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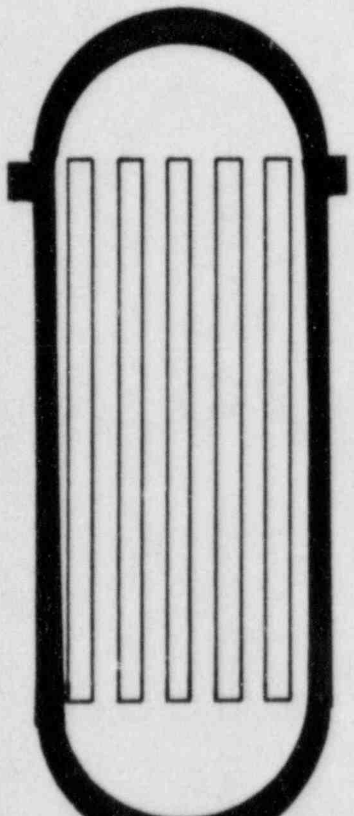
Pacific Gas and Electric Company

DIABLO CANYON NUCLEAR POWER PLANT UNIT I



STARTUP REPORT

LICENSE NUMBER DPR-80

IE26
1/40

FOR THE PERIOD NOVEMBER 15,1983 THROUGH OCTOBER 31,1984

PACIFIC GAS AND ELECTRIC COMPANY

DIABLO CANYON NUCLEAR POWER PLANT
UNIT 1

STARTUP REPORT
TO THE
UNITED STATES
NUCLEAR REGULATORY COMMISSION

LICENSE NUMBER DPR-80

FOR THE PERIOD NOVEMBER 15, 1983
THROUGH OCTOBER 31, 1984

PREPARED BY:

M. J. ANGUS

C. E. BARBEHENN

B. A. LoCONTE

T. S. OHARA

P. G. SARAFIAN

R. A. SAVARD

REVIEWED BY:

M. J. ANGUS

S. G. BANTON

W. E. COLEY

M. N. NOREM

C. G. RAO

L. F. WOMACK

APPROVED FOR ISSUE:

M. N. Norem

M. N. NOREM
GENERAL CONSTRUCTION
LEAD STARTUP SUPERVISOR

L F Womack

L. F. WOMACK
NUCLEAR PLANT OPERATIONS
ENGINEERING MANAGER

DIABLO CANYON NUCLEAR POWER PLANT
UNIT 1 STARTUP REPORT

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SUMMARY

The Diablo Canyon Power Plant Unit 1 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 AFW Pump Endurance Test

5.0 Special Low Power Test Program (Natural Circulation)

Unit 1 fuel loading was performed during the period November 15-24, 1983. After the initial core loading of 193 fuel assemblies was completed, two fuel assemblies had to be removed in order to replace their rod cluster control assemblies, and the fuel assemblies then were reloaded. Except for this delay, no major problems were encountered.

The Pre-Critical Test Program was performed between December 4, 1983 and April 27, 1984. Cold System Tests included Rod Mechanism Timing and Rod Drop Time tests under no flow and full flow conditions. Hot System Tests included Rod Control System tests, Digital Rod Position Indication tests, Rod Mechanism Timing, Rod Drop Time tests under no flow and full flow conditions, Pressurizer Spray and Heater Capacity tests, RTD Bypass Loop Flow tests, Incore Thermocouple/RCS RTD Cross Calibrations, RCS Flow Measurement and RCS 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 licensing hearings, regulatory reviews and equipment problems.

Initial Criticality and Zero Power Physics testing were conducted from April 28 to May 6, 1984. 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 low power testing program.

Following Zero Power Physics testing, a Turbine Driven Auxiliary Feedwater Pump endurance test was performed with the reactor at low power. Next, a series of special low power tests was conducted with the reactor in natural circulation conditions. The first series of tests was conducted from May 19 to May 23, 1984. Portions of the natural circulation test sequence were repeated on July 24 to satisfy previously incomplete operator training requirements.

1.0 FUEL LOADING PROGRAM

Summary

The initial core loading for Diablo Canyon Nuclear Power Plant Uuit 1 consisted of two phases. During the first phase, all of the 193 fuel assemblies were loaded into the reactor vessel within the period November 15-20, 1983. During the second phase, two fuel assemblies, A-35 and A-40, were removed to the Unit 1 Fuel Handling Building for replacement of their rod cluster control assemblies (RCCAs) which failed the drag test. These two fuel assemblies were then reinserted into the reactor vessel, and the second phase was completed within a single eight hour shift on November 24, 1983. Except for the replacement of the two RCCAs, no significant difficulties or delays from equipment problems were encountered. The first phase of the core loading proceeded relatively smoothly with only a few minor equipment problems and short suspensions of operations (less than a single eight hour shift).

1.1 OP B-8D: INITIAL CORE LOADING (PREREQUISITES AND PERIODIC CHECKOFFS)

TEST OBJECTIVE

Operating Procedure B-8D provided a checklist of prerequisites for Unit 1 fuel load along with their scheduling and frequency requirements.

TEST DESCRIPTION

OP B-8D included actual prerequisites for fuel load such as procedures and tests, periodic tests to be completed during fuel load, valve lineup checklists, and chemistry sampling requirements and data sheets.

TEST RESULTS

Preparations began 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.

By retaining signed copies of all prerequisite tests, a complete record of requirements was assembled and verified. This helped to ensure that all open items were completed in an orderly fashion. No major problems or delays were encountered.

1.2 INITIAL FUEL LOADING

Operations

Fuel loading operations commenced at approximately 1545 hours on Tuesday, November 15, 1983. Operations were performed in accordance with plant Operating Procedure OP B-8D, Supplement 1, and commenced with the grappling of the first fuel assembly in the Unit 1 Fuel Handling Building (FHB). The core loading map and core loading sequence that were used had been provided to Pacific Gas and Electric Company (PGandE) by Westinghouse, the NSSS vendor (see Figure 1 for core map).

Prior to being loaded into the core, the fuel assemblies had been wrapped in polyethylene sheaths and dry stored in the FHB spent fuel storage racks and were 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 A, B and C).

Fuel assemblies were placed sequentially into the fuel transfer mechanism and transferred under water along the partially flooded refueling canal into the Containment Building. Assemblies were then grappled and transferred dry to the reactor vessel and lowered into the partially filled vessel. Boron concentration in the vessel was maintained between 2000 and 2100 ppm. Operations were conducted 24 hours a day with three shifts of personnel, and personnel at major work stations were rotated every four hours. All operations were conducted by PGandE personnel with Westinghouse representatives on hand to provide technical support. Fuel loading continued with only minor equipment problems and interruptions until initial loading of all 193 fuel assemblies was completed at approximately 2307 hours on December 20, 1983. This corresponds to an average of 1 1/2 assemblies per hour including all interruptions.

Prior to core loading, the two permanent Source Range Nuclear Instrumentation channels N31 and N32 read only about 0.01 and 0.03 counts per second (cps). After core loading, the count rate had increased to about 7.78 and 10.01 cps respectively. This corresponded to a signal-to-noise ratio of about 778 and 334, well above the required number of 2 for initial criticality. Inverse Count Rate Ratio (ICRR) plots for the fuel load for channels N31 and N32 are shown in Figures 2 and 3.

Three other temporary neutron detectors were also used to continously monitor the neutron count rate. These were lowered into the core and were secured with safety lines. Since the refueling pool was dry (except for some water in the transfer canal area), the top of the reactor vessel was available for personnel access for observation and positioning of the temporary detectors.

After initial loading of the core, a videotape recording was made of the upper nozzle serial numbers on the fuel assemblies to verify proper loading. Burnable poison and thimble plug insert numbers were also verified at this time. An entire row of fuel assemblies was scanned in one pass by movement of the manipulator crane trolley along the bridge and scanning the camera from side to side to record both the fuel assembly number and the insert number. After all initial preparations and equipment adjustments were made, videotaping was completed in about four hours.

Following initial core loading, reactor upper internals were installed and RCCAs were latched with little or no problems. However, during RCCA drag testing measurements, performed to ensure that RCCA fingers would freely pass through the fuel assembly guide tubes, it was found that the RCCAs in fuel assemblies A-35 and A-40 exceeded the drag specifications. After consultations with Westinghouse representatives, it was decided to replace the RCCAs.

Assemblies A-35 and A-40 were unloaded from the core commencing at approximately 0125 hours on November 24, 1983. The two fuel assemblies were transferred to the FHB, and their RCCAs were replaced with Unit 1 spares. Examination of the replaced RCCAs showed numerous vertical scratches and at least one finger visibly bowed. This RCCA was found to have a bowed finger during previous new fuel inspections, however the RCCA had passed drag tests in the FHB prior to fuel load. Following RCCA replacement, the two fuel assemblies were reinserted into the core, and the operation was completed at approximately 0615 hours on November 24. Drag tests were completed successfully on the following day.

Problems

Several minor equipment and related problems caused short delays during Unit 1 fuel loading and are summarized below.

1. Fuel Assemblies - In order to compensate for dimensional tolerances of some fuel assemblies, manual positioning of the manipulator crane was used successfully to mate the assemblies with the core support plate alignment pins.

   The polyethylene sheath that was placed over the fuel assemblies in the FHB, to protect the fuel from construction-related dust and debris, jammed between fuel assembly B-35 and the spent fuel storage rack. An overload indication on the spent fuel pool crane caused operations to be suspended while the sheath was stripped off the fuel assembly by hand. A visual inspection of B-35 revealed no physical damage, and operations were resumed.

2. Manipulator Crane - Manipulator crane hoist motion at slow speeds was somewhat irregular and intermittent and caused momentary overload indications. Lubrication of pulleys with water was successful in alleviating the problem.

   Manipulator operation caused severe electrical disturbances on various circuits around the refueling cavity. Noise spikes were induced in the temporary neutron detectors that were placed in the vessel for fuel loading. The manipulator crane had to be moved from the vessel area and left parked at the extreme end of the refueling cavity near the upender mechanism prior to taking neutron count rate data. This slowed the overall fuel loading operation considerably and added some 5-10 minutes per fuel assembly. A more isolated power source for the neutron detectors is planned for future fuel loadings.

   The Selsyn indicator on the manipulator crane was slightly out of adjustment in certain core locations at the start of fuel loading (approximately 1/8 inch) and appeared to drift further out of adjustment as the fuel loading progressed. The indicator was used for the rough initial alignment of the bridge over a core position, and final adjustment was made using fixed index marks on the operating deck and bridge.

3. Fuel Transfer Mechanism - The carriage drive air motor became frozen, causing operations to be suspended. The air motor was removed, cleaned and lubricated and returned to service. A Design Change Notice was written to modify the air supply lines to maintain positive air pressure on the motor and prevent water from leaking into the air motor.

4. Temporary Neutron Detectors - Three temporary detectors were used during fuel loading to help monitor subcritical multiplication. Several minor problems interfered with operation of these detectors. As mentioned previously, operation of the manipulator crane caused electrical spikes and false indications on the detectors, requiring delays in fuel loading to permit count rates to be determined.

   Minor physical contact with temporary detector cables that were laid around the vessel flange caused false indications on these detectors. Indications included spikes and abrupt continuous changes in count rate. Detector readings had to be renormalized. Placement of one detector close to a hot leg nozzle resulted in flow induced vibration of the detector and cable. A false high count rate indication resulted from this motion. The signal stabilized when the detector and cable were repositioned.

5. Source Range Nuclear Instruments - The two permanently installed source range detectors activated several spurious High Flux at Shutdown alarms with resulting suspension of operations and assemblage of personnel in Containment at the personnel hatch. The causes of several of the spurious alarms were traced to electrical interferences caused by activation of the Unit 1 Code Call System, operation of the Containment Polar Crane and operation of the Movable Incore Detector System. Design changes to alleviate these problems are being evaluated.

6. Underwater Lights - The nominal 1000 watt model had to have several bulbs replaced with lower wattage floodlight bulbs to alleviate a problem with short bulb life.

   The other style of light, a 750 watt model that could not be employed during fuel load, had a plexiglass lens cover that was loosely retained by four cap screws. The cap screws are being replaced with longer screws to prevent loss of the cover into the RCS under high flow conditions. This has occurred at other plants.

   Another problem encountered with moving underwater lights was motion of the cables disturbing the temporary neutron detector cables and causing false count rate indications as discussed previously. Cable placement became an important factor in ensuring efficient operations.

7. Diesel Generators - Because Unit 1 fuel loading was conducted with one operable diesel, DG 1-2, several minor problems with this diesel caused fuel loading to be suspended temporarily. Problems included Low Turbo-Start Air Pressure and Low Fuel Oil Priming Tank Level

alarms. In the former case, a repair to the starting air compressor was required. Although a redundant starting air compressor was available, the Shift Foreman declared Diesel 1-2 inoperable (Ref. Licensee Event Report-LER 83-27).

8. Instrument Power Inverters - Inverter 1-2 was removed from service for testing. This caused a momentary interruption of power to a Source Range Nuclear Instrument during the load transfer. Fuel loading was suspended temporarily during this period to allow a functional test of the instrument to be performed after the load transfer.

9. Manipulator Crane Auxiliary Hoist - While using the manipulator crane auxiliary hoist for latching and handling control rods, it was noticed that the hoist had not been load tested within 100 hours prior to use for performing RCCA drag tests (LER-83-31).

10. Containment Equipment Hatch - Subsequent to fuel loading and while in Mode 5, an NRC Inspector noted a visible gap at the top of the containment equipment hatch. The hatch had been held in place by four (4) bolts during fuel loading, as required by Technical Specifications; however, the bolt placement and torque had been inadequate. The equipment hatch was secured with additional bolting (LER-83-28).

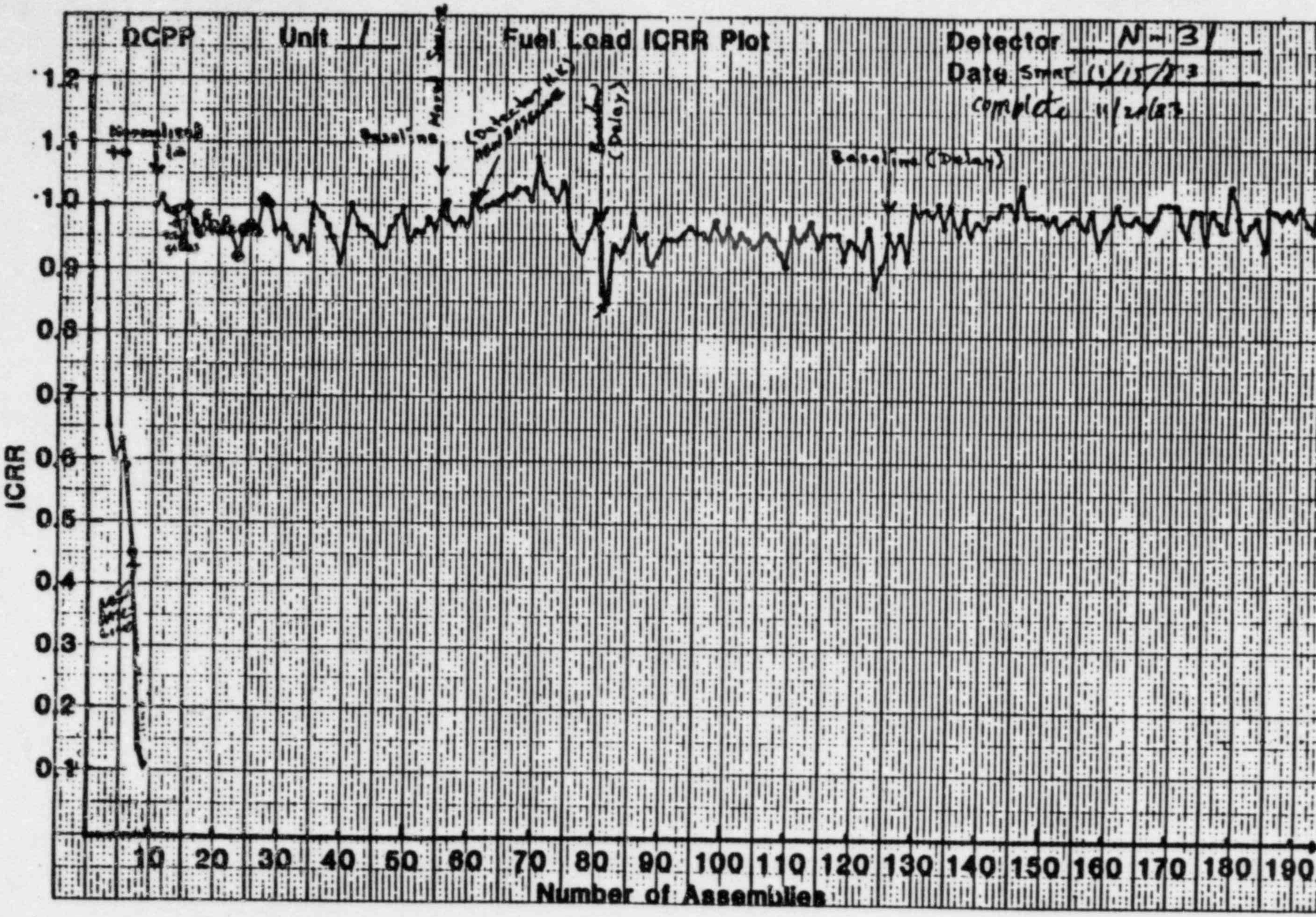
DIABLO CANYON POWER PLANT
UNIT No. 1
CORE MAP

0°

	A	B	C	D	E	F	G	H	J	K	L	M	N	P	R
15					C29	C58	C37	C10	C52	C14	C44				
14			C48	C57	C62	A19	C30	A34	C31	A09	C21	C23	C16		
13		C46	C33	B08	A54	B25	A45	B44	A57	B41	A08	B29	C59	C35	
12		C18	B55	B27	B21	A07	B60	A12	B30	A31	B51	B36	B54	C19	
11	C49	C64	A65	B40	A59	B52	A20	B58	A27	B42	A50	B04	A25	C08	C53
10	C39	A39	B05	A04	B26	A55	B01	A11	B23	A18	B35	A17	B56	A02	C09
9	C24	C11	A48	B64	A56	B02	A44	B17	A47	B33	A64	B50	A58	C61	C20
8	C13	A24	B57	A21	B32	A35	B53	A29	B11	A60	B38	A36	B18	A37	C22
7	C40	C55	A61	B06	A63	B13	A03	B07	A15	B15	A53	B12	A13	C03	C41
6	C34	A62	B14	A46	B63	A10	B31	A26	B10	A41	B62	A05	B37	A06	C38
5	C45	C25	A28	B22	A14	B09	A23	B61	A32	B20	A51	B16	A49	C42	C06
4		C05	B39	B59	B49	A01	B43	A52	B34	A43	B48	B24	B28	C47	
3		C17	C15	B47	A33	B19	A42	B45	A40	B03	A30	B46	C51	C56	
2			C63	C02	C28	A16	C04	A22	C01	A38	C54	C32	C36		
1					C26	C50	C07	C12	C60	C43	C27				

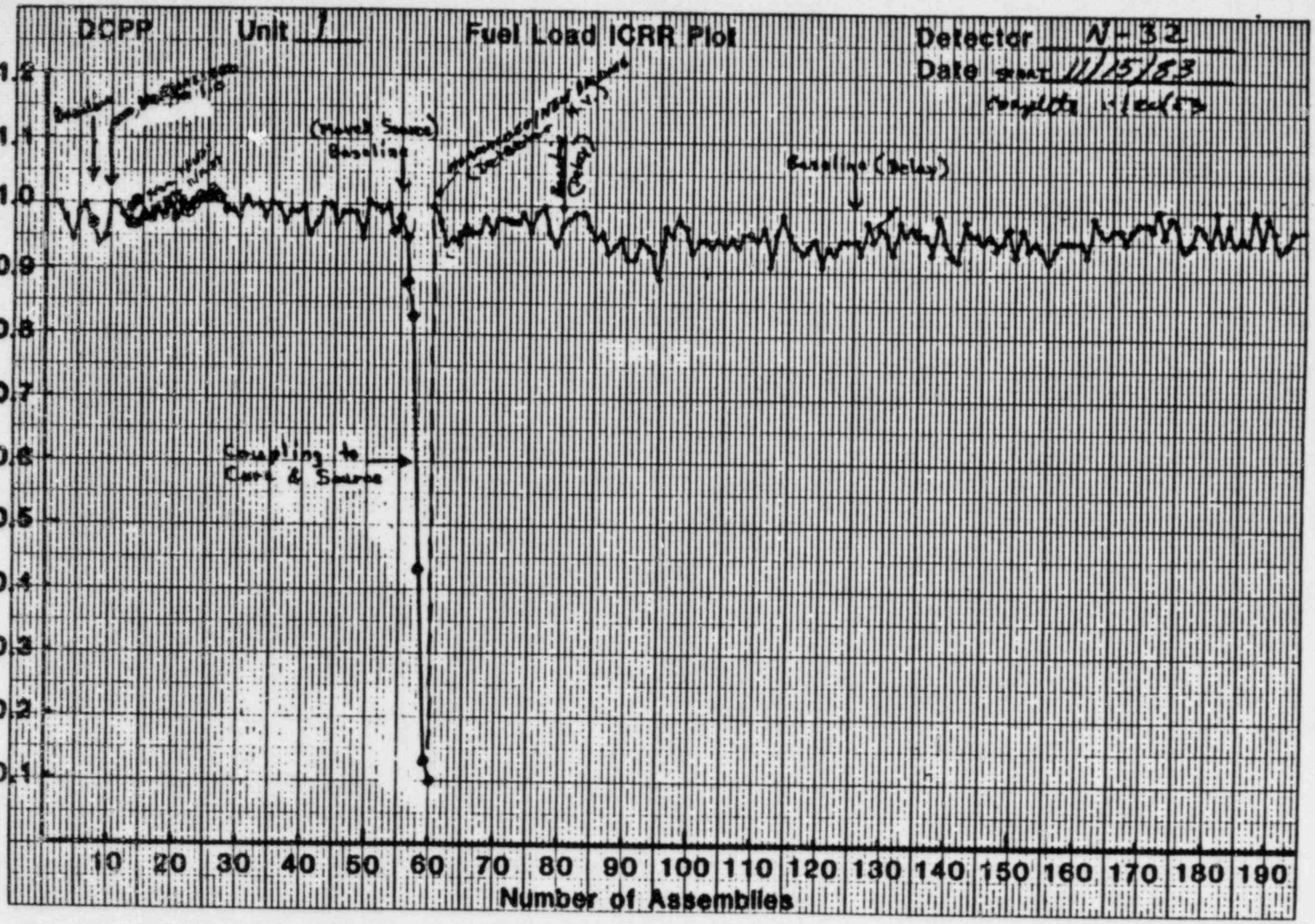
REMARKS Final Fuel Assembly Locations
Core 1
● Primary Source Assembly
⁚⁚ Secondary Source Assembly

Figure 1



DETECTOR N31 ICRR PLOT

Figure 2



DETECTOR N32 ICRR PLOT

2.0 PRE-CRITICAL TEST PROGRAM

Summary

This phase of the Startup Program was divided into three sections.

