

PACIFIC GAS AND ELECTRIC COMPANY

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JAMES D. SHIFFER  
VICE PRESIDENT  
NUCLEAR POWER GENERATION

July 26, 1985

*See Report*

PGandE Letter No.: DCL-85-252

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

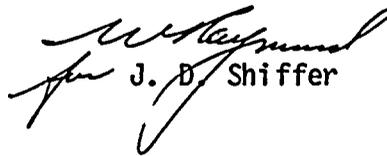
Re: Docket No. 50-275, OL-DPR-80  
Diablo Canyon Unit 1  
Supplement 2 to Startup Report

Dear Mr. Knighton:

As required by the Operating License for Unit 1 (Section 6.9 of the Technical Specifications), enclosed is Supplement 2 to the Startup Report for the period from February 1, 1985 through the completion of power ascension testing on March 29, 1985 and the start of commercial operation on May 7, 1985. This enclosure completes the Unit 1 Startup Reports required by the Technical Specifications.

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

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PGandE Letter No.: DCL-85-252

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DIABLO CANYON POWER PLANT  
UNIT 1 STARTUP REPORT - SUPPLEMENT 2

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UNIT 1 STARTUP REPORT - SUPPLEMENT 2

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## SUMMARY

The Diablo Canyon Power Plant Unit 1 Startup Program activities included in this report cover the period from February 1, 1985 to the completion of Power Ascension testing and start of commercial operation.

This document is the second and final supplement to the Diablo Canyon Power Plant Unit 1 Startup Report. The initial report applied to activities during the period from November 15, 1983 to May 23, 1984, or from Fuel Load through the Special Low Power Test Program. Supplement 1 to the Startup Report covered the period from November 1, 1984 to January 31, 1985, or from receipt of the full power operating license through the 50% power test plateau.

The Maintenance Inspection Program started on January 5, 1985 following a scheduled plant trip from 50% rated thermal power (RTP). On February 7, 1985, the inspection was completed upon the plant's return to 50% power and the ascension to 75% power was begun. Testing at the 75% power plateau was completed on March 8, 1985. The major reasons for delay at this plateau were:

- 1) Plant power had to be reduced to repair secondary steam leaks during the initial power escalation from 50% to 75% power.
- 2) Portable radio transmission near steam-line pressure transmitters caused a safety injection and a reactor trip.
- 3) Conservative thrust bearing wear setpoints on the Main Feedwater pumps caused the pumps to trip, leading to the initiation of a manual reactor trip.

The power level was increased to 90% on March 8, 1985 and testing at this plateau was completed on March 13, 1985.

100% power was achieved on March 13, 1985 and testing was completed on March 29, 1985 after the satisfactory completion of the plant trip from 100% Power and the Natural Circulation Boron Mixing Test.

Following the natural circulation testing, the plant remained shut down for approximately one month. During this outage, turbine stop valve strainers were removed and Regulatory Guide 1.97 commitments were met. Power operation was subsequently resumed and Diablo Canyon Power Plant Unit 1 was declared commercial at 0243 hours on May 7, 1985.

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## 1.0 Test Procedure No. 42.5 - Thermal Power Measurement and Statepoint Data Collection

### TEST OBJECTIVE

The objective of this test was to collect statepoint data and verify core power level at all power ascension test plateaus. The data included temperatures, pressures, and levels related to control and protection instrumentation as well as neutron flux distribution measurements. Core power level was determined by secondary system heat balance calculations.

### TEST DESCRIPTION

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

- Full core flux maps through the use of the Incore Movable Detector System, (except at 15% power),
- Primary plant parameters such as reactor coolant system temperatures and pressures,
- Secondary plant parameters such as steam and feedwater flows; steam generator levels, and various temperatures and pressures.

The collected data had a variety of applications. Full core neutron flux and power distributions were derived from the flux mapping data. Core power was accurately determined by secondary side heat balances on the steam generators. Parameters related to control and safety systems provided input to other tests, which set forth the guidelines for necessary adjustments (if needed) to control systems. The collected information also served as a data base for steady state conditions at each of the power test plateaus during the power ascension test program.



## TEST RESULTS

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

Measured steady state, equilibrium power distributions (i.e., relative fuel assembly power, radial power shape, axial power shape, quadrant power tilt, peaking factors) were within Acceptance Criteria and very close to design predictions. The Acceptance Criteria stipulated that each assembly's relative power (P) must be within 10% ( $P \geq 0.9$ ) or 15% ( $P < 0.9$ ) of design. For all flux maps performed for Test Procedure 42.5, the maximum observed relative assembly power error was 5.8%. Peaking factors were well below safety limits specified by the Technical Specifications. As expected, the radial power distribution was characterized by a small, but acceptable, quadrant tilt. Results of the flux maps are summarized in Table 1 and Figures 2 through 6. At each power plateau, the measured  $F_{\Delta H}^N$  and  $F_Q^T$  were compared to the limiting values at the next power plateau and found acceptable. In a manner consistent with core design methodology and Doppler feedback effects,  $F_{\Delta H}^N$  and  $F_Q^T$  trended lower as power increased.

At each test plateau, test equipment was used to measure steam generator pressure. These pressures were compared to steam line pressure readings taken at the pressure taps at which permanent plant equipment is connected. The steam line pressure drop from the steam generator to the PT taps just outside containment increased with power level. Each loop had a full power pressure drop of just under 20 psi, as shown in Figure 7.

Collected data served as input to other tests discussed elsewhere in this report.



Table 1

Power Distribution Results

ITEM	30% POWER TEST PLATEAU	50% POWER TEST PLATEAU	75% POWER TEST PLATEAU	90% POWER TEST PLATEAU	100% POWER TEST PLATEAU
CONDITIONS* - temperature - boron concentration - burnup	~556 deg. F 1080 ppm 57 MWD/MT	~562 deg. F 1020 ppm 148 MWD/MT	~565 deg. F 939 PPM 620 MWD/MT	~569 deg. F 912 PPM 985 MWD/MT	~571 deg. F 898 PPM 1176 MWD/MT
DATE	11-24-84	12-22-84	2-12-85	3-9-85	3-15-85
$F_{\Delta H}^N$ - measured value - location**	1.427 D04-IJ	1.391 M12-IH	1.358 M12-LE	1.350 M12-LE	1.347 D12-FE
$F_T^Q$ - measured value - location**	2.103 D04-IJ @77"	2.033 M12-IH @74"	2.038 M12-LE @58"	2.007 M12-LE @58"	1.976 D12-FE @55"
$F_Z$ - measured value	1.369	1.359	1.399	1.380	1.358
Quadrant Tilt - measured value	1.008	1.006	1.007	1.007	1.007

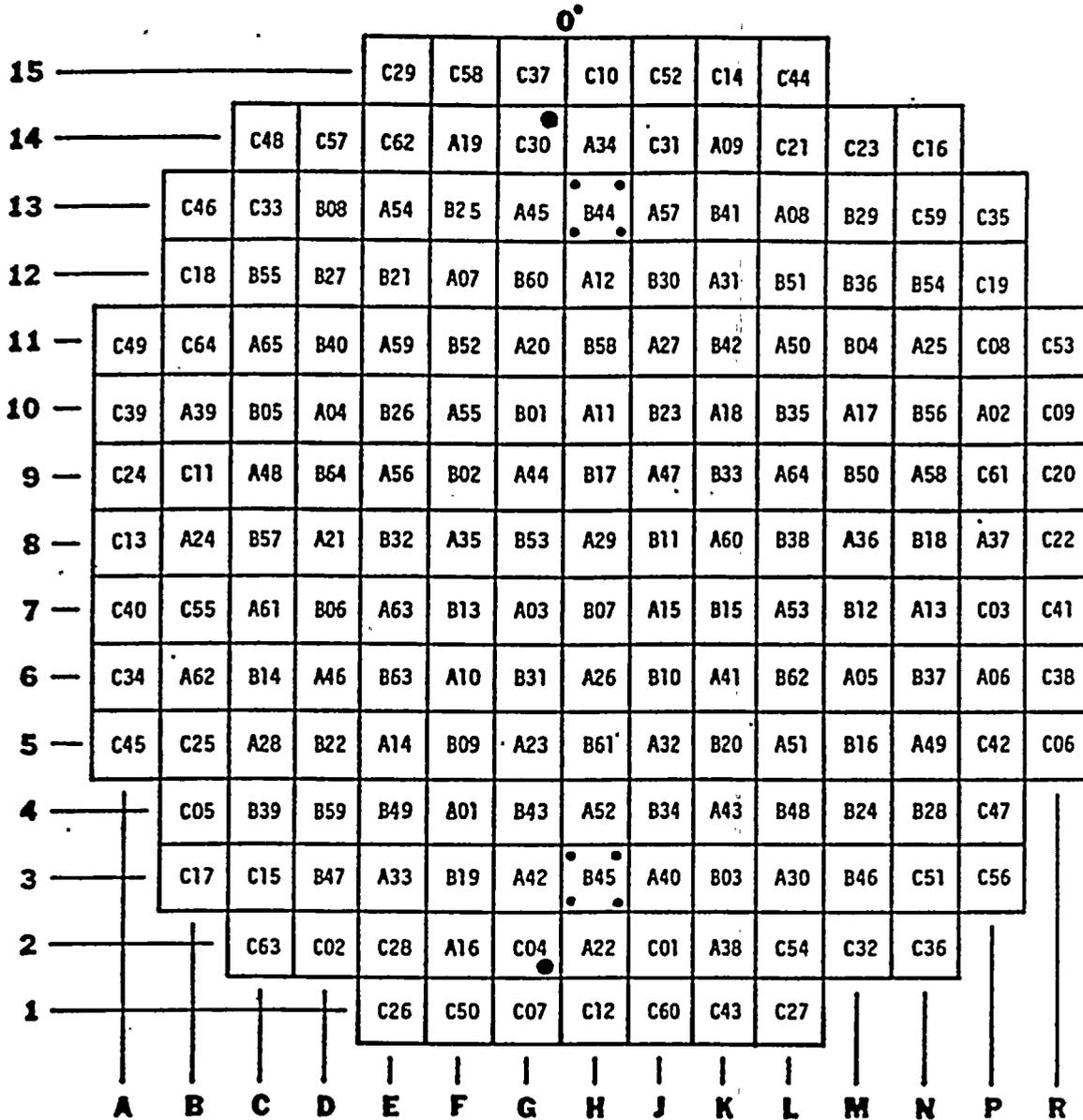
\* Common conditions include stable plant parameters, equilibrium xenon, control rods at or near fully withdrawn positions.

