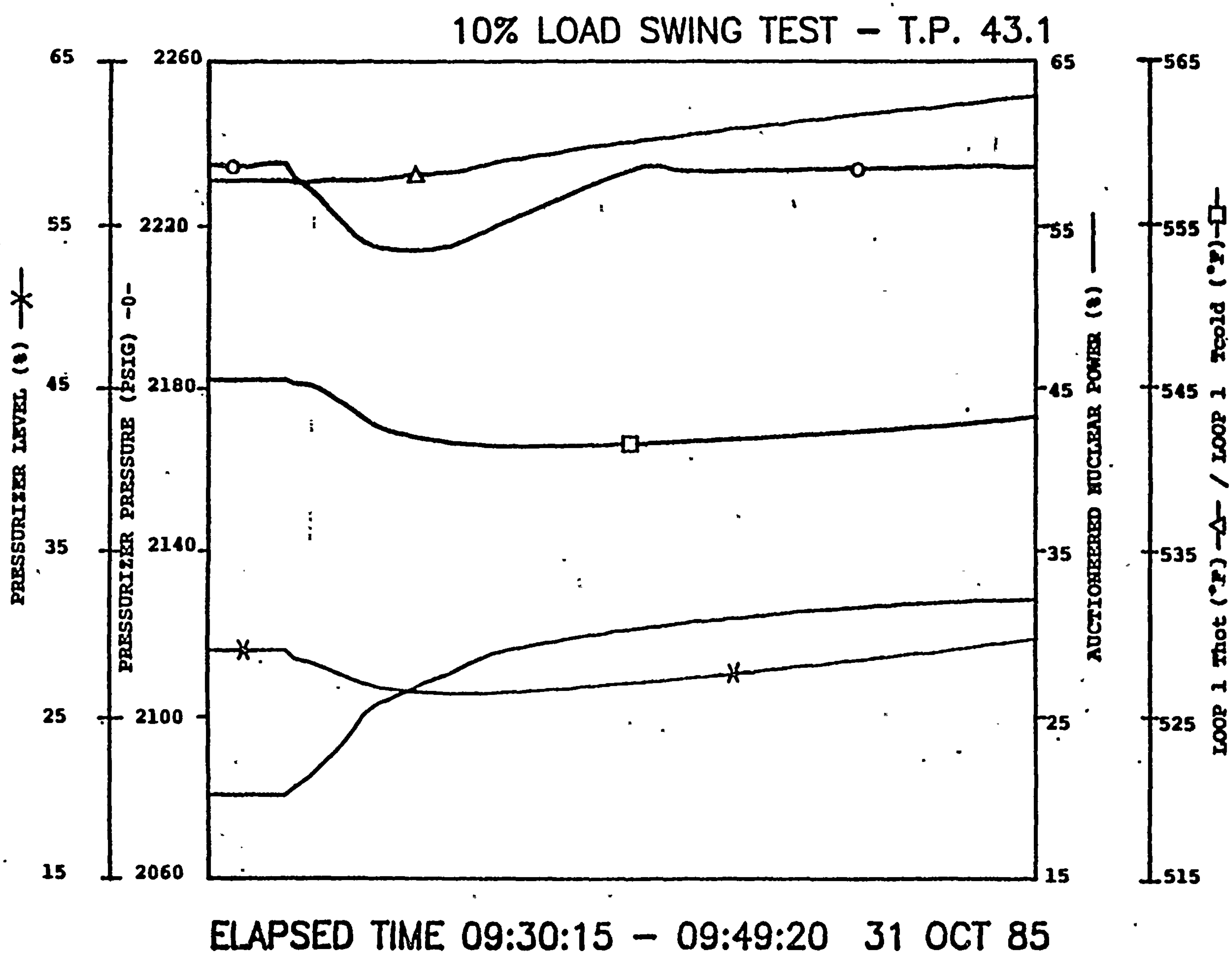

10% LOAD SWING TEST - T.P. 43.1
PRESSURIZER LEVEL (%)
PRESSURIZER PRESSURE (PSIG) -O-
AUCTIONEERED NUCLEAR POWER (%)
LOOP 1 Thot (°F) / LOOP 1 Tcold (°F)
65
55
45
35
25
15
2260
2220
2180
2140
2100
2060
565
555
545
535
525
515
ELAPSED TIME 09:30:15 - 09:49:20 31 OCT 85

FIGURE 37B

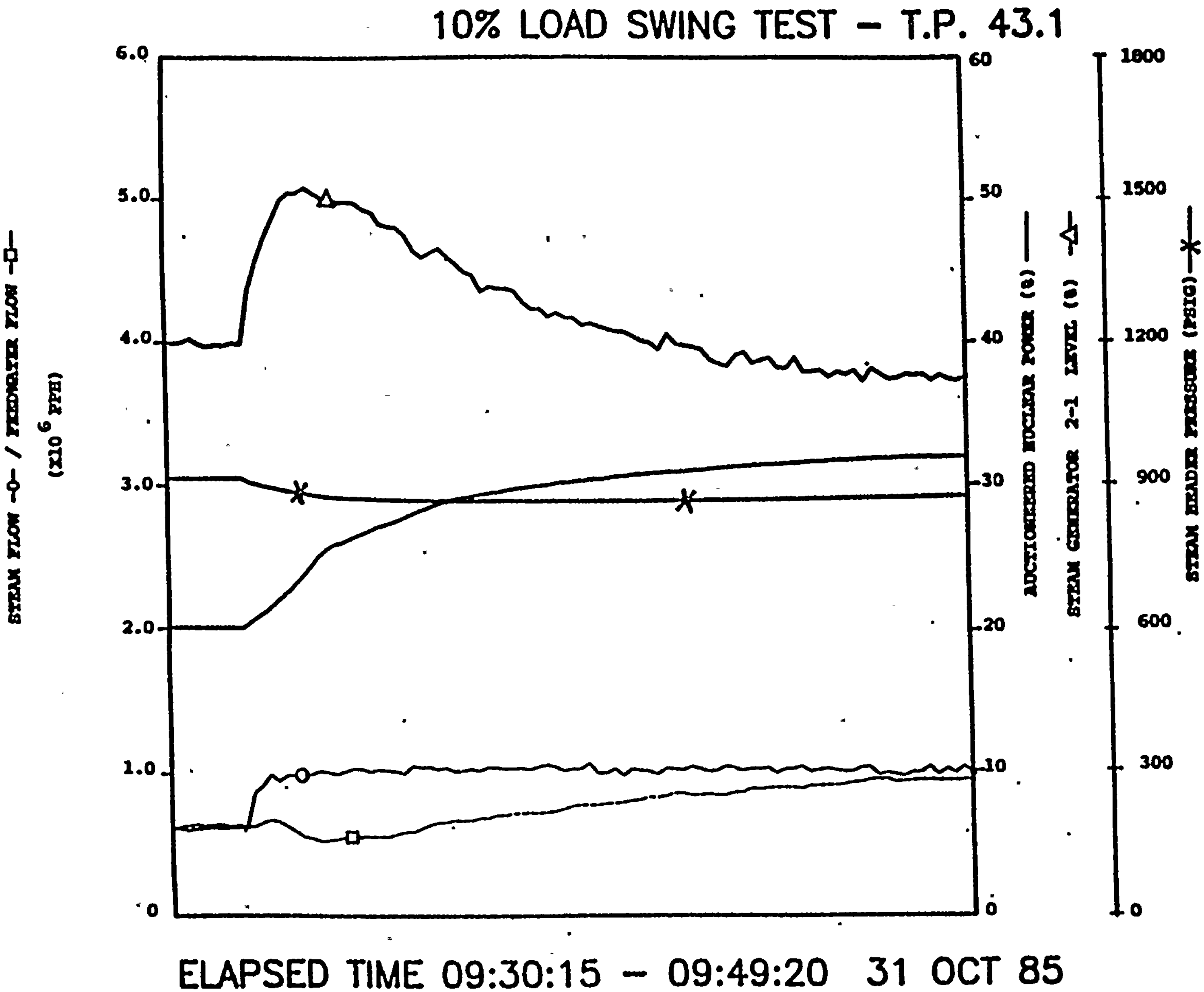

10% LOAD SWING TEST - T.P. 43.1
STEAM FLOW -○- / FEEDWATER FLOW -□-
(X10^6 PPH)
AUCTIONEERED NUCLEAR POWER (%)
STEAM GENERATOR 2-1 LEVEL (%)
STEAM HEADER PRESSURE (PSIG)
6.0
5.0
4.0
3.0
2.0
1.0
0
60
50
40
30
20
10
0
1800
1500
1200
900
600
300
0
ELAPSED TIME 09:30:15 - 09:49:20 31 OCT 85

FIGURE 37C

10% LOAD SWING TEST – T.P. 43.1

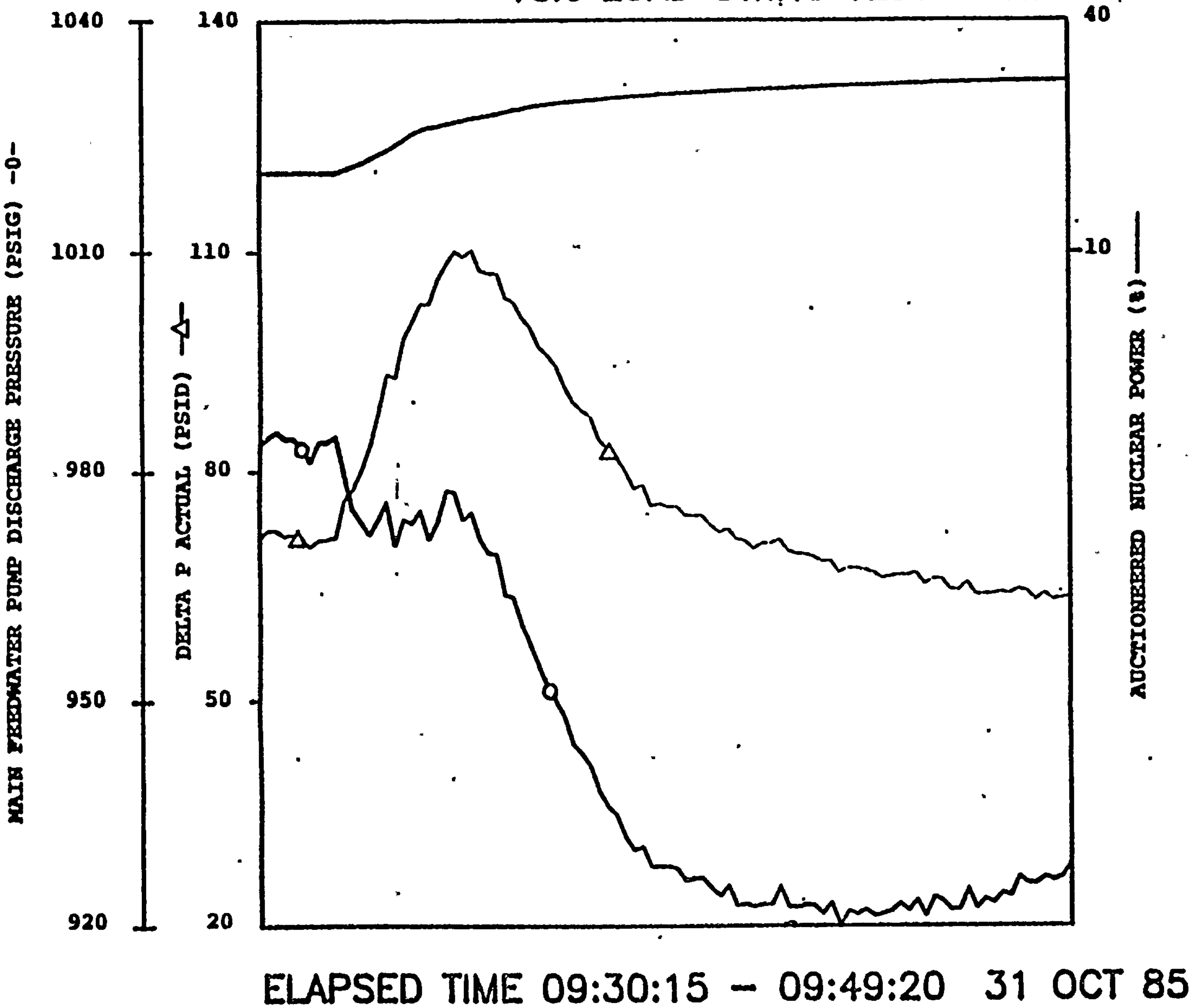


FIGURE 37D

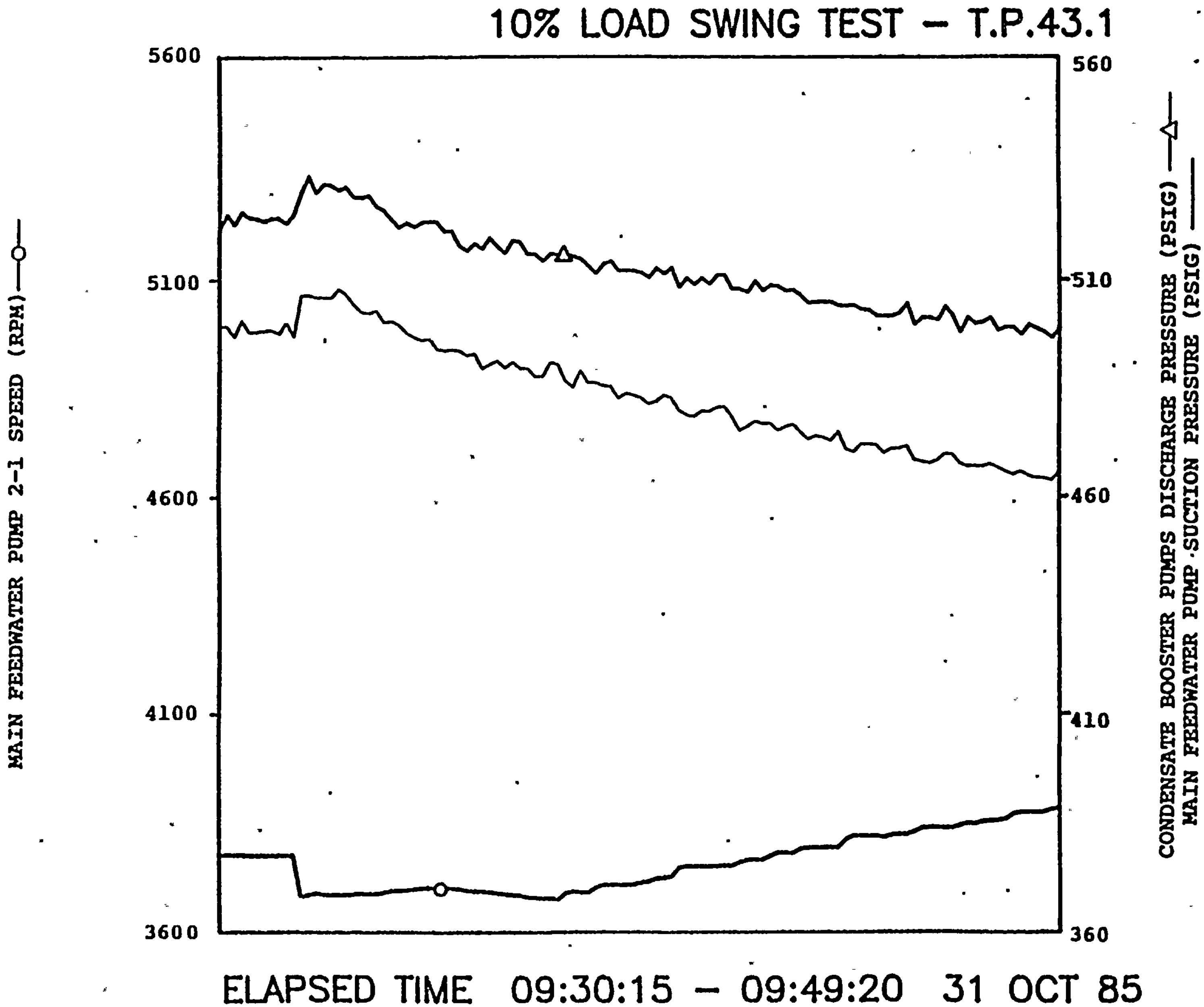

10% LOAD SWING TEST - T.P.43.1
5600
5100
4600
4100
3600
560
510
460
410
360
MAIN FEEDWATER PUMP 2-1 SPEED (RPM)
CONDENSATE BOOSTER PUMPS DISCHARGE PRESSURE (PSIG)
MAIN FEEDWATER PUMP SUCTION PRESSURE (PSIG)
ELAPSED TIME 09:30:15 - 09:49:20 31 OCT 85