Cold System Tests
RCS Heatup
Hot System Tests

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, Phase 1 of Rod Drive Mechanism Timing, and Phase 1 of Rod Drop Time Measurements. These were performed during the period from December 4, 1983 to December 10, 1983.

Mode transition surveillance testing began after approval for heatup was granted. RCS Heatup phase started on February 20, 1984 and was completed on March 8, 1984. RCS Heatup included four temperature plateaus: 250 deg. F, 340 deg. F, 450 deg. F, and 547 deg. F.

At each plateau :

1) Piping walkdowns were performed and interference problems were fixed or evaluated before leaving the plateau.

2) Data was taken for Reactor Vessel Level Indication System (RVLIS).

3) Incore Thermocouples and RCS RTD cross calibration was performed (except at 450 deg. F plateau).

Hot System Tests were performed with the RCS at rated temperature and pressure. The tests included 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 March 25, 1984 and were completed on April 27, 1984.

Major delays during the prescribed test program were as follows:

December 10, 1983 to January 25, 1984:	Waiting for approval to heatup.
March 9, 1984 to March 24, 1984:	Replace RCP 1-3 motor.
April 7, 1984 to April 25, 1984:	RCP 1-4 seal repairs, Pressurizer Spray Valve repair and waiting for issuance of the Low Power Test License.

2.1 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. Finally, leak detection and gas purge systems related to the moveable detectors were tested.

TEST RESULTS

All acceptance criteria were met and the system was ready for standard flux mapping at the conclusion of the test.

The main problem involved sticking of the dummy cable due to excessive friction at various points along the detector paths. The solution consisted of cleaning all thimble tubes, and modifying and aligning the isolation valve rack.

2.2 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 accomplished 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.

Cold System Testing was performed from December 5, 1983 to December 9, 1983 at approximately 350 psig and 110 deg. F. Because the Digital Rod Position Indication (DRPI) system had not been declared operable, Technical Specification Special Test Exception 3.10.5 was declared in effect. Each rod was individually operated to verify mechanism timing, brought to 228 steps out of the core to verify DRPI and then dropped to perform rod drop test (ref. T.P. 36.3). This sequence was repeated until all the rods were tested and Special Test Exception 3.10.5 was no longer necessary for mechanism timing tests.

Hot System Testing began on March 26, 1984 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). This part of the test program went smoothly with no problems.

TEST RESULTS

The traces for each mechanism were evaluated immediately following their performance and were determined to be satisfactory.

Listed below are some of the problems encountered and their associated resolutions during the performance of Cold System Testing:

1) DRPI indication problems/replaced encoder cards

2) Blown stationary fuses/replaced fuses

3) Step counter malfunction/cleaned step counters

4) Rod H-10 would not move/lift pole connector pin at the mechanism had retracted into its housing and was repaired

5) Rod D-12 DRPI problem/loose electrical connector pins from the DRPI coils at the head area were repaired

2.3 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 separate 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 on the rods with the slowest and fastest drop times ("A+B" traces only) under all of the above mentioned conditions a minimum of six times.

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. Figures 4 and 5 are examples of the traces obtained.

Since the DRPI could not be declared operable until it was functionally demonstrated that the DRPI system correctly tracked rod position, it was necessary to perform the cold no flow ("A+B" traces) rod drop test in conjunction with the rod mechanism timing test (see T.P. 36.1 - Rod Mechanism Timing Test). This portion of the test was conducted from December 5, 1983 to December 9, 1983 with several equipment problems delaying the test program (see T.P. 36.1). The cold no flow portion of the test was then repeated for the "A&B" traces. The remainder of the test was performed using a method that allowed the "A+B" and "A&B" traces to be taken on one trace. (Figure 6)

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

Cold No Flow	350 psig/110 deg. F	December 10, 1983
Hot Full Flow	2235 psig/ 547 deg. F	March 27, 1984
Hot No Flow	2235 psig/547 deg. F	March 28, 1984

TEST RESULTS

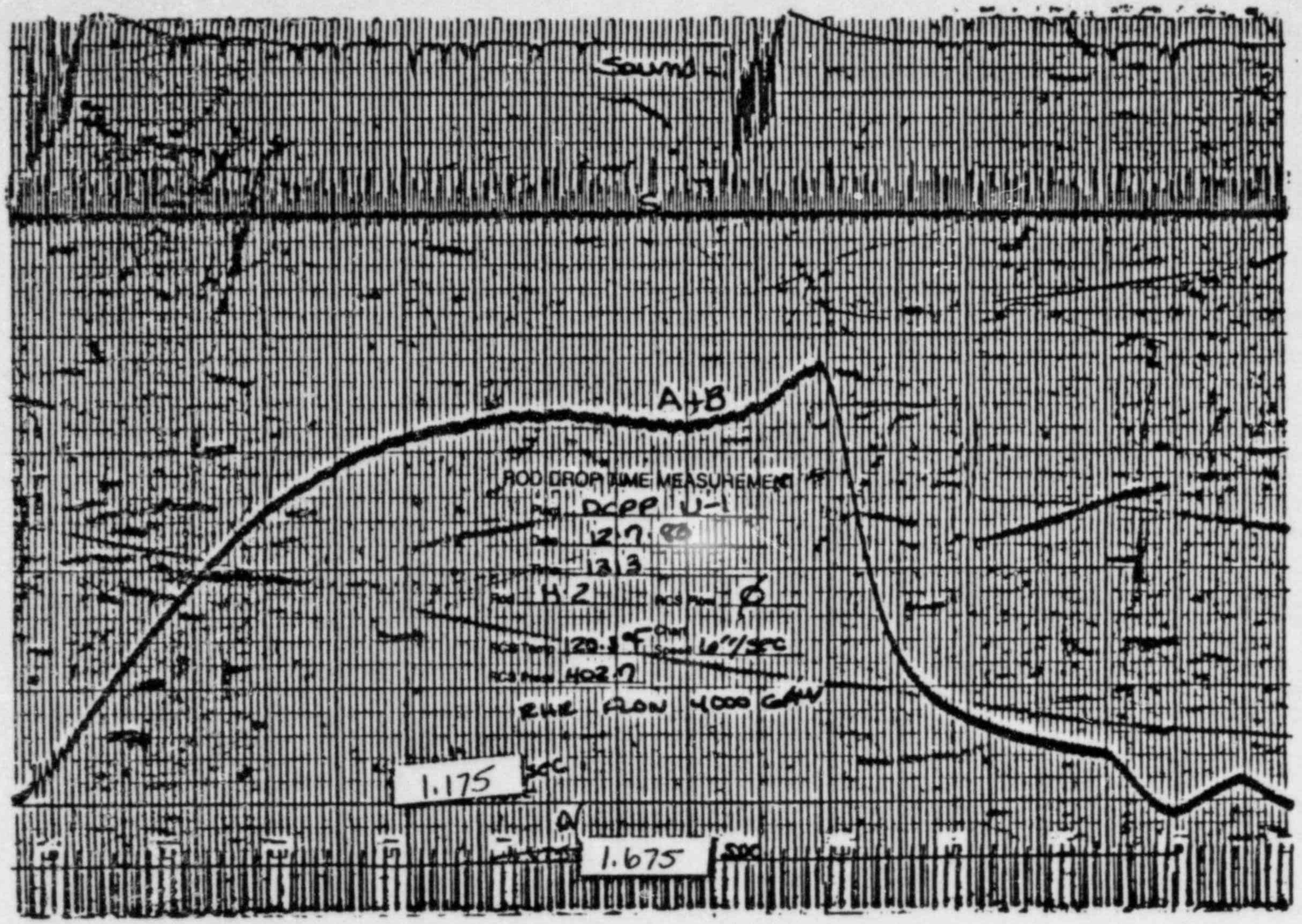
Figures 7 through 10 show the rod drop times for the four plant conditions and Table 1 lists the core average, and slowest and fastest drop times. The traces for each rod drop were evaluated soon after performance and were satisfactory. All rod drop times were well below the Technical Specification Requirements of 2.2 seconds from initiation of event to dashpot entry.

Table 1

Rod Drop Times (Sec.) for Various Plant Conditions

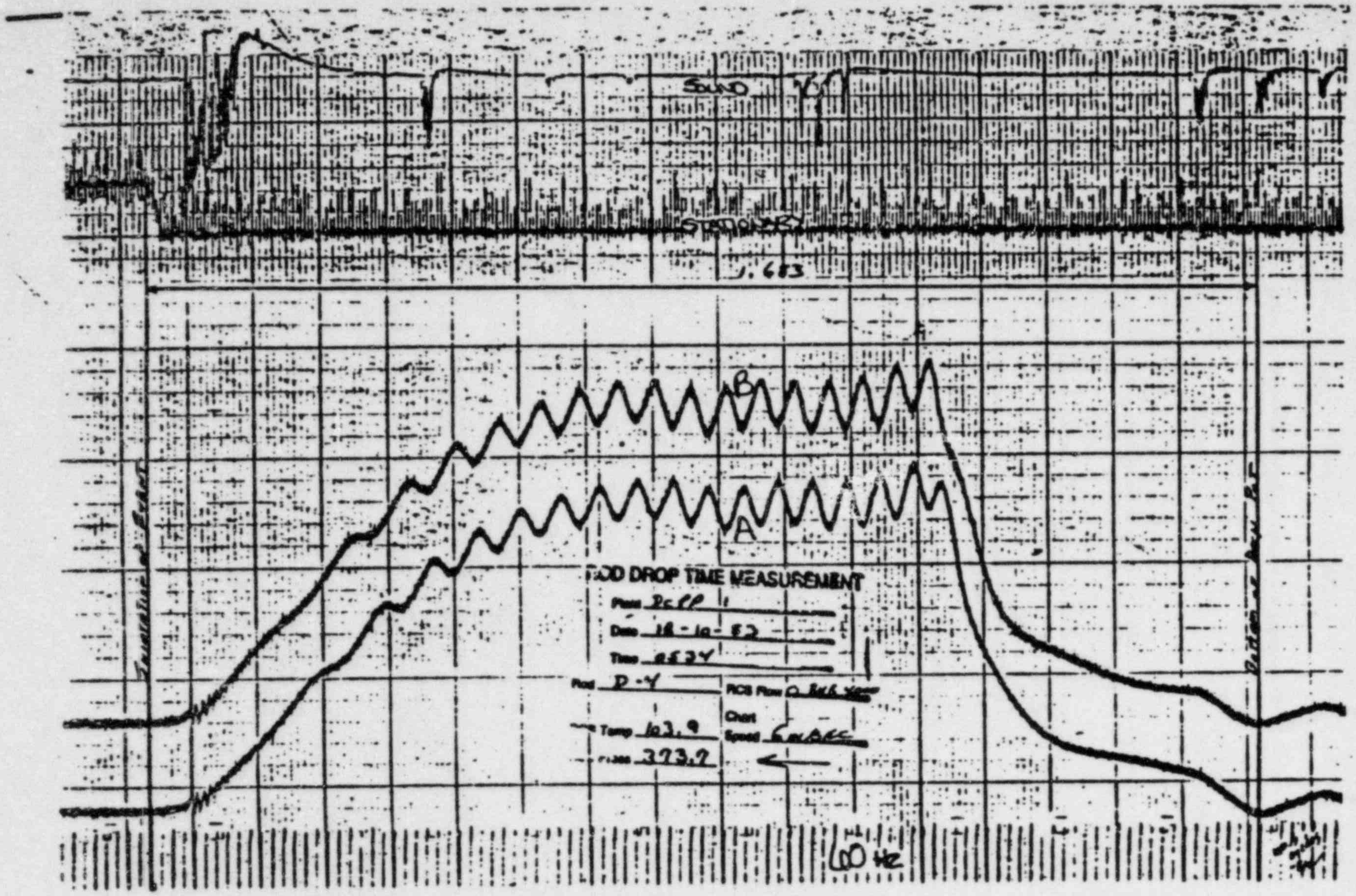
Plant Conditions	Core Average	Slowest Rod	Fastest Rod	Standard Deviation
Cold Shutdown - No Flow	1.178/1.680	1.207/1.722	1.150/1.650	±0.013/0.017
Cold Shutdown - Full Flow	1.435/2.059	1.527/2.183	1.363/1.967	±0.032/0.041
Hot Standby - No Flow	1.114/1.542	1.146/1.563	1.080/1.500	±0.014/0.019
Hot Standby - Full Flow	1.304/1.788	1.352/1.843	1.266/1.745	±0.016/0.024

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



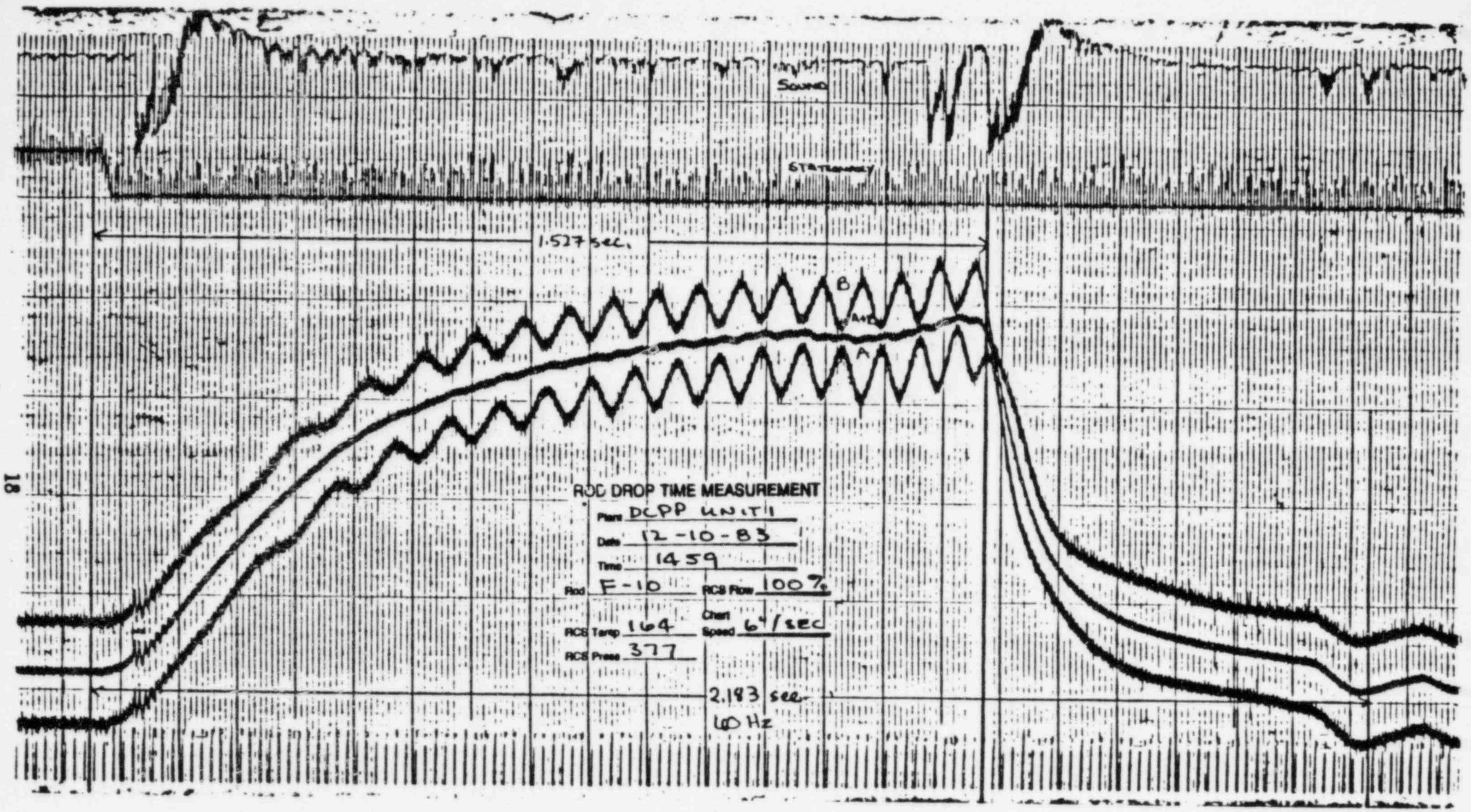
SAMPLE A+B TRACE

Figure 4



SAMPLE A & B TRACE

Figure 5



COMBINED A+B AND A & B TRACES

Figure 6

Temperature 164 °F   Pressure 360 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						1.183 1.680		1.175 1.675		1.187 1.690					
3			1.198 1.702		1.183 1.683		1.187 1.694		1.187 1.690		1.183 1.690		1.158 1.650		
4				1.180 1.690				1.183 1.698				1.180 1.680			
5			1.207 1.722						1.198 1.693				1.173 1.683		
6		1.181 1.680				1.170 1.675		1.160 1.661		1.160 1.661				1.190 1.700	
7			1.167 1.670		1.200 1.707								1.197 1.700		
8		1.175 1.675		1.183 1.683		1.183 1.690		1.150 1.650		1.157 1.640		1.183 1.675		1.160 1.661	
9			1.170 1.670								1.167 1.683		1.160 1.650		
10		1.175 1.683				1.187 1.702		1.167 1.670		1.200 1.713				1.163 1.667	
11			1.170 1.680				1.187 1.698						1.183 1.697		
12				1.163 1.667				1.197 1.697				1.167 1.667			
13			1.180 1.680		1.190 1.690		1.180 1.680		1.167 1.663		1.170 1.670		1.170 1.663		
14						1.167 1.667		1.183 1.680		1.167 1.675					
15															