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



FIGURE 1

**DIABLO CANYON POWER PLANT  
UNIT No. 1  
CORE MAP**



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

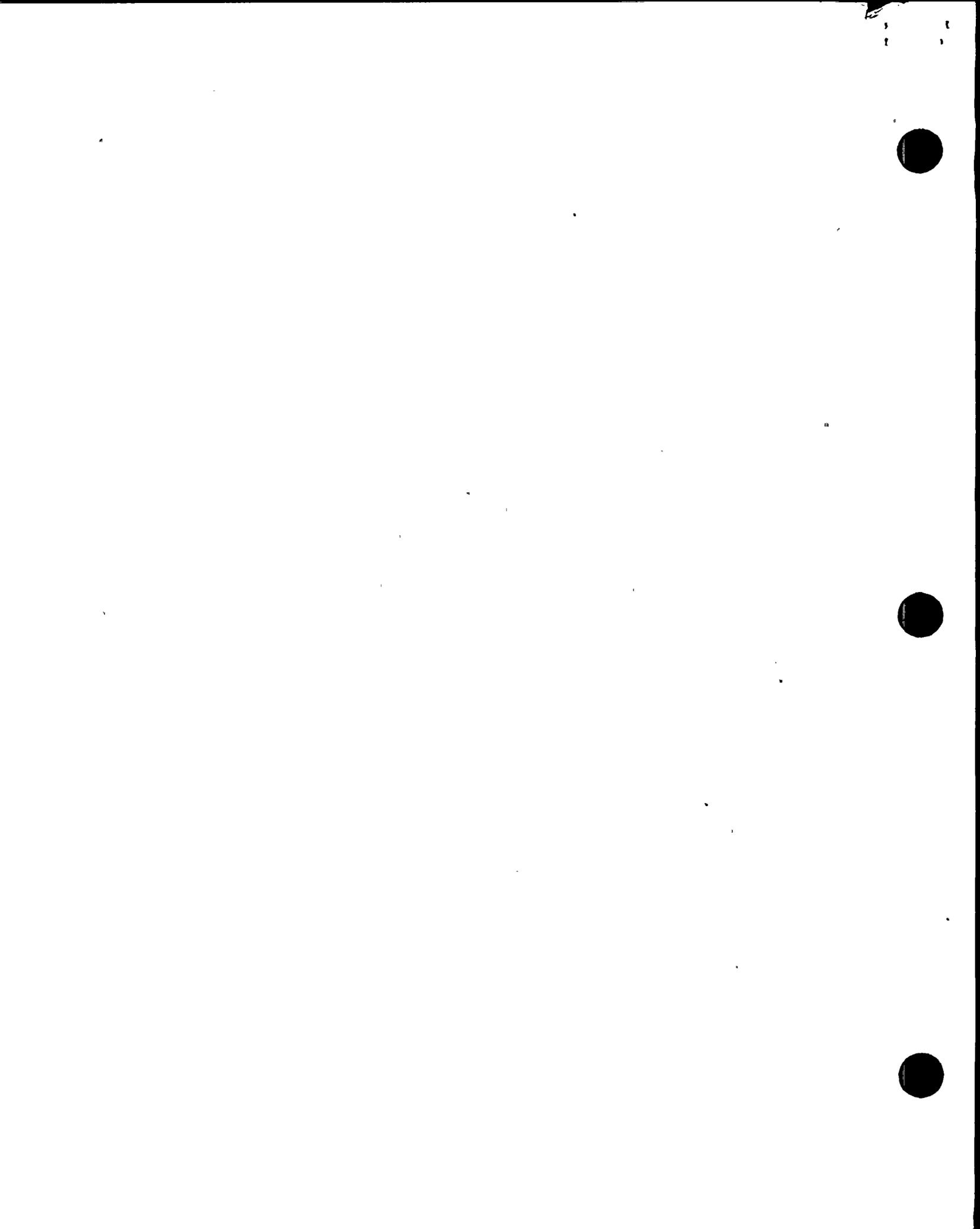


FIGURE 2

**CORE AVERAGE RADIAL POWER DISTRIBUTION - 30% TEST PLATEAU**  
**Assembly Average Powers From Unrodded Flux Map**

RELATIVE ASSEMBLY  
 POWER (Pi)  
 MEASURED Pi - EXPECTED Pi  
 ----- X 100  
 EXPECTED Pi

1					.604	.692	.820	.741	.817	.679	.577								
					3.7	.9	1.2	.3	.8	-.9	-.9								
2					.503	.909	1.057	1.023	1.064	1.045	1.058	1.002	1.010	.869	.520				
					.8	3.7	3.7	1.2	1.5	.6	.9	-.9	-.9	-.9	2.6				
3					.512	1.063	.592	1.148	1.138	1.184	1.139	1.177	1.088	1.097	.985	1.081	.520		
					1.0	.9	-.8	3.7	3.7	2.5	1.9	1.9	-.9	-.9	2.6	2.6	2.6		
4					.886	.967	1.276	1.057	1.212	1.157	1.186	1.135	1.137	1.095	1.322	.986	.901		
					1.0	.8	-.8	-.8	3.7	1.2	-.5	-.7	-2.7	2.8	2.8	2.7	2.7		
5					.587	1.017	1.106	1.050	1.171	1.133	1.169	1.115	1.155	1.096	1.196	1.095	1.138	1.047	.599
					.8	-.2	-.1	-1.4	.6	.6	.8	-.3	-.5	-2.7	2.8	2.8	2.8	2.8	2.8
6					.680	.996	1.071	1.140	1.114	1.133	1.016	1.077	.994	1.094	1.087	1.181	1.109	1.022	.704
					-.8	-1.5	-2.4	2.5	-1.1	.3	.4	-.6	-1.8	-3.1	-3.5	1.1	1.1	1.1	2.8
7					.807	1.038	1.127	1.115	1.146	1.009	1.041	.952	1.027	.977	1.145	1.146	1.169	1.063	.839
					-.5	-1.0	-2.5	-2.5	-1.2	-.3	-.1	-.3	-1.5	-3.5	-1.3	.2	1.2	1.4	3.5
8					.738	1.038	1.086	1.158	1.086	1.075	.947	1.008	.941	1.067	1.105	1.196	1.133	1.050	.750
					-.1	-.1	-2.9	-2.9	-2.9	-.8	-.9	-.4	-1.5	-1.6	-1.1	.3	1.3	1.1	1.6
9					.810	1.047	1.116	1.104	1.120	.984	1.017	.941	1.047	1.017	1.170	1.153	1.176	1.066	.822
					-.1	-.1	-3.5	-3.5	-3.5	-2.8	-2.5	-1.5	.4	.4	.8	.8	1.7	1.6	1.4
10					.711	1.050	1.060	1.128	1.087	1.091	.982	1.064	1.002	1.134	1.133	1.176	1.104	1.023	.693
					3.8	3.8	-3.5	-3.5	-3.5	-3.4	-3.0	-1.9	-1.0	.4	.6	.6	.6	1.2	1.2
11					.605	1.058	1.150	1.029	1.144	1.101	1.137	1.084	1.127	1.089	1.146	1.050	1.091	1.045	.597
					3.8	3.8	3.8	-3.5	-1.7	-2.2	-2.0	-3.0	-2.9	-3.2	-1.5	-1.5	1.5	2.6	2.6
12					.902	.978	1.309	1.075	1.178	1.131	1.158	1.110	1.031	1.028	1.241	.958	.904		
					2.9	1.9	1.8	.9	.8	-1.2	-2.9	-2.9	-3.2	-3.6	-3.6	-.1	3.1		
13					.516	1.074	.977	1.118	1.093	1.137	1.082	1.150	1.094	1.111	.953	1.062	.524		
					1.9	1.9	1.8	1.0	-.5	-1.6	-3.3	-.5	-.3	.4	-.7	.8	3.5		
14					.516	.891	1.034	1.009	1.044	1.038	1.078	1.035	1.043	.897	.521				
					1.9	1.6	1.5	-.2	-.4	-.1	2.8	2.4	2.3	2.2	2.8				
15					.590	.679	.810	.740	.833	.702	.596								
					1.3	-.9	-.1	.3	2.8	2.4	2.3								
	R	P	N	M	L	K	J	H	G	F	E	D	C	B	A				

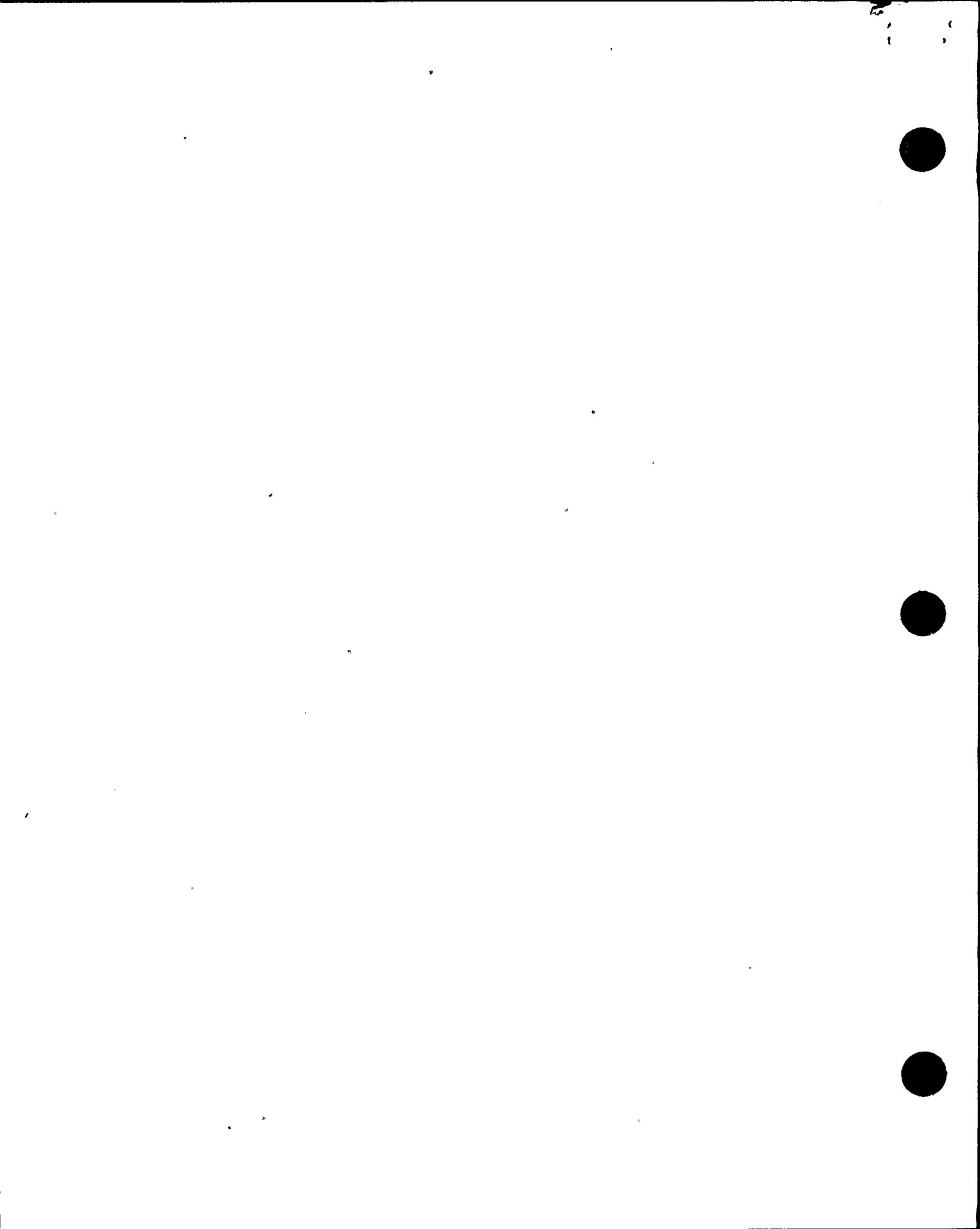


FIGURE 3

**CORE AVERAGE RADIAL POWER DISTRIBUTION - 50% TEST PLATEAU**  
 Assembly Average Powers From Unrodded Flux Map

RELATIVE ASSEMBLY  
 POWER (P<sub>i</sub>)

MEASURED P<sub>i</sub> - EXPECTED P<sub>i</sub>  
 ----- X100  
 EXPECTED P<sub>i</sub>

					.586	.686	.812	.741	.812	.683	.581									
1					.9	.5	.8	.8	.7	-.0	-.0									
						.517	.892	1.020	1.013	1.036	1.026	1.026	.933	.997	.859	.503				
2					2.6	2.6	.9	.5	-.5	-1.0	-1.5	-1.5	-1.4	-1.2	-.1					
						.511	1.056	.962	1.114	1.105	1.154	1.110	1.143	1.063	1.081	.940	1.040	.509		
3					1.5	1.5	1.1	.9	.9	-.2	-.6	-1.1	-3.0	-2.1	-1.2	-.1	1.1			
						.879	.960	1.271	1.059	1.167	1.132	1.195	1.136	1.164	1.047	1.209	.952	.875		
4					1.1	.9	-.4	-.5	-.5	-1.3	-.2	-.9	-.7	-1.7	-.6	.1	.7			
						.587	1.014	1.108	1.059	1.173	1.138	1.198	1.154	1.198	1.136	1.158	1.061	1.103	1.010	.578
5					1.0	.3	.4	-.5	.4	.5	2.5	2.5	2.5	.4	-.9	-.3	-.1	-.1	-.6	
						.681	1.003	1.077	1.156	1.125	1.149	1.044	1.122	1.038	1.138	1.120	1.167	1.091	1.004	.679
6					-.2	-.5	-1.7	-1.4	-.5	.8	2.1	2.2	1.4	-.2	-1.1	-.5	-.4	-.4	-.6	
						.798	1.035	1.142	1.133	1.161	1.031	1.076	.987	1.066	1.016	1.172	1.139	1.155	1.038	.816
7					-1.0	-.7	-1.2	-1.2	-.7	.8	1.8	2.0	.8	-.7	.2	-.7	-.0	-.4	1.2	
						.721	1.022	1.099	1.183	1.107	1.103	.979	1.037	.965	1.094	1.127	1.191	1.118	1.034	.746
8					-1.9	-1.4	-1.6	-1.2	-1.6	.5	1.1	.8	-.3	-.3	.2	-.5	.0	-.1	1.5	
						.790	1.027	1.142	1.136	1.153	1.015	1.057	.968	1.059	1.023	1.179	1.148	1.165	1.042	.817
9					-1.9	-1.4	-1.2	-1.0	-1.4	-.7	-.1	.0	.1	.1	.8	.1	.8	-.0	1.4	
						.683	1.007	1.071	1.160	1.119	1.127	1.015	1.092	1.023	1.146	1.143	1.184	1.114	1.022	.692
10					.1	-.0	-2.2	-1.0	-1.1	-1.1	-.8	-.4	.0	.5	1.0	1.0	1.7	1.4	1.4	
						.594	1.034	1.129	1.054	1.158	1.120	1.160	1.113	1.156	1.128	1.179	1.074	1.112	1.039	.597
11					2.3	2.3	2.3	-1.0	-.9	-1.0	-.8	-1.1	-1.1	-.3	.9	.9	.7	2.8	2.8	
						.880	.954	1.290	1.071	1.179	1.131	1.176	1.126	1.167	1.070	1.282	.967	.898		
12					1.2	.2	1.0	.7	.6	-1.4	-1.8	-1.8	-.4	.5	.4	1.6	3.2			
						.507	1.047	.965	1.117	1.094	1.130	1.088	1.137	1.088	1.112	.960	1.060	.522		
13					.6	.6	1.4	1.2	-.1	-2.2	-2.6	-1.6	-.7	.8	.9	1.8	3.7			
						.507	.877	1.023	1.007	1.022	1.017	1.034	1.013	1.024	.887	.518				
14					.7	.9	1.2	-.1	-1.9	-1.9	-.8	.6	1.3	2.0	2.8					
15						.578	.673	.796	.727	.809	.686	.588								
						-.4	-1.4	-1.2	-1.0	.4	.6	1.3								
	R	P	N	M	L	K	J	H	G	F	E	D	C	B	A					