FIGURE 37E

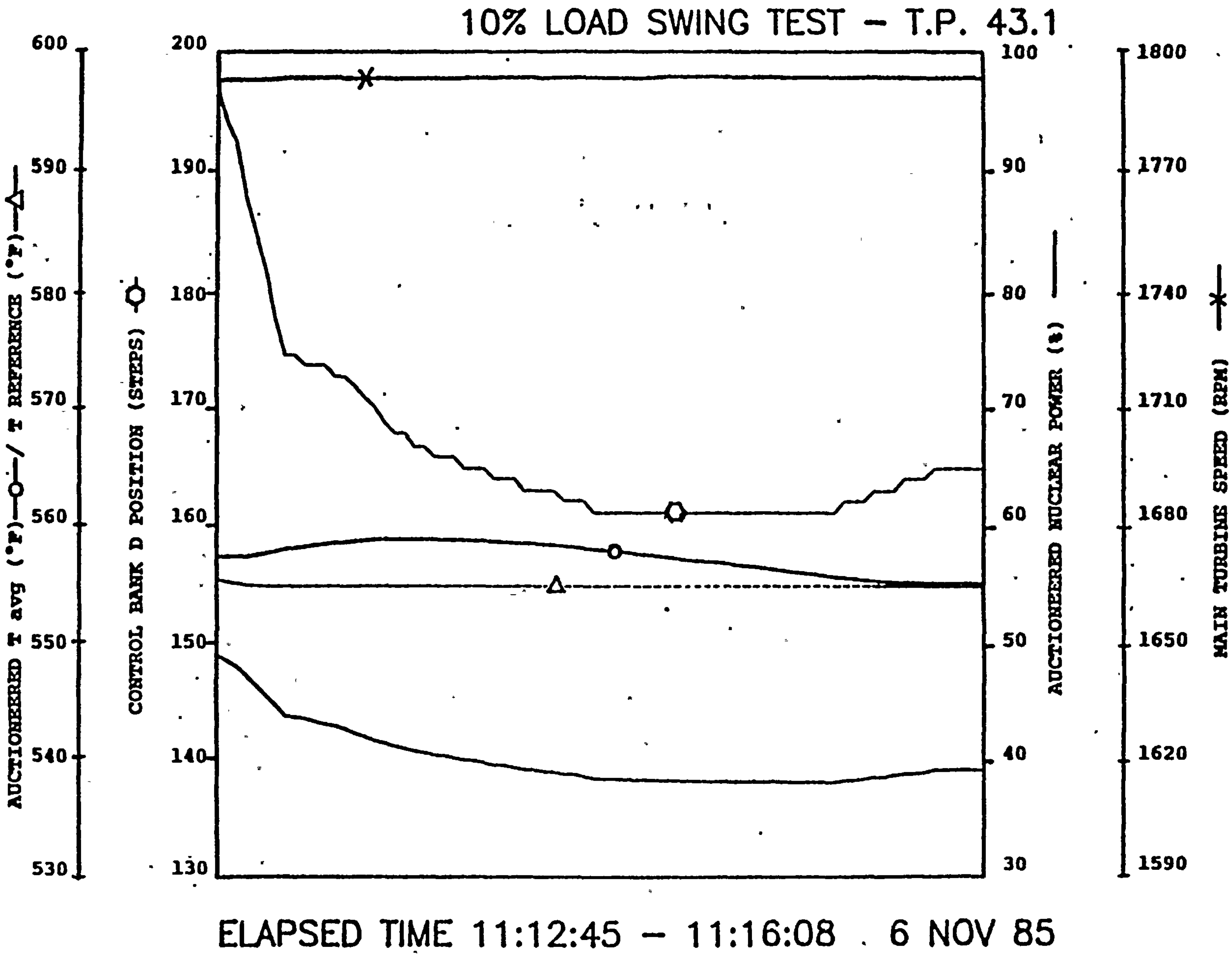

10% LOAD SWING TEST - T.P. 43.1
AUCTIONEERED T avg (°F)—O—/ T REFERENCE (°F)—△—
600
590
580
570
560
550
540
530
CONTROL BANK D POSITION (STEPS)
200
190
180
170
160
150
140
130
AUCTIONEERED NUCLEAR POWER (%)
100
90
80
70
60
50
40
30
MAIN TURBINE SPEED (RPM)
1800
1770
1740
1710
1680
1650
1620
1590
ELAPSED TIME 11:12:45 - 11:16:08 . 6 NOV 85

FIGURE 38A

137

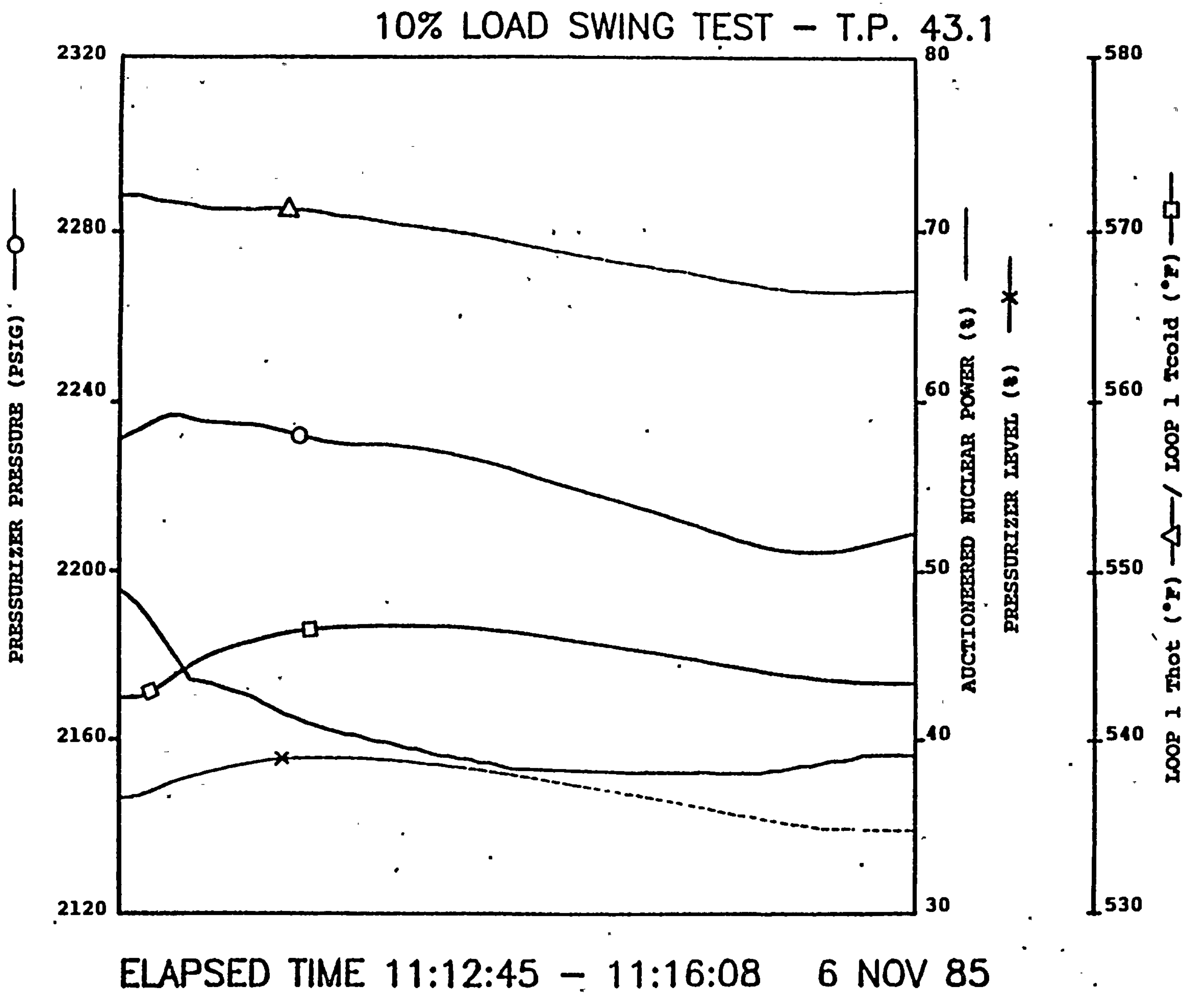
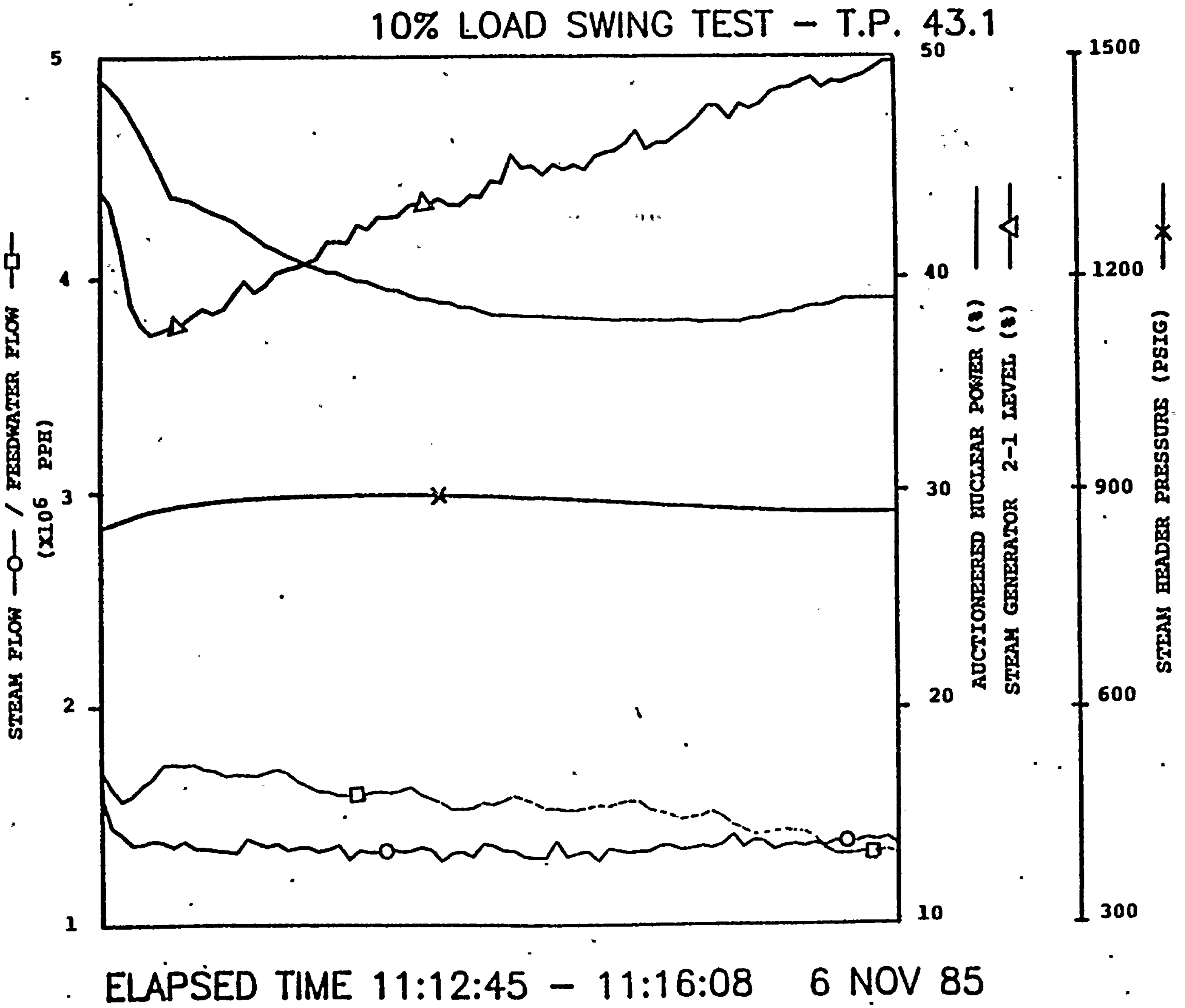


FIGURE 38B



10% LOAD SWING TEST - T.P. 43.1
STEAM FLOW / FEEDWATER FLOW (X10^6 PPH)
5
4
3
2
1
50
40
30
20
10
AUCTIONEERED NUCLEAR POWER (%)
STEAM GENERATOR 2-1 LEVEL (%)
1500
1200
900
600
300
STEAM HEADER PRESSURE (PSIG)
ELAPSED TIME 11:12:45 - 11:16:08 6 NOV 85

FIGURE 38C

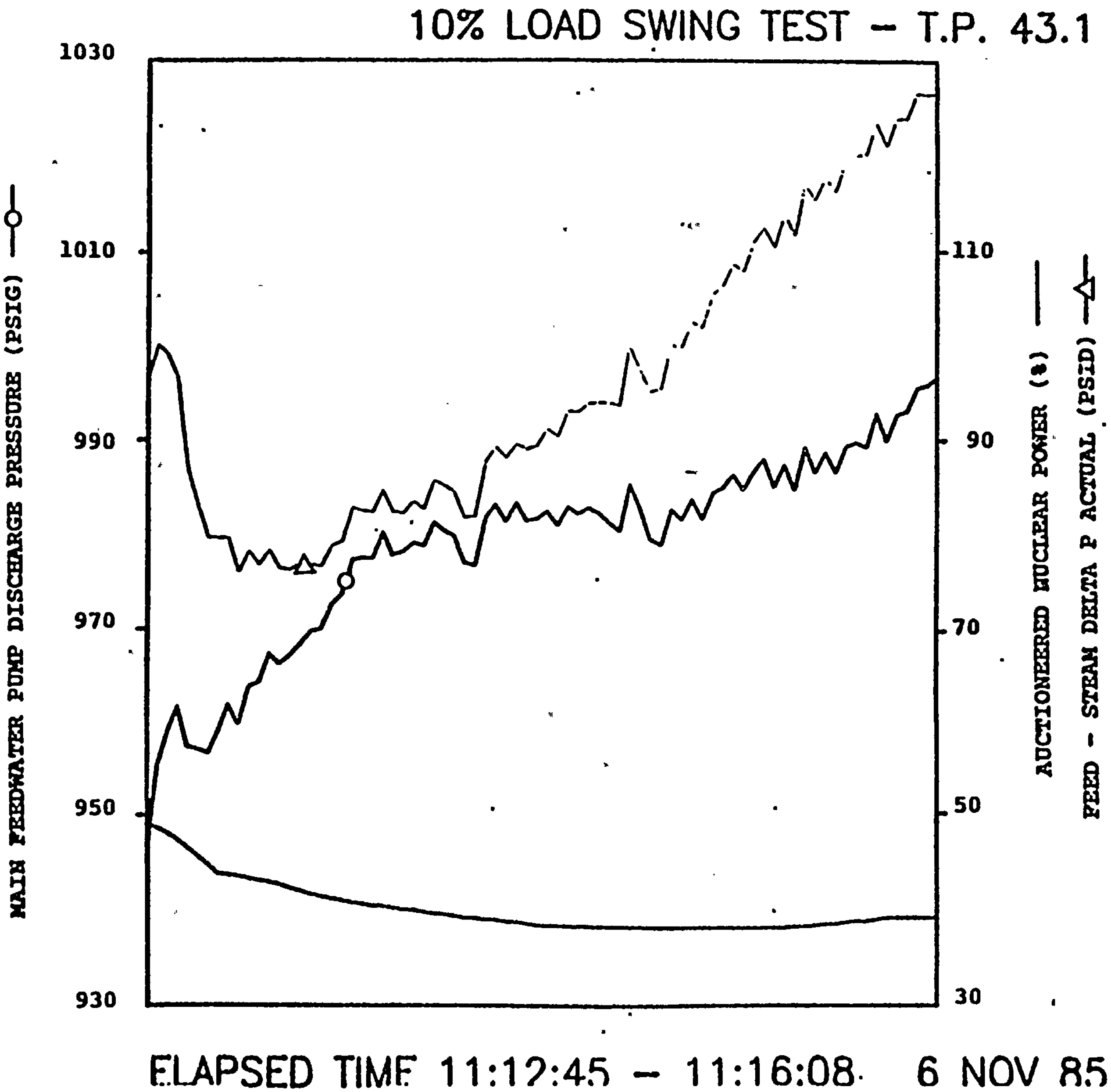

10% LOAD SWING TEST - T.P. 43.1
MAIN FEEDWATER PUMP DISCHARGE PRESSURE (PSIG)
1030
1010
990
970
950
930
AUCTIONEERED NUCLEAR POWER (%)
FEED - STEAM DELTA P ACTUAL (PSID)
110
90
70
50
30
ELAPSED TIME 11:12:45 - 11:16:08 6 NOV 85