N

DIABLO CANYON POWER PLANT - UNIT 1

CONTROL ROD DROP TIME - COLD, NO FLOW

Figure 7

Temperature 164 °F   Pressure 360 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						1.452 2.080		1.413 2.040		1.458 2.100					
3			1.467 2.137		1.433 2.073		1.520 2.167		1.430 2.083		1.427 2.076		1.363 1.967		
4				1.433 2.067				1.400 2.033				1.410 2.033			
5			1.507 2.147						1.483 2.117				1.413 2.033		
6		1.450 2.058				1.430 2.050		1.408 2.002		1.403 2.017				1.442 2.058	
7			1.430 2.050		1.483 2.123								1.437 2.030		
8		1.447 2.043		1.467 2.050		1.458 2.071		1.400 2.025		1.433 2.008		1.458 2.067		1.430 2.037	
9			1.417 2.023								1.433 2.060		1.393 2.000		
10		1.417 2.033				1.457 2.083		1.416 2.020		1.527 2.183				1.417 2.033	
11			1.427 2.080				1.433 2.060						1.437 2.077		
12				1.410 2.033				1.467 2.075				1.400 2.037			
13			1.425 2.028		1.437 2.052		1.397 2.037		1.433 2.067		1.427 2.063		1.400 2.033		
14						1.425 2.038		1.437 2.057		1.417 2.067					
15															

N

DIABLO CANYON POWER PLANT - UNIT 1

CONTROL ROD DROP TIME - COLD, FULL FLOW

Figure 8

Temperature 547 **°F** **Pressure** 2235 **psig** **% Flow** 0 **%**

X.XXX X.XXX	
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						1.125 1.552		1.104 1.531		1.114 1.542					
3			1.125 1.562		1.086 1.510		1.115 1.536		1.104 1.542		1.104 1.542		1.080 1.500		
4				1.120 1.573				1.104 1.542				1.115 1.563			
5			1.125 1.552						1.125 1.563				1.104 1.531		
6		1.114 1.542				1.104 1.542		1.093 1.515		1.115 1.542				1.114 1.552	
7			1.115 1.552		1.146 1.563								1.135 1.563		
8		1.115 1.542		1.125 1.562		1.125 1.542		1.093 1.510		1.093 1.500		1.125 1.542		1.104 1.542	
9			1.125 1.526								1.125 1.573		1.099 1.510		
10		1.125 1.552				1.110 1.552		1.125 1.542		1.125 1.563				1.093 1.521	
11			1.104 1.562				1.125 1.563						1.114 1.552		
12				1.125 1.563				1.135 1.562				1.115 1.552			
13			1.108 1.521		1.135 1.562		1.104 1.531		1.104 1.521		1.104 1.521		1.100 1.520		
14						1.125 1.531		1.125 1.552		1.104 1.521					
15															

N

DIABLO CANYON POWER PLANT - UNIT 1

CONTROL ROD DROP TIME - HOT, NO FLOW

Figure 9

Temperature 547 °F   Pressure 2235 psig   % Flow 100 %

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

	R	P	N	M	L	K	J	H	G	F	E	D	C	B	A
1															
2						1.320 1.810		1.285 1.767		1.310 1.805					
3			1.325 1.820		1.316 1.818		1.316 1.810		1.308 1.800		1.302 1.800		1.266 1.745		
4				1.316 1.816				1.292 1.775				1.295 1.767			
5			1.352 1.843						1.292 1.783				1.300 1.783		
6		1.315 1.785				1.285 1.767		1.280 1.755		1.280 1.750				1.312 1.800	
7			1.320 1.810		1.316 1.816								1.316 1.816		
8		1.310 1.785		1.316 1.810		1.315 1.805		1.280 1.769		1.286 1.742		1.302 1.775		1.285 1.780	
9			1.285 1.767								1.300 1.783		1.283 1.750		
10		1.302 1.780				1.300 1.783		1.310 1.783		1.324 1.827				1.283 1.760	
11			1.300 1.783				1.316 1.800						1.305 1.800		
12				1.300 1.780				1.340 1.810				1.297 1.783			
13			1.310 1.783		1.327 1.816		1.283 1.783		1.305 1.800		1.314 1.712		1.300 1.791		
14						1.302 1.783		1.302 1.785		1.290 1.788					
15															

N

DIABLO CANYON POWER PLANT - UNIT 1

CONTROL ROD DROP TIME - HOT, FULL FLOW

Figure 10

2.4 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

On March 27, 1984, with the plant in Hot Standby conditions, the control rod system was operated and Rod Bottom LEDs (Light Emitting Diodes) indication 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, pulse-to-analog and step counter readings agreed exceptionally well. However, the P-250 printout did not agree between rod positions of 96 steps and 168 steps out. This was determined to be a software problem in the P-250 program. The problem was corrected and further testing verified that the P-250 computer tracked the rod position properly. Rod bottom LED indication operated satisfactorily.

2.5 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.

2.6 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 in Hot Standby conditions with the spray flow bypass valves (valves 8050 and 8051) 3/4 turn open. 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 (500 deg. F ±5 deg. F). The resulting valve positions represented the final settings.

To initiate the pressurizer spray effectiveness portion of this test, the plant was stabilized in Hot Standby conditions and all pressurizer heaters were de-energized. 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 in 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

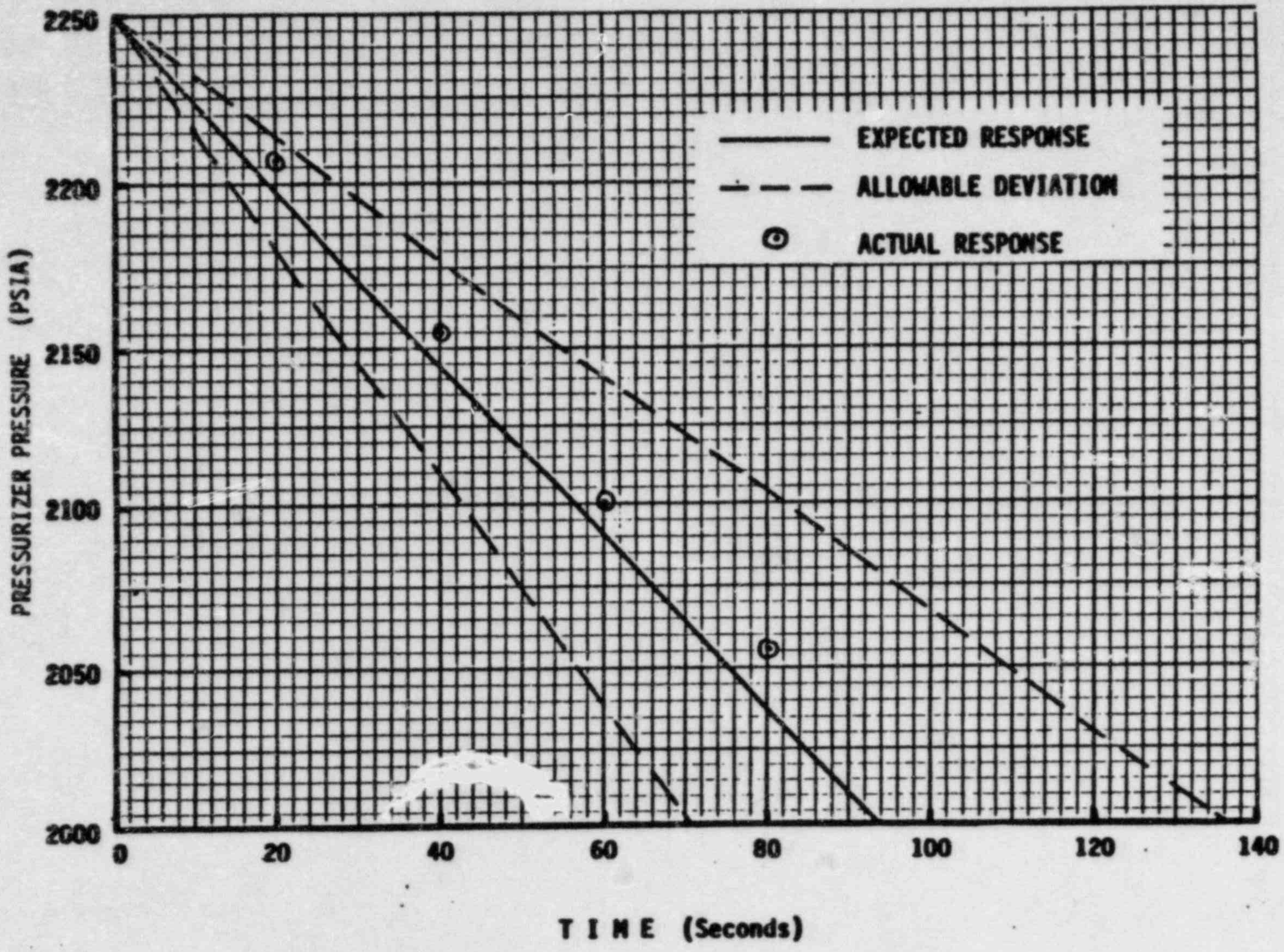
The pressurizer continuous spray flow bypass valves were set as follows:

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

During the adjustment of valve 8051, its valve stem broke; valve 8051 was subsequently replaced.

Pressurizer spray effectiveness was determined to be approximately -150 psi/minute. This rate was well within limits, as shown by Figure 11.

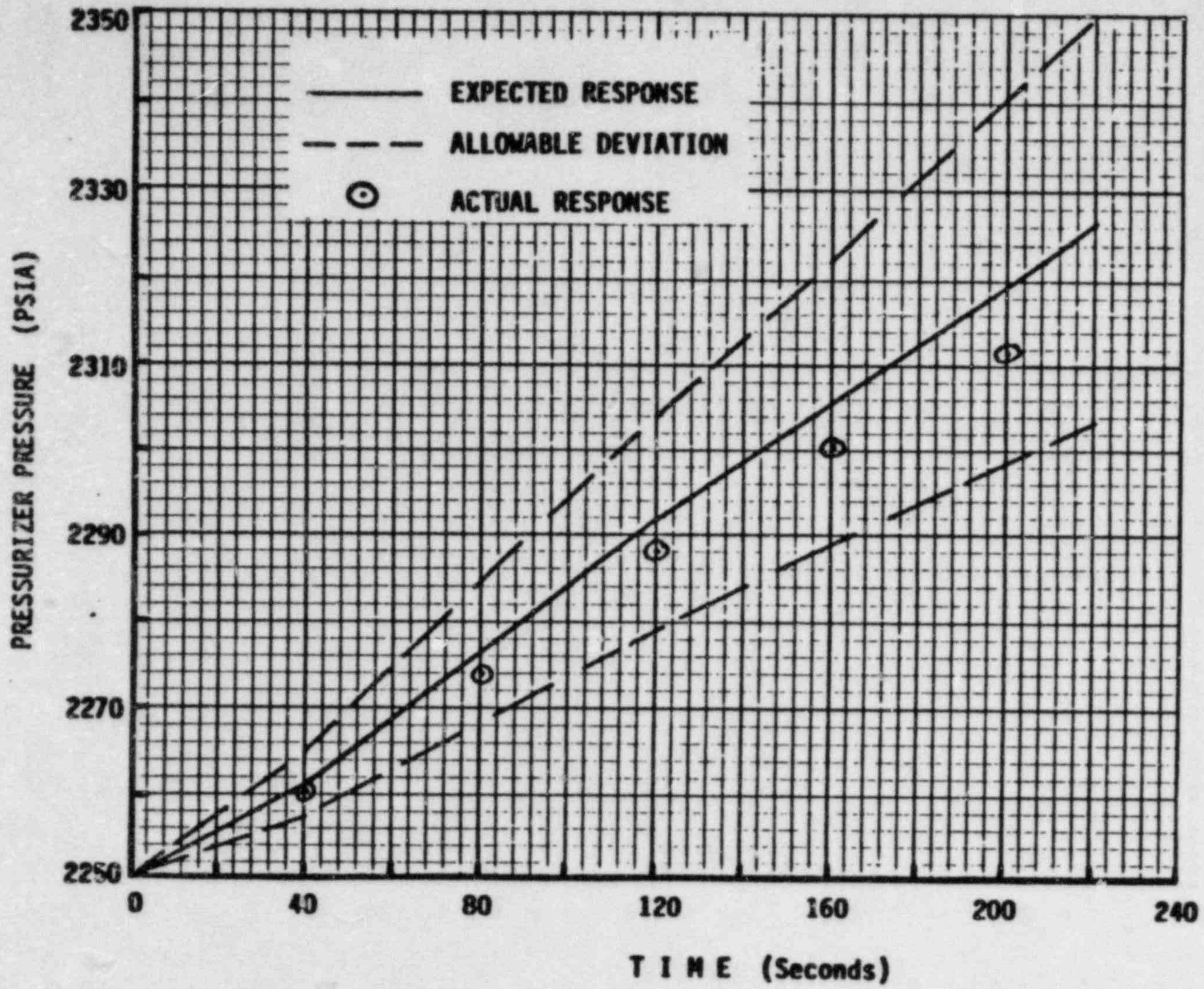
Pressurizer heater effectiveness was determined to be approximately 20 psi/minute. Again, the transient response was well within the limits, as shown in Figure 12.



DIABLO CANYON POWER PLANT - UNIT 1

PRESSURIZER SPRAY EFFECTIVENESS

Figure 11



DIABLO CANYON POWER PLANT - UNIT 1

PRESSURIZER HEATER EFFECTIVENESS

Figure 12

2.7 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 RTD. In addition, the RTD low flow alarms were set and verified.

TEST RESULTS

During the conduct of the flow measurements, restricting orifices (RO) 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 ± 3 gpm of 90% of the total measured loop RTD 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 Bypass Loop RO/Size	Cold Leg Flow (gpm)		Hot Leg Flow (gpm)		Total Flow (gpm)
		Minimum	Actual	Minimum	Actual	
1-1	RO-406/0.77"	56.0	120.3	63.6	138.8	259.0
1-2	RO-407/0.739"	70.3	132.7	61.2	136.3	269.0
1-3	RO-408/0.745"	70.1	121.8	64.7	133.2	255.0
1-4	RO-409/0.74"	64.6	135.2	64.5	127.8	263.0

2.8 STP R-27/TB-8403: Incore Thermocouples and RCS RTD Cross Calibration

TEST OBJECTIVE

The combination of Surveillance Test Procedure R-27 and Temporary Procedure TB-8403 provided a means to calibrate the incore thermocouples using the RCS loop RTDs as a reference over the temperature range 250 - 547 deg. F. The procedures also allowed cross calibration of the RTDs themselves.

TEST DESCRIPTION

Temporary procedure TB-8403 was performed at 250 deg. F and 340 deg. F. This test consisted of establishing a stable, full-flow, isothermal temperature in the RCS using the Residual Heat Removal System and recording spare narrow range RTD resistance and incore thermocouple temperature and millivolt readings. Thermocouple temperature readings were obtained from the output of the P-250 process computer, front panel of the two Post Accident Monitors (PAMs), the Emergency Response Facility Data System, and the Subcooled Margin Monitor. Millivolt readings were recorded at the input terminal boards to the PAM panels.

Surveillance Test Procedure R-27 was performed at 547 deg. F. This test consisted of establishing stable, full-flow isothermal RCS temperature using condenser steam dumps and simultaneously recording all RTD resistance readings in one RCS loop along with all thermocouple temperatures and millivolt readings and steam generator pressures. 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 from service at any time and measured. Therefore, the test was repeated for each remaining RCS loop after restoring the previous loop to service and re-establishing isothermal temperature in the RCS by operating the steam dump system in pressure control mode.

TEST RESULTS

Thermocouple readings at the various panel readouts initially were uncalibrated and therefore contained substantial errors. After calibration, errors generally were reduced to acceptable levels (±2 deg. at 547 deg. F).

RTD readings were consistent and only a few required small temperature corrections, all less than ±1 deg. at 547 deg. F. All but about 10% of the RTDs met the ±0.3 deg. F accuracy specification for narrow range RTDs at 547 deg. F with no correction.

The P-250 computer thermocouple readings were inconsistent as a result of excessive temperature sensitivity of the isolation amplifiers. A design change had been initiated previously to solve this problem but had not been completed at the time of the test.

2.9 Test Procedure No. 7.5 - 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 was 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. The pressures were then converted to reactor coolant loop flow 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 in the event that assessments of possible core damage become necessary.

TEST RESULTS

The total RCS flow rate was 390,000 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
1-1	97,500	0.0
1-2	94,900	-2.7
1-3	98,400	0.9
1-4	99,200	1.7
Total Flow	390,000	
Loop Average	97,500	

$$* \ \frac{\text{Loop Flow} - 97{,}500}{97{,}500} \times 100$$

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 363,000 GPM.

2.10 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

Four 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),

3) Three pumps operating initially, two pumps coasting down (2/3),

4) Three pumps operating initially, three pumps coasting down (3/3).

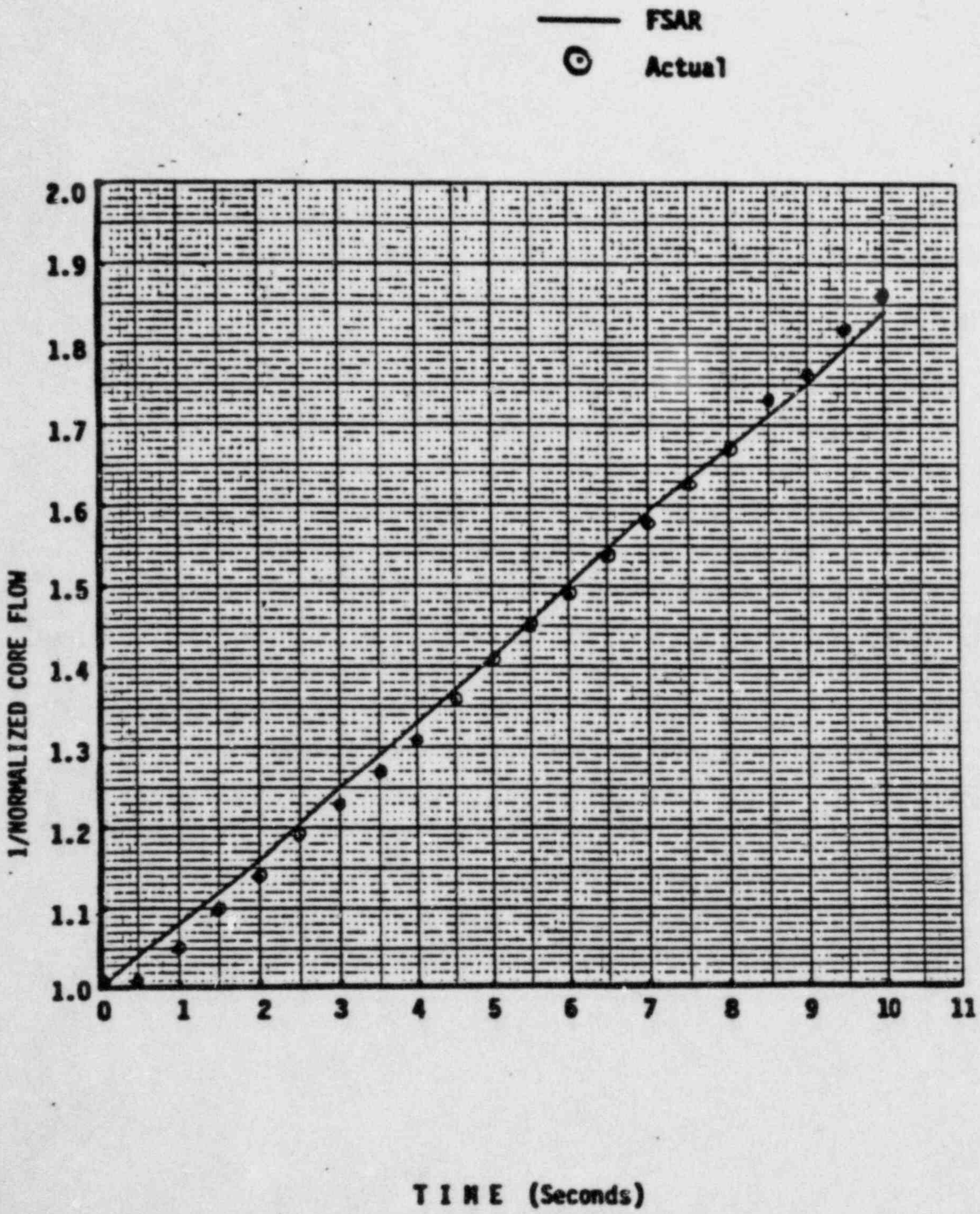
In each case, the pumps coasting down were tripped within 100 msec of one-another under Hot Standby conditions. The resulting coastdowns, i.e., flow as a function of time, were compared to coastdowns in the FSAR.