FIGURE 4

CORE AVERAGE RADIAL POWER DISTRIBUTION - 75% TEST PLATEAU  
 Assembly Average Powers From Unrodded Flux Map

RELATIVE ASSEMBLY  
 POWER (P<sub>i</sub>)  
 MEASURED P<sub>i</sub> - EXPECTED P<sub>i</sub>  
 ----- X 100  
 EXPECTED P<sub>i</sub>

1				.567	.649	.779	.721	.784	.643	.549					
				1.7	-.4	.8	.9	1.4	-1.4	-1.4					
2		.505	.858	.978	.947	1.003	1.005	.990	.917	.945	.838	.495			
		3.0	3.1	1.6	-.0	-.4	-.9	-1.8	-3.2	-1.9	.7	1.0			
3	.498	1.005	.941	1.096	1.100	1.162	1.131	1.143	1.029	1.057	.931	1.000	.497		
	1.6	1.6	1.6	1.3	1.4	.9	.5	-.8	-5.2	-2.3	.6	1.0	1.5		
4	.841	.934	1.227	1.061	1.177	1.169	1.213	1.151	1.149	1.047	1.229	.936	.844		
	1.1	.9	-.5	-1.3	-1.3	-.8	-1.2	-2.3	-3.6	-2.6	-.4	1.1	1.4		
5	.559	.966	1.088	1.069	1.189	1.171	1.230	1.182	1.217	1.145	1.160	1.064	1.092	.973	.563
	.2	.3	.6	-.5	-.6	-.5	1.0	.0	-.1	-2.7	-3.0	-.9	.9	1.1	1.0
6	.643	.943	1.079	1.181	1.169	1.207	1.100	1.163	1.163	1.159	1.138	1.183	1.091	.956	.659
	-1.3	-.5	-.5	-.9	-.7	.9	1.1	.1	-2.4	-3.1	-3.4	-.7	.5	.9	1.1
7	.782	1.013	1.144	1.158	1.194	1.081	1.128	1.034	1.112	1.076	1.222	1.185	1.168	1.021	.797
	1.3	.5	-.6	-1.8	-2.0	-.7	.4	.4	-1.0	-1.2	.3	.5	1.4	1.3	3.1
8	.728	1.023	1.116	1.200	1.143	1.147	1.037	1.084	1.041	1.159	1.174	1.223	1.143	1.024	.731
	1.8	.8	-.9	-2.3	-3.3	-1.2	.7	2.4	1.0	-.2	-.6	-.4	1.6	1.0	2.2
9	.785	1.016	1.143	1.145	1.170	1.052	1.093	1.029	1.160	1.098	1.228	1.182	1.180	1.020	.788
	1.6	.8	-.7	-2.9	-3.9	-3.3	-2.7	-.1	3.3	.8	.7	.3	2.5	1.2	2.0
10	.688	.970	1.077	1.180	1.164	1.175	1.050	1.149	1.091	1.205	1.181	1.198	1.118	.959	.659
	2.5	2.4	-.7	-1.0	-1.1	-1.8	-3.6	-1.1	.2	.8	.3	.5	3.1	1.2	1.2
11	.587	1.013	1.138	1.065	1.202	1.174	1.202	1.139	1.173	1.168	1.216	1.093	1.101	.990	.573
	5.3	5.2	5.2	-.9	.5	-.3	-1.3	-3.6	-3.8	-.8	1.7	1.7	1.8	2.8	2.9
12	.865	.951	1.258	1.098	1.219	1.164	1.183	1.136	1.181	1.087	1.247	.936	.851		
	4.0	2.8	1.9	2.2	2.3	-1.2	-3.6	-3.6	-.9	1.1	1.1	1.1	2.2		
13	.502	1.014	.947	1.112	1.091	1.135	1.084	1.115	1.051	1.095	.937	1.003	.499		
	2.5	2.5	2.3	2.8	.6	-1.4	-3.7	-3.2	-3.2	1.2	1.2	1.4	1.8		
14	.502	.847	.974	.942	.988	.995	1.003	.969	.983	.848	.499				
	2.4	1.8	1.2	-.6	-2.0	-1.9	-.5	2.3	2.1	1.9	1.8				
15				.558	.639	.764	.709	.789	.666	.569					
				.2	-1.9	-1.1	-.8	2.1	2.2	2.1					
	R	P	N	M	L	K	J	H	G	F	E	D	C	B	A

DIABLO CANYON POWER PLANT - UNIT I



FIGURE 5

**CORE AVERAGE RADIAL POWER DISTRIBUTION - 90% TEST PLATEAU**  
**Assembly Average Powers From Unrodded Flux Map**

RELATIVE ASSEMBLY  
 POWER (Pi)  

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

1				.559 1.9	.653 1.4	.771 1.2	.710 .5	.759 -.4	.642 -.3	.547 -.4					
2		.499 3.7	.850 3.8	.971 1.9	.959 1.6	1.000 -.4	.989 -1.8	.980 -2.3	.930 -1.4	.944 -1.1	.816 -.2	.485 .8			
3	.494 2.7	.998 2.7	.943 2.4	1.094 1.7	1.106 1.7	1.145 -.5	1.109 -1.7	1.127 -2.0	1.056 -2.8	1.059 -1.6	.917 -.3	.980 .8	.490 1.9		
4	.837 2.3	.939 2.1	1.217 -.6	1.052 -2.5	1.164 -2.5	1.169 -1.5	1.214 -1.4	1.165 -1.9	1.163 -2.6	1.055 -2.2	1.221 -.4	.933 1.4	.837 2.3		
5	.569 1.9	.967 1.4	1.093 1.6	1.074 -.5	1.181 -1.5	1.172 -1.4	1.224 -.2	1.187 -.7	1.218 -.8	1.160 -2.4	1.161 -3.2	1.075 -.4	1.090 1.3	.966 1.3	.549 .1
6	.647 .4	.950 .7	1.079 -.7	1.176 -1.5	1.169 -1.7	1.198 -.7	1.103 -.3	1.167 -.8	1.084 -2.1	1.174 -2.8	1.149 -3.3	1.190 -.3	1.096 .8	.955 1.2	.646 .2
7	.775 1.8	1.013 1.0	1.142 -.7	1.171 -1.4	1.205 -1.8	1.096 -.9	1.135 -.5	1.047 -.5	1.120 -1.8	1.082 -2.2	1.216 -.9	1.187 .0	1.159 .7	1.011 .7	.777 2.0
8	.717 1.5	1.014 .7	1.116 -1.1	1.209 -1.8	1.165 -2.5	1.162 -1.2	1.049 -.3	1.095 1.1	1.050 -.3	1.160 -1.4	1.176 -1.6	1.220 -.9	1.139 .9	1.015 .8	.721 2.0
9	.772 1.4	1.011 .7	1.143 -.6	1.161 -2.1	1.190 -3.0	1.078 -2.6	1.117 -2.0	1.050 -.3	1.154 1.2	1.098 -.8	1.221 -.5	1.183 -.4	1.169 1.7	1.014 1.0	.776 1.9
10	.655 1.7	.959 1.7	1.079 -.7	1.177 -1.4	1.170 -1.6	1.184 -1.9	1.080 -2.4	1.168 -.7	1.112 .5	1.204 -.2	1.185 -.3	1.193 -.1	1.115 2.6	.959 1.6	.654 1.5
11	.569 3.6	.958 3.6	1.114 3.5	1.065 -1.3	1.192 -.6	1.175 -1.1	1.211 -1.3	1.165 -2.5	1.197 -2.5	1.176 -1.0	1.215 1.3	1.094 1.4	1.095 1.8	.978 2.5	.563 2.6
12	.845 3.3	.948 3.0	1.248 1.9	1.092 1.2	1.208 1.2	1.170 -1.4	1.198 -2.7	1.154 -2.7	1.181 -1.1	1.089 .9	1.236 .9	.949 3.1	.851 4.0		
13	.494 2.7	.999 2.7	.944 2.6	1.094 1.7	1.095 .8	1.137 -1.2	1.100 -2.5	1.116 -3.0	1.057 -2.7	1.100 2.2	.940 2.1	1.004 3.2	.507 5.4		
14	.494 2.7	.837 2.2	.971 1.8	.953 1.0	.998 -.5	1.001 -.6	1.002 -.1	.979 3.8	.967 3.5	.846 3.3	.502 4.3				
15				.555 1.2	.645 .2	.768 .8	.713 .9	.782 2.7	.662 2.8	.568 3.6					
	R	P	N	M	L	K	J	H	G	F	E	D	C	B	A

NORTH



FIGURE 6

**CORE AVERAGE RADIAL POWER DISTRIBUTION - 100% TEST PLATEAU**  
**Assembly Average Powers From Unrodded Flux Map**

RELATIVE ASSEMBLY  
 POWER (Pi)  
  