FIGURE 38D

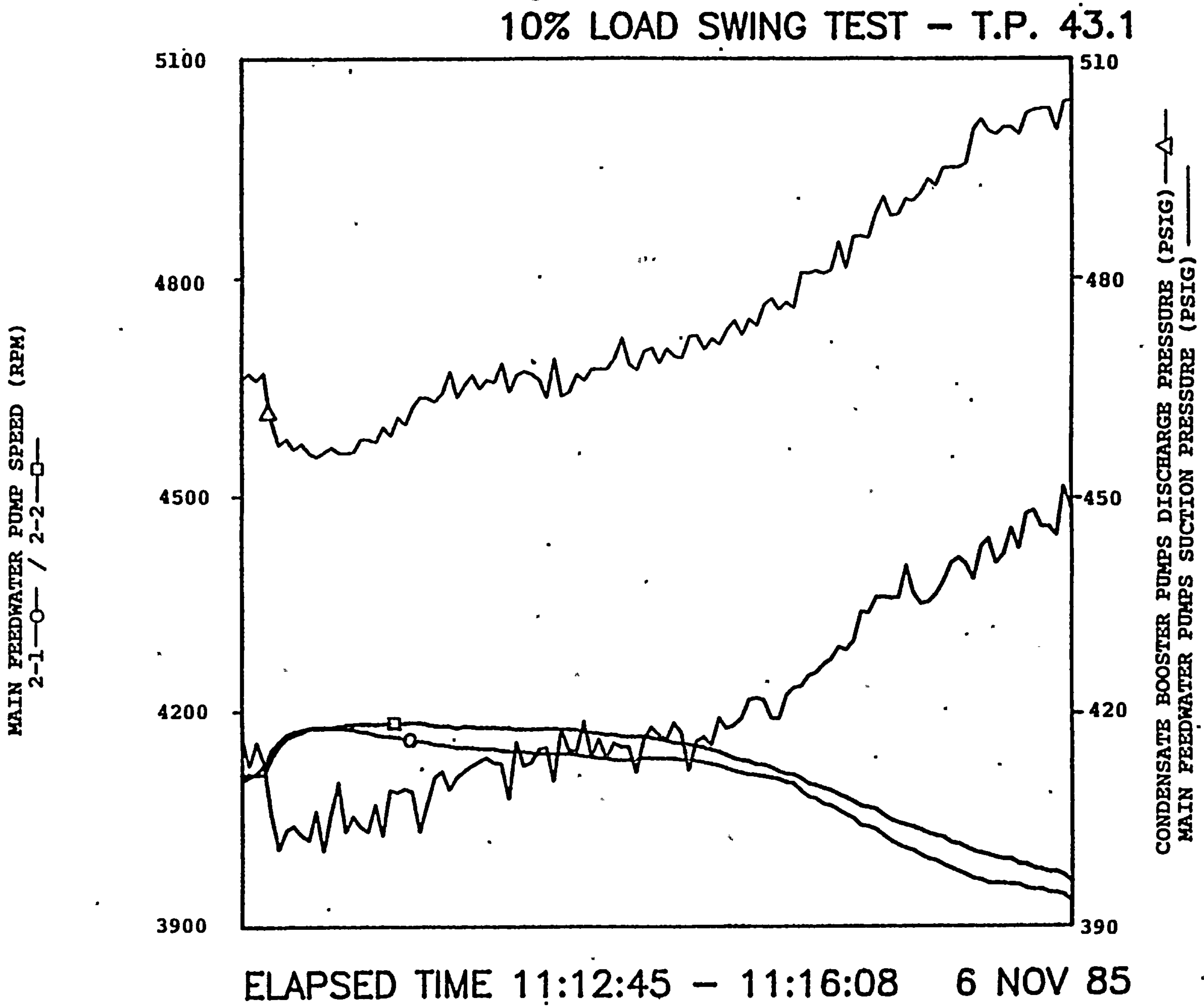
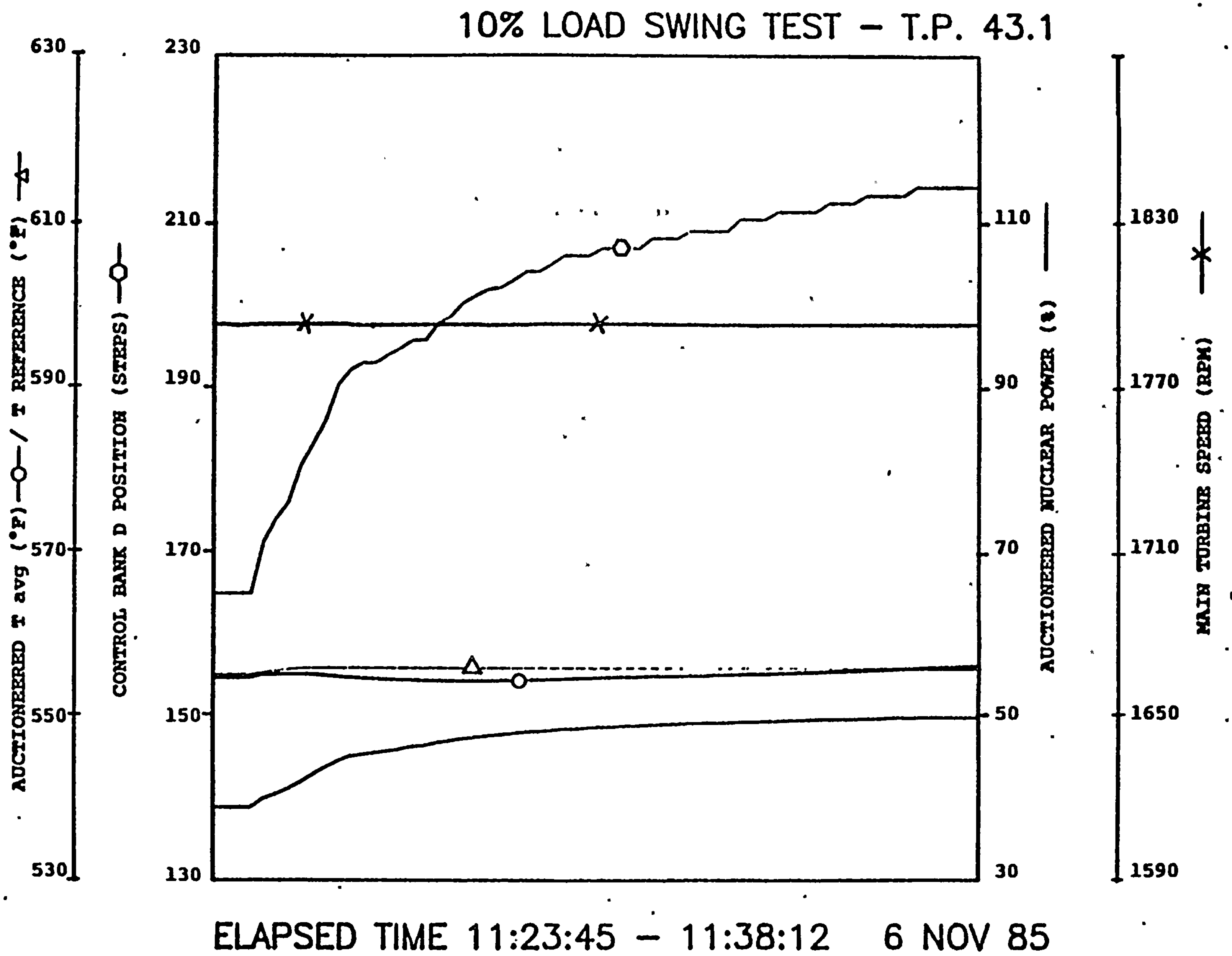
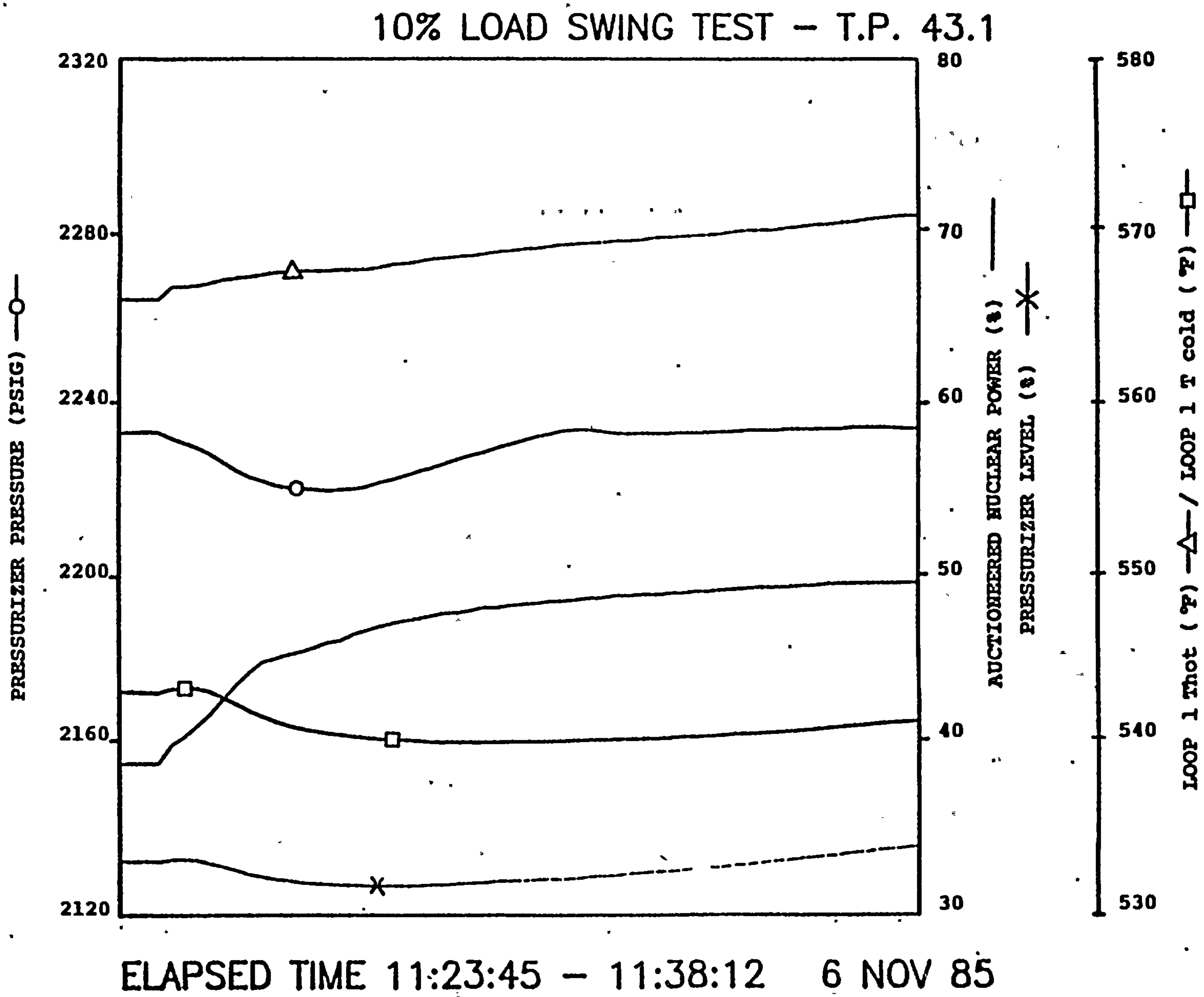

10% LOAD SWING TEST - T.P. 43.1
5100
4800
4500
4200
3900
510
480
450
420
390
MAIN FEEDWATER PUMP SPEED (RPM)
2-1 / 2-2
CONDENSATE BOOSTER PUMPS DISCHARGE PRESSURE (PSIG)
MAIN FEEDWATER PUMPS SUCTION PRESSURE (PSIG)
ELAPSED TIME 11:12:45 - 11:16:08 6 NOV 85

FIGURE 38E



10% LOAD SWING TEST - T.P. 43.1
AUCTIONEERED T avg (°F) / T REFERENCE (°F)
630
610
590
570
550
530
CONTROL BANK D POSITION (STEPS)
230
210
190
170
150
130
AUCTIONEERED NUCLEAR POWER (%)
110
90
70
50
30
MAIN TURBINE SPEED (RPM)
1830
1770
1710
1650
1590
ELAPSED TIME 11:23:45 - 11:38:12 6 NOV 85

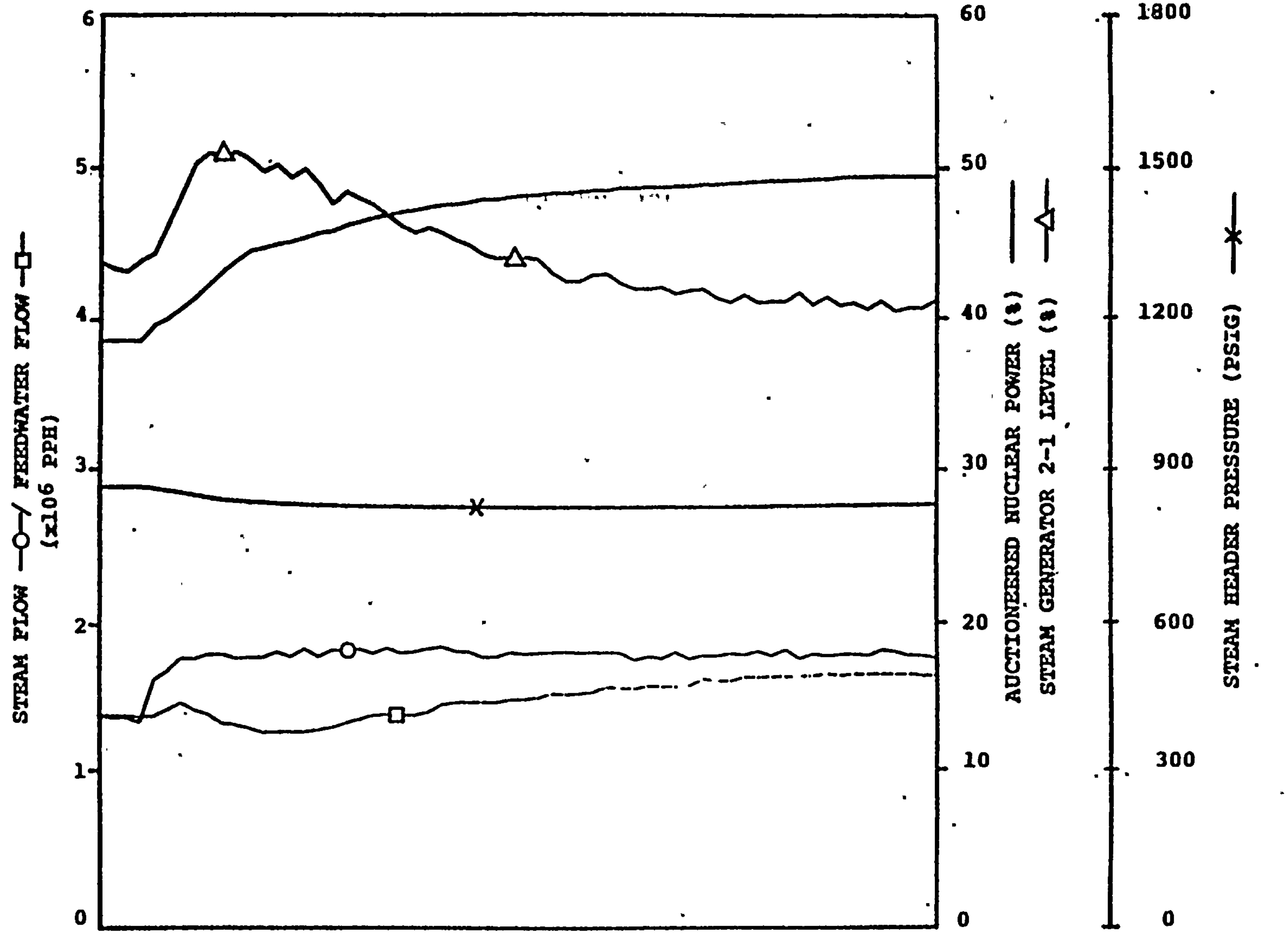
FIGURE 39A



10% LOAD SWING TEST - T.P. 43.1
2320
2280
2240
2200
2160
2120
PRESSURIZER PRESSURE (PSIG)
80
70
60
50
40
30
AUCTIONEERED NUCLEAR POWER (%)
PRESSURIZER LEVEL (%)
580
570
560
550
540
530
LOOP 1 Thot (°F) / LOOP 1 T cold (°F)
ELAPSED TIME 11:23:45 - 11:38:12 6 NOV 85

FIGURE 39B

10% LOAD SWING TEST – T.P. 43.1



ELAPSED TIME 11:23:45 – 11:38:12 6 NOV 85

FIGURE 39C

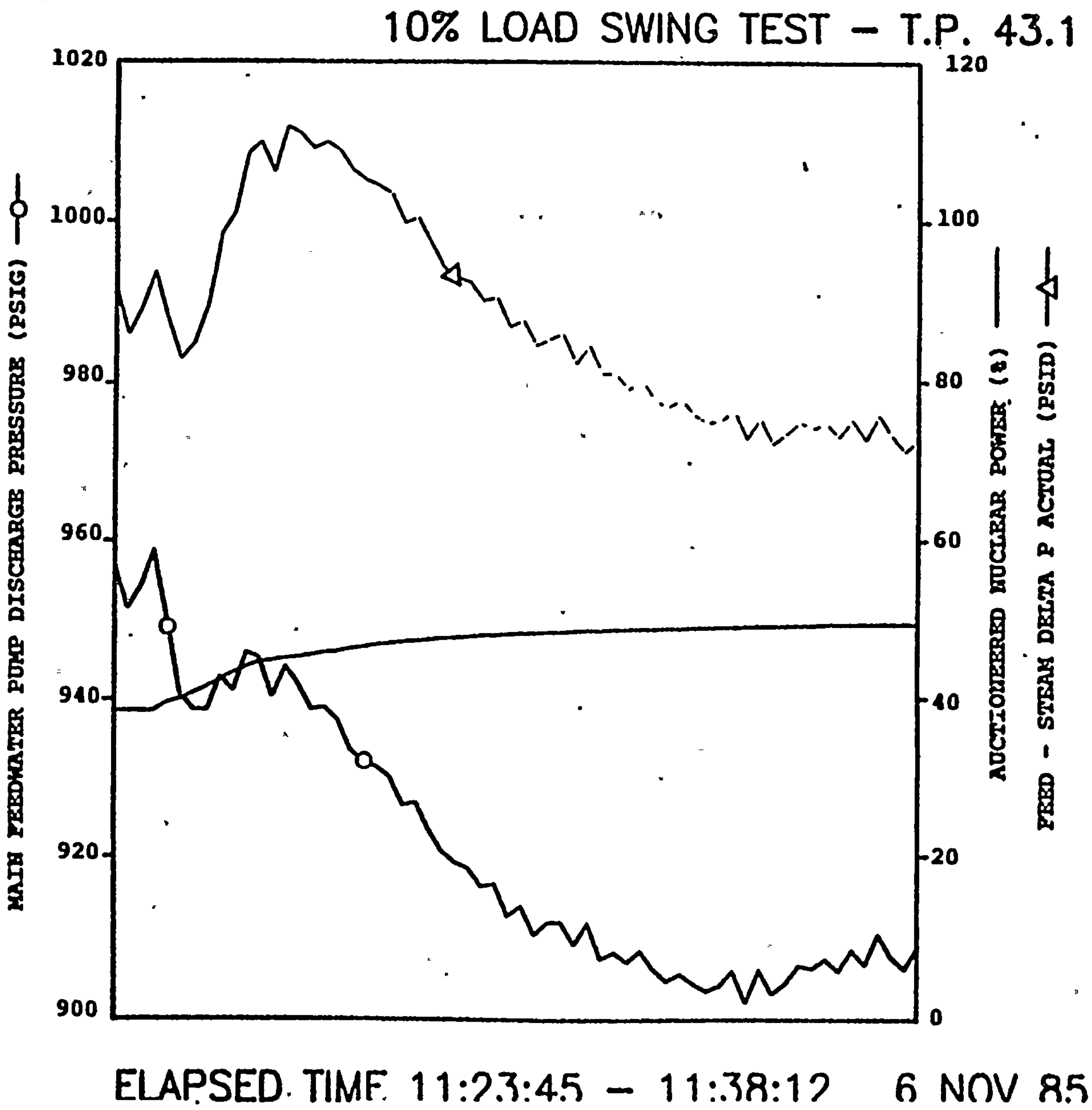
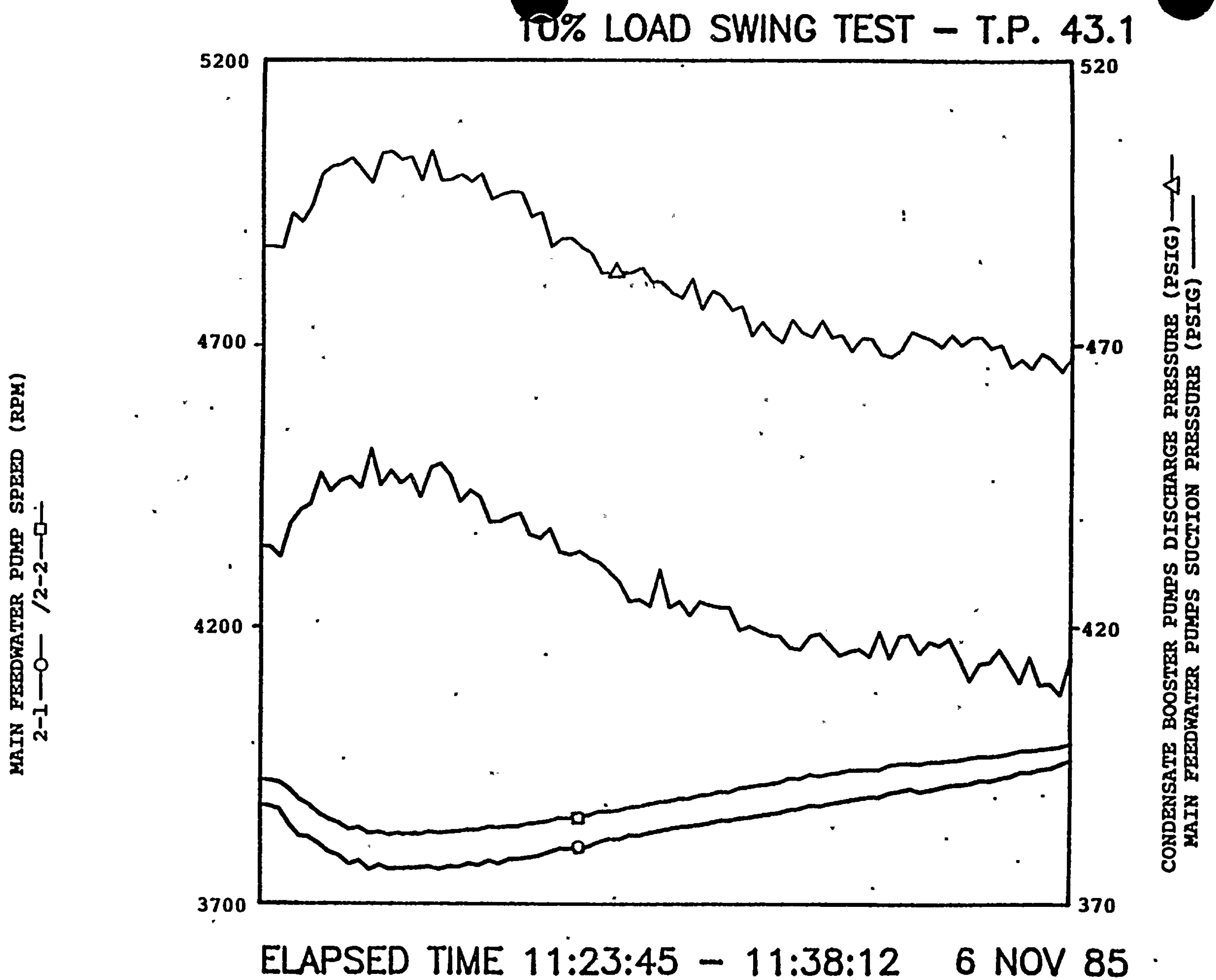