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 13. 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 all four coastdowns, the actual flow, corrected for flow sensor delay, was compared to the flow in the FSAR. Results are shown in Figure 14. 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 procedures, actual flow curves typically lie slightly below the FSAR curves for Westinghouse plants. The results were evaluated and declared acceptable by Westinghouse.

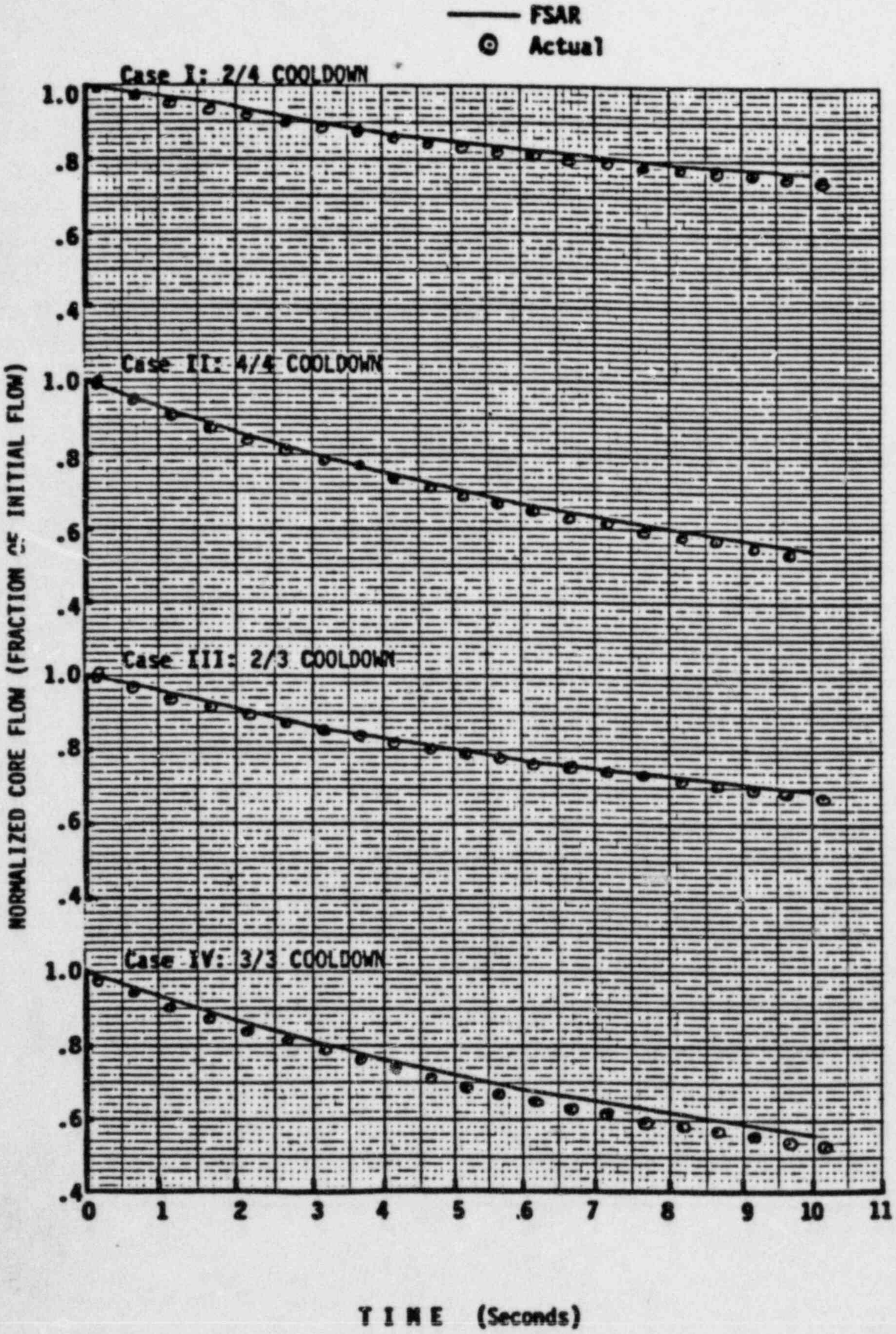
Data from the 2/4 coastdown was used to calculate the low flow time delay, the undervoltage trip delay time, and the under frequency trip delay time. All three parameters met their respective Acceptance Criteria (A.C.). The low flow time delay, defined as the time from beginning of coastdown until rod motion, was calculated to be 1.44 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 0.08 second (A.C. of $\leq$1.2 seconds). The under frequency 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 0.17 second (A.C. of $\leq$0.6 second).



DIABLO CANYON POWER PLANT - UNIT 1

INVERSE CORE FLOW CURVE

Figure 13



DIABLO CANYON POWER PLANT - UNIT 1

RCS FLOW COASTDOWN

Figure 14

2.11 Test Procedure No. 7.9 - Pressurizer Safety Valve Loop Seal Temperature Profile

TEST OBJECTIVE

The objective of this test was to verify acceptable performance of the insulation on the pressurizer safety valve loop seals and show compliance with NUREG-0737.

TEST DESCRIPTION

This test involved temperature measurements associated with the modified insulation on the pressurizer safety valve loop seals. As shown in Figure 15, thermocouples were located on the inside of the insulation of each loop seal and on the body of each safety valve. Data were collected over a thirty minute period with the primary system at Hot Standby conditions.

Safety valve temperatures were required to be less than 350 deg. F while the loop seal pipe temperatures were required to be greater than 260 deg. F.

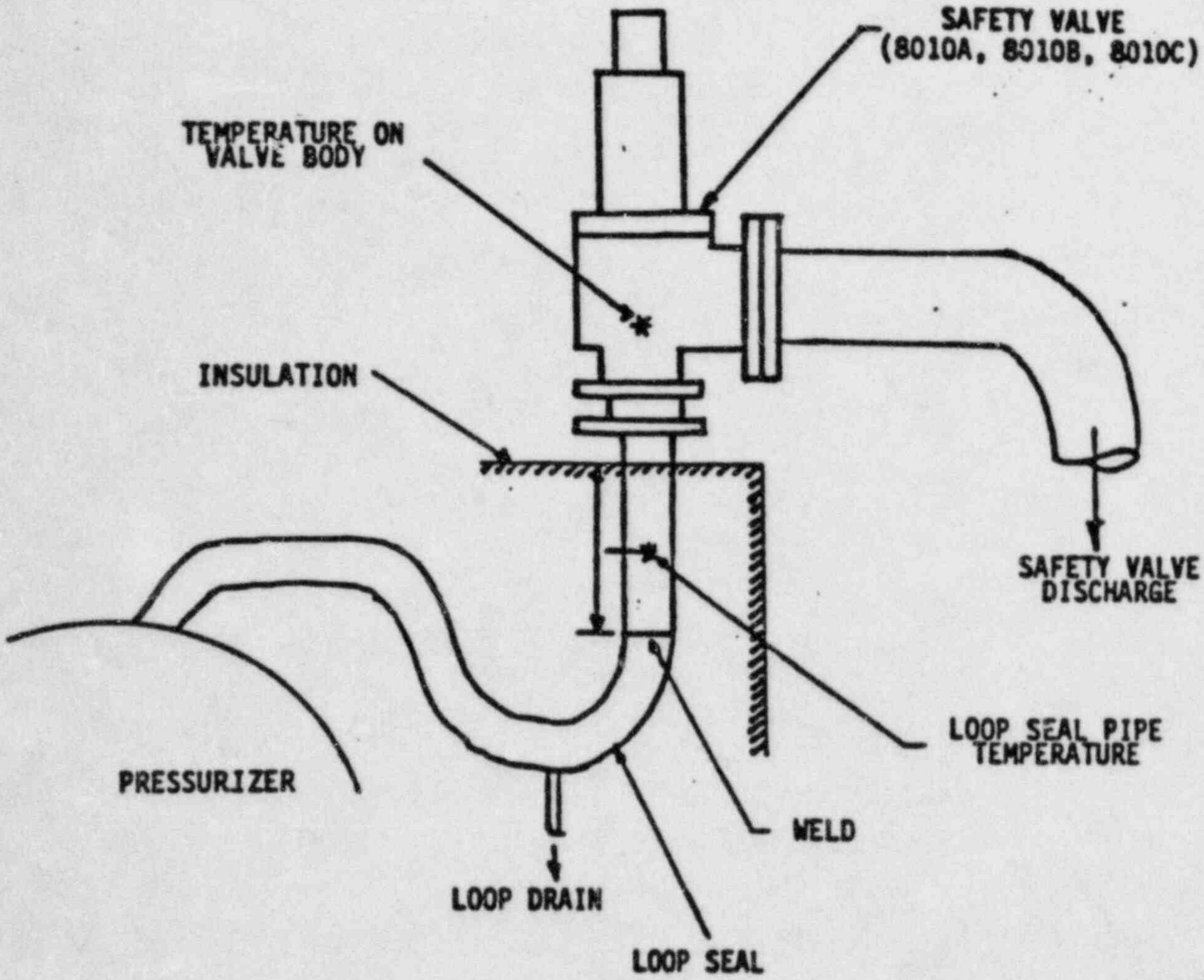
TEST RESULTS

Initial temperature measurements yielded satisfactory results for the valve temperatures, but loop seal piping temperatures were too low. Additional insulation was installed and existing insulation was modified to eliminate the excessive heat loss. After these corrections, all acceptance criteria were met as shown in Table 4.

Table 4

Pressurizer Safety Valve Loop Seal Temperatures

	Acceptance Criteria (deg. F)	Pressurizer Safety Valve		
		8010A (deg. F)	8010B (deg. F)	8010 (deg. F)
Loop Seal Piping Temp.	>260	328	322	316
Safety Valve Temp.	<350	148	184	171



DIABLO CANYON POWER PLANT - UNIT 1

PRESSURIZER SAFETY VALVE LOOP SEAL PIPING

Figure 15

3.0 INITIAL CRITICALITY AND ZERO POWER PHYSICS TEST PROGRAM

Summary

This portion of the Startup Program consisted of Initial Criticality and Zero Power Physics Testing. The approach to criticality started on April 28, 1984 and the low power physics measurements were completed on May 6, 1984. No major problems were encountered during the conduct of these tests.

Initial criticality was achieved on April 29, 1984 at 0007 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.

3.1 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 1895 ppm, RCS temperature at 547 deg. F, and RCS pressure at 2235 psig.

The control banks were withdrawn in 50 step intervals until Control Bank D reached 160 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 0.85.

Dilution to criticality was then commenced at approximately 1000 pcm/hr. Again, an ICRR was tracked and plotted. When the ICRR reached 0.2, the dilution was stopped to allow RCS mixing. At 0007 hours on April 29, 1984 criticality was achieved.

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.

Following recording of the initial criticality data, power was increased toward the point of adding heat (POAH). The POAH was found and the approach repeated three times to ensure data reliability. From the POAH data, (1×10^{-6} amp as indicated on the reactivity computer), the zero power test range (ZPTR) was established. This was set from 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. A 35 pcm positive reactivity addition was made and the reactor doubling time recorded. The results were checked against Westinghouse design criteria and found to be satisfactory.

TEST RESULTS

All parameters measured during this testing fell within the Acceptance Criteria provided by Westinghouse. Critical boron concentration was measured at 1335 ppm. The estimated critical condition was 1300 ppm. The difference was well within the design data allowance.

The POAH was measured at 1.03×10^{-6} amp on the intermediate range. Recording the same data for each of the three approaches to the POAH verified the value was indeed correct and repeatable.

The last test to verify proper operation of the reactivity computer indicated proper response for reactivity changes. All test cases fell within the $\pm 4\%$ Acceptance Criteria provided by Westinghouse. This test was repeated on a nominal daily basis to ensure continued proper operation of the reactivity computer throughout testing.

3.2 Test Procedure No. 41.3 - Nuclear Design Checks

TEST OBJECTIVE

The objective of this test was to measure the Boron Endpoint, Isothermal Temperature Coefficient and the Zero Power Neutron Flux Distribution and compare with predicted values.

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 Measurement

This measurement was performed to determine the boron concentration at which the reactor would be just critical at the control rod endpoint configuration.

The endpoint configurations at which this 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.

This measurement was performed with the reactor just critical and within 60 pcm of the endpoint configuration. The critical RCS boron concentration was determined. The controlling bank was then withdrawn/inserted to the endpoint configuration and the reactivity change measured. The corresponding critical boron endpoint concentration was determined using the following equation:

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

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.

Isothermal Temperature Coefficient Measurement

This measurement determined the reactivity change due to the overall temperature change of the core. This measurement was performed at the following endpoint configurations:

1) All rods out.
2) Control bank D fully inserted.
3) Control bank D and C fully inserted.
4) Control bank D, C, and B fully inserted.
5) Control bank D, C, B, and A fully inserted.

With the output from the reactivity computer and an average RCL 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.

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

$$\alpha\ iso = \alpha mod + \alpha doppler$$

where:

α iso = Isothermal Temperature Coefficient

α mod = Moderator Temperature Coefficient

α doppler = Doppler (Fuel) Temperature Coefficient (from Nuclear Design Report)

Zero Power Flux Distribution

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 neutron flux 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 distribution is shown in Figures 16 and 17 for the two cases.

The Movable Detector Flux Mapping System was used to collect data from the 56 fuel assemblies with instrument paths. Due to small detector currents during zero power testing, the movable detector system required a special set up 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.

TEST RESULTS

Boron Endpoint Measurement

The results of the Boron Endpoint Measurements are shown in Table 5. The measured values agreed very well with predicted values.

Isothermal Temperture Coefficient Measurement

The results of the Isothermal Temperature Coefficient measurements are summarized in Table 6. It was determined that the moderator temperature coefficient was positive at the ARO endpoint configuration. Rod withdrawal limits were established using an extrapolation 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 18. They shall 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 negative. (The Technical Specifications require only that the moderator temperature coefficient be more negative than 0.0 pcm/deg.F).

Zero Power Flux Distribution

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 16 for relative assembly powers. Insertion of the Control Bank D caused a slight increase in flux peaking, as shown in Figure 17. The radial distribution was also characterized by a small, but acceptable, flux tilt. Peaking factors are summarized in Table 7.

One of the six detectors failed (detector C) during the ARO Flux Mapping. Investigations determined that a high resistance short had developed in the detector cabling between the drive unit and the detector. Detector C remained out of service for the remainder of Zero-Power physics testing and Low-Power testing. This doubled the amount of time required to obtain a full core flux map.

Table 5

Measured Versus Predicted Boron Endpoint Concentrations

Endpoint Configuration	Critical Boron Concentration	
	Actual (ppm)	Predicted (ppm)
ARO	1344	1310 ± 50
CD in	1254	1255 ± 15.3
CD,CC in	1160	1162 ± 15.8
CD,CC,CB in	1083	1081 ± 13.4
CD,CC,CB,CA in	967	971 ± 16.8
CD,CC,CB,CA,SDD in	874	876 ± 13.4
CD,CC,CB,CA,SDD,SDC, in	789	791 ± 12.9
ARI, N-1	619	611 ± 50

Table 6

Measured Versus Predicted Isothermal Temperature Coefficient and Derived Moderator Temperature Coefficient

Endpoint Configuration	Isothermal Temperature Coefficient (α iso) Measured (pcm/deg. F)	Isothermal Temperature Coefficient (α iso) Predicted (pcm/deg. F)	Derived * Moderator Temperature Coefficient (α mod) (pcm/deg. F)
ARO	-.77	-1.7 ±3.0	+1.09
CD in	-2.25	-2.5 ±3.0	-0.39
CD,CC in	-4.75	-4.3 ±3.0	-2.89
CD,CC,CB in	-6.58	-6.4 ±3.0	-4.72
CD,CC,CB,CA in	-8.78	-7.9 ±3.0	-6.92

* From the Design Report, $\alpha^{HZP}_{doppler} = -1.86$ pcm/deg.F

$$\alpha_{mod} = \alpha_{iso} - \alpha_{doppler} = (\alpha_{iso} + 1.86) \text{ pcm/deg. F}$$

Table 7

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.	1342 PPM	1255 PPM
- power	0%	0%
- burnup	0 MWD/MTU	0 MWD/MTU
DATE	April 30, 1984	May 1, 1984
ROD CONFIGURATION	Bank D @ 218 steps Bank C @ 228 steps	Bank D @ 13 steps Bank C @ 228 steps
$F^N_{\Delta H}$ - measured value	1.438	1.619
- location *	D12-IH	D12-FE
- acceptance criteria	1.40 $\pm$10%	1.58 $\pm$10%
F^T_Q - measured value	2.362	2.639
- location *	D12-IH @ 77"	D12-FE @ 77"
F_Z - measured value	1.522	1.504
QUADRANT TILT-measured value	1.004	1.008
- acceptance criteria	$\leq$1.020	$\leq$1.020
- by quadrant	1.000 | 1.002 .995 | 1.003	.998 | 1.005 .994 | 1.003

* 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.

RELATIVE ASSEMBLY POWER (P1)
$\frac{\text{MEASURED P1} - \text{EXPECTED P1}}{\text{EXPECTED P1}} \times 100$

	R	P	N	M	L	K	J	H	G	F	E	D	C	B	A
1					.591 .5	.696 1.1	.827 1.4	.755 1.8	.815 0.0	.675 -1.9	.577 -1.8				
2			.528 1.4	.914 1.4	1.043 .5	1.020 1.0	1.056 .2	1.039 .1	1.045 -.9	1.000 -1.1	1.035 -.3	.912 1.2	.527 1.4		
3		.528 1.5	1.105 1.4	.993 1.0	1.126 .5	1.109 .5	1.148 -.5	1.109 -.9	1.146 -.7	1.100 -.3	1.125 .5	.995 1.2	1.105 1.4	.529 1.7	
4		.915 1.5	.997 1.4	1.322 1.4	1.099 2.2	1.191 2.2	1.116 -2.1	1.166 -1.3	1.127 -1.0	1.175 .8	1.088 1.2	1.331 2.1	1.001 1.8	.920 2.1	
5	.588 0.0	1.037 -.1	1.125 .5	1.080 .5	1.177 1.2	1.130 1.1	1.144 .3	1.105 .4	1.145 .4	1.134 1.4	1.193 2.6	1.102 2.5	1.143 2.0	1.058 1.9	.596 1.3
6	.674 -2.1	.993 -1.8	1.080 -2.1	1.156 -.8	1.115 -.2	1.111 .4	.988 .3	1.054 .4	.977 -.9	1.097 -.8	1.107 -1.0	1.183 1.6	1.116 1.2	1.022 1.1	.697 1.3
7	.806 -1.1	1.042 -1.2	1.133 -1.8	1.120 -1.7	1.125 -1.4	.974 -1.2	.996 -.6	.911 -.6	.990 -1.2	.960 -2.6	1.121 -1.7	1.125 -1.2	1.150 -.4	1.051 -.4	.824 1.1
8	.734 -1.0	1.027 -1.1	1.098 -2.0	1.161 -1.7	1.080 -1.9	1.033 -1.6	.899 -1.9	.947 -1.5	.900 -1.8	1.035 -1.4	1.088 -1.1	1.171 -.9	1.110 -.9	1.029 -.8	.742 .1
9	.807 -1.1	1.042 -1.2	1.136 -1.6	1.121 -1.6	1.119 -1.8	.960 -2.6	.968 -3.3	.894 -2.5	.996 -.5	.983 -.3	1.142 .2	1.132 -.6	1.147 -.6	1.043 -1.1	.816 0.0
10	.688 0.0	1.009 -.2	1.081 -2.0	1.164 -.1	1.116 -.2	1.091 -1.3	.956 -3.0	1.025 -2.3	.971 -1.5	1.110 .4	1.129 1.0	1.176 .9	1.111 .7	1.006 -.4	.685 -.4
11	.599 1.9	1.057 1.9	1.142 1.9	1.076 .1	1.165 .2	1.108 -.9	1.118 -2.0	1.066 -3.2	1.106 -3.0	1.108 -.9	1.185 2.0	1.096 2.0	1.142 2.0	1.036 -.2	.587 -.2
12		.916 1.7	.997 1.4	1.332 2.1	1.093 1.7	1.183 1.5	1.116 -2.0	1.142 -3.3	1.100 -3.4	1.153 -1.0	1.103 2.6	1.339 2.7	1.014 3.2	.915 1.6	
13		.526 1.1	1.102 1.1	.998 1.5	1.134 1.3	1.104 .1	1.130 -2.1	1.081 -3.4	1.134 -1.7	1.100 -.3	1.150 2.7	1.008 2.5	1.119 2.7	.537 3.2	
14			.526 1.1	.913 1.3	1.053 1.5	1.012 .1	1.036 -1.8	1.020 -1.7	1.055 .1	1.039 2.8	1.062 2.4	.917 1.8	.533 2.5		
15					.592 .7	.683 -.8	.811 -.6	.739 -.4	.832 2.0	.708 2.9	.602 2.4				