 MEASURED Pi - EXPECTED Pi  
 ----- X 100  
 EXPECTED Pi

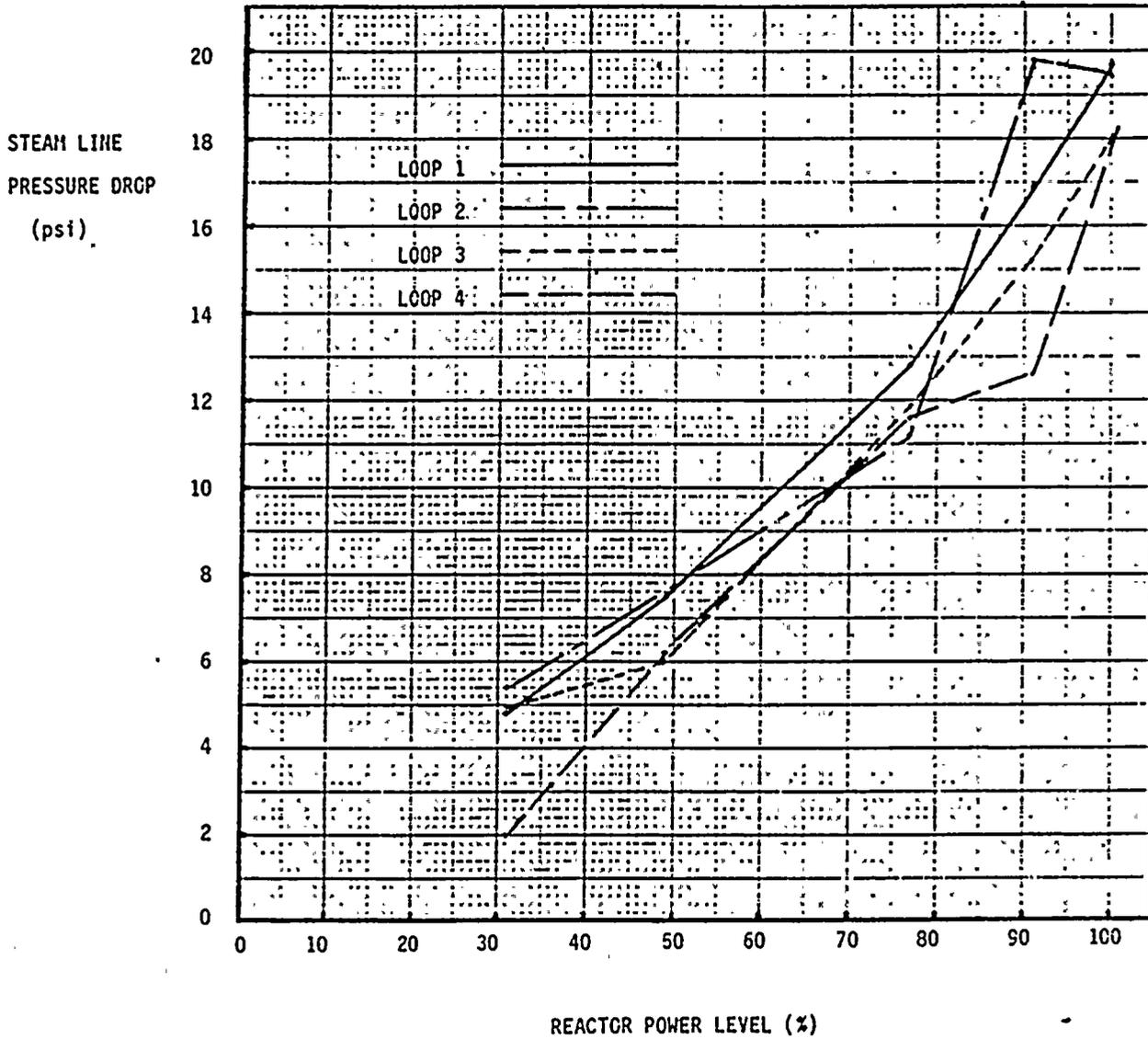
1				.566 1.7	.666 1.2	.778 1.1	.715 .6	.758 -1.6	.647 -1.6	.547 -1.6					
2		.491 2.5	.837 2.5	.981 1.7	.986 1.2	1.012 -.3	.995 -1.5	.991 -2.4	.957 -1.8	.950 -1.4	.811 -.7	.478 -.1			
3	.485 1.4	.979 1.4	.928 1.2	1.094 1.7	1.113 1.7	1.142 -.6	1.106 -1.9	1.128 -1.9	1.071 -2.1	1.061 -1.4	.911 -.7	.964 -.1	.481 .5		
4	.824 1.0	.925 .8	1.226 .9	1.096 2.1	1.209 2.1	1.160 -1.7	1.205 -1.2	1.163 -1.4	1.167 -1.5	1.056 -1.6	1.207 -.6	.920 .3	.822 .7		
5	.559 .5	.966 .2	1.080 .4	1.079 .5	1.206 1.7	1.197 1.6	1.227 1.1	1.192 .6	1.220 .6	1.164 -1.2	1.162 -2.1	1.067 -.6	1.085 .9	.973 .9	.561 .9
6	.653 -.8	.969 -.6	1.076 -1.7	1.178 -.6	1.179 .1	1.208 1.3	1.114 1.4	1.178 1.0	1.096 -.4	1.171 -1.9	1.148 -2.6	1.181 -.3	1.102 .7	.982 .7	.664 .9
7	.768 -.3	1.012 -.4	1.134 -1.4	1.162 -1.5	1.199 -1.2	1.097 -.2	1.144 .6	1.065 .5	1.134 -.3	1.083 -1.5	1.209 -.3	1.183 .3	1.158 .7	1.020 .5	.782 1.5
8	.708 -.5	1.005 -.5	1.109 -1.6	1.199 -1.7	1.159 -2.2	1.157 -.8	1.054 -.5	1.111 -.5	1.048 -1.1	1.159 -.6	1.185 .1	1.226 .5	1.135 .7	1.012 .1	.716 .7
9	.766 -.6	1.009 -.6	1.139 -.9	1.162 -1.5	1.189 -2.0	1.077 -2.0	1.113 -2.1	1.043 -1.6	1.135 -.2	1.101 .1	1.230 1.4	1.191 1.0	1.164 1.2	1.015 -.0	.774 .5
10	.656 -.3	.971 -.4	1.079 -1.4	1.180 -.4	1.172 -.5	1.180 -1.1	1.073 -2.5	1.145 -1.9	1.083 -1.5	1.201 .7	1.198 1.7	1.206 1.8	1.121 2.5	.977 .3	.660 .3
11	.560 .7	.971 .7	1.083 .7	1.071 -.2	1.191 .4	1.174 -.3	1.199 -1.1	1.154 -2.6	1.180 -2.7	1.172 -.5	1.215 2.4	1.100 2.4	1.104 2.7	.976 1.3	.563 1.3
12	.825 1.0	.930 1.3	1.238 1.9	1.095 2.0	1.208 2.0	1.164 -1.3	1.184 -2.9	1.145 -2.9	1.177 -.6	1.100 2.4	1.245 2.5	.957 4.3	.846 3.6		
13	.487 1.8	.982 1.8	.936 2.0	1.098 2.1	1.102 .8	1.135 -1.3	1.095 -2.8	1.113 -3.2	1.061 -3.0	1.103 2.6	.941 2.6	1.000 3.7	.506 5.8		
14	.487 1.9	.828 1.4	.972 .9	.973 -.1	.997 -1.8	.992 -1.8	.998 -1.7	.996 2.2	.985 2.2	.834 2.2	.497 3.9				
15				.552 -.7	.649 -1.3	.762 -1.0	.705 -.9	.769 -.1	.657 -.1	.568 2.2					
	R	P	N	M	L	K	J	H	G	F	E	D	C	B	A

↓  
NORTH



**FIGURE 7**  
**STEAM LINE PRESSURE DROP**

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



POWER *	PRESSURE DROP (psi)			
	LOOP 1	LOOP 2	LOOP 3	LOOP 4
31.0	4.8	5.4	4.9	2.0
49.0	7.5	N/A	6.0	6.2
77.1	12.8	11.2	11.9	11.6
90.7	16.9	19.8	15.2	12.6
100.1	19.7	19.5	18.0	18.0

\*DATA UNRELIABLE BELOW 30% POWER



## 2.0 Test Procedure No. 1.15 - Radiation Surveys and Shielding Effectiveness

### TEST OBJECTIVE

The objective of this procedure was to verify the adequacy of the radiation surveys and shielding effectiveness program as prescribed by Nuclear Plant Operations (NPO) Procedure TC 8401. The main objective of the test program was to measure radiation levels in accessible areas of Unit 1 at various power levels and identify any locations where shielding may be deficient.

### TEST DESCRIPTION

Radiation measurement points, or radiation base points (RBPs) were located throughout the DCPD site. The purpose of the RBPs was to provide fixed points outside radiation shields from which the neutron and gamma radiation levels could be measured. The radiation levels at shield wall pipe penetrations and the area close to the Steam Generators were also measured. The measured radiation levels were then compared to the FSAR design criterion to determine the adequacy of the shield. Most RBPs were located outside the secondary shield wall of Unit 1. Secondary shielding is defined as the shielding in the reactor building designed to attenuate the gamma radiation emanating from the primary coolant system external to the reactor vessel. Labyrinth entrances and shielding penetrations were closely monitored to determine the adequacy of the shielding. Most RBPs were selected to verify that the radiation levels at labyrinth entrances and penetrations met design radiation levels. Radiation base points were also located to test shielding thickness adequacy by measuring the radiation levels on the shield side farthest away from the radiation source.

Neutron and gamma radiation measurements were taken at each fixed radiation base point unless dose rates precluded measurement. The radiation dose rates measured during the early stages of the testing program were linearly extrapolated to the 100% power level. Background radiation measurements were taken prior to the start-up of the Unit 1 reactor. Background radiation measurements were taken to provide values against which measurements made during the start-up phase of Unit 1 could be compared. Radiation measurements during the start-up phase of DCPD Unit 1 were taken at 0%, 5%, 15%, 30%, 50%, 75%, and 100% power levels.

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

### TEST RESULTS

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



### 3.0 Test Procedure No. 42.9 - Operational Alignment of Nuclear Instrumentation

#### TEST OBJECTIVE

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

#### TEST DESCRIPTION

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

Prior to Startup, the initial trip setpoint for the instrumentation power range channels was determined. In addition, the trip settings for the flux deviation averaging amplifiers were determined. During Startup, the overlaps between source range and intermediate range and between the intermediate range and power range channels were determined. During power ascension, the power range detector currents vs. core power were determined and the flux deviation alarm settings were monitored. At the 50% power test plateau, the intermediate and power range detector voltages were checked.

After shutdown from power operations at the 50% power test plateau, the source range operating voltages were checked, the intermediate range detectors' compensation voltages were set and the currents for each power range channel which gives 100% power level indication were obtained.

#### TEST RESULTS

Required adjustments, calibrations, and setpoint determinations were accomplished without significant problems using standard I&C procedures. Selected results of the nuclear instrumentation overlap data taken prior to criticality and at all major testing plateaus are shown in Table 2. Results during and prior to the 50% power test plateau had been presented in Supplement 1 to the Startup Report.

Power range detector high level trip setpoints were reset prior to power increase to the next power plateau as listed in Table 3.



Table 2

## Nuclear Instrumentation Overlap Data

DETECTOR	PRECITICAL READINGS	0% POWER	~3% POWER	~15% POWER	~30% POWER	~50% POWER	~75% POWER	~90% POWER	~100% POWER
SOURCE RANGE (cps)									
N31 - Control Board	45	$3.8 \times 10^4$	Blocked						
N31 - NI Drawer	45	$3.1 \times 10^4$	Blocked						
N32 - Control Board	55	$4.8 \times 10^4$	Blocked						
N32 - NI Drawer	55	$4.8 \times 10^4$	Blocked						
INTERMEDIATE RANGE (amps)									
N35 - Control Board	$1.0 \times 10^{-11}$	$2.0 \times 10^{-10}$	$1.2 \times 10^{-5}$	$8.0 \times 10^{-5}$	$1.0 \times 10^{-4}$	$2.5 \times 10^{-4}$	$3.8 \times 10^{-4}$	$4.5 \times 10^{-4}$	$4.9 \times 10^{-4}$
N35 - NI Drawer	$1.0 \times 10^{-11}$	$2.0 \times 10^{-10}$	$9.9 \times 10^{-6}$	$6.0 \times 10^{-5}$	$1.0 \times 10^{-4}$	$1.4 \times 10^{-4}$	$2.7 \times 10^{-4}$	$3.1 \times 10^{-4}$	$3.3 \times 10^{-4}$
N36 - Control Board	$1.0 \times 10^{-11}$	$2.0 \times 10^{-10}$	$1.3 \times 10^{-5}$	$8.0 \times 10^{-5}$	$1.0 \times 10^{-4}$	$2.5 \times 10^{-4}$	$4.0 \times 10^{-4}$	$4.5 \times 10^{-4}$	$5.0 \times 10^{-4}$
N36 - NI Drawer	$1.0 \times 10^{-11}$	$2.0 \times 10^{-10}$	$1.0 \times 10^{-5}$	$6.0 \times 10^{-5}$	$1.0 \times 10^{-4}$	$1.4 \times 10^{-4}$	$2.8 \times 10^{-4}$	$3.2 \times 10^{-4}$	$3.4 \times 10^{-4}$
POWER RANGE (%)									
N41 - Control Board	0	0	3.0	15.0	32.0	51.0	76.0	91.0	100.5
N41 - NI Drawer	0	0	2.9	14.5	31.0	50.0	74.8	90.5	99.8
N42 - Control Board	0	0	3.0	14.0	31.0	46.0	75.8	90.5	99.9
N42 - NI Drawer	0	0	3.0	14.0	31.5	46.5	75.0	90.5	99.1
N43 - Control Board	0	0	1.0	13.0	30.0	50.0	76.0	90.5	100.2
N43 - NI Drawer	0	0	2.9	14.0	31.0	50.5	75.5	91.0	100.6
N44 - Control Board	0	0	2.1	14.0	31.0	57.0	75.8	90.0	99.7
N44 - NI Drawer	0	0	2.9	14.5	32.5	47.5	76.2	91.0	100.0



Table 3

## Power Range High Level Trip Setpoints

Power Plateaus (% RTP)	Desired Setpoint (%)	Actual Setpoint (%)			
		N41	N42	N43	N44
0 to 5	25 + 0.5 - 1.0	24.6	24.8	24.5	24.1
15	25 + 0 - 1.0	25.0	25.0	25.0	24.9
30	40 + 0.5 - 1.0	40.1	39.8	40.1	40.0
50	60 + 0 - 1	59.9	59.7	60.0	60.0
75	90 + 0 - 1	90.0	90.0	90.0	90.0
90	109 + 0 - 1	108.8	109.0	108.8	109.0
100	109 + 0 - 1	108.8	109.0	108.8	109.0



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

TEST OBJECTIVE

The purpose of this test procedure was to align the  $\Delta T$  and Tave instrumentation channels during power ascension.