10% LOAD SWING TEST - T.P. 43.1
MAIN FEEDWATER PUMP DISCHARGE PRESSURE (PSIG)
AUCTIONEERED NUCLEAR POWER (%)
FEED - STEAM DELTA P ACTUAL (PSID)
1020
1000
980
960
940
920
900
120
100
80
60
40
20
0
ELAPSED TIME 11:23:45 - 11:38:12 6 NOV 85

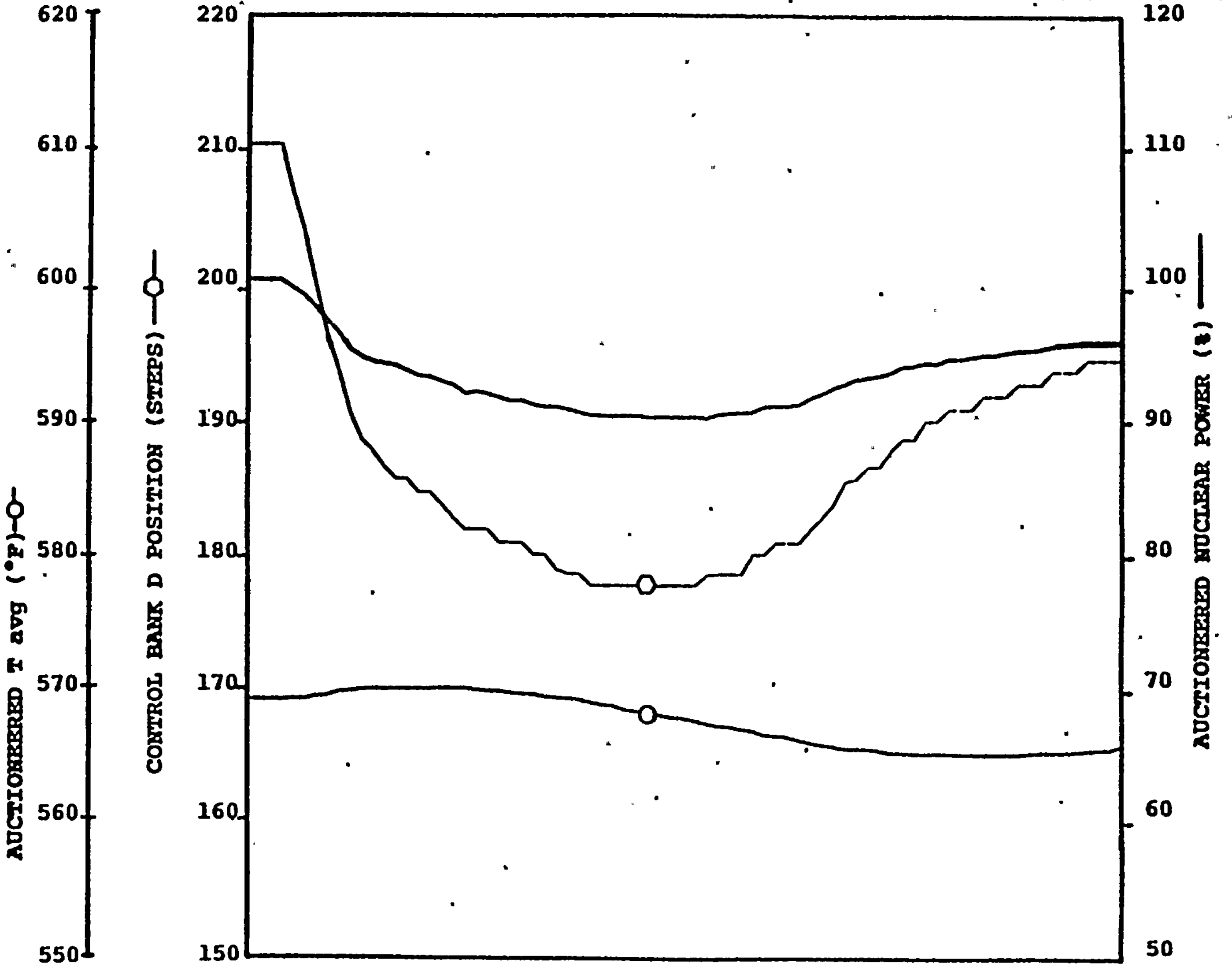
FIGURE 39D



10% LOAD SWING TEST - T.P. 43.1
5200
4700
4200
3700
520
470
420
370
MAIN FEEDWATER PUMP SPEED (RPM)
2-1 —o— /2-2 —□—
CONDENSATE BOOSTER PUMPS DISCHARGE PRESSURE (PSIG) —△—
MAIN FEEDWATER PUMPS SUCTION PRESSURE (PSIG) ——
ELAPSED TIME 11:23:45 - 11:38:12 6 NOV 85

FIGURE 39E

10% LOAD SWING TEST - T.P. 43.1



ELAPSED TIME 18:42:55 - 18:46:36 20 DEC 85

FIGURE 40A

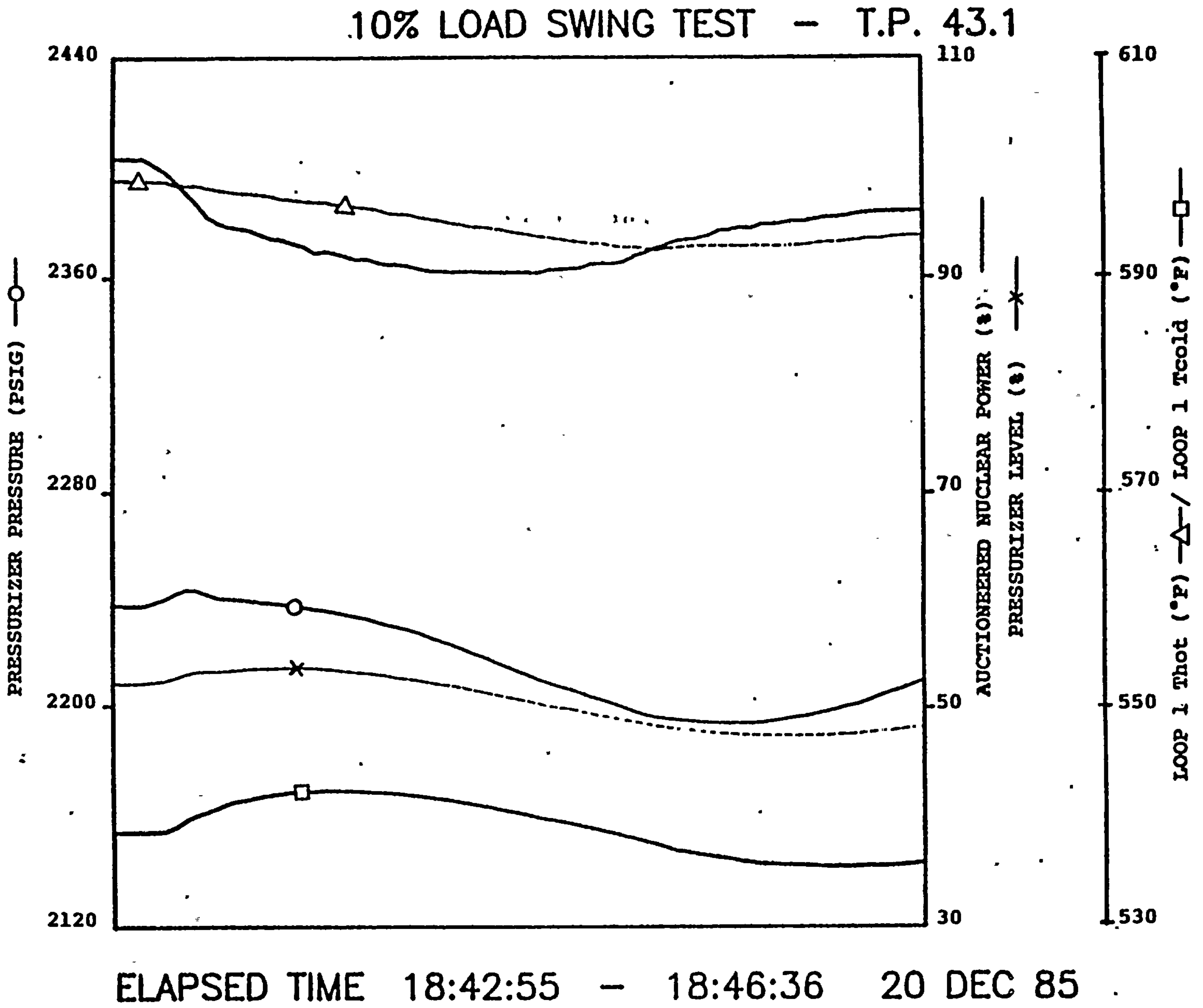

10% LOAD SWING TEST - T.P. 43.1
2440
2360
2280
2200
2120
PRESSURIZER PRESSURE (PSIG)
110
90
70
50
30
AUCTIONEERED NUCLEAR POWER (%)
PRESSURIZER LEVEL (%)
610
590
570
550
530
LOOP 1 Thot (°F) / LOOP 1 Tcold (°F)
ELAPSED TIME 18:42:55 - 18:46:36 20 DEC 85

FIGURE 40B

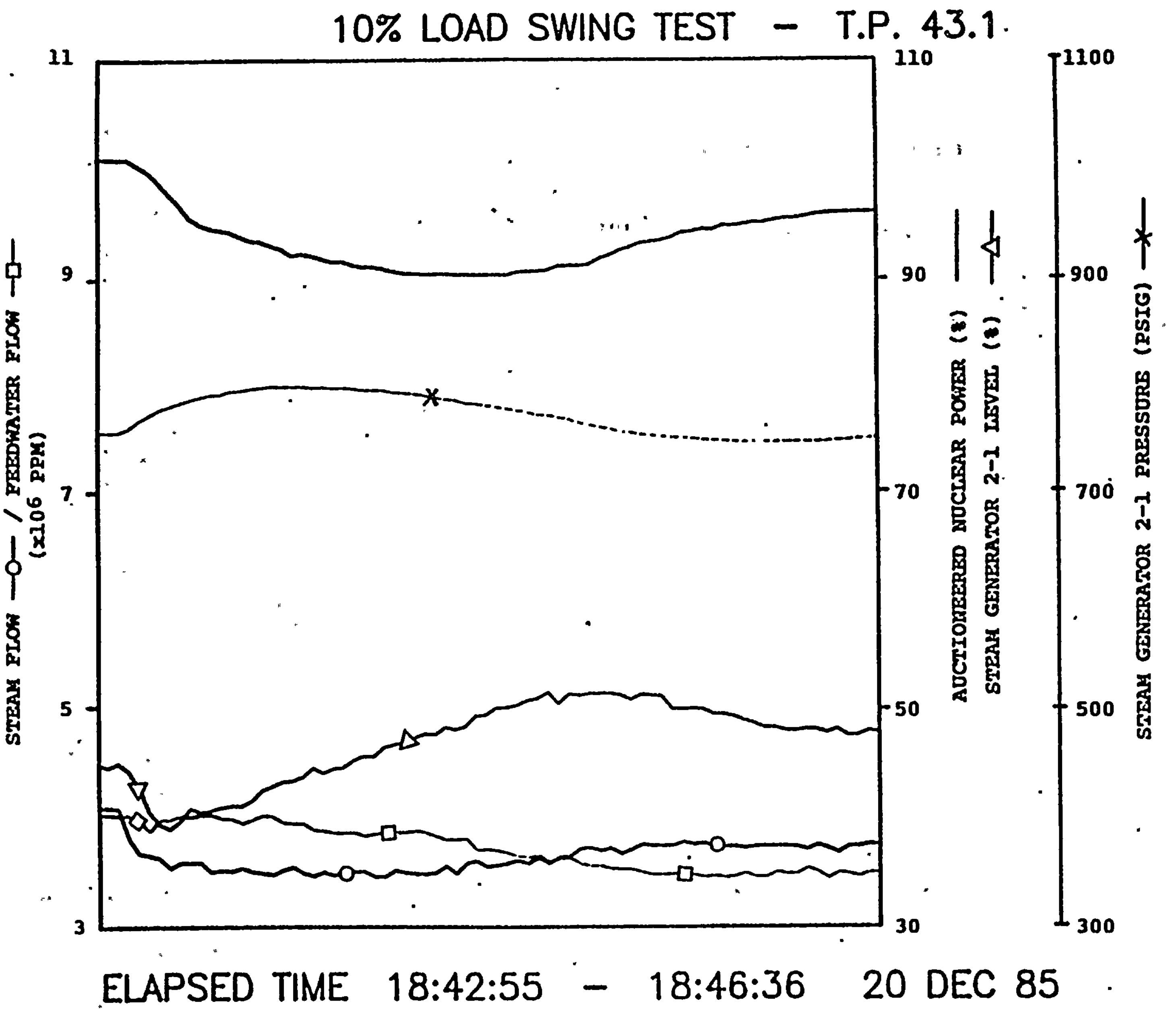

10% LOAD SWING TEST - T.P. 43.1
STEAM FLOW —O— / FEEDWATER FLOW —□—
(x106 PPM)
AUCTIONEERED NUCLEAR POWER (%)
STEAM GENERATOR 2-1 LEVEL (%)
STEAM GENERATOR 2-1 PRESSURE (PSIG)
11
9
7
5
3
110
90
70
50
30
1100
900
700
500
300
ELAPSED TIME 18:42:55 - 18:46:36 20 DEC 85