NORTH

DIABLO CANYON POWER PLANT - UNIT 1

<u>CORE AVERAGE RADIAL POWER DISTRIBUTION - ALL RODS OUT</u>

Figure 16

RELATIVE ASSEMBLY
POWER (P1)

$$\frac{\text{MEASURED P1 - EXPECTED P1}}{\text{EXPECTED P1}} \times 100$$

	R	P	N	M	L	K	J	H	G	F	E	D	C	B	A
1					.391 -6.4	.405 -4.1	.585 -2.0	.599 .2	.591 -1.0	.405 -3.9	.402 -3.8				
2			.563 2.2	.899 2.2	.761 -6.3	.430 -4.5	.791 -2.2	.899 -.6	.799 -1.2	.439 -2.4	.805 -.8	.896 1.9	.564 2.3		
3		.563 2.1	1.200 2.1	1.052 1.7	1.019 -6.3	.921 -6.3	1.096 -1.2	1.127 .3	1.111 .1	.978 -.6	1.095 .6	1.054 1.9	1.203 2.3	.566 2.7	
4		.899 2.1	1.055 2.0	1.478 2.0	1.240 2.9	1.350 2.9	2.292 -.3	1.366 .3	1.297 .1	1.322 .8	1.217 1.0	1.487 2.6	1.066 3.0	.911 3.5	
5	.416 -.6	.809 -.3	1.095 .7	1.214 .8	1.428 2.3	1.401 2.2	1.437 1.8	1.383 1.8	1.436 1.8	1.393 1.6	1.420 1.7	1.241 3.0	1.093 .5	.812 0.0	.400 -4.3
6	.406 -3.8	.437 -3.0	.962 -2.1	1.304 -.5	1.381 .7	1.413 1.3	1.235 1.0	1.302 1.0	1.220 -.3	1.387 -.6	1.354 -1.3	1.347 2.8	.990 .7	.451 .3	.404 -4.3
7	.579 -3.0	.792 -2.1	1.092 -1.6	1.278 -1.4	1.403 -.6	1.225 .2	1.164 .7	.965 .6	1.154 -.1	1.206 -1.4	1.407 -.3	1.310 1.1	1.112 .2	.806 -.3	.588 -1.5
8	.582 -2.6	.887 -1.9	1.103 -1.9	1.347 -1.1	1.339 -1.4	1.285 -.3	.958 -.1	.573 -.1	.951 -.8	1.283 -.4	1.358 0.0	1.375 .9	1.136 1.1	.908 .5	.598 0.0
9	.581 -2.7	.792 -2.0	1.091 -1.7	1.280 -1.2	1.389 -1.6	1.205 -1.5	1.140 -1.4	.949 -1.0	1.165 .8	1.233 .8	1.431 1.4	1.306 .8	1.121 1.0	.802 -.2	.596 -.2
10	.415 -1.7	.441 -2.0	.953 -3.0	1.313 .1	1.371 -.1	1.381 -1.0	1.199 -2.0	1.271 -1.4	1.217 -.5	1.404 .6	1.389 1.3	1.329 1.3	.999 1.6	.444 -1.4	.417 -1.1
11	.417 -.4	.810 -.3	1.085 -.2	1.208 .3	1.405 .6	1.367 -.3	1.399 -.9	1.331 -2.0	1.388 -1.7	1.366 -.4	1.429 2.3	1.233 2.4	1.117 2.7	.807 -.6	.415 -.6
12		.883 .3	1.043 .8	1.475 1.9	1.228 2.0	1.336 1.9	1.283 -1.0	1.330 -2.3	1.265 -2.4	1.305 -.5	1.238 2.8	1.490 2.9	1.074 3.8	.895 1.7	
13		.554 .5	1.182 .5	1.044 .9	1.103 1.5	.984 .1	1.090 -1.8	1.089 -3.1	1.067 -3.9	.943 -4.0	1.091 .3	1.067 3.1	1.216 3.4	.572 3.8	
14			.554 .5	.883 .3	.813 .1	.445 -1.1	.788 -2.6	.881 -2.6	.780 -3.5	.425 -5.5	.801 -1.4	.903 2.6	.569 3.2		
15					.412 -1.4	.413 -2.1	.585 -2.0	.586 -1.9	.577 -3.3	.399 -5.4	.412 -1.4				

NORTH

DIABLO CANYON POWER PLANT - UNIT 1

CORE AVERAGE RADIAL POWER DISTRIBUTION - CONTROL BANK D INSERTED

Figure 17

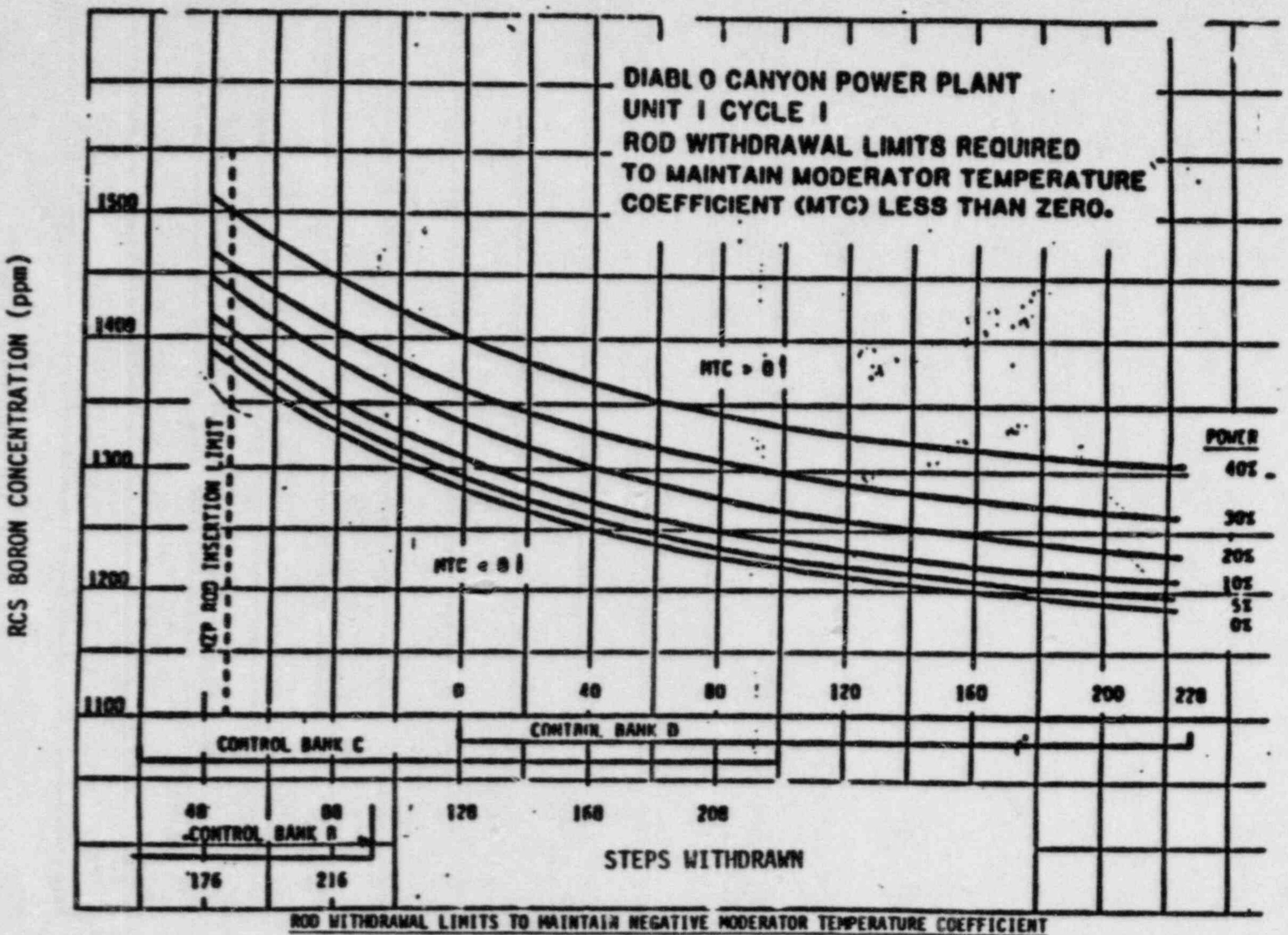

DIABLO CANYON POWER PLANT
UNIT I CYCLE I
ROD WITHDRAWAL LIMITS REQUIRED
TO MAINTAIN MODERATOR TEMPERATURE
COEFFICIENT (MTC) LESS THAN ZERO.
RCS BORON CONCENTRATION (ppm)
1500
1400
1300
1200
1100
HZP ROD INSERTION LIMIT
MTC > 0
MTC < 0
POWER
40%
30%
20%
10%
5%
0%
0
40
80
120
160
200
228
CONTROL BANK C
CONTROL BANK D
48
88
128
168
208
CONTROL BANK B
176
216
STEPS WITHDRAWN
ROD WITHDRAWAL LIMITS TO MAINTAIN NEGATIVE MODERATOR TEMPERATURE COEFFICIENT

Figure 18

3.3 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 was used to develop integral and differential bank worth curves. Figures 19 through 22 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.

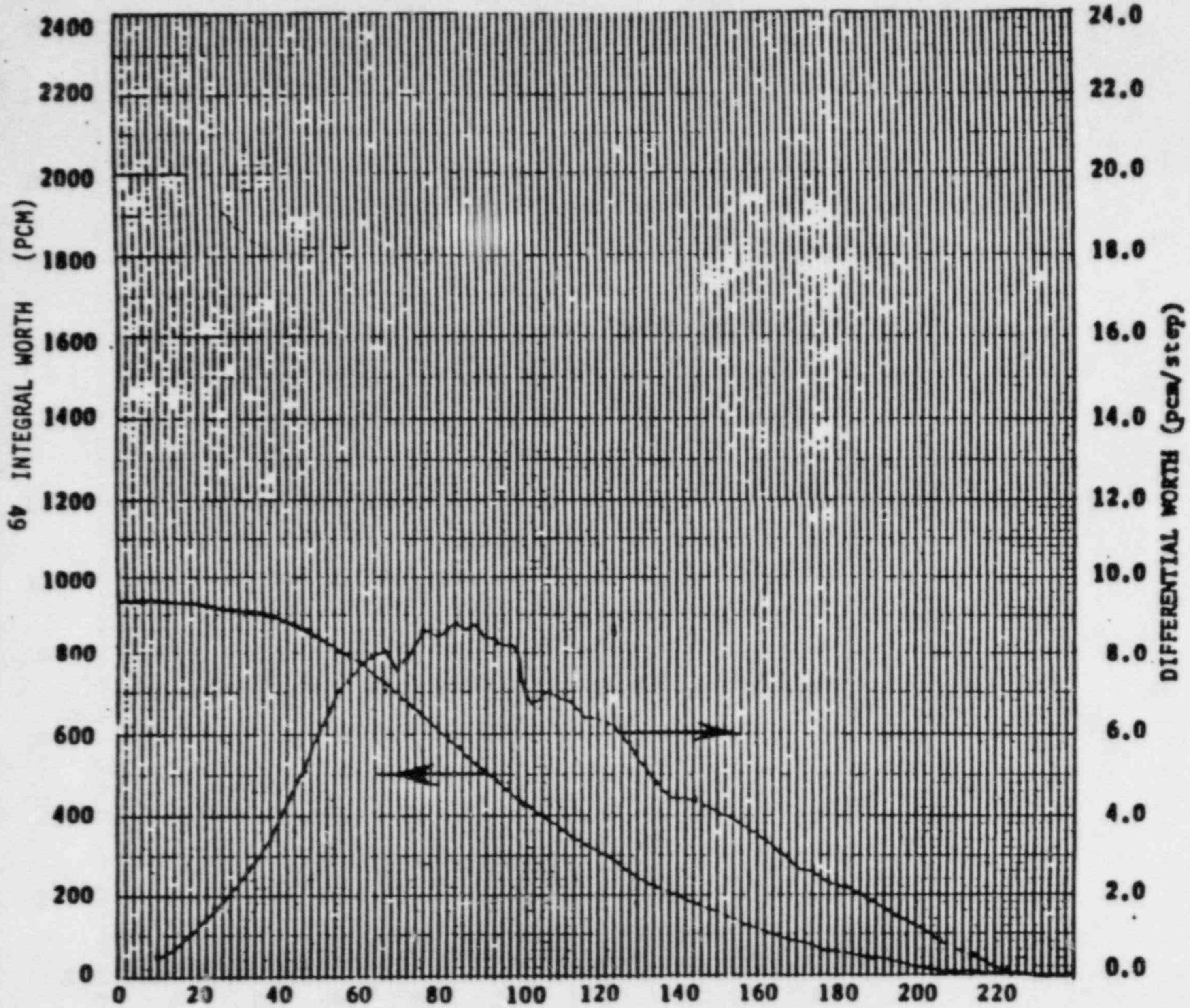
The integral and differential worth curves are shown in Figure 23.

Boron Worth

The average boron reactivity worth was determined using the reactivity data obtained during the control bank worth measurements and the change in boron endpoint concentration from the ARO configuration to the all control banks inserted configuration.

TEST RESULTS

The measured values agreed well with predictions as can be seen in Tables 8 and 9.



Unit: 1

T. P.: 41.4

Date: 4/30/84

Test Conditions:

1. RCC Bank Positions:

SDA	228
SDB	228
SDC	228
SDD	228
CA	228
CB	228
CC	228
CD	Moving
RCCA	N/A

2. Power Level:

   ZPTR % RTP

3. RCS Temperature:

   Initial 546.75 °F

   Final 546.9 °F

4. RCS Pressure:

   Initial 2235 psig

   Final 2235 psig

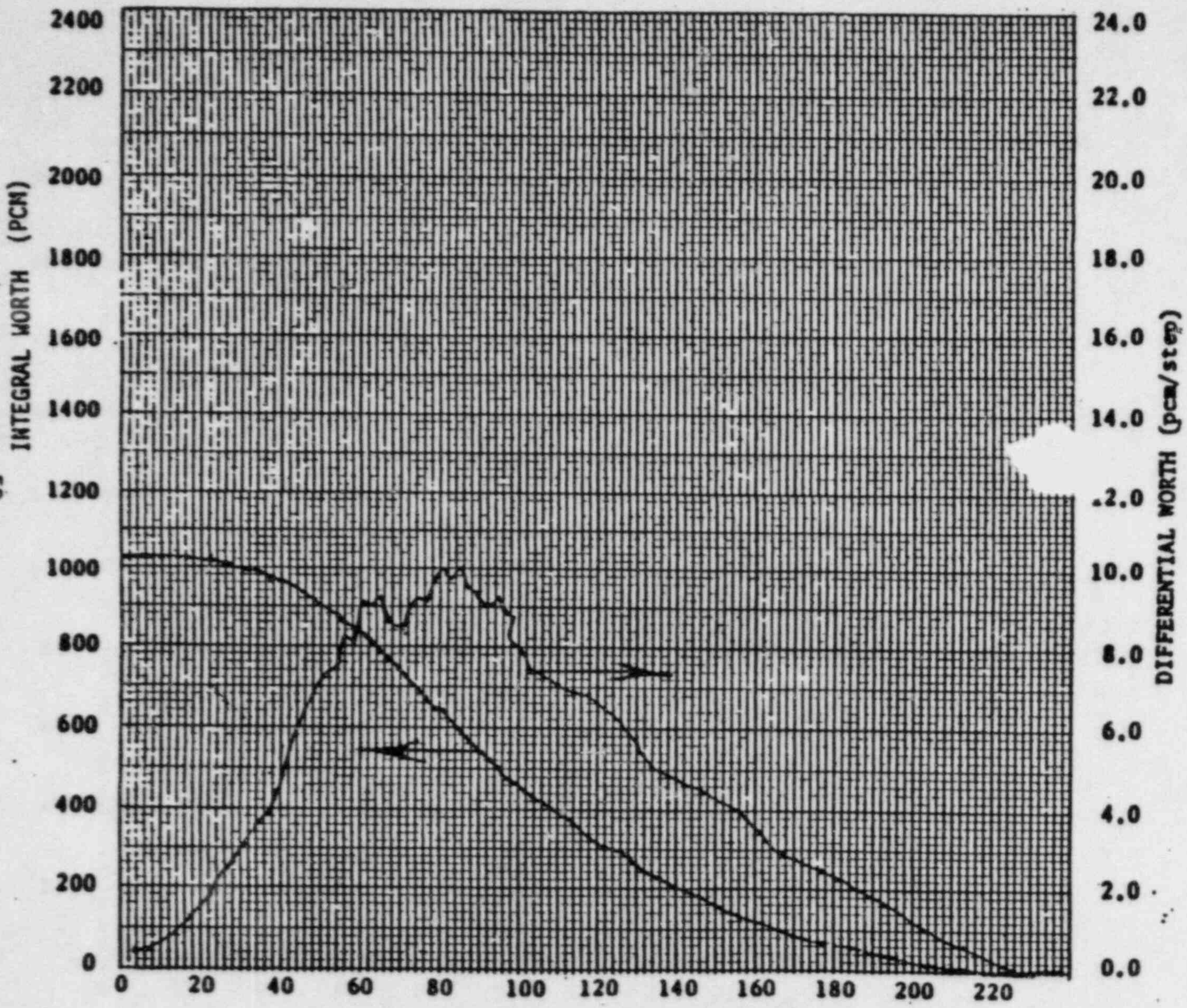
5. Avg. Core Burnup:

   0 MWD/MTU

DIABLO CANYON POWER PLANT - UNIT 1

CONTROL BANK D DIFFERENTIAL AND INTEGRAL WORTH

Figure 19



DIABLO CANYON POWER PLANT - UNIT 1

CONTROL BANK C DIFFERENTIAL AND INTEGRAL WORTH

Figure 20

Unit: 1

T. P.: 41.4

Date: 5/1/84

Test Conditions:

1. RCC Bank Positions:

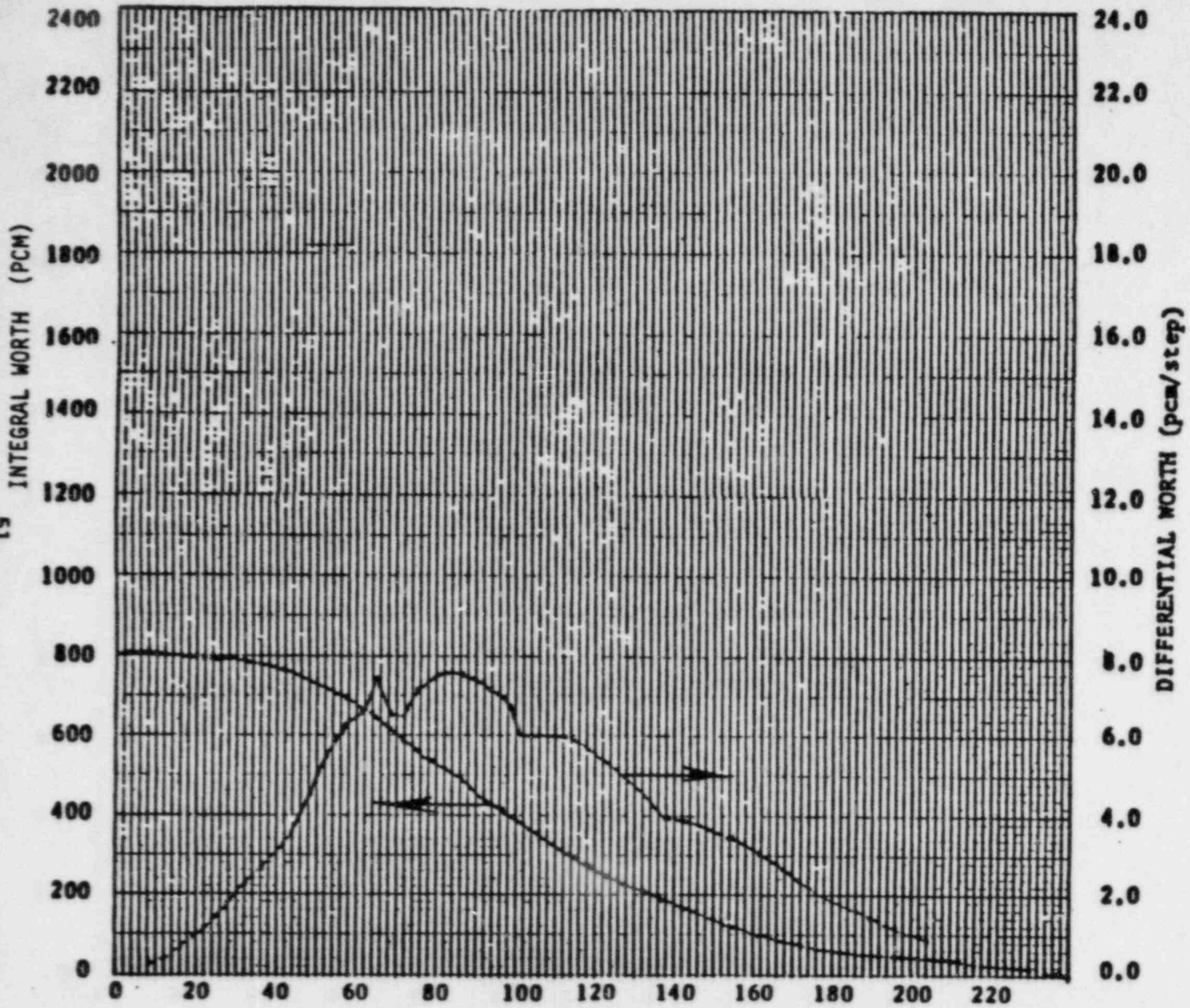
SDA	228
SDB	228
SDC	228
SDD	228
CA	228
CB	228
CC	Moving
CD	Ø
RCCA	N/A

2. Power Level: ZPTR % RTP

3. RCS Temperature:
   Initial 546 °F
   Final 547 °F

4. RCS Pressure:
   Initial 2235 psig
   Final 2235 psig

5. Avg. Core Burnup: 0 MWD/MTU



DIABLO CANYON POWER PLANT - UNIT 1

CONTROL BANK B DIFFERENTIAL AND INTEGRAL WORTH

Figure 21

Unit: 1

T. P.: 41.4

Date: 5/2/84

Test Conditions:

1. RCC Bank Positions:

SDA	228
SDB	228
SDC	228
SDD	228
CA	228
CB	Moving
CC	ø
CD	ø
RCCA	N/A

2. Power Level:

   ø % RTP

3. RCS Temperature:

   Initial 546.4 °F

   Final 546.9 °F

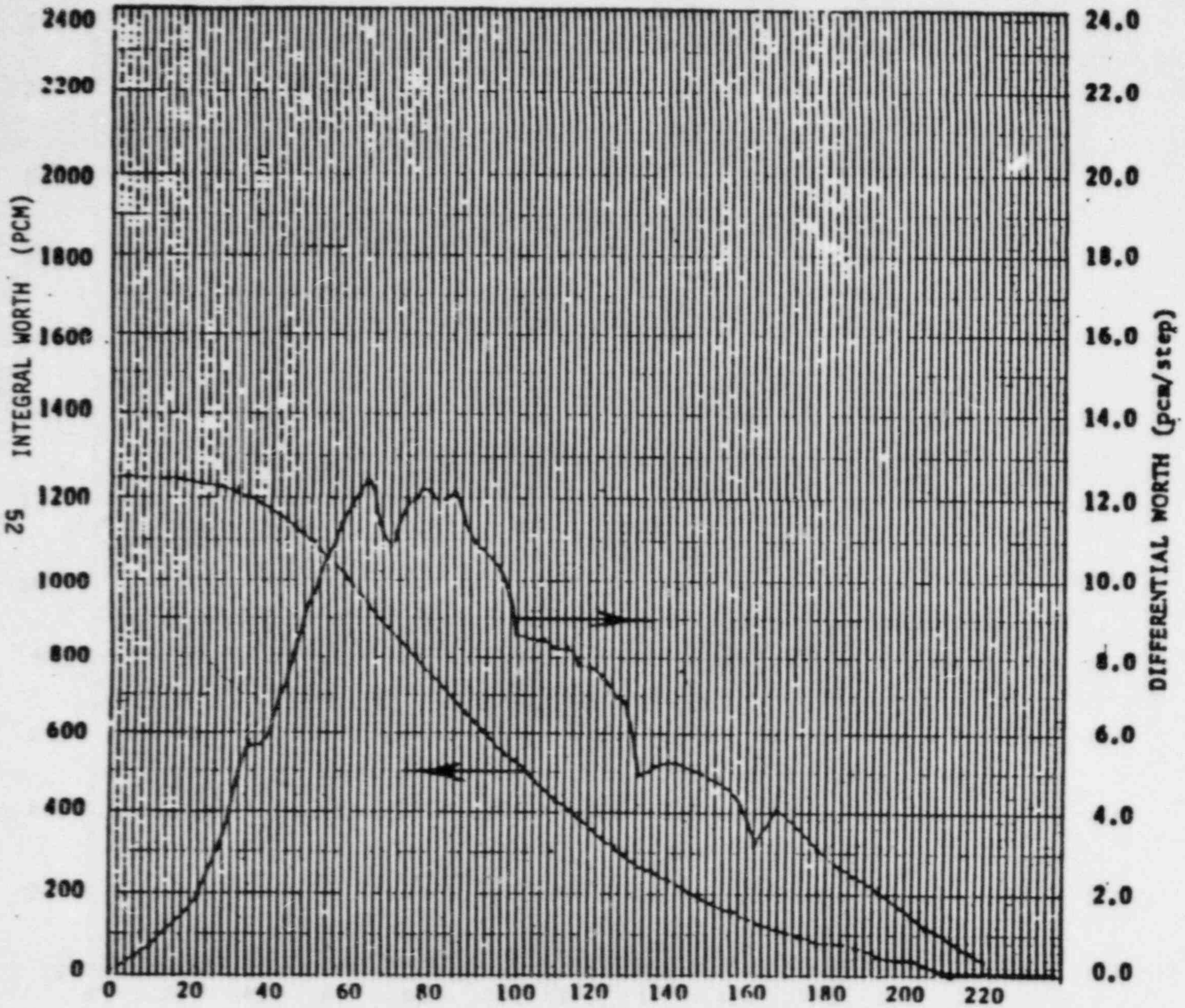
4. RCS Pressure:

   Initial 2235 psig

   Final 2235 psig

5. Avg. Core Burnup:

   0 MWD/MTU



DIABLO CANYON POWER PLANT - UNIT 1

CONTROL BANK A DIFFERENTIAL AND INTEGRAL WORTH

Figure 22

Unit: 1

T. P.: 41.4

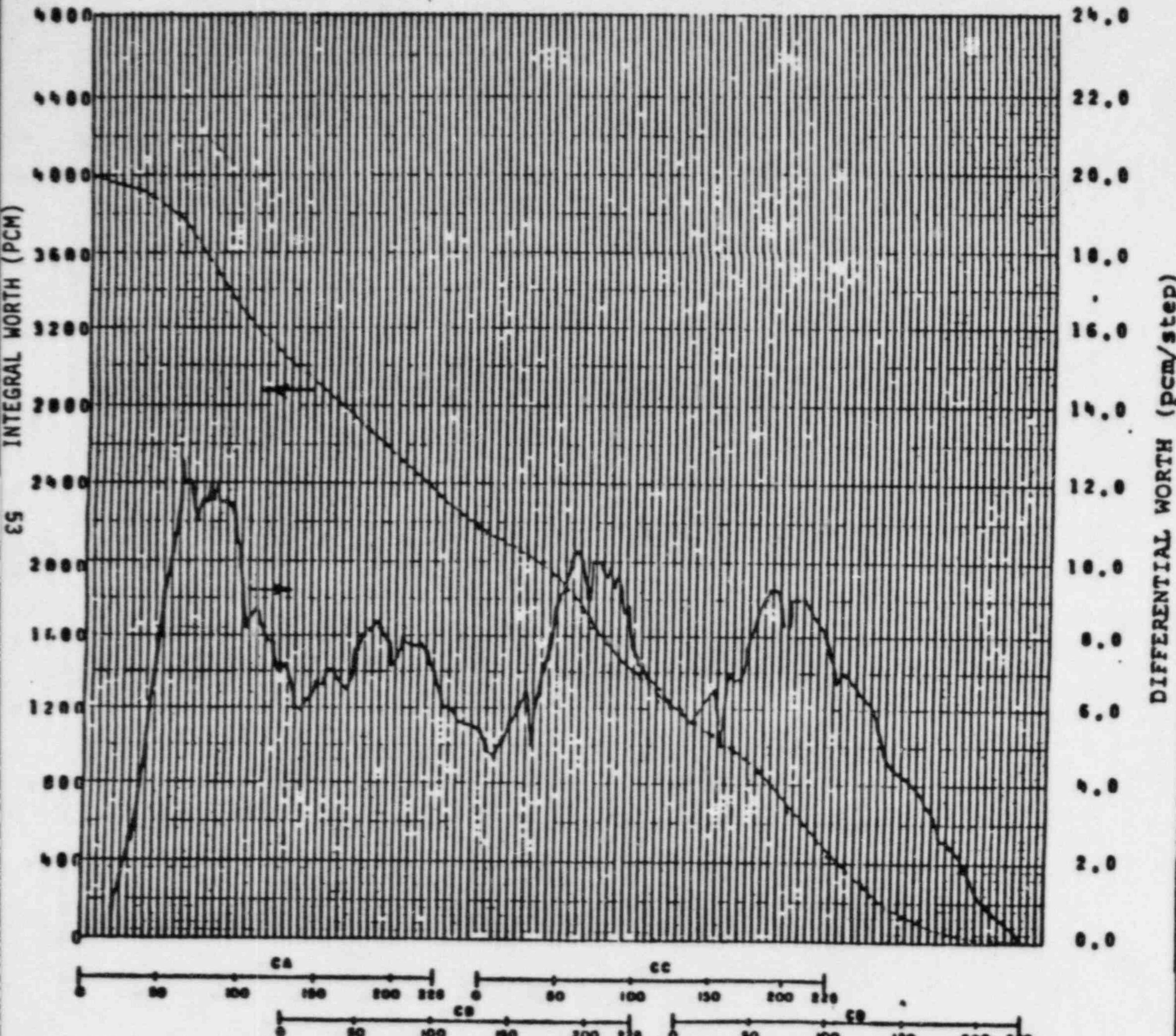
Date: 5/3/84

Test Conditions:

1. RCC Bank Positions:

SDA	228
SDB	228
SDC	228
SDD	228
CA	Moving
CB	0
CC	0
CD	0
RCCA	N/A

2. Power Level: 0 % RTP
3. RCS Temperature:
   Initial 547.5 °F
   Final 547.2 °F
4. RCS Pressure:
   Initial 2232 psig
   Final 2245 psig
5. Avg. Core Burnup: 0 MWD/MTU



Unit:	1
T. P.:	41.5
Date:	5/6/84

Test Conditions:

1. **RCC Bank Positions:**

SDA	228
SDB	228
SDC	228
SDD	228
CA	Moving
CB	Moving
CC	Moving
CD	Moving
RCCA	

2. **Power Level:** ZPTR % RTP
3. **RCS Temperature:**
   - Initial 547.2 °F
   - Final 547.0 °F
4. **RCS Pressure:**
   - Initial 2235 psig
   - Final 2235 psig
5. **Avg. Core Burnup:** 0 MWD/MTU

DIABLO CANYON POWER PLANT - UNIT 1

CONTROL BANKS WITH OVERLAP DIFFERENTIAL AND INTEGRAL WORTH

Figure 23

Table 8

Measured Versus Predicted Control Bank Worth

Control Bank	Measured Worth (pcm)	Predicted Worth (pcm)
CD	937	918 ± 92
CC	1031	978 ± 98
CB	805	795 ± 80
CA	1250	1195 ± 120
Total	4023	

Control Banks in Overlap	3987 pcm	Within ±4% of the Total Measured Worth of Individual Banks
		4023 ± 161 pcm

Table 9

Measured Versus Predicted Average Boron Worth

Control Bank	Bank Worth (pcm)	C_B (ppm)	Boron Worth (pcm/ppm)
CD	-937	89.5	-10.47
CC	-1031	94.0	-10.97
CB	-805	77.5	-10.39
CA	-1250	115.5	-10.83
CD+CC+CB+CA	-4023	376.5	-10.69
CD,CC,CB,CA in overlap	-3987	376.5	-10.59
		Average	-10.64
		Predicted	-10.4 ±1.0

3.4 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 (B06) with control banks at the zero power insertion limit. Integral worth of Rod B06 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 (10^{-8} to 10^{-7} amp). Control rods were positioned at 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...183 steps
Control Bank C.. 55 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 B06 were opened and a 300 pcm/hr continuous boron addition commenced. Criticality was maintained by withdrawal of the 'ejected rod', B06. During withdrawal, integral rod worth for B06 was measured. With B06 near the full out position, boron addition was stopped. B06 was then fully withdrawn and reactor control shifted to Bank B. The ejected rod incore flux map was then taken and analyzed.

Following the flux map, a dilution was commenced and Rod B06 was realigned with the other rods in Control Bank D. During the dilution, additional rod worth measurements for B06 were taken. Lift coil disconnect switches were then closed and normal rod configuration maintained.

TEST RESULTS

The reactivity worth measured during withdrawal of control rod B06 was 275 pcm. The measured value plus 10% for uncertainty was 303 pcm. This was considerably less than the design value of 480 ±48 pcm but well within the FSAR limitation of less than 785 pcm. The rod worth measurement taken during the dilution phase resulted in a measured value of 275 pcm. Due to the excellent repeatability of the two measurements, the results are considered to be accurate. Westinghouse was informed of the discrepancy between predicted and measured worth and accepted the results. Since the rod worth is lower than predicted, no safety review criteria have been violated.

Summaries of the power distributions are shown in Figure 24 and Table 10. The 'ejected rod' flux map showed the heat flux hot channel factor, F_Q^T, to be 5.63. This is well within the FSAR acceptance criteria limit of less than or equal to 13.

TABLE 10

Power Distribution Results (Pre and Post Pseudo RCCA Ejection)

ITEM	PRE-EJECTED FLUX MAP	POST-EJECTED FLUX MAP
CONDITIONS - temperature	~545 deg. F	~545 deg. F
- boron conc.	1200 ppm	1200 ppm
- power	0%	0%
- burnup	0 MWD/MTU	0 MWD/MTU
DATE	May 5, 1984	May 5, 1984
$F_{\Delta H}^N$ - measured value	1.683	3.357
- location*	E06-IA	B05-AQ
F_Q^T - measured value	2.974	5.633
- location*	M12-LE @ 36"	B07-AA @ 55"
QUADRANT TILT - measured value	1.004	1.660
- by quadrant	0.997 | 1.003 0.999 | 1.002	0.708 | 1.624 0.603 | 1.066

* Assembly locations (i.e., E06) as shown in Figure 1

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

	R	P	N	M	L	K	J	H	G	F	E	D	C	B	A
1					.421 .332	.474 .398	.728 .640	.758 .698	.743 .771	.482 .603	.440 .614				
2			.378 .284	.785 .593	.788 .605	.487 .416	.964 .846	1.119 .994	.977 1.017	.501 .643	.836 1.149	.781 1.097	.381 .634		
3		.374 .253	.632 .448	.905 .662	1.067 .786	1.032 .845	1.283 1.062	1.322 1.225	1.302 1.322	1.098 1.344	1.140 1.530	.909 1.406	.644 1.195	.384 .846	
4		.766 .486	.892 .601	1.430 .926	1.312 .908	1.471 1.057	1.408 1.168	1.436 1.298	1.410 1.528	1.469 1.766	1.301 1.892	1.464 2.370	.928 1.903	.804 1.914	
5	.442 .283	.831 .525	1.129 .702	1.288 .837	1.547 1.000	1.473 1.067	1.397 1.132	1.233 1.198	1.392 1.538	1.478 1.884	1.563 2.233	1.335 2.371	1.121 2.540	.822 2.551	.411 1.478
6	.482 .303	.503 .318	1.096 .658	1.442 .866	1.452 .948	1.385 .940	1.024 .834	.715 .710	1.006 1.207	1.368 1.759	1.445 2.213	1.488 2.576	1.094 2.700	.500 2.532	.449 1.778
7	.730 .438	.976 .574	1.283 .720	1.400 .838	1.383 .844	1.026 .704	.883 .677	.663 .645	.855 1.018	.981 1.377	1.366 2.052	1.418 2.463	1.285 2.648	.964 2.593	.718 2.017
8	.747 .425	1.118 .608	1.304 .733	1.427 .798	1.213 .734	.715 .477	.665 .495	.411 .383	.653 .736	.706 .901	1.218 1.731	1.443 2.164	1.324 2.342	1.123 2.144	.764 1.596
9	.737 .413	.983 .551	1.299 .694	1.405 .794	1.367 .781	1.004 .637	.864 .590	.660 .552	.870 .853	1.010 1.146	1.389 1.687	1.420 1.943	1.314 1.931	.985 1.665	.752 1.316
10	.494 .269	.510 .289	1.105 .609	1.466 .784	1.454 .834	1.357 .802	.998 .676	.708 .559	1.018 .911	1.376 1.323	1.460 1.600	1.471 1.695	1.134 1.443	.511 .742	.495 .749
11	.453 .246	.848 .465	1.149 .623	1.309 .729	1.528 .845	1.441 .872	1.363 .865	1.205 .884	1.363 1.092	1.443 1.325	1.542 1.461	1.318 1.398	1.173 1.255	.845 1.008	.451 .578
12		.764 .446	.864 .575	1.438 .829	1.319 .756	1.478 .853	1.394 .897	1.410 .963	1.387 1.078	1.451 1.207	1.318 1.203	1.462 1.376	.925 1.008	.785 .894	
13		.368 .229	.620 .397	.900 .560	1.160 .643	1.111 .678	1.276 .807	1.284 .898	1.250 .924	1.064 .902	1.132 .989	.924 .867	.649 .666	.381 .438	
14			.368 .234	.776 .458	.854 .490	.510 .326	.955 .657	1.085 .752	.938 .724	.475 .414	.820 .736	.786 .716	.381 .382		
15					.452 .263	.489 .308	.718 .504	.727 .536	.700 .546	.460 .376	.438 .378				

DIABLO CANYON POWER PLANT - UNIT 1

MEASURED FDHN (PRE AND POST PSEUDO RCCA EJECTION)

Figure 24

Pre Ejection
Post Ejection

3.5 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 full length 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. A 500 pcm/hr RCS dilution was then initiated. Individual shutdown banks were inserted to compensate for the dilution and their reactivity worths measured.

Upon completion of the worth measurements for Shutdown Bank D, preparations were made to enter the Technical Specification Special Test Exception for minimum shutdown margin. This required demonstration of tripability from at least 50% withdrawn of each full length rod not fully inserted within 24 hours prior to reducing the shutdown margin to less than 1.6% $\Delta k/k$. Therefore, with Shutdown Banks A, B and C fully withdrawn, the reactor was tripped. Shutdown Bank D 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 rods inserted except Shutdown Banks A, B and C fully withdrawn.

RCS boron dilution was then commenced and Shutdown Bank C rod worth was measured. The dilution was stopped.

Shutdown Bank D (the bank containing the most reactive rod, F10) was then pulled to 5 steps. Lift coil disconnect switches were opened for all rods on Shutdown Bank D with the exception of F10.