TEST DESCRIPTION

At isothermal conditions,  $\Delta T$  and Tave values were determined from Thot and Tcold readings. At each power ascension test plateau,  $\Delta T$  and Tave data were collected in Test Procedure 42.5 - Thermal Power Measurement and State Point Data Collection, and transcribed to this test. At the 75% power plateau, linear regression analysis was used to determine extrapolated Tave and  $\Delta T$  for each loop at 100% power. These extrapolated full power values were averaged, and the average  $\Delta T$  and Tave were used to make the necessary adjustments to  $\Delta T$ , OT  $\Delta T$ , and OP  $\Delta T$  instrumentation.

At 100% power, the calibrations were refined by adjusting the instrumentation based on the actual  $\Delta T$  values of each reactor coolant loop. A final verification for Tave consisted of comparing loop Tave values to the average Tave. A final verification for  $\Delta T$  involved comparing the power level inferred from each loop's  $\Delta T$  to core-average power based on a secondary side heat balance (i.e., Surveillance Test Procedure R-2B).

TEST RESULTS

At isothermal conditions,  $\Delta T$  and Tave values agreed with the values calculated from Thot and Tcold readings within the specified tolerance as shown in Table 4. Instrumentation adjustments were not needed.

Both the extrapolated  $\Delta T$  and Tave values and the actual values at 100% power were below the respective upper limits of 64.4 deg. F and 576.6 deg. F. Core-average  $\Delta T$  at 100% power was 60.7 deg. F and core-average Tave was 569.82 deg. F. These values are consistent with the fact that the measured RCS flow rate is slightly greater than design.

At 100% power, adjustment of the instrumentation was based on each loop's actual  $\Delta T$  value. Following this adjustment, the power levels inferred from the loop  $\Delta T$ 's were all in tolerance, as summarized in Table 5.



Table 4

 $\Delta T$  and Tave at Isothermal Conditions

Parameter (deg. F)	Loop 1	Loop 2	Loop 3	Loop 4
Thot	546.17	546.14	546.2	546.08
Tcold	546.42	546.45	546.36	546.54
$\Delta T$ (calculated)	-0.25	-0.31	-0.16	-0.46
$\Delta T$ (measured)	-0.19	-0.227	-0.14	-0.37
Tave (calculated)	546.3	546.3	546.28	546.31
Tave (measured)	546.45	546.43	546.45	546.6

Acceptance Criteria

$\Delta T$  (measured) =  $\Delta T$  (calculated)  $\pm 0.3$  deg. F

Tave (measured) = Tave (calculated)  $\pm 1.0$  deg. F



Table 5

ΔT and Tave at Full Power  
(data recorded at 99.58% of RTP)

Parameter	Loop 1	Loop 2	Loop 3	Loop 4
Tave (deg. F)	570.6	570.2	570.5	570.3
ΔT (deg. F)	60.53	59.94	61.31	59.67
Power-based on ΔT (%)	99.18	99.78	99.53	99.75
Power-based on R-2B(%)	99.58	99.58	99.58	99.58
Loop power deviation (%)	-0.40	+0.20	+0.05	-0.17
Tave - Avg Tave (deg. F)	+0.2	-0.2	+0.1	-0.1

Acceptance Criteria

Upper Limits: Tave <576.6, ΔT <64.4 deg. F

Tave: Loop Tave within 1 deg. F of average Tave

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



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

### TEST OBJECTIVE

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

### TEST DESCRIPTION

The feedwater and steam flow instrumentation output signals were checked against the reference feedwater flow (Barton gauges) at steady state power levels of 30%, 50%, 75%, 90% and 100% RTP. Test data collected as part of Startup Test Procedure 42.5-Thermal Power Measurement and Statepoint Data Collection were analyzed by this procedure to determine the deviation of steam and feedwater flow compared to the reference feedwater flow. Any transmitted signal data found to be outside the allowable tolerance was submitted to the Instrumentation Department for evaluation and recalibration as required. If any adjustments were made, verification data were collected and analyzed prior to ascending to the next power plateau.

### TEST RESULTS

At the completion of power ascension testing, all feedwater transmitters were within their tolerance of 1.5% of reference flow. Only one feedwater flow transmitter (FT-540) was a problem, and that transmitter was replaced at the 75% test plateau. Several other feedwater flow transmitters were just outside the allowed 2.0% deviation at low power levels. This discrepancy was attributed to noise in the data, as the transmitters were well within tolerance at the 100% test plateau.

The eight steam flow transmitters required rescaling as follows:

- 30% power - Transmitters 512, 513, 522, 523, 532, 533, 542, 543
- 50% power - Transmitters 522, 523
- 75% power - Transmitters 512, 513
- 90% power - Transmitters 522, 523, 532, 533, 542
- 100% power - Transmitters 512, 513, 543
- 100% power outage - Transmitters 512, 513, 522, 523, 532, 533, 542, 543

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

At the 100% test plateau, the transmitter steam flows were observed to drift slightly outside the tolerance of  $\pm 2.0\%$  of reference steam flow. Recalibration would bring the steam flows back within tolerance, but the slow drift was observed to continue and will be monitored following power ascension testing. From a control systems standpoint, this reference/transmitter discrepancy between steam flows was too small to cause any adverse effects.



## 6.0 Surveillance Test Procedure R-3A - Incore Power Distribution

### TEST OBJECTIVE

The purpose of this procedure was to obtain flux maps using the Moveable Incore Detector System (MIDS). The detector outputs were used to determine such core parameters as axial flux distributions, peaking factors, and core tilts for several startup tests during the 75%, 90%, and 100% power plateaus. The flux maps were also used to fulfill the routine Technical Specification surveillance requirements.

### TEST DESCRIPTION

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

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

- 75% power, all rods out (ARO), equilibrium xenon; provided base line data for T.P. 42.5 - Thermal Power Measurement and Statepoint Data Collection.
- 75% power, ARO, equilibrium xenon; provide reference data for STP R-13 - Nuclear Power Range Incore-Excore Detector Calibration.
- 7 quarter-core maps at 75% power provided data for STP R-13-Nuclear Power Range Incore-Excore Detector Calibration.
- 90% power, ARO, equilibrium xenon; provided base line data for Test Procedure 42.5 - Thermal Power Measurement and Statepoint Data Collection.
- 100% power, ARO, equilibrium xenon; provided base line data for Test Procedure 42.5 - Thermal Power Measurement and Statepoint Data Collection.

### TEST RESULTS

Each of the flux maps listed above was analyzed and determined to be satisfactory. Results are discussed in more detail in other sections of this report (i.e., reports related to Test Procedures 42.5 and R-13).



## 7.0 Test Procedure No. 1.16 - Effluents and Effluents Monitoring

### TEST OBJECTIVE

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

### TEST DESCRIPTION

Effluent monitoring is an ongoing program by Nuclear Plant Operations (NPO) which involves following the procedures in the Plant Manual. The test collects data to verify the effluent monitoring program and from these data verifies the calibration of the effluent monitors. The intent was to perform these verifications at the 30, 50, 75, and 100% power test plateaus.

### TEST RESULTS

A minimum activity level is required to adequately judge the calibration of each monitor. However, through the 90% power test plateau, the activity levels at each monitor had not been large enough to verify the calibration of the monitors.

During the 100% power test plateau, only rad monitor RE-18 (i.e., the rad monitor for the Liquid Batch Tanks to Outfall) had a sufficiently high activity level. The collected data have provided an empirical and increasingly consistent relationship between rad monitor counts and sample activity. However, as more data become available, this correlation will be further refined. With respect to all other rad monitors, the activity had been too low for meaningful analysis. With continued operation of the plant and a corresponding increase in effluent inventories, rad monitor readings will be correlated and verified against sample activities through the use of NPO procedures.



## 8.0 Test Procedure No. 1.17 - Chemical and Radiochemical Analysis

### TEST OBJECTIVE

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

### TEST DESCRIPTION

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

### TEST RESULTS

A review of the performance of NPO Chemistry and Radiation Protection Department analyses found them to be in accordance with the approved NPO procedures. Samples taken and analyzed at 50%, 75%, 90%, and 100% rated thermal power levels were checked against the ongoing program results and found to be comparable. Plant chemistry was being maintained within the limits established and specified in NPO Operating Procedure F-5, or corrective action was taken to bring the system back within specifications.

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



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

### TEST OBJECTIVE

The main objective of this test procedure was to determine the reactor coolant average temperature program required to maintain the design full load Steam Generator pressure.

### TEST DESCRIPTION

Reactor Coolant Tave, Steam Generator pressure and Turbine Impulse Chamber pressure were recorded at 0%, 30% and 50% RTP. Each of these parameters was extrapolated to 100% RTP. A temperature program correction was then computed from the difference between the saturation temperature of the extrapolated Steam Generator pressure and the design full power Tave. This correction was applied to the design temperature program generated by the Reactor Control System, Steam Dump Control System and plant computer. With Tave controlled at the new Tref, Turbine Impulse Chamber pressure was compared to the 50% load design value and agreement was verified. This entire process was repeated at 75% RTP to obtain a further refinement in the temperature program. Upon reaching 100% RTP, the temperature program was adjusted to obtain the design value of Steam Generator pressure. Throughout this procedure, changes in the temperature program were verified to maintain the 100% RTP value for Tref below 576.6 deg. F.

### TEST RESULTS

Test data was taken at 0%, 30% and 50% RTP and the results were plotted and extrapolated to 100% RTP. A Tref correction of -3.46 deg. F was subtracted from the maximum design Tref of 576.6 to yield the new projected 100% RTP Tref of 573.14 deg. F. This value correlated closely with the plotted and extrapolated 100% Tave of 573.4 deg. F. Tref as a linear function of percent load was used to determine the desired voltage as a linear function of power for recalibration of the Turbine Impulse Chamber Pressure Controllers TC-505 and TC-505A. The calibration was done during the 50% RTP outage and verified upon returning to 50% power. Performance of the test at 75% power led to a slight adjustment of the 100% RTP Tref to 571.3 deg. F.

At 100% power, steam generator pressures were 806.7, 807.3, 808.1 and 807.8 psia for Loops 1, 2, 3, and 4, respectively, well within the Acceptance Criteria of 805  $\pm$ 10 psia. Tave values were 570.0, 569.3, 569.7, and 569.5 deg. F for Loops 1, 2, 3, and 4, all below the safety limit of 576.6 deg. F. However, turbine impulse chamber pressure was approximately 30 psi below the original full load design value of 566 psia. Upon a Westinghouse review of the collected data and the design information, 540 psia (measured during turbine benchmark testing) was judged to be consistent with design. Because the steam generators agreed so well with design, no further adjustments were made as part of this test procedure. However, turbine impulse chamber pressure instrumentation will be rescaled at a convenient time in the future (i.e., an outage). This will necessitate a small adjustment to the Tref program in order to maintain Steam Generator pressure within specification.



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



## 10.0 Test Procedure No. 43.1 - Load Swing Tests

### TEST OBJECTIVE

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

### TEST DESCRIPTION

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

- Nuclear Power
- Controlling bank position
- Reactor Coolant Temperatures
- Steam Flow and Pressure
- Feedwater Flow and Control Parameters
- Steam Generator Level
- Pressurizer Pressure and Level

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

### TEST RESULTS

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

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

A final acceptance criteria, no manual intervention to bring the plant to steady state conditions, was satisfied, except as noted below.

At the 30% plateau, the 10% load decrease was initiated and all systems responded correctly. The plant stabilized within 15 minutes. Then the 10% load increase was initiated and all systems responded correctly except that Feedwater Pump 1-1 and the four feedwater regulating valves were oscillating. The plant stabilized within 15 minutes but the oscillations in feedwater control persisted until the Feedwater Pump Master Controller was placed in manual. During the load decrease, the pressurizer pressure swing exceeded the expected value of less than 50 psid (actual was 56.4 psid) and steam pressure overshoot exceeded the expected value of 25 psid (actual was 32.5 psid). During the load increase, pressurizer pressure again exceeded the expected value with a 54 psid swing, as indicated in Table 6.