FIGURE 40C

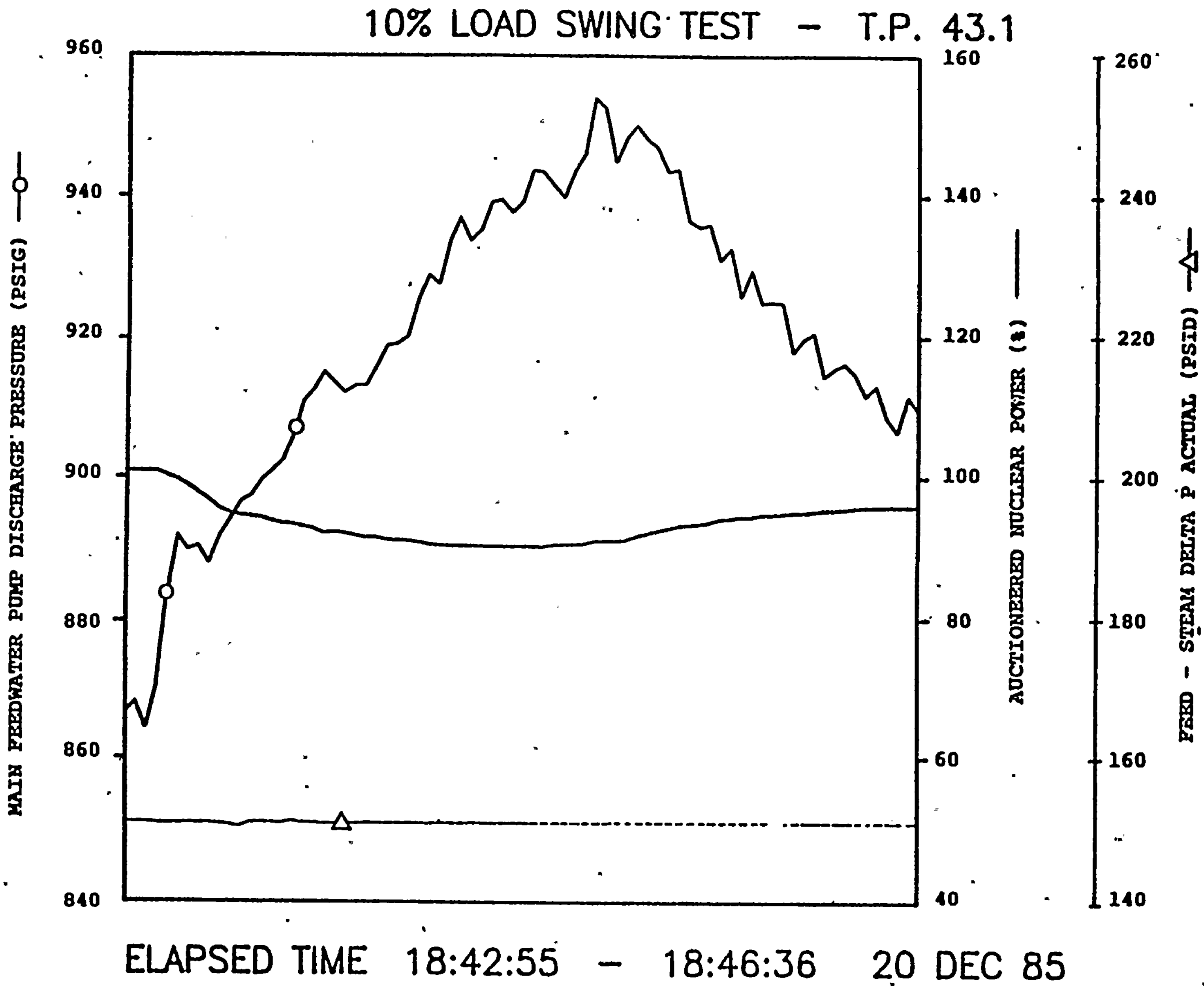
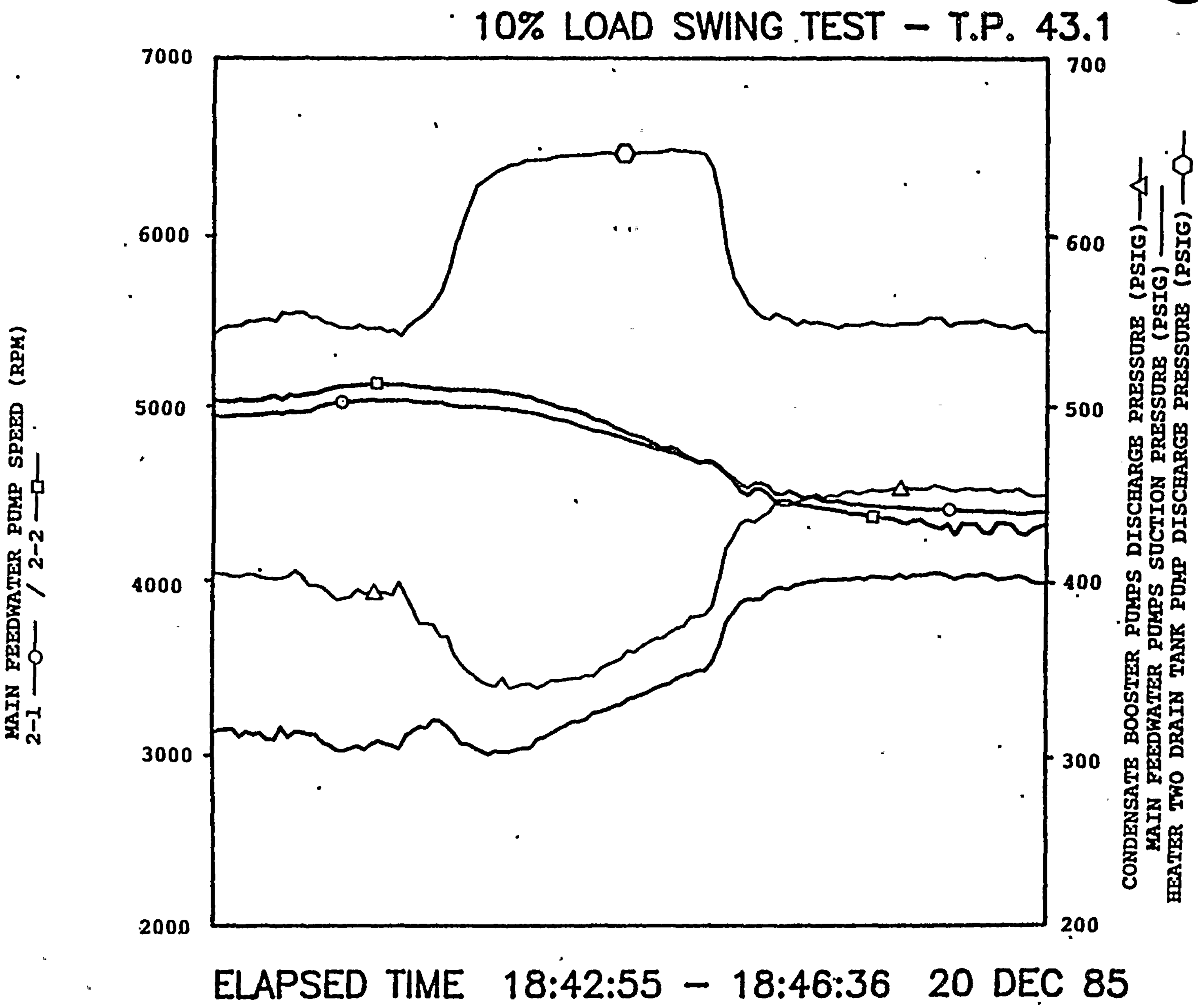

10% LOAD SWING TEST - T.P. 43.1
MAIN FEEDWATER PUMP DISCHARGE PRESSURE (PSIG)
960
940
920
900
880
860
840
AUCTIONEERED NUCLEAR POWER (%)
160
140
120
100
80
60
40
FEED - STEAM DELTA P ACTUAL (PSID)
260
240
220
200
180
160
140
ELAPSED TIME 18:42:55 - 18:46:36 20 DEC 85

FIGURE 40D



10% LOAD SWING TEST - T.P. 43.1
7000
6000
5000
4000
3000
2000
700
600
500
400
300
200
MAIN FEEDWATER PUMP SPEED (RPM)
2-1 / 2-2
CONDENSATE BOOSTER PUMPS DISCHARGE PRESSURE (PSIG)
MAIN FEEDWATER PUMPS SUCTION PRESSURE (PSIG)
HEATER TWO DRAIN TANK PUMP DISCHARGE PRESSURE (PSIG)
ELAPSED TIME 18:42:55 - 18:46:36 20 DEC 85

FIGURE 40E

5.23 Test Procedure No. 4.6 - Steam Generator Moisture Carryover Test

TEST OBJECTIVE

The objective of this test was to determine the moisture carryover (MCO) from the Steam Generators at full power.

TEST DESCRIPTION

The feedwater to the Steam Generators was charged with a radioactive tracer, Sodium-24, in the form of sodium carbonate. The percent moisture carryover at 100% RTP was determined by measuring the activity of sodium in the main steam lines, Heater 2 Drain Tank Pump discharge, condensate lines and steam generator blowdown lines with the Condensate Polishers in service.

The Sodium-24 tracer was mixed with demineralized water and the injection tank's contents injected downstream of the H.P. Feedwater Heaters as close as possible to the steam generators and still get adequate mixing. The feedwater sampling point was temporarily isolated for the duration of this test and sampling provision made upstream of the Sodium-24 tracer injection point.

After injection of the tracer source was completed, approximately one hour was allowed to obtain good mixing of the tracer within the Steam Generators and the secondary system. Steam Generator blowdown via the Blowdown Demineralyzers to the condenser was established for approximately 15 minutes to ensure that representative blowdown samples were obtained. Three sets of simultaneous samples were collected approximately 10 minutes apart as follows:

- All four Steam Generator Main Steam lines
- All four Steam Generator Blowdown lines
- Condensate
- Heater Two Drain Tank Pump Discharge
- Feedwater Heater 2-1B inlet drain

Simultaneously with sample collection, three sets of Operator Heat Balances, STP R-2B, were performed and some additional plant data taken to verify stable plant conditions. The three sets of samples obtained were analyzed to determine Na-24 activity.

TEST RESULTS

The MCO and calorimetric RTP were determined at 100% RTP and tabulated below:

Sample Set	MCO (%)	RTP (%)
1	0.099	98.43
2	0.105	98.50
3	0.106	98.52
Average	0.103	98.48

The 0.103% MCO easily met the required Acceptance Criterion of less than 0.25% at 100% RTP.

5.24 Test Procedure No. 43.3 - Large Load Reduction Tests

TEST OBJECTIVE

The objective of this test was to verify the ability of the primary and secondary plant, and the automatic reactor control systems, to sustain a 50% step load reduction from 75% and 100% of full power.

TEST DESCRIPTION

This test procedure was performed at the 75% power and 100% power test plateaus. In each case, the plant was verified to be at steady state and on automatic control prior to the start of the transient. A large load reduction (50% of full load or approximately 560 MW) was then initiated at the maximum turbine load controller rate of 2200 MWe per minute.

The transient was evaluated for acceptable dynamic response. In addition, the interactions between the control systems were studied for possible setpoint changes to improve transient response.

TEST RESULTS

The large load reduction test at 75% power resulted in reactor trips in the first two attempts. The following is a listing of the sequence of events explaining the problems experienced during this test and their resolutions:

November 26, 1985: Approximately five seconds into the initiation of the transient, an actuation of the 35% Atmospheric Steam Dump valves caused a reactor trip and safety injection actuation due to high steam flow coincident with low steam header pressure. Appropriate emergency procedures were followed and the unit was stabilized in Mode 3. Subsequently, a design change was implemented to modify the steam dump program.

December 1, 1985: Upon initiation of 50% load rejection transient from 75% power level, the feedwater control system did not respond properly, so the control room operators took manual control of the Main Feedwater Pumps' Master Controller and S/G 2-2 feedwater regulating valve to maintain steam generator level. The subsequent heating of the cool feedwater introduced into the steam generators during the transient caused Steam Generator 2-2 level to swell to the high level trip setpoint, resulting in a turbine and reactor trip. Appropriate emergency procedures were followed and the unit was stabilized in Mode 3. Subsequently, setpoint changes to the feedwater control system were implemented.

December 7, 1985: Prior to making a third attempt at retest, a new procedure was developed which would test the plants' ability to sustain a much smaller load reduction of 30%. The 30% load reduction transient was initiated from 75% power at 1838 hours and equilibrium conditions reached at 1843 hours.

5.24 (Continued)

Once verification was made that all Acceptance Criteria of the test were met, the plant was returned to 75% power and a 50% load reduction transient was initiated at 2105 hours and the plant stabilized at 2118 hours. All Acceptance Criteria listed below were met:

1. Reactor and Turbine did not trip.

2. Safety injection did not initiate.

3. Pressurizer safety valves did not lift.

4. Main steam safety valves did not lift.

5. No manual intervention was required to bring plant conditions to equilibrium values following the transient.

Finally, plant response was within expectations, except for Tavg oscillations which were 7 deg. F (peak to valley) versus expected value of less than 5 deg. F. Steam generator level 2-1 undershoot was 20.2% versus an expected value of $\leq$15%. Also, the control rod time of maximum speed was 40.4 seconds versus the expected value of 30 seconds. These variances were not considered to be a problem by Westinghouse and PG&E Engineering.

December 23, 1985: With the plant stable at 100% power, a 50% load reduction transient was initiated at 2000 hours and stable plant conditions achieved at 2012 hours. The large load reduction test at 100% power met all Acceptance Criteria satisfactorly. Also, the plant response was within expectations except for the following variables:

(i) Steam Generator 2-1 level undershoot was 15.5% versus an expected value of $\leq$15.0%.

(ii) Steam dump shutoff time was 11 minutes versus an expected value of less than 8 minutes. This was attributed to steam dump valves PCV-49, 50 & 51 not closing fully.

(iii) Control rod time of maximum speed was 44.7 seconds versus the expected value of 30 seconds.

These variances were not considered to be a problem by Westinghouse and PG&E Engineering.

The response of key plant parameters are tabulated in Table 26 and illustrated in Figures 41A through 41E.

Table 26

Key Plant Parameters During Large Load Reduction Tests

	From 75% Power	From 100% Power
Tave:		
Initial Tave (deg. F)	563.7	569.5
Peak Tave (deg. F)	567.7	572.4
Final Tave (deg. F)	554.2	556.9
Tave Undershoot (deg. F)	1.3	0.4
Tave Oscillations (deg. F)	7 *	very small
Pressurizer Pressure:		
Initial Pressure (psig)	2231	2230
Maximum Pressure (psig)	2262	2275.8
Minimum Pressure (psig)	2160	2167.8
Maximum - Initial (psid)	31	45.8
Initial - Minimum (psid)	71	62.2
Steam Generator Level:		
Initial S/G 2-1 Level (%)	45	44
Maximum S/G 2-1 Level (%)	55	57
Minimum S/G 2-1 Level (%)	24.8	28.5
Maximum - Initial (%)	10.0	13.0
Initial - Minimum (%)	20.2 *	15.5 *
Control Rod Speed:		
Time of Maximum Speed (sec)	40.4 *	44.7 *
Steam Dumps:		
Actuation and Modulation	yes	yes
Cycling	no	no
Shutoff Time (min)	5	11 *

Expected Responses:

1. The Tave peak should be less than 5 deg. F above the initial value, while the Tave undershoot should be less than 3 deg. F below the final value. Tave oscillations should be small and decreasing.
2. Pressurizer pressure should not vary more than +80 psi and -100 psi from the initial pressure.
3. Steam Generator levels should not vary more than ±15% from the initial value.
4. Maximum control rod speed should exist for approximately 30 seconds.
5. Steam dumps should actuate and modulate and shut off within 8 minutes, with no cycling.

* As described in Section 5.24, these deviations from the expected responses were judged to be acceptable.

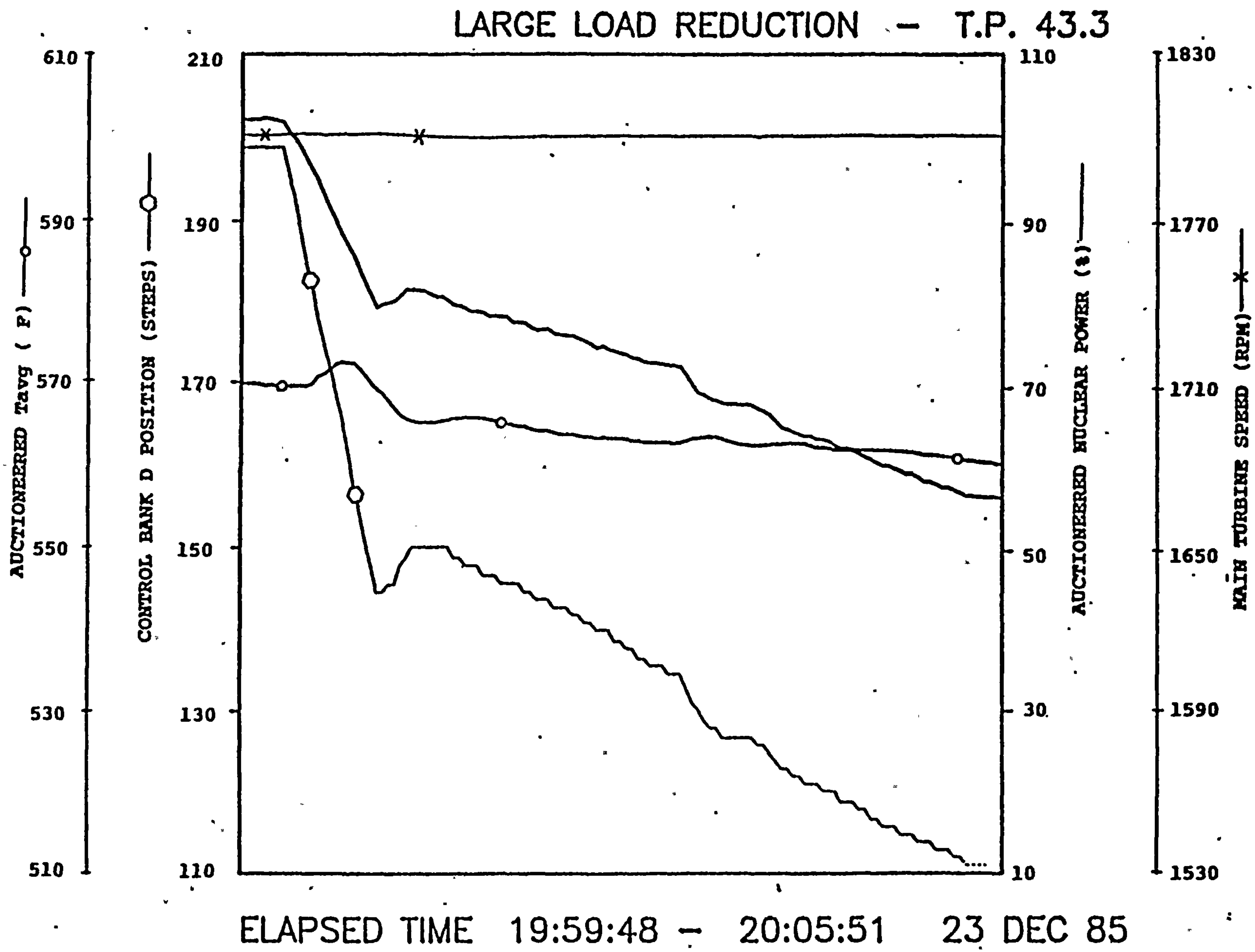

LARGE LOAD REDUCTION - T.P. 43.3
AUCTIONEERED Tavg (F)
610
590
570
550
530
510
CONTROL BANK D POSITION (STEPS)
210
190
170
150
130
110
AUCTIONEERED NUCLEAR POWER (%)
110
90
70
50
30
10
MAIN TURBINE SPEED (RPM)
1830
1770
1710
1650
1590
1530
ELAPSED TIME 19:59:48 - 20:05:51 23 DEC 85