While maintaining criticality, Shutdown Bank B was exchanged with Rod F10. As Shutdown Bank B reached the fully inserted condition, the exchange with F10 was continued using Shutdown Bank A. When Rod F10 reached its fully withdrawn condition a dilution was commenced to allow the insertion of the remainder of Shutdown Bank A. At this point the reactor was critical with all rods inserted with the exception of the most reactive rod, F10, and Shutdown Bank A within 20 to 24 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 so as to observe the reactivity insertion associated with tripping F10. Once ready, the stationary gripper coil fuses for Rod F10 were pulled, dropping F10 into the core.

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

TEST RESULTS

No abnormalities were observed during this testing. The measured integral rod worth for Shutdown Bank D of 948 pcm was slightly outside the design value of 854pcm ±10%. Westinghouse later reevaluated the worth of Shutdown Bank D and provided a new design value of 941 pcm.

The measured integral rod worth for Shutdown Bank C of 905 pcm agreed well with the design value of 892 pcm ±10%.

The critical boron concentration for All-Rods-In Minus F10 configuration was measured at 627 ±4 ppm. This agreed well with the design value of 611 ±50 ppm.

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 1-1) 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 4.4% power, sufficient to support rated flow from the Turbine Driven Auxiliary Feedwater Pump (1-1), AFW Pump 1-1 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 temperature, vibration readings, pump room temperature and humidity were monitored. Pump flow was allowed to remain above rated flow of 880 GPM during the test. At the end of the 48 hour run flow was adjusted 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 1-1 operated satisfactorily and all acceptance criteria were met. During the 48 hour run, pump flow remained above its 880 GPM minimum, pump suction pressure varied between 16.9 and 21.5 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 SPECIAL LOW POWER TEST PROGRAM

SUMMARY

A series of Special Low Power Tests, conducted to provide supplementary technical information and operator training, were performed at the Diablo Canyon Nuclear Power Plant, Unit 1 between May 14-23, 1984. These tests, as listed below, fulfilled certain post-TMI action items of NUREG-0737 and consisted of the seven tests outlined in Supplement 10 of the Diablo Canyon Safety Evaluation Report (SER) dated August 1980 - and evaluated in detail in Supplement 14 of the SER dated April 1981.

Procedure No.	Test No.	Title
44.1	1.1	Natural Circulation
44.1	1.2	Natural Circulation with Loss of Pressurizer Heaters
44.1	1.3	Natural Circulation at Reduced Pressure
44.1	1.4	Natural Circulation with Simulated Loss of Off-Site AC Power
44.1	1.5	Effect of Steam Generator Isolation on Natural Circulation
44.2	---	Cooldown Capability of the Charging and Letdown System
44.3	---	Simulated Loss of All On-Site and Off-Site Power

The last two tests were conducted prior to establishing natural circulation conditions and T.P. 44.2 was repeated for operator training. Of the five natural circulation tests, 1.1 through 1.4 were conducted in succession and then repeated for further operator training. Test 1.5 was completed once at the conclusion of the test program.

Results of the Natural Circulation Testing demonstrated the stability and cooling capability of Diablo Canyon Unit 1 under natural circulation conditions. The program also demonstrated the ability of the licensed operators to establish, maintain, and recover from natural circulation in an orderly fashion.

Startup Test Procedure No. 44.1 was the governing document for the five natural circulation tests, Numbers 1.1 through 1.5. The procedure outlined each test and gave the prerequisites, initial conditions, and detailed instructions for establishing, maintaining, and recovering from the various configurations or conditions. It also provided a detailed procedure for temporary modifications to plant safety systems that were required in order to operate the reactor at low power without reactor coolant pumps (RCPs) running. Safety aspects of the procedure were evaluated in Supplement 14 of the Diablo Canyon SER.

5.1 Test Procedure No. 44.1
Test 1.1 - Natural Circulation

TEST OBJECTIVE

The objective of this test was to establish, maintain and recover from natural circulation conditions while at low power.

TEST DESCRIPTION

With the reactor at about 3% power, the reactor coolant system (RCS) pressure at a reduced value of approximately 2135 psig, and steam being dumped to the condenser, all four reactor coolant pumps were tripped in rapid succession. Pressurizer spray valves were placed in manual and opened, and one power operated relief valve controller was temporarily placed in the closed position to prevent its opening during the expected pressure rise.

Temporary modifications had previously been made to plant safety systems to prevent automatic initiation of safety injection due to the off-normal operating conditions encountered during natural circulation. Plant conditions were monitored and recorded during the subsequent stabilization of plant parameters. Adjustments were made to RCP seal water flow rate, charging flow rate and auxiliary feedwater flow rate to maintain stable conditions. When all plant parameters had stabilized and training was completed, one of the other five natural circulation tests was begun or the plant was returned to forced circulation.

Recovery from natural circulation was achieved from stable natural circulation conditions with pressurizer heaters operating. The controlling bank of control rods was fully inserted, and the RCPs were restarted one at a time. Pressurizer spray valves were realigned for normal operation and the plant was stabilized at normal operating conditions prior to withdrawing the controlling bank to a critical configuration and returning the reactor to about 3% power.

TEST RESULTS

Results were consistent with the response of other Westinghouse pressurized water reactors. Following the trip of the RCPs, loop cold leg temperatures dropped a few degrees while hot leg temperatures rose roughly 35-40 deg. F, as shown in Figure 25. RCS loop temperatures stabilized within about 15 minutes. Calculated loop $\Delta T(T_{hot} - T_{cold})$ never approached the limit of 65 deg. F maximum. The hotter RCS temperatures, combined with a slightly negative temperature coefficient of reactivity, caused a small reduction in power. This was compensated for by the operators manually withdrawing control rods to maintain constant power. The entire test was conducted with Control Bank D rods at about 150-190 steps withdrawn.

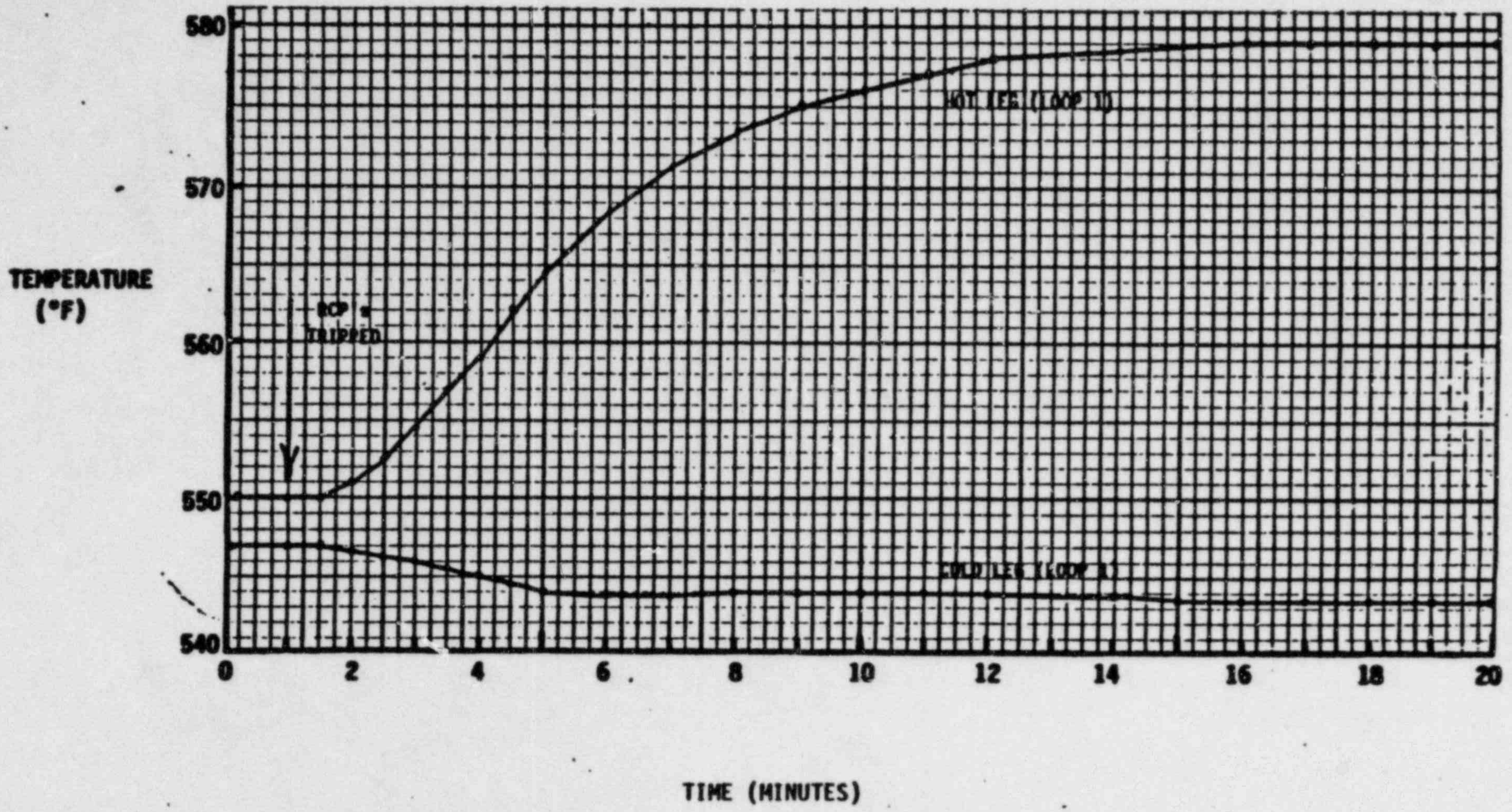
Pressurizer pressure increased during the initial transient as expected and peaked at about 2195 psig after about 7-8 minutes, as shown in Figure 26. Pressure slowly decreased thereafter without any requirement for initiating auxiliary spray. Pressurizer level required approximately 8 minutes to reach a peak of 33.5% from an initial level of 25%.

Core exit temperatures, as monitored by thermocouples, stabilized and showed no evidence of hot spots in the upper plenum or head region. Temperatures remained below the limit of 610 deg. F maximum. Subcooling margin in the reactor vessel stabilized at more than 60 deg. F below saturation.

The full core flux map was obtained using movable incore detectors with the plant stabilized in natural circulation. Results were consistent with plant conditions, and peaking factors all were acceptable. There was no appreciable flux tilt nor evidence of poor mixing or hot spots.

Recovery from natural circulation was achieved with only the expected minor perturbations in plant pressures and levels, well within the capability of automatic control systems.

Test 1.1 demonstrated that licensed plant operators could establish, maintain and recover from stable natural circulation conditions with the reactor operating at low power. The planning and execution of these evolutions provided ample opportunity for training all operators in the core cooling capability of Diablo Canyon Unit 1 without the benefit of reactor coolant pumps.



DIABLO CANYON POWER PLANT - UNIT 1

REACTOR COOLANT LOOP TEMPERATURE ON NATURAL CIRCULATION

Figure 25

5.2 Test Procedure No. 44.1

Test 1.2 - Natural Circulation With Loss of Pressurizer Heaters

Test 1.3 - Natural Circulation at Reduced Pressure

TEST OBJECTIVES

These procedures had a number of objectives:

1) To determine the rate of decrease of margin to saturation,

2) To verify recovery of margin through cooldown and make-up,

3) To verify operation and accuracy of the subcooling margin monitor,

4) To verify that changes in margin will not affect natural circulation, provided natural circulation exists,

5) To determine the effect of auxiliary spray on pressurizer pressure,

6) To determine the position of the normal spray valves to obtain a sufficient back pressure for the auxiliary spray valves.

TEST DESCRIPTION

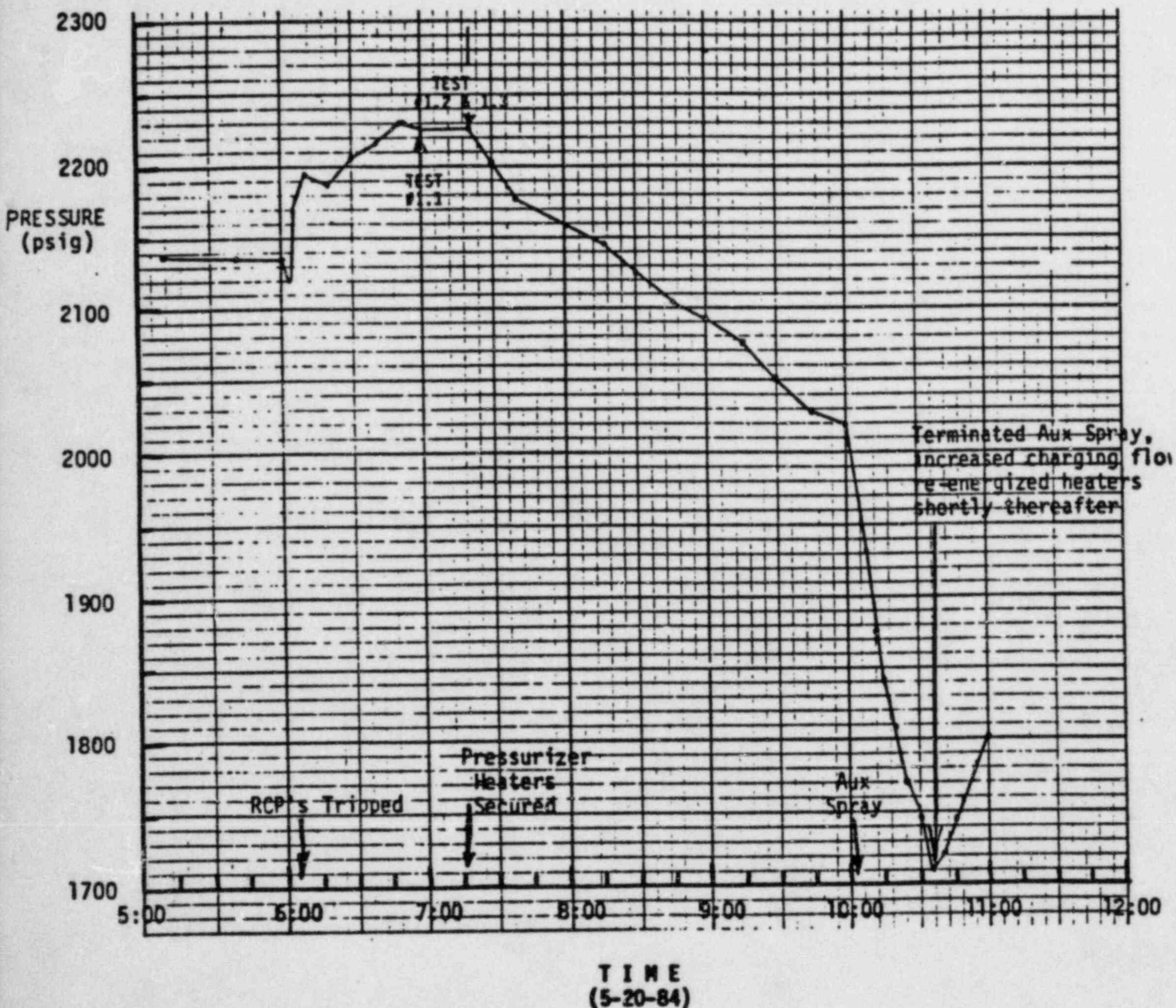
Natural Circulation with Loss of Pressurizer Heaters and Natural Circulation at Reduced Pressure tests were performed concurrently. The description and results given below apply to both tests.

Initial conditions required that the reactor be critical at approximately 3% power under natural circulation conditions. A transient was then initiated by deenergizing the pressurizer heaters. Reactor coolant system pressure was monitored to determine the rate of depressurization and resulting loss of margin to saturation.

The auxiliary pressurizer spray valves and, as required, the normal spray valves were employed to accelerate the rate of depressurization. Core exit thermocouples were monitored to determine core flow distribution. The operation of the subcooling margin monitor was verified through hand calculations. Finally, prior to reaching saturation conditions, primary system charging flow was used to recover margin prior to restoring the pressurizer heaters to service.

TEST RESULTS

Upon de-energizing the pressurizer heaters, the plant responded as expected with a steady decrease in primary system pressure. Over a period of approximately three hours, pressure dropped from approximately 2235 psig to approximately 2020 psig, a rate of 72 psi/hr (corresponding to approximately a 4 deg. F/hr of decrease of margin to saturation). When auxiliary spray was introduced, the depressurization rate increased to roughly 600 psi/hr. Consequently, pressure control through the use of the normal spray valves was not necessary. The transient was permitted to continue to an RCS pressure of 1710 psig (a subcooling margin of 29 deg. F) before charging flow was successfully used to demonstrate the ability to restore margin. Restoration was then completed by using the pressurizer heaters. Pressurizer pressure for the entire transient is illustrated in Figure 26. The primary system subcooling margin monitor was found to be both reliable and accurate; hand calculations showed excellent agreement with the monitor readings throughout the depressurization.



DIABLO CANYON POWER PLANT - UNIT 1

PRESSURIZER PRESSURE VS. TIME

Figure 26

5.3 Test Procedure No. 44.1

Test 1.4 - Natural Circulation With Simulated Loss of Off-Site AC Power

TEST OBJECTIVE

The objective of this test was to verify that natural circulation cooling can be adequately maintained following a simulated loss of off-site AC power.

TEST DESCRIPTION

Initial conditions were achieved with the reactor critical and operating at approximately 1% power under natural circulation conditions. Loss of off-site power to the vital 4160 Vac buses was then simulated by opening the main supply breaker, 52-HG-15, from the main control board. That initiating event caused the diesel generators to automatically start and pickup loads. Natural circulation conditions were maintained and monitored. Core exit thermocouples were used to monitor core flow distribution.

TEST RESULTS

The plant response to the simulated power loss proceeded as expected. All diesel generators automatically started and picked up associated loads. Steam generator levels were adequately controlled through the use of the auxiliary feedwater system. Pressurizer pressure was controlled with vitally powered heater groups, while pressurizer level was maintained using normal letdown and charging. Finally, steady state natural circulation conditions were maintained without any problems.

5.4 Test Procedure No. 44.1

Test 1.5 - Effect of Steam Generator Isolation on Natural Circulation

TEST OBJECTIVE

There were two objectives associated with this test.

1) Monitor the effects of isolating two steam generators while on natural circulation.

2) Determine the effect of reduced reactor coolant system average temperature (Tavg) on the nuclear instrumentation system (NIS).

The first objective was to ensure the plant would respond as designed should one or more steam generators require isolation while the RCS was in natural circulation. This configuration would be required in the event of a steam generator tube rupture. Primary concerns centered on, 1) maintaining natural circulation cooling in the non-affected loops, 2) correct RCS parameter response in the affected loops and 3) control of the pressure on the secondary side of the isolated steam generators.

The second objective was to observe and quantify the change in coupling between the reactor core and NIS with reduced RCS Tavg. With cooler water in the reactor vessel downcomer region, fewer neutrons are sensed by the NIS. This results in indicated reactor power being less than actual power. This effect required measurement prior to commencement of natural circulation with reduced Tavg to establish limits for indicated reactor power.

TEST DESCRIPTION

Initial conditions required the reactor to be critical at 3% power with all reactor coolant pumps running. RCS temperature and pressure were maintained at 547 deg. F and 2235 psig respectively.

At the outset, RCS pressure was reduced to 2000 psig to minimize the potential for exceeding the 1600 psid limitation across the steam generator tubes. Subsequent steps would reduce the secondary side steam pressure to approximately 735 psig. If normal operating pressure were maintained, an approximate 1500 psi differential across the steam generator tubes would result. The reduced RCS pressure thus effected an increased margin to the 1600 psid limit.

A 5 deg. F/step cooldown was then initiated from 547 deg. F to 510 deg. F. At each plateau, RCS parameters were stabilized at the core delta-T corresponding to 3% reactor power. RCS temperatures and NIS data were then recorded. Though the reactor power level was maintained at a constant 3%, at each temperature plateau, NIS indicated power decreased at a rate of 0.036% per 1.0 deg. F. This number was consistent with Westinghouse predictions. At the end of the cooldown, all data was collected and reduced. From the information gathered, a plot was generated and provided for the operators showing NIS indicated power level vs temperature for the 5% licensed power limitation.

While maintaining RCS temperature at 510 deg. F and pressure at 2000 psig, reactor power was reduced to 1%. Once stable, initial conditions for entry into natural circulation were verified and RCPs secured. Natural circulation was then established and stabilized.