At the 50% plateau, the plant response to the load decrease and increase was satisfactory except that Condensate Bypass Valve FCV-723 failed at about 15% open, requiring manual intervention on the feedwater valves. In addition, only one main feedwater pump was on-line. The load swings were reperformed satisfactorily after manually opening FCV-723 and placing the second main feedwater pump in service. During the load decrease, pressurizer pressure failed to meet the expected value (actual swing was 58.5 psid) and steam pressure overshoot was higher than expected at 32.5 psid. During the load increase, only the steam pressure exceeded expectations with a swing of 36.2 psid, as shown in Table 6.

At the 75% plateau, the 10% load decrease was performed and the Load Transient Bypass system actuated during the transient. The plant response was satisfactory. The steam pressure overshoot was 50 psid. Equilibrium conditions were reached in approximately 10 minutes. During the 10% load increase, it was necessary to take manual control of the feedwater regulating valves after the valves went to a full open position. Inadequate feedwater pump response was the cause. Thereafter, the plant response was satisfactory, and equilibrium conditions were reached in approximately 10 minutes. The feedwater pump response was corrected by adjustments in speed control settings. A subsequent test was performed from 65% to 75% power and feedwater pump/feedwater regulating valve response was satisfactory.

At the 100% plateau, the Load Transient Bypass system actuated during the 10% load decrease. Plant response was satisfactory and equilibrium conditions were reached in approximately 10 minutes. The steam pressure overshoot was 68.75 psid.

Deviations from the expected values were reviewed by Westinghouse at each test plateau. Westinghouse identified control system setting changes which were implemented prior to load swing tests at 75% power. The steam pressure transients encountered during the 75% and 100% power tests were judged acceptable by Westinghouse and PGandE Engineering. No further control system adjustments were made.



Table 6

## Key Plant Parameters During 10% Load Swings

	30% Pwr to 20% Pwr	20% Pwr to 30% Pwr	50% Pwr to 40% Pwr	40% Pwr to 50% Pwr	75% Pwr to 65% Pwr	65% Pwr to 75% Pwr	100% Pwr to 90% Pwr
<b>Pressurizer Pressure:</b>							
Initial Pressure (psig)	2214	2226	2229	2235	2236	2244	2240.4
Maximum Pressure (psig)	2238	2264.4	2238	2238	2250	2244	2250.6
Minimum Pressure (psig)	2181.6	2210.4	2179.5	2221.5	2219	2199	2205.0
Maximum-Minimum (psid)	56.4	54	58.5	16.5	31	45	45.6
<b>Steam Generator Level:</b>							
Initial S/G 1-1 Level (%)	44	43	44	44	44	44	45
Maximum S/G 1-1 Level (%)	49	52	51	51	53	47	54
Minimum S/G 1-1 Level (%)	38	37	38	39	34	38	40
Maximum - Initial (%)	5	9	7	7	9	3	9
Initial - Minimum (%)	6	6	6	5	10	6	5
<b>Steam Pressure:</b>							
Initial Pressure (psig)	890	900	866.3	867	784	802	737.5
Final Pressure (psig)	900	885	881.3	862.5	780	772.5	721.3
Maximum Pressure (psig)	932.5	904.4	913.8	867.5	830	802	790.0
Minimum Pressure (psig)	893.8	862.5	864.3	826.3	780	757	721.3
Final-Minimum (psid)	N/A	22.5	N/A	36.2	N/A	15.5	N/A
Maximum-Final (psid)	32.5	N/A	32.5	N/A	50	N/A	68.7

Expected Response

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



## 11.0 Test Procedure No. 42.1 - Power Coefficient Measurement

### TEST OBJECTIVE

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

### TEST DESCRIPTION

At the 30%, 50%, 75% and 90% test plateaus, after establishing stable plant conditions with equilibrium xenon and axial flux difference at or near its target value, the turbine load was decreased approximately 22 MWe at a rate of 2200 MWe/min. This action caused about 2% drop in reactor power level. Subsequent 44 MWe load swings were performed in order to vary reactor power by about 4% in each case, and a final load swing of 22 MWe returned power to its initial value. This sequence is shown on Figure 8. The period between each load swing was long enough to allow stabilization of  $T_{ave}$  and  $\Delta T$ . Boron concentration and control rod position were maintained constant throughout the test.

For each increase in turbine load,  $\Delta T$  increases and  $T_{ave}$  decreases. The increased  $\Delta T$  causes a negative reactivity effect due to the fuel's negative Doppler coefficient. This is offset by the positive reactivity due to the negative isothermal temperature coefficient (ITC) and the decreased  $T_{ave}$ . In a similar manner, load decreases involve a decrease in  $\Delta T$  and an associated increase in  $T_{ave}$ .

The load swings done in this test directly measured the change in core average coolant temperature required to offset a change in  $\Delta T$ . By relating the changes in  $\Delta T$  to changes in reactor power level, ratios of Doppler coefficient to ITC were calculated by dividing the change in  $T_{ave}$  by the change in power for each load swing. The acceptance criterion was that this ratio (Doppler coefficient/ITC) must be within 0.5 deg. F/% of the design value. The Doppler coefficient was then inferred by multiplying the average ratio by an ITC which was based on design calculations.

### TEST RESULTS

At each test plateau, the ratios of Doppler coefficient to ITC were within the acceptance criterion. As shown in Table 7, the inferred Doppler coefficients ranged from -11.84 pcm/% power at 30% power to -10.10 pcm/% power at 90% power. This trend of decreasing doppler coefficient with power was expected because higher fuel temperatures cause a reduced degree of broadening of U-238 resonance peaks. The fact that the inferred coefficients were always less than design values can be partially attributed to xenon burnout (build-in) caused by reactor power increases (decreases).



Table 7

Power Coefficient Measurements

TEST PLATEAU (% RTP) (NOMINAL)	MEASURED RATIO* (deg.F/% power)	DESIGN RATIO* (deg.F/% power)	INFERRED DOPPLER COEFFICIENT (pcm/% power)	DESIGN DOPPLER COEFFICIENT (pcm/% power)
30	2.83	3.22	-11.84	-13.5
50	2.05	2.40	-10.94	-12.8
75	1.50	1.61	-11.02	-11.8
90	1.16	1.29	-10.10	-11.3

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

Acceptance criterion: measured ratio = design ratio  $\pm 0.5$  deg.F/% power

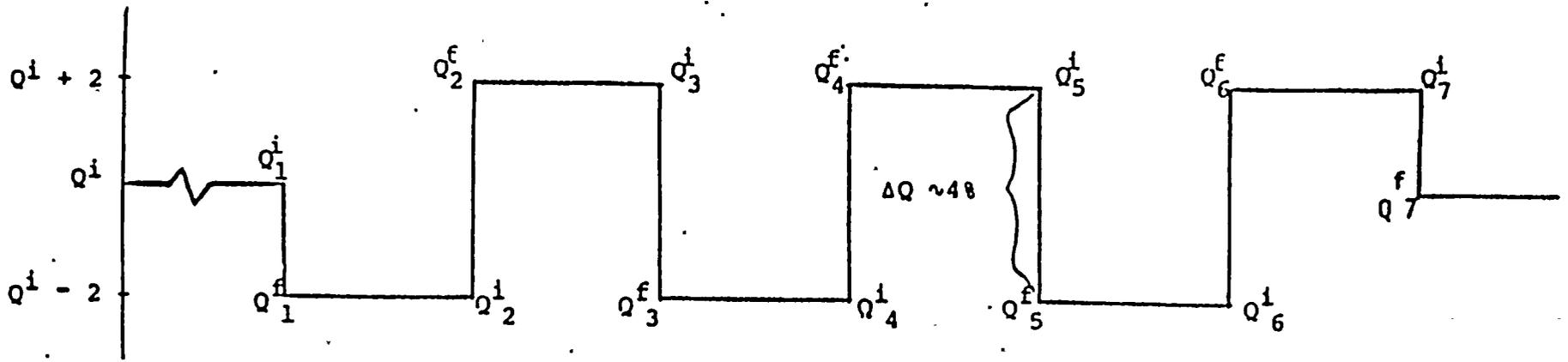


FIGURE 8

T.P.42.1

LOAD CYCLING PATTERN FOR POWER COEFFICIENT VERIFICATION

Turbine Generator Load (% RTP)



Time

30



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

### TEST OBJECTIVE

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

### TEST DESCRIPTION

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

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

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

\*  $AFD (\%) = AO (\%) \times \% \text{ core power} / 100$

\*\*  $AO (\%) = (\text{Upper detector current or core power} - \text{Lower detector current or core power}) \times 100 / (\text{Upper} + \text{Lower})$



### TEST RESULTS (75% Power Plateau)

Control Bank D rods were inserted in 2 increments of approximately 10 steps each. Incore A0 shifted from about -9.0% initially (Bank D at 187 steps) to about -15.5% (173 1/2 steps) and then to about -21.2% (164 1/2 steps). Upon holding the latter configuration for approximately 2 hours, rods were returned to their original position. During the subsequent axial xenon oscillation, six more quarter-core flux maps were produced at incore A0's ranging from about -11% to -1%. Thus, the total span of minimum to maximum A0 was from about -21% to about -1%. The entire test required approximately 23 hours to complete.

Slopes (gains) of the incore vs. excore A0 plots were approximately 1.6 for all channels. Offsets ranged from about +0.16% for channel N44 to +6.8% for channel N42. The upper detector in channel N42 had a somewhat lower current (sensitivity) than the other detectors, accounting for the slightly different calibration constants for this channel. A sample of each type of plot is enclosed for channel N41 (Figures 9 and 10).