FIGURE 41A

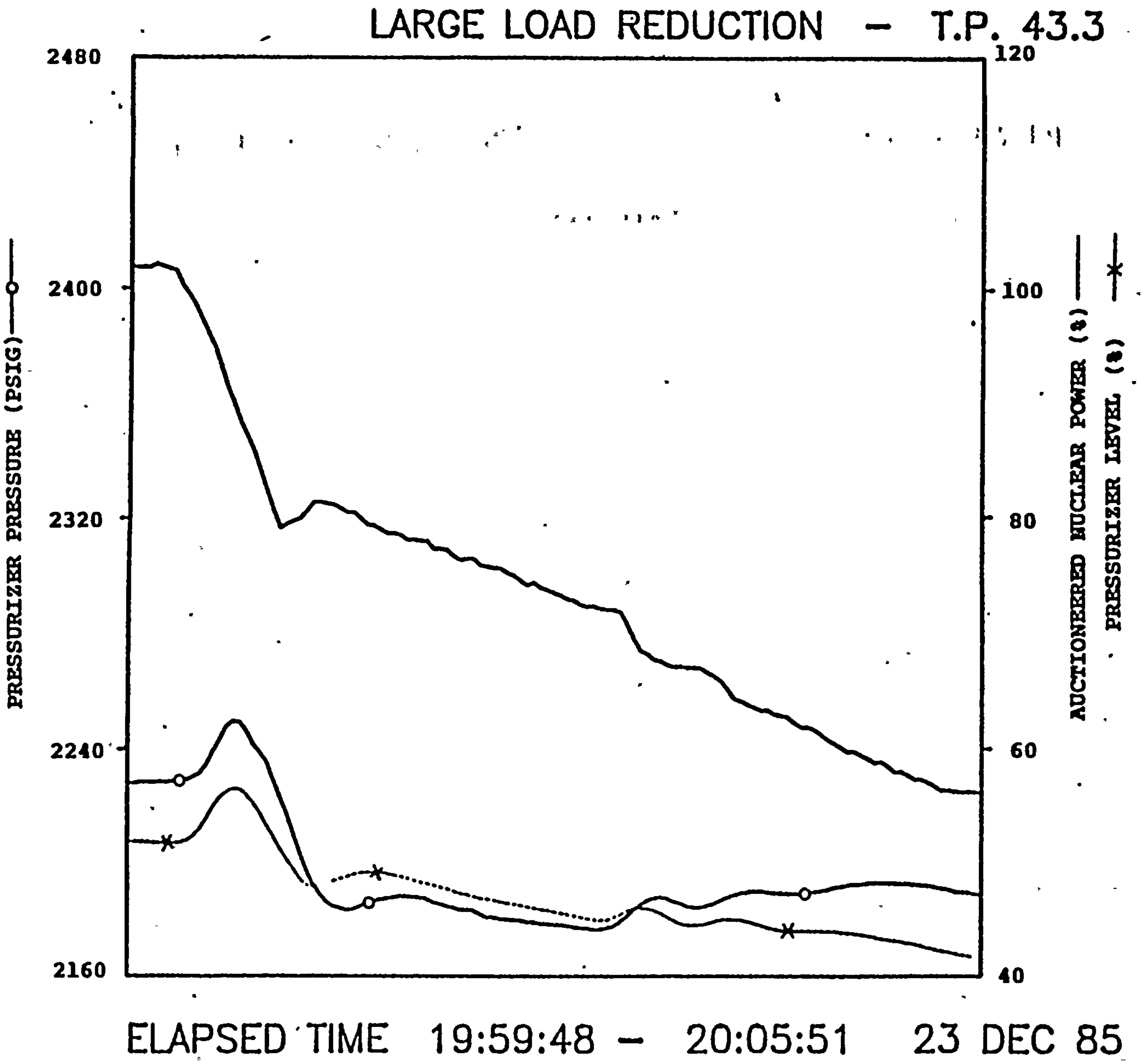

LARGE LOAD REDUCTION - T.P. 43.3
2480
2400
2320
2240
2160
PRESSURIZER PRESSURE (PSIG)
120
100
80
60
40
AUCTIONEERED NUCLEAR POWER (%)
PRESSURIZER LEVEL (%)
ELAPSED TIME 19:59:48 - 20:05:51 23 DEC 85

FIGURE 41B

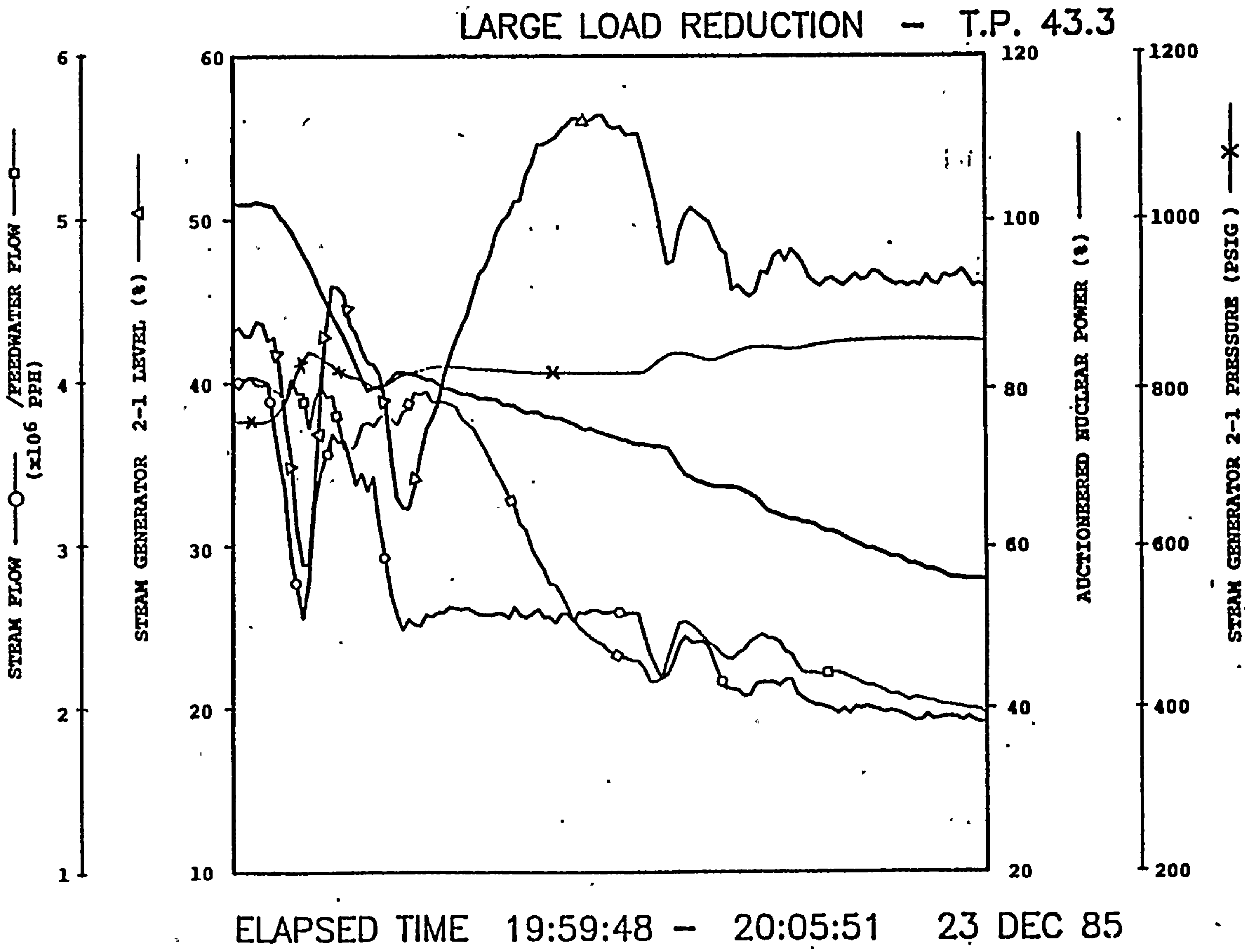

LARGE LOAD REDUCTION - T.P. 43.3
STEAM FLOW /FEEDWATER FLOW (x10^6 PPH)
STEAM GENERATOR 2-1 LEVEL (%)
AUCTIONEERED NUCLEAR POWER (%)
STEAM GENERATOR 2-1 PRESSURE (PSIG)
1
2
3
4
5
6
10
20
30
40
50
60
20
40
60
80
100
120
200
400
600
800
1000
1200
ELAPSED TIME 19:59:48 - 20:05:51 23 DEC 85

FIGURE 41C

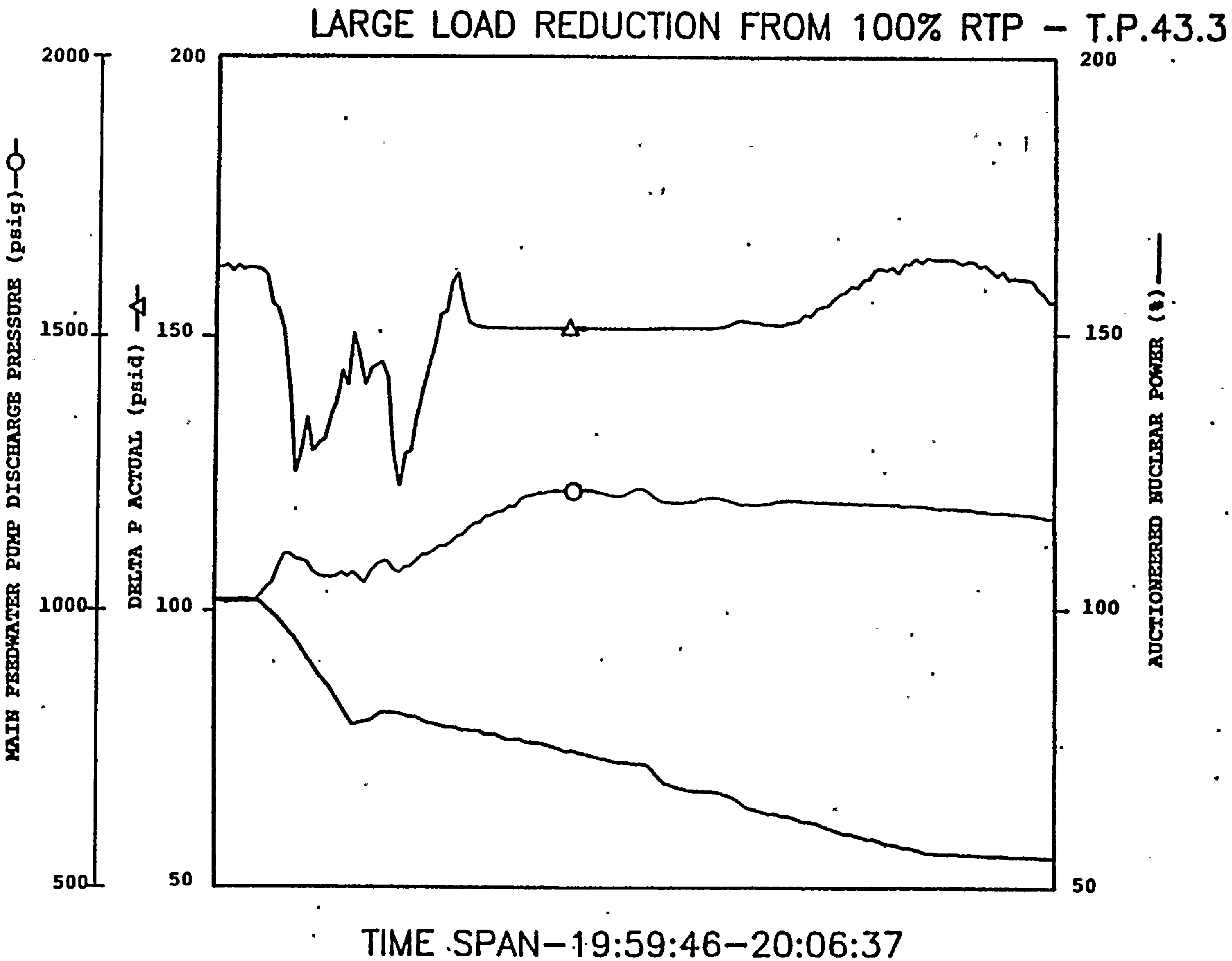

LARGE LOAD REDUCTION FROM 100% RTP – T.P.43.3
MAIN FEEDWATER PUMP DISCHARGE PRESSURE (psig)
2000
1500
1000
500
DELTA P ACTUAL (psid)
200
150
100
50
AUCTIONEERED NUCLEAR POWER (%)
200
150
100
50
TIME SPAN–19:59:46–20:06:37

FIGURE 41D

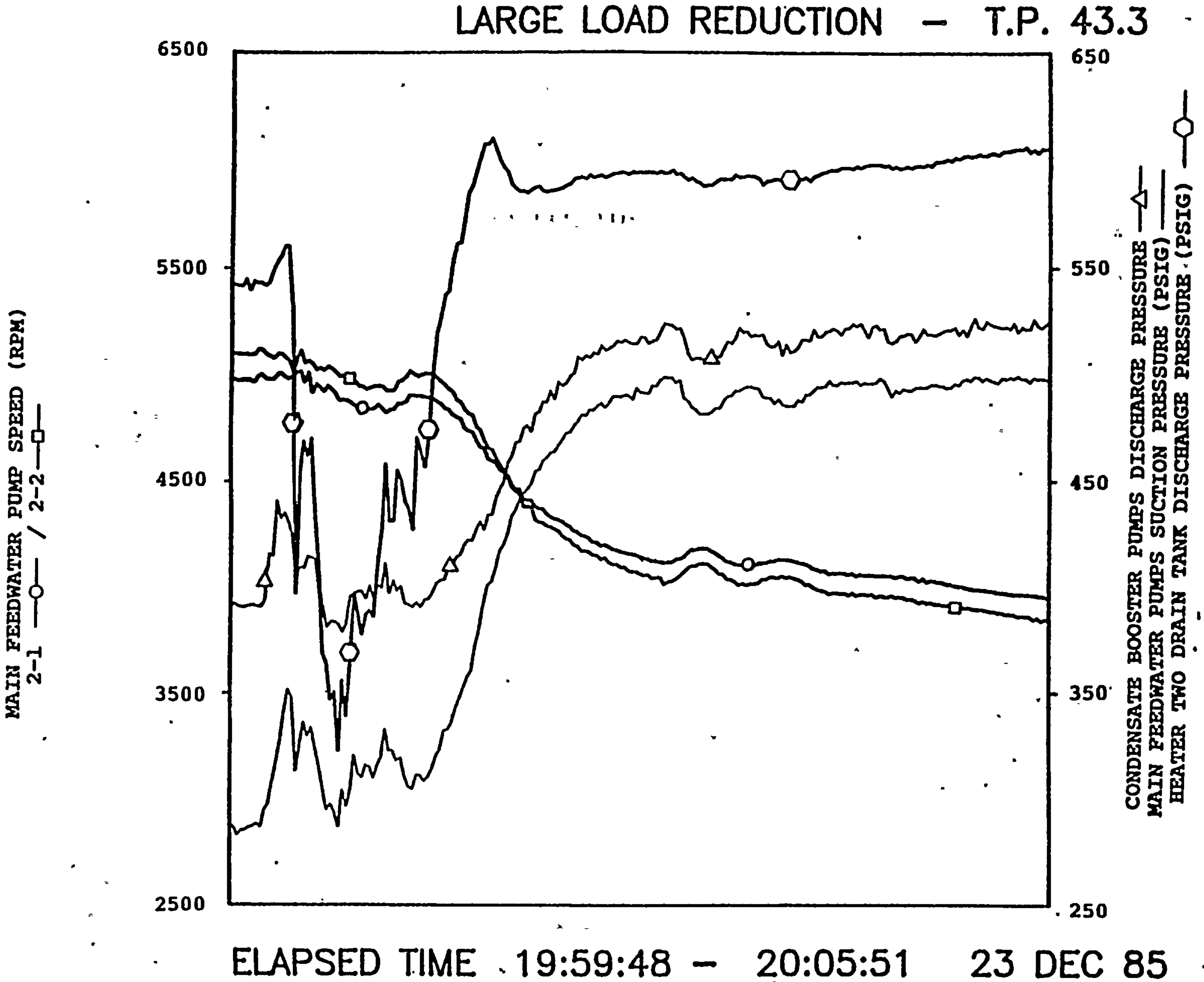

LARGE LOAD REDUCTION - T.P. 43.3
6500
5500
4500
3500
2500
650
550
450
350
250
MAIN FEEDWATER PUMP SPEED (RPM)
2-1 / 2-2
CONDENSATE BOOSTER PUMPS DISCHARGE PRESSURE
MAIN FEEDWATER PUMPS SUCTION PRESSURE (PSIG)
HEATER TWO DRAIN TANK DISCHARGE PRESSURE (PSIG)
ELAPSED TIME 19:59:48 - 20:05:51 23 DEC 85

FIGURE 41E

5.25 Test Procedure No. 43.8 - Plant Trip with Loss of Offsite Power Test

TEST OBJECTIVE

The objective of this test was to verify the ability of the primary plant to sustain a turbine trip from approximately 20% power coincident with a loss of offsite power and to bring the reactor to stable conditions following the transient.