Steam Generator 1-3 was then isolated by closing the main steam isolation and bypass valves, securing auxiliary feedwater, and isolating steam generator blowdown. As expected, Tcold for loop 1-3 began to increase toward Thot resulting in a corresponding increase in Steam Generator 1-3 pressure. Parameters gradually stabilized until the delta-T across the isolated steam generator decreased to less than 10 deg. F, as shown in Figure 27.

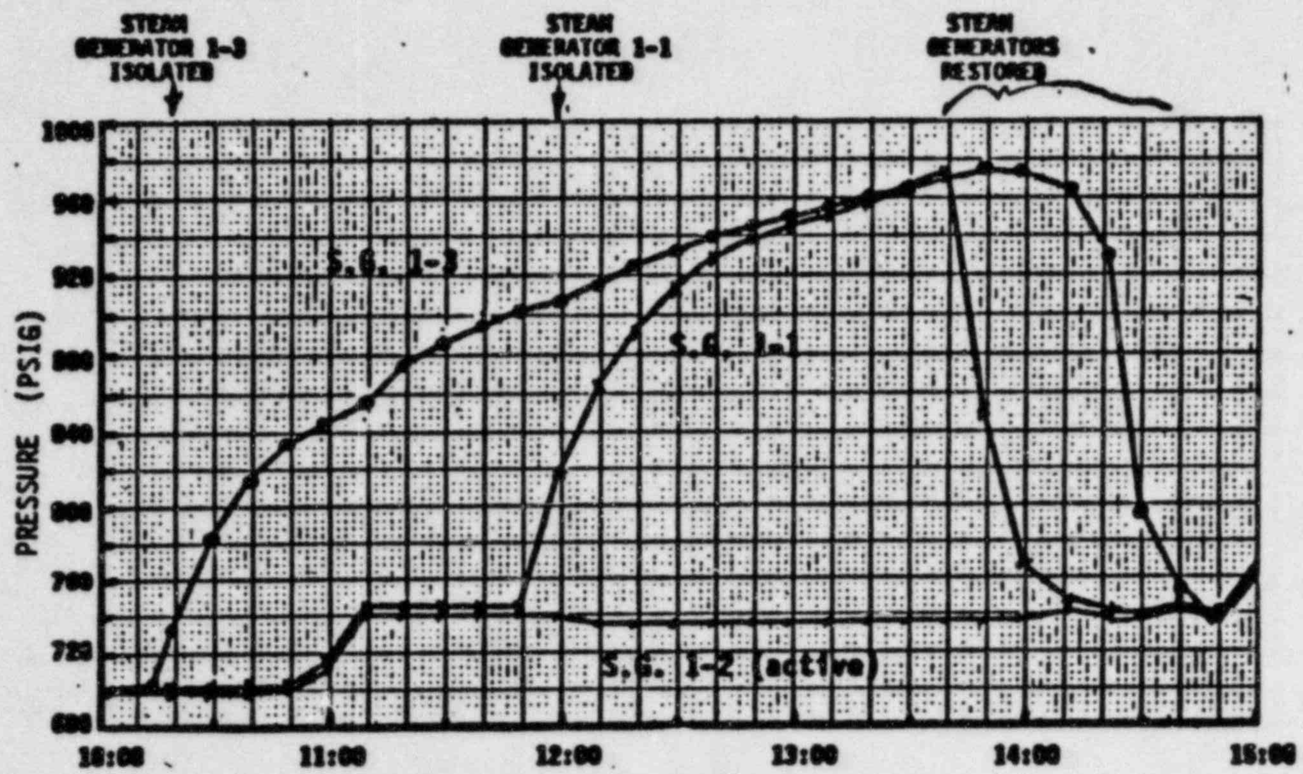
After the 10 deg. F stabilization criteria for the first steam generator was met, the second steam generator, 1-1, was isolated. Again, plant parameters responded as expected with no anomalies noted.

Following the required stabilization period, the steam generators were unisolated in the reverse order that they were isolated; that is, Steam Generator 1-1 was returned to service first followed by Steam Generator 1-3. This was done by equalizing across and opening the main steam isolation valve and restoring auxiliary feedwater. The entire transient is covered by Figures 28 and 29.

TEST RESULTS

The entire evolution progressed in a slow and controlled manner. At no time were any anomalous plant responses or conditions observed. By using the steam dump system, operating personnel were able to control RCS temperatures and thus control isolated steam generator pressures.

Test 1.5 demonstrated that natural circulation could be maintained following isolation of one or more steam generators. All plant responses were consistent with expectations and results obtained at similar Westinghouse pressurized water reactors. Operating personnel demonstrated their ability to initiate, maintain and recover from a steam generator isolation during natural circulation conditions.

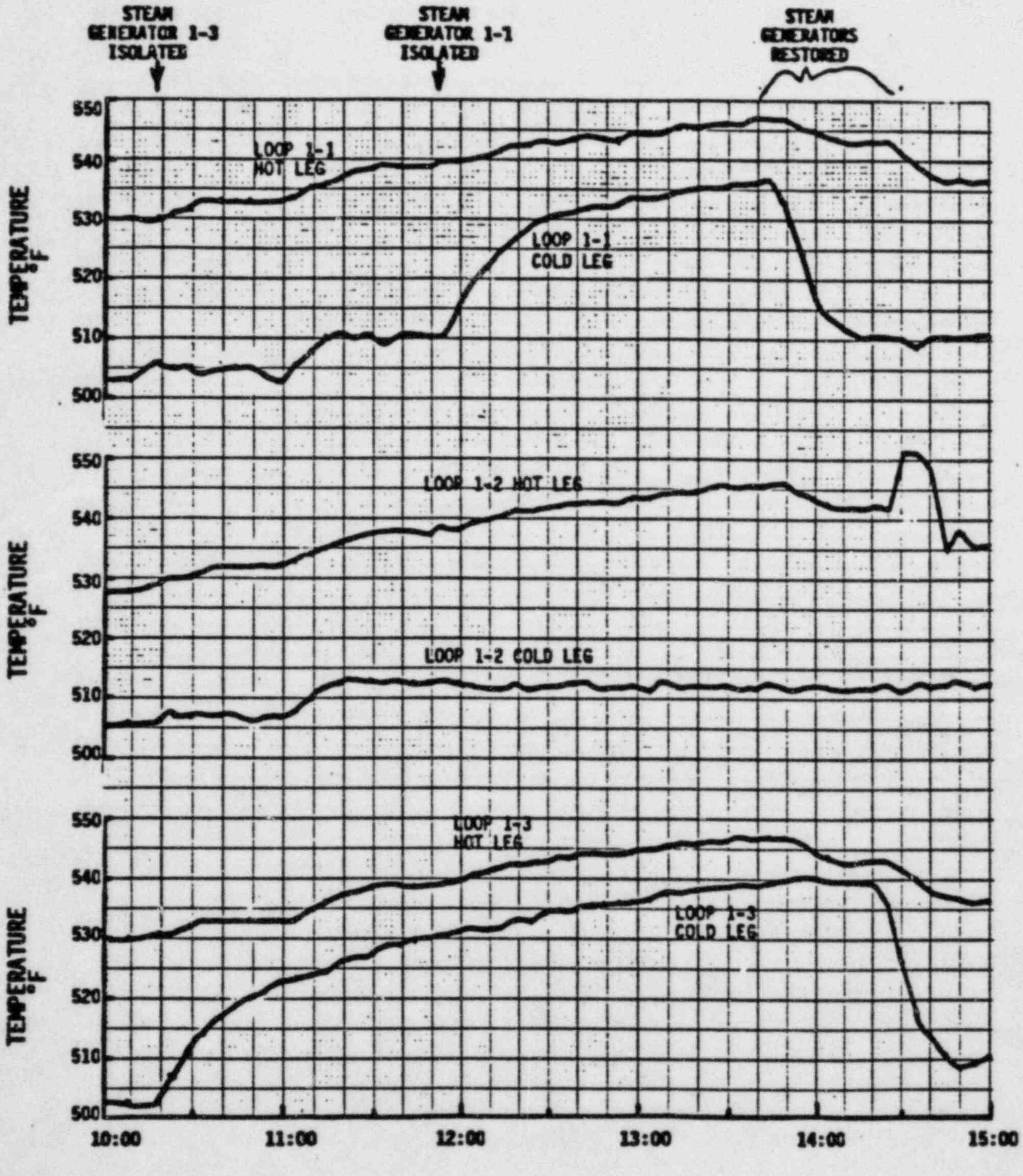


TIME
(5-23-84)

DIABLO CANYON POWER PLANT - UNIT 1

STEAM GENERATOR ISOLATION DURING NATURAL CIRCULATION

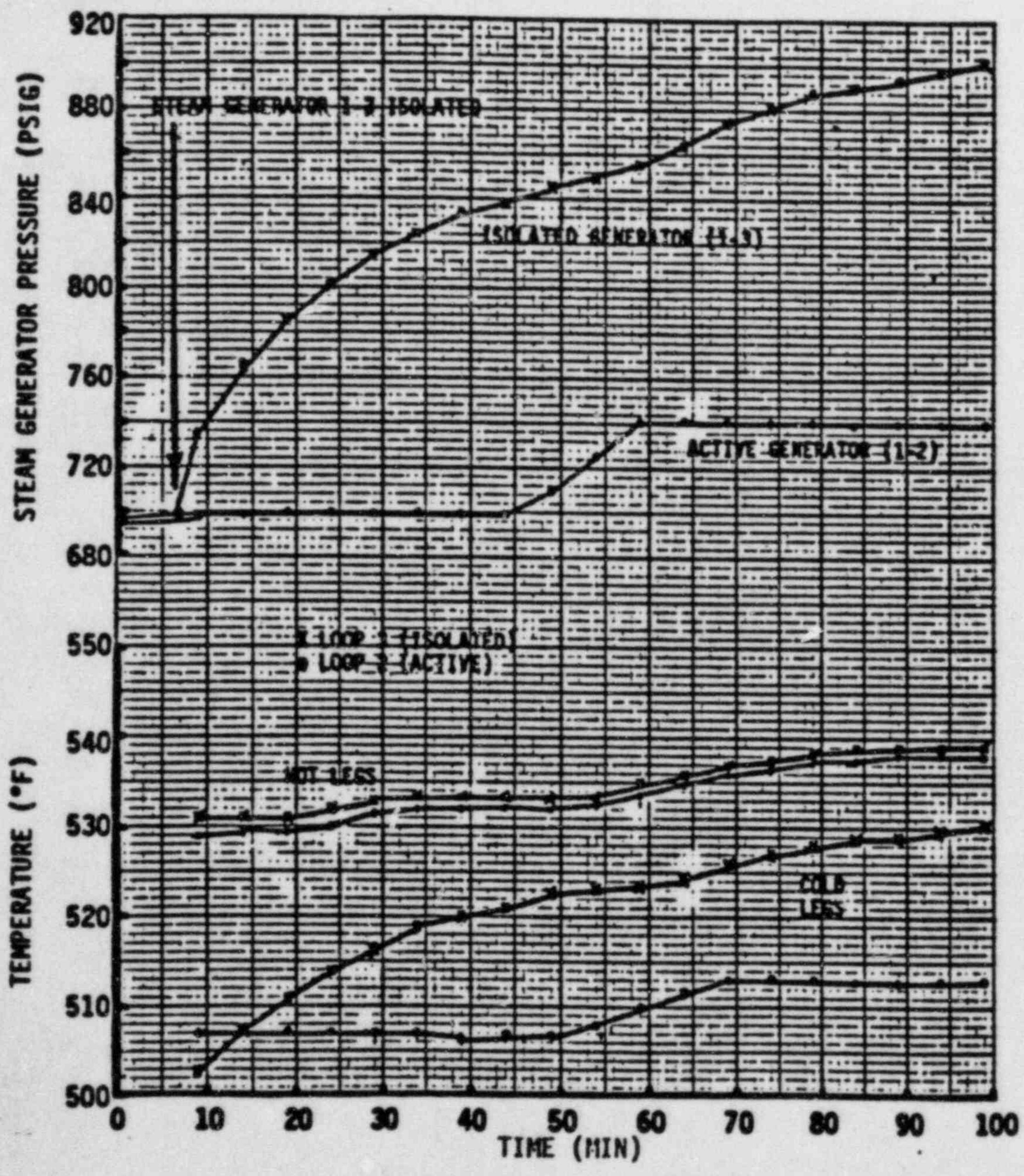
Figure 27



DIABLO CANYON POWER PLANT - UNIT 1

RCL TEMPERATURES DURING STEAM GENERATOR ISOLATION

Figure 28



DIABLO CANYON POWER PLANT - UNIT 1

STEAM GENERATOR PRESSURES DURING STEAM GENERATOR ISOLATION

Figure 29

5.5 Test Procedure No. 44.2 - Cooldown Capability of the Charging and Letdown System

TEST OBJECTIVE

The objective of this test was to determine the capability of the chemical and volume control system (CVCS) charging and letdown system to cooldown the reactor coolant system (RCS).

TEST DESCRIPTION

With the RCS in Mode 3, (RCS temperature of approximately 540 deg. F and RCS pressure of approximately 2235 psig) three of the four reactor coolant pumps were tripped and all steam generators were isolated. The cooldown capability of the CVCS charging and letdown system was determined from the hot and cold leg temperatures of the active loop at maximum (120 gpm) and minimum letdown flow (45 gpm). In addition, core exit thermocouples were monitored to assess core flow distribution.

TEST RESULTS

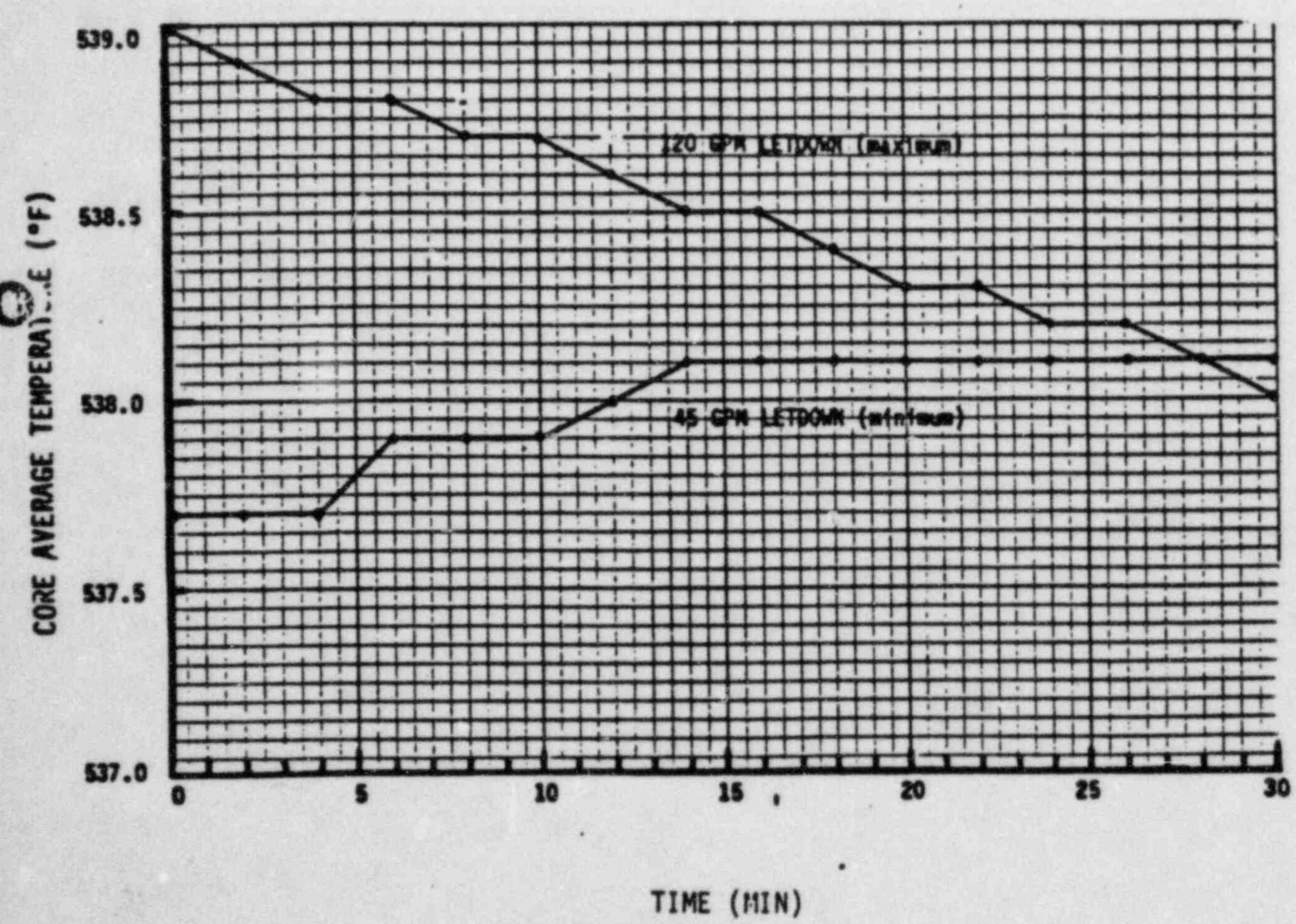
Plant response consisted of an average cooldown rate of about 2.1 deg. F/hr with maximum letdown flow and an average heatup rate of about 0.2 deg. F/hr with minimum flow.

Results from the initial performance of the test are shown in Figure 30. Additional test performances were done for operator training, and average results are summarized in Table 11.

Table 11

CVCS Cooldown Capability

	Maximum Flow Conditions (gpm)	Minimum Flow Conditions (gpm)
Target Charging/Letdown Flow Rate	120	45
Actual Charging Flow Rate	125.6	53.6
Actual Letdown Flow Rate	116.6	44.8
Cooldown Rate	2.1 deg. F/hr	________
Heatup Rate	________	0.2 deg. F/hr



DIABLO CANYON POWER PLANT - UNIT 1

<u>CVCS COOLDOWN CURVE</u>

Figure 30

5.6 Test Procedure No. 44.3 - Simulated Loss of All On-Site and Off-Site AC Power

TEST OBJECTIVE

The objective of this test was to demonstrate the ability to maintain Hot Standby conditions under simulated loss of all on-site and off-site AC power.

TEST DESCRIPTION

The initial conditions for the test were 1) reactor shutdown and the RCS in Hot Standby conditions (i.e., reactor coolant system at approximately 2235 psig and 547 deg. F), 2) all four reactor coolant pumps running, and 3) steam generator levels and steam pressure being maintained with auxiliary feedwater and steam dump respectively. Selected equipment were then tripped or realigned to simulate loss of all AC power. The final initiating event involved tripping the RCP bus undervoltage relays which automatically started the turbine driven auxiliary feedwater pump.

The reactor coolant system was maintained in Hot Standby conditions for two hours using atmospheric dump valves and controlling the steam generator levels by manual control of the turbine driven auxiliary feedwater pump and/or its discharge level control valves.

TEST RESULTS

Plant response was satisfactory in that the turbine driven auxiliary feedwater pump started automatically and the operators were able to successfully maintain steam generator levels within normal operating range during the two hour period. Reactor coolant loop average temperatures were all maintained at or below 551 deg. F. Voltages for all vital batteries remained above the minimum allowable value of 109 Vdc.

PACIFIC GAS AND ELECTRIC COMPANY

PGandE — 77 BEALE STREET • SAN FRANCISCO, CALIFORNIA 94106 • (415) 781-4211 • TWX 910-372-6587

JAMES D. SHIFFER
VICE PRESIDENT
NUCLEAR POWER GENERATION

January 29, 1985

PGandE Letter No.: DCL-85-031

Mr. George W. Knighton, Chief
Licensing Branch No. 3
Division of Licensing
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, D.C. 20555

Re: Docket No. 50-275, OL-DPR-80
Diablo Canyon Unit 1
Startup Test Report

Dear Mr. Knighton:

As required by the Operating License for Unit 1 (Section 6.9 of the Technical Specifications), the Startup Test Report for the period from fuel load to completion of special low power testing is transmitted herewith.

Kindly acknowledge receipt of this material on the enclosed copy of this letter and return it in the enclosed addressed envelope.

Sincerely,

J D Shiffer

Enclosure

cc: J. B. Martin
H. E. Schierling
Service List

RETURN ORig
TO Reg FiLE

IE26
1/40

PGandE Letter No.: DCL-85-031

ENCLOSURE

DIABLO CANYON UNIT 1

STARTUP REPORT

0098S/0025K