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



Figure 9

STP R-13 SAMPLE PLOT: H-41 INCORE vs. EXCORE AXIAL OFFSET  
75% RTP, STP R-13, 3-1-85

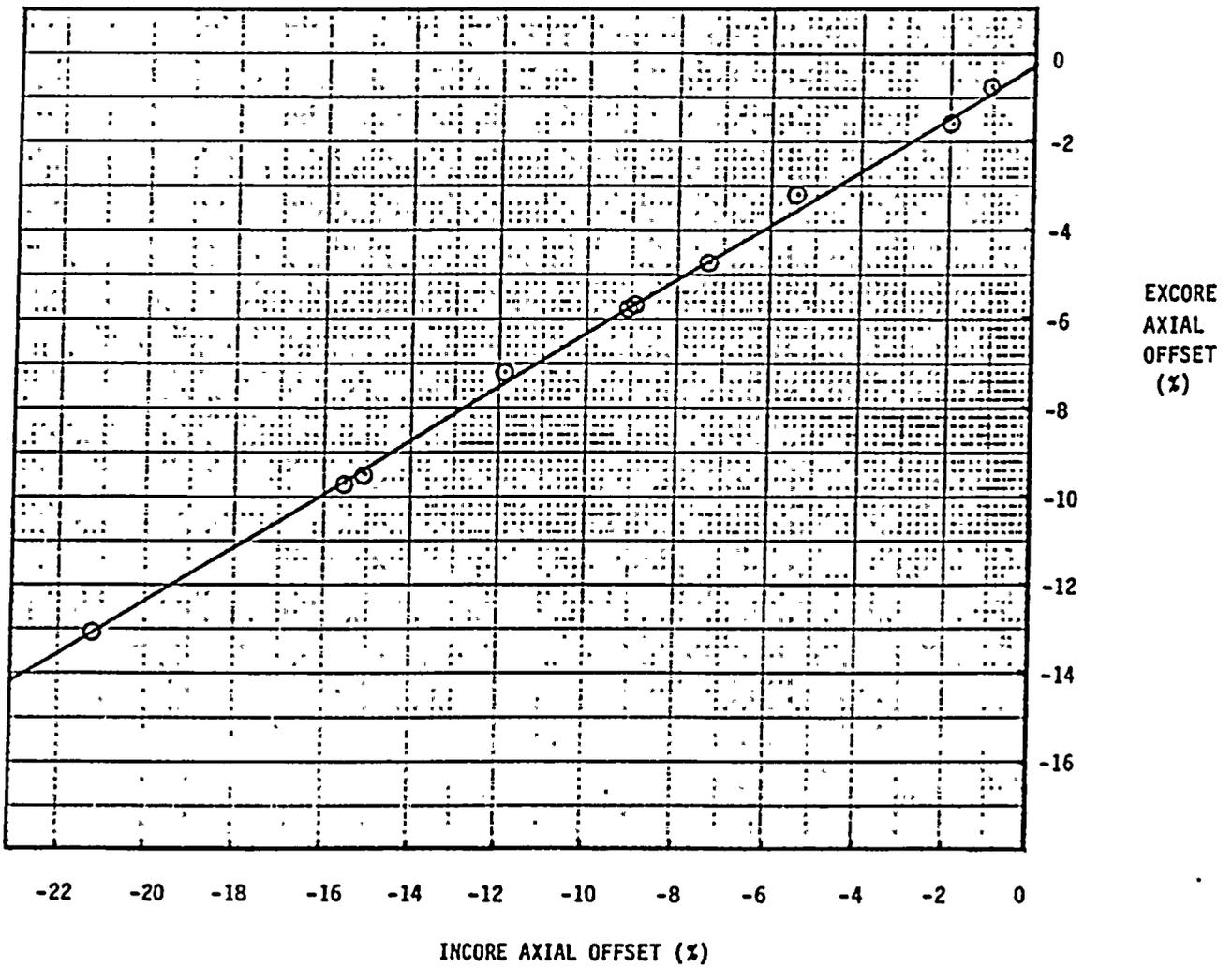
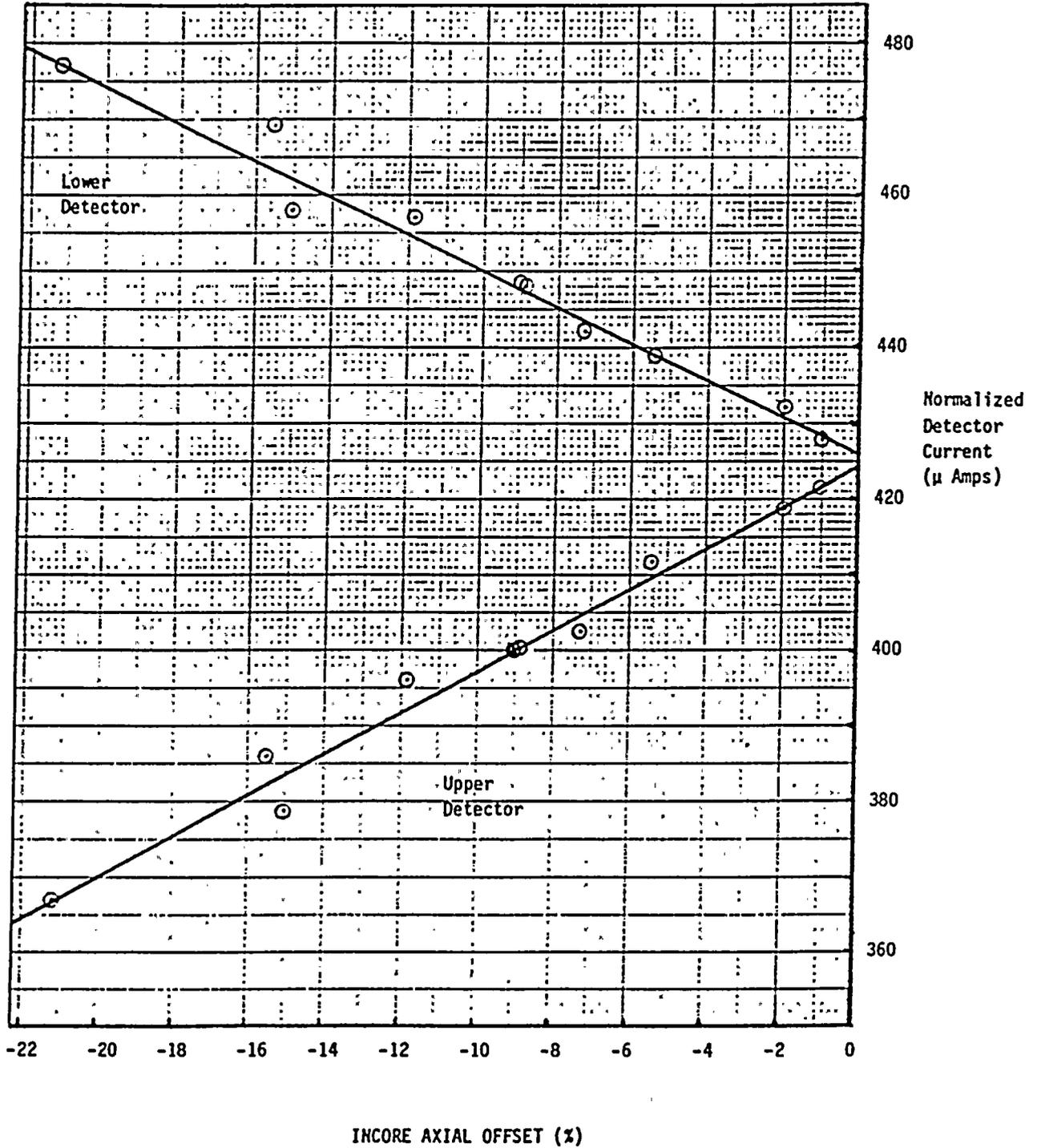




Figure 10

STP R-13 SAMPLE PLOT: N-41 INCORE AXIAL OFFSET vs. NORMALIZED DETECTOR CURRENT  
75% RTP, STP R-13, 3-1-85





### 13.0 Surveillance Test Procedure R-26 - RCS Primary Coolant Flow Measurement

#### TEST OBJECTIVE

The objective of this test was to verify the calibration of RCS flow instruments at 75%, 90%, and 100% of full power and to confirm that the total flow of all four loops is greater than the Technical Specification requirement of 363,000 gpm.

#### TEST DESCRIPTION

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

Reactor coolant flow was determined by measuring  $\Delta T$  and performing a heat balance on the RCS. This was done by using the gross steam generator thermal output calculated in the high accuracy secondary heat balance test (Surveillance Test Procedure R-2A) and narrow range hot-leg and cold-leg RCS RTD temperature measurements.

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

#### TEST RESULTS

At 75% RTP, total RCS flow was measured as 375,069 gpm which is approximately 3.3% more than required by Technical Specifications. No recalibrations of loop flow meters were required; however loop flow constants (in Surveillance Test Procedure I-1A, done each shift by Operations) were updated.

At 90% RTP, total RCS flow was measured at 375,653 gpm which is 3.5% more than that required by Technical Specifications. No recalibrations of loop flow meters were required. Since the flow change was in the conservative direction and this test was to be reformed at 100%, no changes were required for the Surveillance Test Procedure flow constants mentioned above.

At 100% RTP, total RCS flow was measured as 376,848 gpm which is 3.8% more than required by Technical Specifications. Eleven loop flow meter indicators were calibrated. Loop flow constants in Surveillance Test Procedure I-1A were updated.



## 14.0 Test Procedure No. 43.3 - Large Load Reduction Tests

### TEST OBJECTIVE

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

### TEST DESCRIPTION

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

- Reactor Coolant Temperature and Pressure
- Control Rod Bank Position
- Pressurizer Level and Pressure
- Steam Flow and Pressure (Loop 1 only)
- Feedwater Flow and Pressure (Loop 1 only)
- Feedwater Pumps speed and control Variables.

The above data were evaluated for acceptable dynamic response. In addition, the interactions between the control systems were studied for possible setpoint changes to improve transient response based on actual plant operations.

### TEST RESULTS

The large load reduction at 75% power met all acceptance criteria, except for no manual intervention, as follows:

1. Reactor and turbine did not trip,
2. Safety injection did not initiate,
3. Pressurizer safety values did not lift,
4. Main steam safety values did not lift.

Manual intervention for Feedwater Pump Speed was required 3 minutes into the transient, but only for 10 to 15 seconds before being returned to automatic. This was considered acceptable by the Plant Staff Review Committee. In general, plant response was within expectations, except for the duration of maximum rod speed which was approximately 45 seconds instead of 30 seconds. This was not considered to be a problem by Westinghouse and PGandE Engineering and no further changes were made.

The large load reduction at 100% power met all acceptance criteria satisfactorily without any manual intervention. With regard to expected plant response, the maximum rod speed duration was 45 seconds instead of 30 seconds and primary pressure swing exceeded the -100 psi from initial pressure by 0.5 psi, an acceptably small deviation.

The responses of key plant parameters are shown in Table 8.



Table 8

Key Plant Parameters During Large Load Reduction Tests

	From 75% Power	From 100% Power
<b>Tave:</b>		
Initial Tave (deg. F)	565.5	570.5
Peak Tave (deg. F)	568.0	574.9
Final Tave (deg. F)	552.0	559.0
Tave Undershoot (deg. F)	0	1.7
Tave Oscillations	very small	very small
<b>Pressurizer Pressure:</b>		
Initial Pressure (psig)	2238	2223
Maximum Pressure (psig)	2262	2290
Minimum Pressure (psig)	2169	2122.5
Maximum - Initial (psid)	24	67
Initial - Minimum (psid)	69	100.5
<b>Steam Generator Level:</b>		
Initial S/G 1-1 Level (%)	44	44.8
Maximum S/G 1-1 Level (%)	47	55.2
Minimum S/G 1-1 Level (%)	33	32.0
Maximum - Initial (%)	3	10.4
Initial - Minimum (%)	11	12.8
<b>Control Rod Speed:</b>		
Time of Maximum Speed (sec)	45	45
<b>Steam Dumps:</b>		
Actuation and Modulation	yes	yes
Cycling	no	no
Shutoff Time (min)	6	5.7

Expected Responses:

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



## 15.0 Test Procedure No. 4.6 - Steam Generator Moisture Carryover Test

### TEST OBJECTIVE

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

### TEST DESCRIPTION

The MCO at 100% RTP was determined by the use of a radioactive tracer in the form of Na<sup>24</sup>. The tracer was mixed with water in four injection tanks and then injected into each Steam Generator (SG) via its feedwater line. The MCO was calculated based on the activity measurements of the tracer in feedwater (condensate/reheater), main steam, and blowdown line samples.

Normally, a 1 to 1 1/2 hour stabilization period is adequate to achieve good tracer mixing prior to obtaining the required samples. However, due to the configuration of the SG blowdown piping (isolated for the test) which has a large volume of water and is only purged by the small sample tubing, stabilization required a large period of time. Pretest SG blowdown samples were taken at various intervals until representative samples were obtained before taking the required samples. The required samples (4 sample sets) were taken 6 1/2 hours after tracer injection at 15-minute intervals. The following samples were taken:

1. Each SG blowdown line
2. Each SG main steam line
3. Condensate
4. Reheater drips (heater 2 drain tank pump discharge)

Due to secondary plant water chemistry, the full flow demineralizers were in service. Normally for MCO testing, the demineralizers are bypassed because they extract the tracer out of the feedwater train. Condensate and reheater drip samples were taken to substitute for feedwater samples. For sample sets 3 and 4, mainsteam line samples were not taken, because it was observed that there was no significant difference in MCO values when data from these samples were neglected. Simultaneously with the sample sets, calorimetric and MCO data were obtained. Other data were also taken to verify stable plant conditions.

Another set of 3 samples was taken (but not analyzed) with a 10 gpm SG blowdown flow. This was done under the assumption that the blowdown tracer samples taken earlier were not taken at stable conditions, but the previous samples were analyzed and showed that they had stabilized.

The power was then reduced to 96% RTP and after a 1-hour stabilization period, 4 sample sets were taken, with the intent of verifying the 100% RTP results. The first set was analyzed, but it was determined that the activity was too low to obtain meaningful data, therefore the rest of the sample sets were not analyzed.



TEST RESULTS

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

Sample Set .	MCO (%)	RTP (%)
1	0.106	99.8 .
2	0.092	99.6
3	0.078	99.6 .
4	0.090	99.2
Average	0.092	99.6

The 0.092% MCO easily met the required Acceptance Criterion of less than 0.25% at 100% RTP.



## 16.0 Test Procedure No. 43.2 - Net Load Trip From 100% Power

### TEST OBJECTIVE

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

### TEST DESCRIPTION

With the plant at stable 100% power conditions and on automatic control, the load rejection was initiated by opening the main transformer high side breakers (PCB 632 and 532). Operator intervention was kept to a minimum and the plant was stabilized to approximately 20% power using Emergency Operating Procedure OP-AP-2 - Full Load Rejection.

### TEST RESULTS

At 0630 hours on March 23, 1985, PCB-632 and PCB-532 were opened, causing a net load trip from 100% power. A detailed listing of the sequence of events and operator actions is given in Table 9.