TEST DESCRIPTION

With the plant at approximately 20% power and on automatic control, plant trip was initiated by sequentially tripping the turbine from the control console, all four Reactor Coolant Pumps, Pressurizer Heater and the Startup Feeder Breaker to the vital busses.

TEST RESULTS

Approximately thirty seconds after the plant trip was initiated, the main generator tripped, all necessary loads aligned to the Startup bus and the diesel generators auto started and connected to the vital 4160 volt buses. Establishment and effectiveness of Natural Circulation was verified during the conduct of the test by monitoring key primary plant parameters at periodic intervals. Stable Natural Circulation was declared established approximately 30 minutes after the initiation of the plant trip. All Acceptance Criteria listed below were met:

1. Pressurizer Safety Valves did not lift.
2. Safety Injection did not initiate.
3. All control rods were released and were fully inserted.
4. Diesel generators auto started and connected to the vital 4160 volt buses.
5. Post transient equilibrium conditions were satisfied.

5.26 Test Procedure No. 43.4 - Plant Trip From 100% Power

TEST OBJECTIVE

The main objective of this test was to verify the ability of the primary and secondary plant to sustain a unit trip from 100% power and to bring the plant to stable conditions following the transient. In addition, the test determined the overall response time of the Reactor Coolant Hot Leg Bypass Resistance Temperature Detectors (RTDs).

TEST DESCRIPTION

With the plant at 100% power, stable, and on automatic control, the event was initiated by tripping the turbine from the control console.

TEST RESULTS

The following acceptance criteria were met satisfactorily:

1. Pressurizer Safety Valves did not lift.
2. Main Steam Safety Valves did not lift.
3. Safety Injection did not initiate.
4. The Reactor Coolant Pumps did not trip.
5. All control rods released and were fully inserted.
6. The auto power system transfer took place.
7. Nuclear Flux reduced to 15% (or less) of its initial value within 2 seconds after initiation of the Turbine trip.
8. The overall RTD response time was less than 7.3 seconds (actual observed response time was 7.1 seconds).

Responses of key plant parameters are tabulated in Table 27 and illustrated in Figures 42A through 42E.

Actual plant response for pressurizer level, pressurizer pressure and time delay between turbine trip and generator trip did not meet expectations. Minimum pressurizer level was 17.7% despite being expected to remain greater than or equal to 20%. Pressurizer pressure varied between 1970-2232 psig; anticipated variation was between 2000 psig and initial pressurizer pressure of 2232 psig. The low limit was exceeded because Pressurizer Heater group 2-4 was not available during the test.

The time delay between turbine trip and generator trip was 45.0 seconds as opposed to an expected time of 30 seconds.

These deviations from expected values were reviewed by Westinghouse and PGandE Engineering. Both organizations deemed the results acceptable and no setpoint changes were recommended.

Table 27

Key Plant Parameters During Plant Trip from 100% Power

Pressurizer Level: Minimum Level (%)	17.7 *
Pressurizer Pressure: Initial Pressure (psig) Minimum Pressure (psig) Maximum Pressure (psig) Maximum - Initial (psid) Initial - Minimum (psid)	 2232 1970 * 2232 0 262
Tave: At Feedwater Isolation (deg. F) Need for intervention	 554 none
Steam Generator Levels: Minimum S/G 2-1 Level (%) Minimum S/G 2-2 Level (%) Minimum S/G 2-3 Level (%) Minimum S/G 2-4 Level (%)	 <0 (narrow range) <0 (narrow range) <0 (narrow range) <0 (narrow range)
Steam Dump Valves:	modulated closed
Time Delay Between Turbine Trip and Generator Trip (sec.)	45.0 *

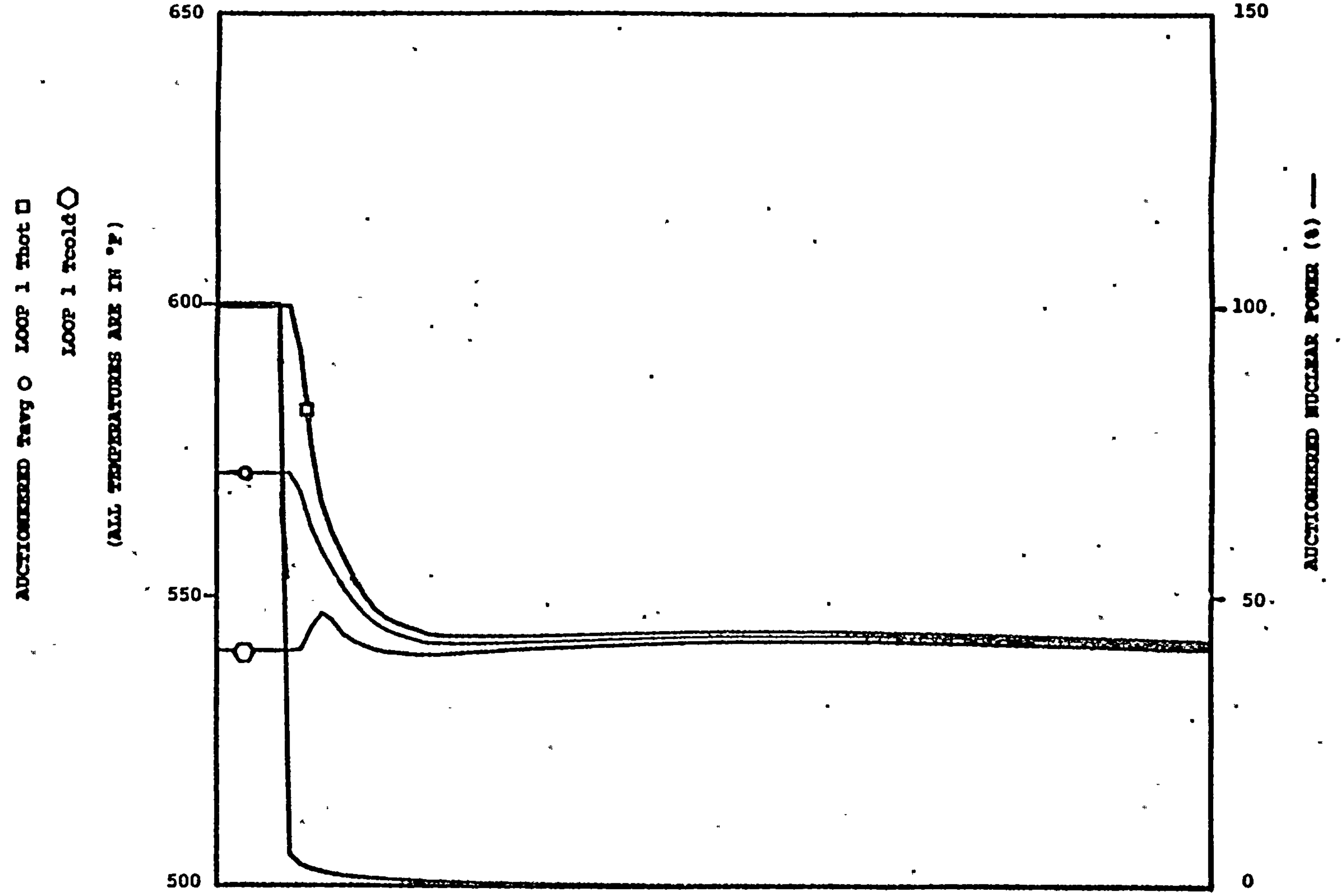
Expected Responses

1. Minimum pressurizer level of 20%
2. Pressurizer pressure less than initial value, but greater than 2000 psig.
3. Tave greater than 547 deg. F at feedwater isolation and no need for manual intervention to steady Tave.
4. Narrow range steam generator levels may drop out of range.
5. Steam dump valves modulating closed.
6. Time delay between turbine trip and generator trip approximately 30 seconds.

* As described in Section 5.26, these deviations from the expected responses were judged to be acceptable.

TURBINE TRIP TEST – T.P. 43:4

13 JAN 86

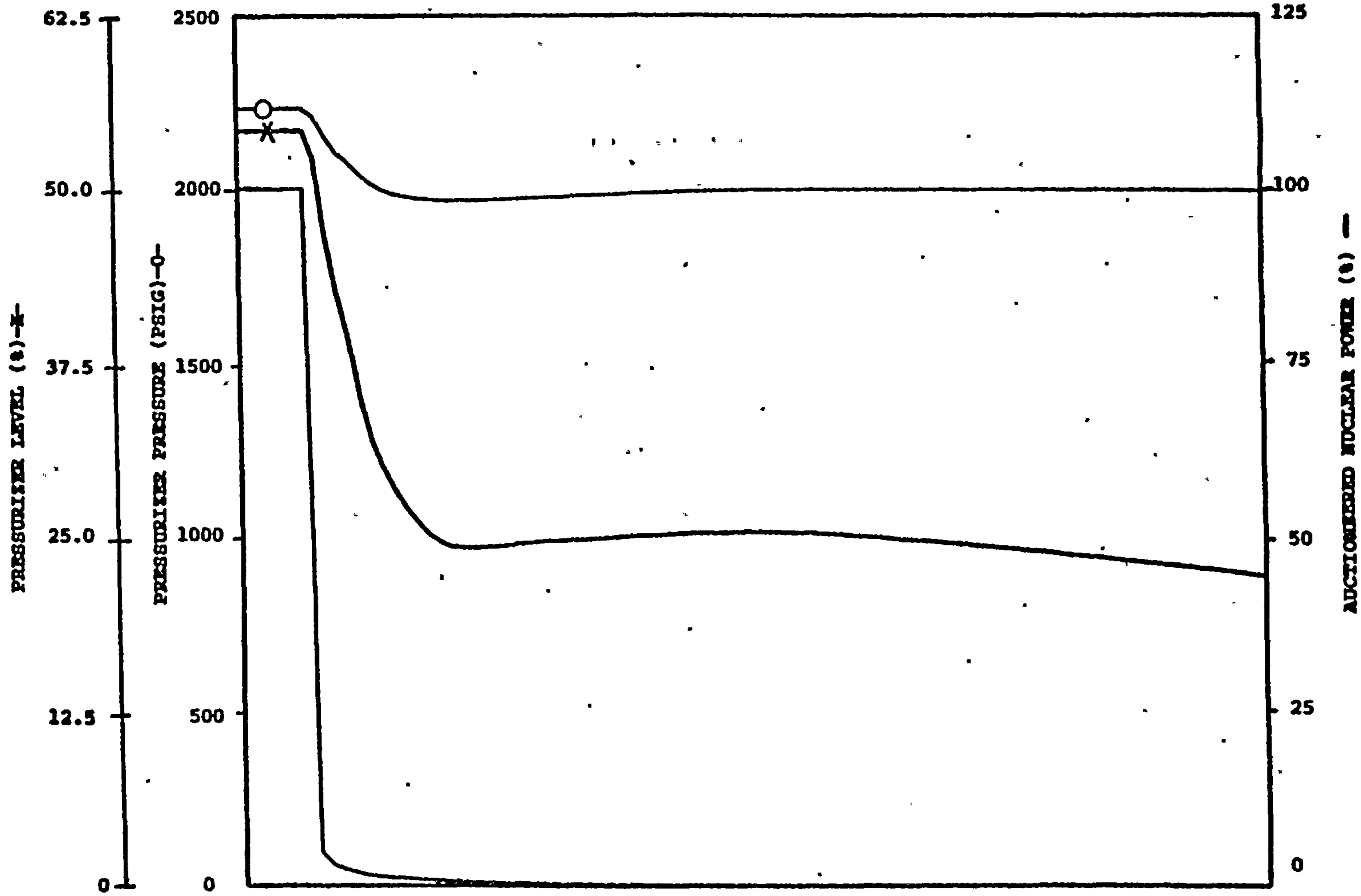


TIME SPAN: 03:29:01 – 03:33:31

FIGURE 42A

TURBINE TRIP TEST – T.P. 43.4

13 JAN 86



TIME SPAN– 03:29:01 – 03:33:31

FIGURE 42B

TURBINE TRIP TEST - T.P. 43.4

13 JAN 86

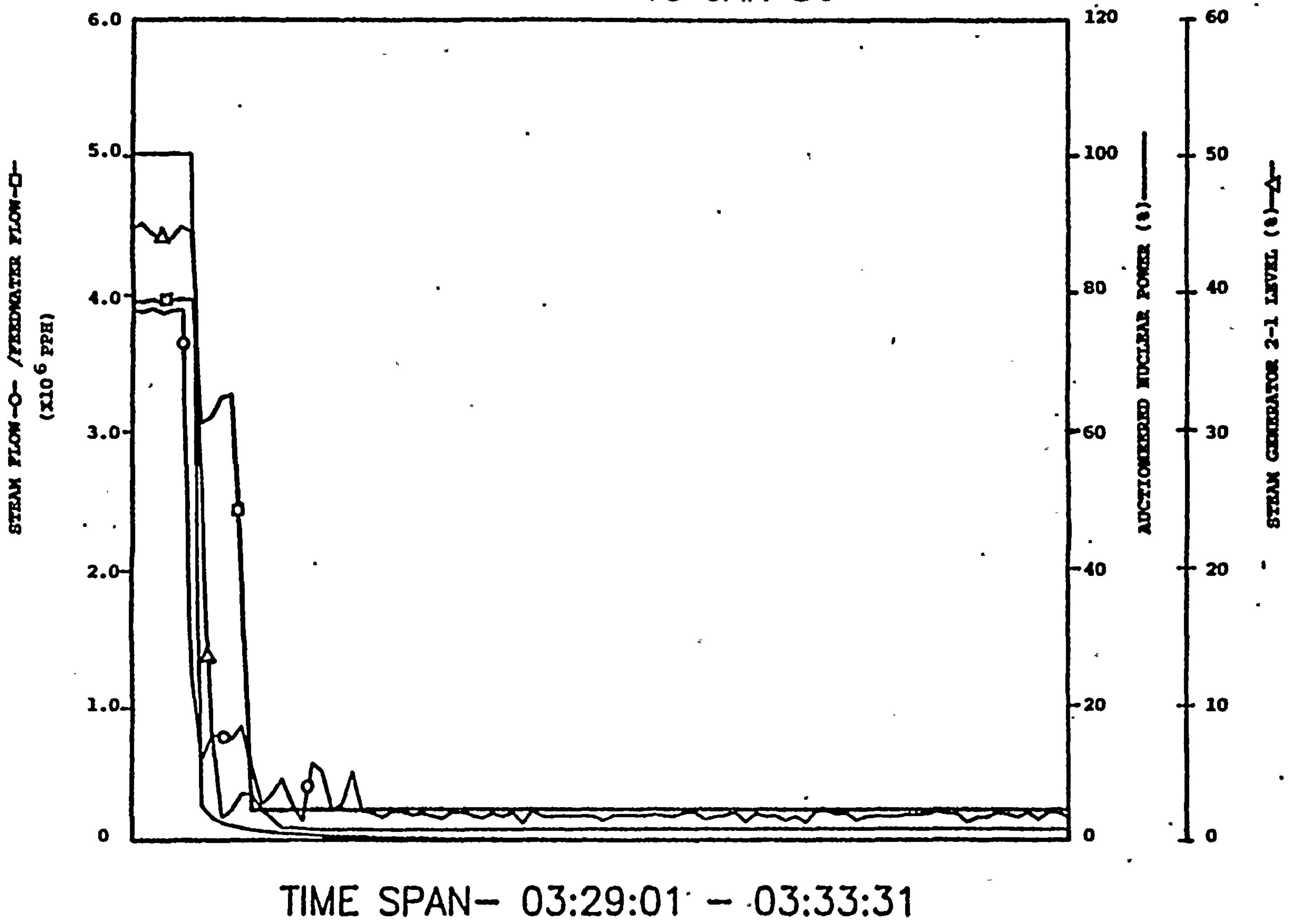
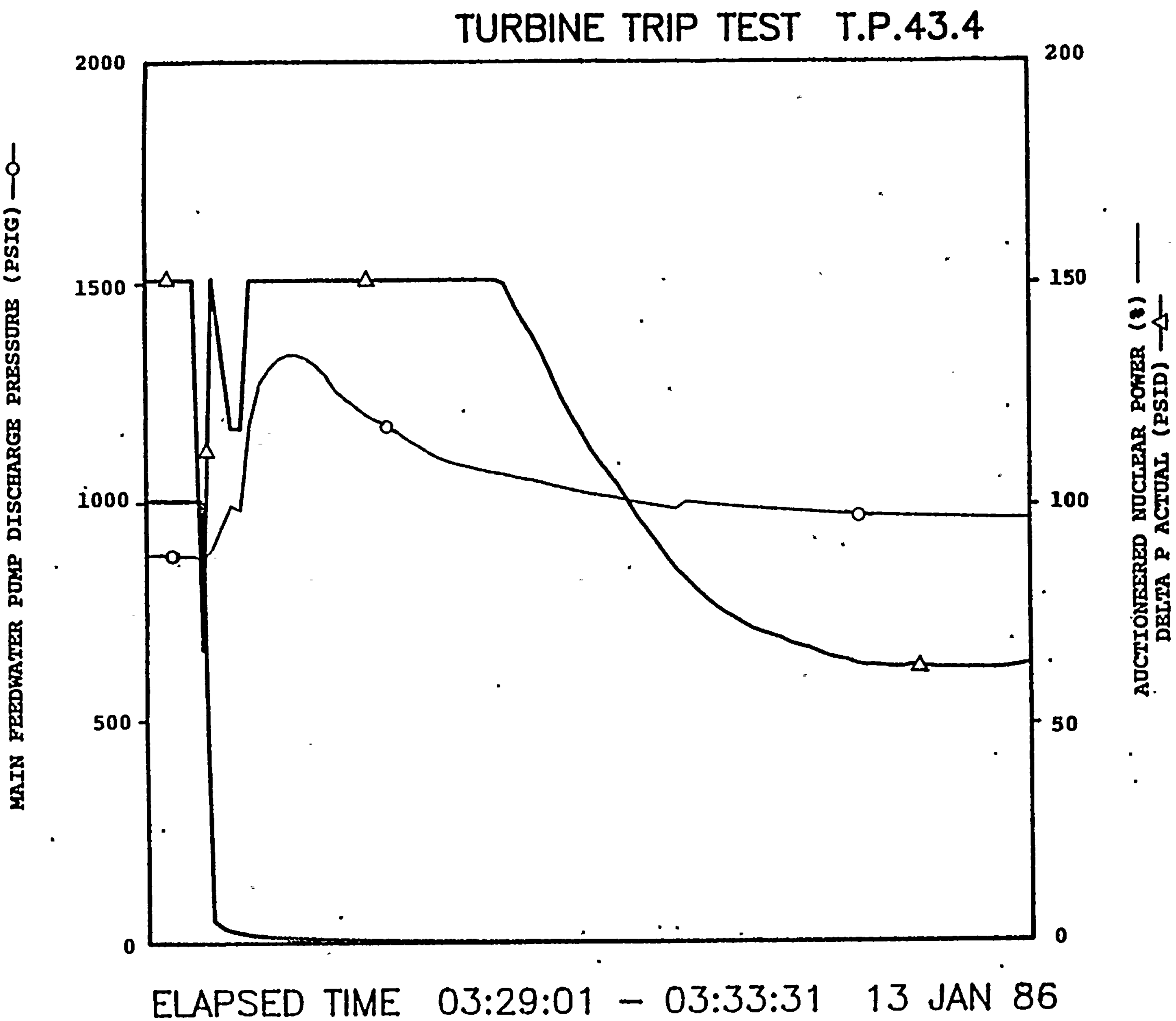


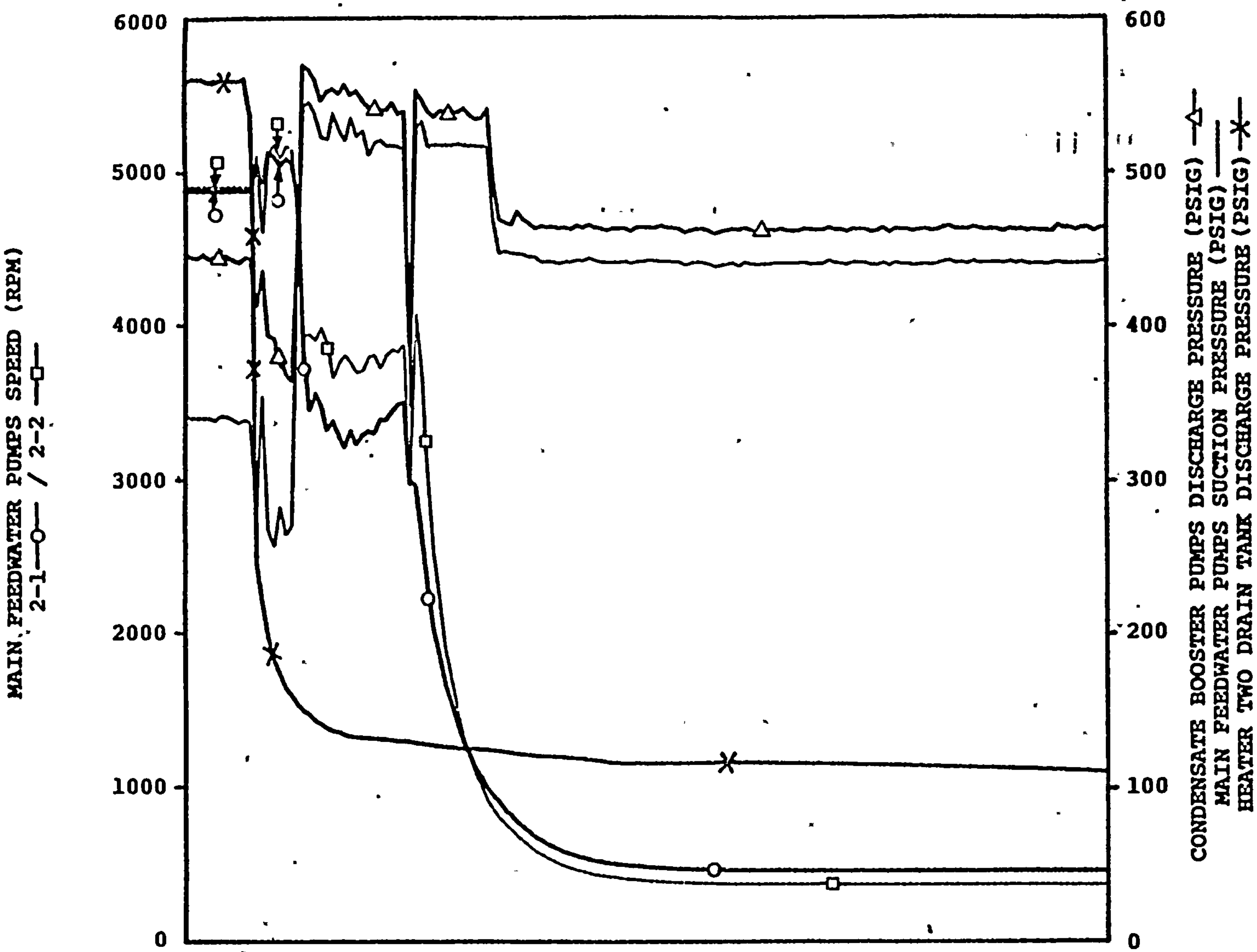
FIGURE 42C



TURBINE TRIP TEST T.P.43.4
2000
1500
1000
500
0
200
150
100
50
0
MAIN FEEDWATER PUMP DISCHARGE PRESSURE (PSIG)
AUCTIONEERED NUCLEAR POWER (%)
DELTA P ACTUAL (PSID)
ELAPSED TIME 03:29:01 – 03:33:31 13 JAN 86

FIGURE 42D

TURBINE TRIP TEST T.P.43.4



ELAPSED TIME 03:29:01 – 03:33:31 13 JAN 86

FIGURE 42E

5.27 Test Procedure No. 43.6 - Nuclear Steam Supply System Acceptance Test

TEST OBJECTIVE

This test procedure had two objectives: first, to demonstrate the reliability of the Nuclear Steam Supply System (NSSS) by maintaining the plant at rated output for 100 consecutive hours without a load reduction or plant trip resulting from an NSSS malfunction, and second, to accurately verify the NSSS warranted output.

TEST DESCRIPTION

The NSSS was stabilized and maintained above 95% of rated thermal output for 100 hours. During this period, log sheets were maintained and various plant parameters were monitored, including main generator electrical output, reactor power level, feedwater flow, steam flow, steam generator pressure, steam generator level, and feedwater temperature. At approximately 50 hours into the 100 hour test, a series of high accuracy secondary heat balance calculations was performed in order to verify that warranty output was being achieved. These heat balances were performed every half-hour over a 4-hour period.

TEST RESULTS

The first attempt at the 100-hour NSSS began on January 10, 1986. After approximately 72 hours, the test had to be terminated due to salt water intrusion into the feedwater system (see Section 5.1).

The second attempt at the 100-hour NSSS Acceptance Test began on March 1, 1986, and was completed without problems on March 5, 1986. As shown in Table 28, key plant parameters remained steady over the full 100 hours. Average NSSS output was 3437.6 MWt, well within the Acceptance Criteria of 3423 MWt (+2%, -5%). During the 4-hour performance test; NSSS output averaged 3434.9 MWt, easily meeting the 3423 MWt ±2% Acceptance Criteria. In addition, steam generator pressures and feedwater temperatures were very close to their respective design values of 805 psia and 432 deg. F, while the average reactor power level was almost 100% RTP.

Table 28

Nuclear Steam Supply System Acceptance Test Data

	Date	Time	Reactor Power (%)	NSSS Power (%)	NSSS Output (MWt)	Gen. Output (MWe)	Steam Generator Pressure (psia) Loop 1	Loop 2	Loop 3	Loop 4	Feedwater Temperature (deg. F) Loop 1	Loop 2	Loop 3	Loop 4
1	3/01/86	1020	100.85	100.8	3452.1	1136	803	804	802	804	429	429	428	429
2	3/01/86	2100	100.32	100.3	3433.9	1117	808	809	808	810	428	428	428	428
3	3/02/86	0300	100.46	100.5	3438.9	1132	811	813	811	813	428	428	428	428
4	3/02/86	1015	100.39	100.4	3436.4	1131	809	810	809	811	428	428	428	428
5	3/02/86	2100	100.37	100.4	3435.6	1130	808	809	808	810	428	428	428	428
6	3/03/86	0300	100.36	100.4	3435.3	1132	813	814	813	814	428	428	428	428
7	3/03/86	1000	100.28	100.3	3432.6	1130	811	813	811	813	428	428	428	428
8	3/03/86	1030	100.38	100.4	3436.1	1120	813	815	813	815	428	428	428	428
9	3/03/86	1100	100.35	100.4	3435.1	1120	814	815	813	815	428	428	428	428
10	3/03/86	1130	100.28	100.3	3432.7	1120	811	812	811	813	428	428	428	428
11	3/03/86	1200	100.36	100.4	3435.2	1120	811	813	811	813	428	428	428	428
12	3/03/86	1230	100.21	100.2	3430.4	1120	811	812	811	812	428	428	428	428
13	3/03/86	1300	100.49	100.5	3439.7	1120	810	811	810	811	428	428	428	428

Table 28

Nuclear Steam Supply System Acceptance Test Data

	Date	Time	Reactor Power (%)	NSSS Power (%)	NSSS Output (MWt)	Gen. Output (MWe)	Steam Generator Pressure (psia)				Feedwater Temperature (deg. F)			
							Loop 1	Loop 2	Loop 3	Loop 4	Loop 1	Loop 2	Loop 3	Loop 4
14	3/03/86	1330	100.43	100.4	3437.6	1120	811	813	811	813	428	428	428	428
15	3/03/86	2115	100.33	100.3	3434.4	1120	815	816	815	815	428	428	428	428
16	3/04/86	0300	100.25	100.2	3431.4	1137	815	817	815	817	428	428	428	428
17	3/04/86	0930	100.82	100.8	3451.2	1144	810	811	809	811	429	429	429	429
18	3/04/86	2120	100.69	100.6	3446.5	1144	809	810	812	811	429	429	428	428
19	3/05/86	0300	100.62	100.6	3444.3	1132	815	815	814	816	429	429	429	429
20	3/05/86	0900	100.29	100.3	3432.9	1137	813	814	813	815	428	428	428	428
AVERAGE			100.43	100.4	3437.6	1128	811	812	811	813	428	428	428	428

5.28 Test Procedure No. 43.2 - Net Load Trip from 100% Power

TEST OBJECTIVE

The main objective of the test was to demonstrate the ability of the primary plant, the secondary plant and the automatic control systems to sustain a net load loss from 100% rated load.

TEST DESCRIPTION

With the plant stable at 100% power conditions, the load rejection was initiated by opening the main transformer high side breakers. Various plant parameters were monitored for analysis of the plant response to the transient. The plant was stabilized using DCPP Emergency Procedure OP AP-2, Full Load Rejection.

TEST RESULTS

Three unsuccessfull attempts were made to conduct this test during the Unit 2 Power Ascension Program and each time the reactor tripped. The following is a chronological summary of the three attempts:

(1) December 25, 1985: At 0100 hours, main generator output breakers were opened to initiate the transient and approximately six seconds later the reactor tripped due to steam generator levels falling below the low low trip setpoint. The unit was stabilized at 0135 in Mode 3 by using the appropriate emergency procedures.

    Post trip transient data analysis indicated that the slow response of the 35 and 40 percent steam dump valves resulted in an increase in steam generator pressure which in turn caused a shrink in steam generator levels below the low low trip setpoint. This slow response of the steam dump valves was corrected by installation of volume boosters on all of the 35 and 40 percent steam dump valves.

(2) January 2, 1986: A second attempt at Net Load trip from 100% nominal power was initiated at 1400 hours. Main Feedwater Pump 2-1 tripped on overspeed approximately six seconds later resulting in a decrease of flow to the steam generators. This decrease in flow resulted in steam generator 2-3 level dropping to the low-low level setpoint and tripping the reactor. When the turbine tripped, the operators initiated a unit trip. Following the unit trip, all four Reactor Coolant Pumps and the two circulating Water Pumps tripped due to stripping of the 12KV buses by startup bus undervoltage relays. Appropriate emergency procedures were followed and the unit was stabilized in Mode 3 at approximately 1435.

    Subsequently, the MFW pump 2-1 control system was flushed, the speed control system was adjusted to improve system response and the manual crossover point was lowered. To defeat the stripping feature of the 12KV buses on unit trip during conduct of the subsequent test, the startup bus undervoltage tripping relays were defeated and the startup bus voltage was monitored.

5.28 (Continued)

(3) March 11, 1986: A third attempt was initiated at 2212 hours and approximately seven seconds later, reactor tripped due to the NIS negative rate trip setpoint being exceeded. Appropriate emergency procedures were followed and the unit was stabilized in Mode 3 at 2305 hours.

Subsequent discussions with Westinghouse indicated that the NIS negative rate trip setpoint is very conservative and was reset by calibrating with a step change of 5% rated thermal power (instead of 2.5% rated thermal power).

Test equipment also indicated that the startup bus voltage dropped below the undervoltage relay's setpoint during the plant's trip. Engineering is investigating the problem.

After performing a safety evaluation, this test was deleted from the Startup Program for the following reasons:

- As the end result of the unsuccessful test is a reactor trip, there is no safety concern both for the plant and the public.
- Very few of the Westinghouse designed plants having 100% load rejection capability have completed this test.
- Subjecting the plant to severe transients repeatedly for a test which has no safety significance is undesirable.

PG and E notified the NRC of the deletion of the 100% Net Load Trip Test from the Unit 2 power ascension test program by letter dated April 11, 1986 (DCL-86-098).

While not all original test objectives of the 100% Net Load Trip Test were satisfied, these tests 1) demonstrated that the responding reactor trip systems functioned as designed and that the reactor can be placed in a safe condition following a 100% load rejection transient, and 2) subjected the turbine generator to a condition of maximum potential for overspeed demonstrating that turbine control systems responded as designed with no overspeed problems identified.

Unscheduled Reactor Trips

Trip	Date	Time	Power	Cause	Comments
1	8/24/85	2034	0%	Control Rod Motor Generator Failure	During Low Power Physics Testing, the voltage relay for the Motor Generator set failed and subsequently tripped the unit.
2	8/29/85	0536	3%	Low S/G Level	S/G levels drifted low during the steam driven AFW Pp endurance run.
3	10/22/85	1617	30%	Unit Differential Relay Tripped	Technician incorrectly actuated the unit differential relay during Power Relay testing.
4	10/24/85	1127	30%	High S/G Level	Loss of feedwater pump suction pressure followed by the start of the stand-by condensate/condensate booster pump caused the S/G levels to increase.
	11/6/85	1918	50%	Low S/G Level	During the performance of T.P. 43.7, Net Load Trip from 50% power, S/G 2-4 level decreased to below 15%.
6	11/9/85	0255	15%	High S/G Level	Closed feedwater pumps recirculation valve with S/G 2-1 feedwater regulating valve in manual. Subsequent increase in feed flow raised the level in S/G 2-1.
7	11/26/85	2310	75%	High Steam Flow coincident with low S/G pressure.	During the performance of T.P. 43.3, Load Rejection from 75% to 25% Power, the 35% Atmospheric Steam Dump actuated causing a steam generator pressure decrease and subsequently a reactor trip and safety injection.

5.29 (Continued)

Trip	Date	Time	Power	Cause	Comments
8	11/28/85	1330	0.5%	Low S/G level coincident with Steam/Feed Flow mismatch.	With one set of S/G level bistables picked up due to a routine test, the main turbine was latched causing the Steam/Feed Flow bistables to momentarily energize thereby subsequently actuating a reactor trip.
9	12/1/85	1021	75%	High/High S/G Level	During the performance of T.P. 43.3, load rejection from 75% to 25% power, a reactor/turbine trip occurred on S/G 2-2 high level.
10	12/2/85	2119	20%	DRPI Data A failure	While doing an emergency power decrease from 50% to 20% power due to kelp blockage at the Intake Structure, DRPI indicated a Data A failure on all of the control rods. Operations subsequently initiated a manual reactor trip.
11	12/21/85	1324	50%	Feed/Steam flow mismatch coincident with Low Level on S/G 2-2	Feedwater regulating valve to S/G 2-2 failed closed due to its solenoid valve being accidentally de-energized by a contractor working near the valve.
12	12/25/85	0101	100%	Low/Low S/G Level	During the performance of T.P. 43.2, Net Load Rejection, the reactor tripped due to Low/Low Steam Generator Levels.
13	12/28/85	2100	2%	Letdown Isolation	While the operators were compensating for Xenon Transients, a letdown isolation occurred thereby minimizing their dilution capabilities. Subsequently, the reactor was manually tripped.

5.29 (Continued)

Trip	Date	Time	Power	Cause	Comments
14	1/2/86	1400	100%	S/G Low/Low Level	During the performance of T.P. 43.2, Net Load Rejection, main feedwater pump 2-1 tripped on over-speed thereby subsequently causing a reactor trip on steam generator 2-3 Low/Low Level.
15	2/22/86	1802	90%	Generator Loss of Field	While testing the main generator voltage regulator, voltage fluctuations activated the generator loss of electrical field relay.
16	3/11/86	2215	100%	Negative Rate Trip	During the performance of T.P. 43.2, Net Load Rejection, the unit tripped due to the negative rate trip bistables being activated during the load reduction.