The test was successful as the plant sustained the net load trip and Acceptance Criteria were met as follows:

1. Reactor and Turbine did not trip.
2. Safety Injection was not initiated.
3. Main steam safety valves did not lift.
4. Pressurizer safety valves did not lift.
5. Minimal operational intervention was necessary for feedwater, steam dump, pressurizer pressure or level control until power level decreases below approximately 20% nuclear power.

The responses of key plant parameters are provided in Table 10 and Figures 11 through 16. The trends depicted by these figures are consistent with expectations. The parameters in Table 10 also met expectations with the exceptions of maximum pressurizer pressure, peak Tave, and steam generator level. The pressure swings involved an increase of 103 psi above the initial value, the peak Tave was 7.5 deg. F above the initial value, and the minimum steam generator level was approximately 10% less than expectations. However, the plant responses were judged to be acceptable upon review by Westinghouse and PGandE Engineering. No setpoint changes were recommended or implemented based on the results of this test.



Table 9

## Sequence of Events and Summary of Operator Actions (T.P. 43.2)

<u>Time (Approx.)</u> (min:sec)	<u>Activity/Event</u>
00:00	Initiate the net load trip. 40% condenser steam dumps (Group I & II) have tripped fully open.
00:01	35% atmospheric steam dumps are modulating open.
00:16	10% atmospheric steam dumps are modulating open.
00:30	Load Transient Bypass (LTB) feature has initiated (See NOTE 1)
00:37	35% atmospheric steam dumps are beginning to modulate closed.
00:38	10% atmospheric steam dumps are fully closed.
01:37	35% atmospheric steam dumps are fully closed.
02:08	40% condenser steam dump (Group II) valves are modulating closed.
04:00	Nuclear power at approximately 50% power and decreasing. Main Feedwater Pump (MFW Pp) 1-1 has been placed in manual minimum load operation.
07:00	40% condenser steam dump (Group II) valves are closed.
07:20	Nuclear power is at approximately 28% and decreasing. Operators placed the reactor rod control system in manual to hold reactor power at approximately 20% (See NOTE 2).
07:30	40% condenser steam dump (Group I) valves are modulating.
09:00	MFW Pp 1-2 is oscillating. MFW Pp 1-1 brought on line and MFW Pp 1-2 placed in manual minimum load operation.
12:00	MFW Pp 1-1 is oscillating. Oscillations were due to a 35% atmospheric steam dump valve not seating correctly. Steam dump valve was isolated. MFW Pp 1-1 operation was stabilized.



Table 9 (Cont.)

## Sequence of Events and Summary of Operator Actions (T.P. 43.2)

<u>Time (Approx.)</u> (min:sec)	<u>Activity/Event</u>
15:00	Nuclear power is still being maintained at 20% power. Main Feedwater Pumps master speed controller has been placed in manual.
18:00	Nuclear power is still being maintained at 20% power with the steam dumps to the condenser. Plant has stabilized. Test is complete. Operators are preparing to place the unit on the grid and increase plant load to 100% power.

NOTE 1: The Load Transient Bypass (LTB) is a feature which actuates with a significantly large step load decrease (>10%) during power operation (>70%). It causes the condensate/feedwater heaters (5 of the total of 6 feedwater trains) between the condensate booster and main feedwater pumps to be bypassed, thereby allowing more flow to the suction of the main feedwater pumps during the transient.

NOTE 2: The rod control system was placed in manual at about the 28% reactor power level so that the steam generator levels could be controlled by the main feedwater regulating valves (The main feedwater bypass valves do not have automatic controls).



Table 10

## Key Plant Parameters During Net Load Trip From 100% Power

<b>Pressurizer Pressure:</b> Initial Pressure (psig) Maximum Pressure (psig) Minimum Pressure (psig) Maximum - Initial (psid) Initial - Minimum (psid)	2241 2344 2100 103 141
<b>Pressurizer Level:</b> Initial Level (%) Maximum Level (%) Minimum Level (%) Maximum - Initial (%) Initial - Minimum (%)	52.0 60.9 30.1 8.9 21.9
<b>Tave:</b> Initial Tave (deg. F) Peak Tave (deg. F) Oscillations (deg. F)	571 578.5 Slightly >2
<b>Steam Generator Level:</b> Initial S/G 1-1 Level (%) Maximum S/G 1-1 Level (%) Minimum S/G 1-1 Level (%) Maximum - Initial (%) Initial - Minimum (%)	43.6 52.8 18.2 9.2 25.4
<b>Nuclear Power:</b> Time to reach 50% RTP (sec.)	244

Expected Responses:

1. Pressurizer pressure swings less than 100 psi above and 150 psi below initial value.
2. Pressurizer level swings less than 10% above and 30% below initial value.
3. Tave peak less than 5 deg. F above initial value, with peak-to-valley oscillations of less than 2 deg. F.
4. Steam generator level swings less than 15% of initial value.
5. Nuclear power below 50% RTP after approximately 200 seconds.



Figure 11

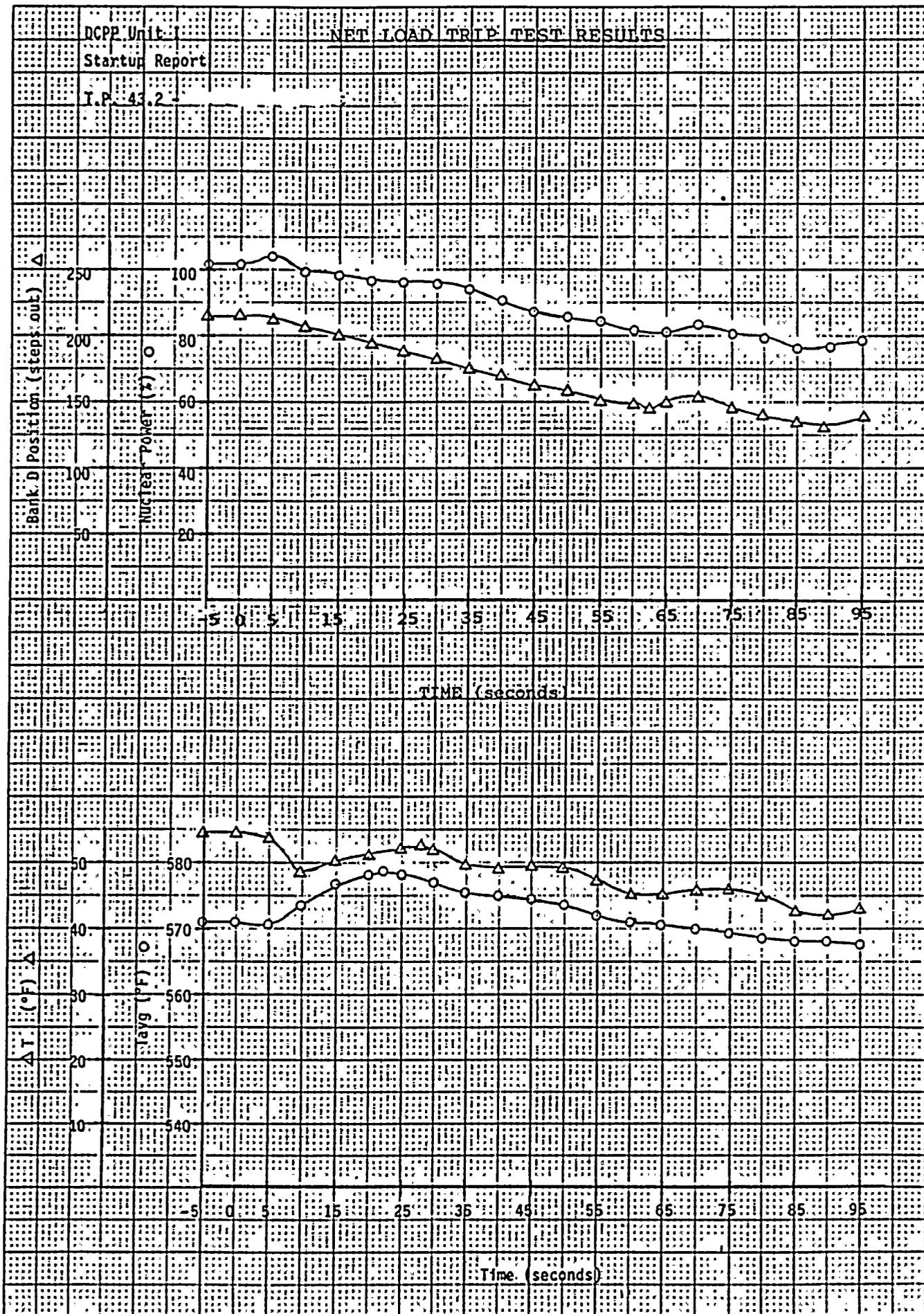




Figure 12

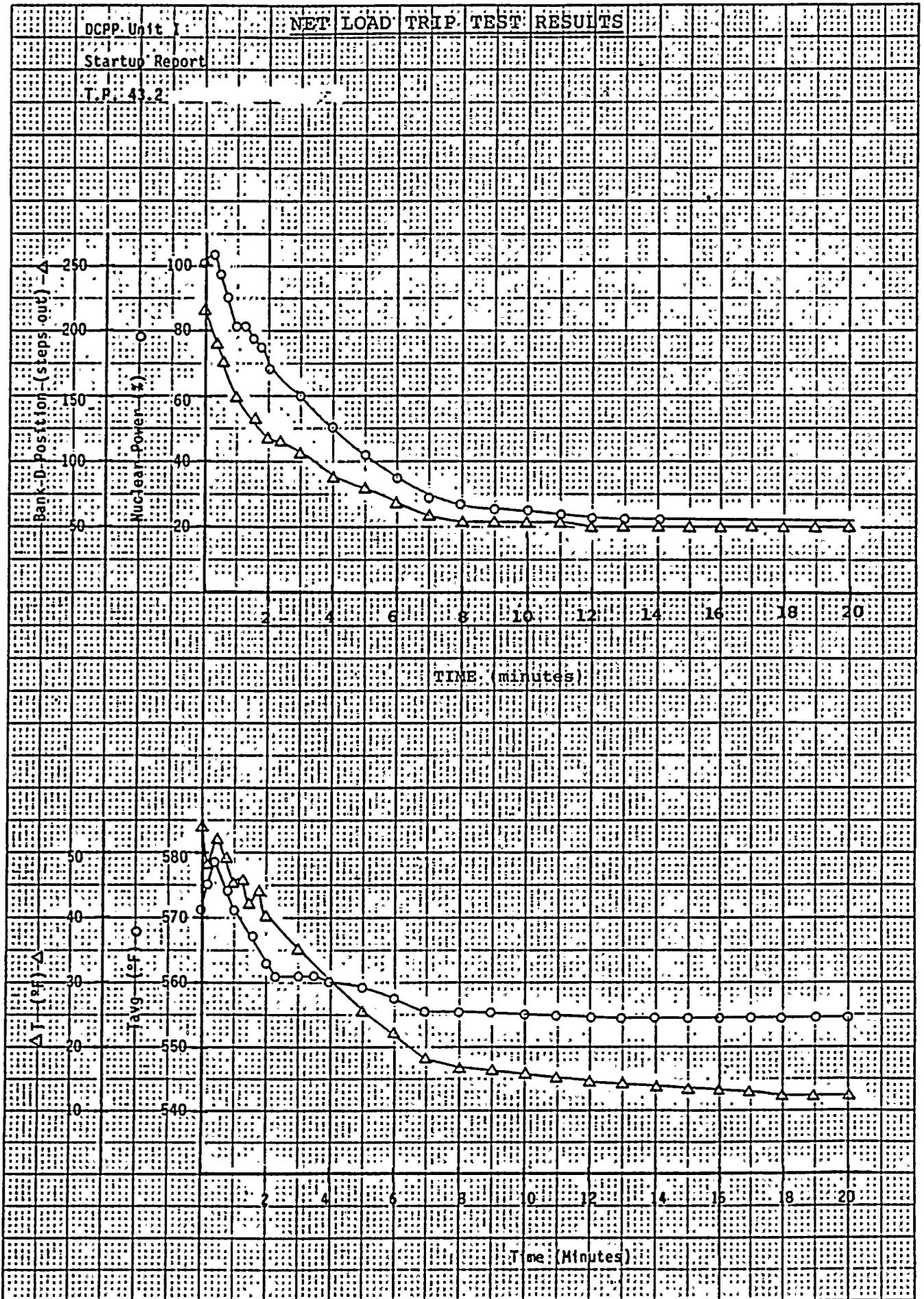




Figure 13

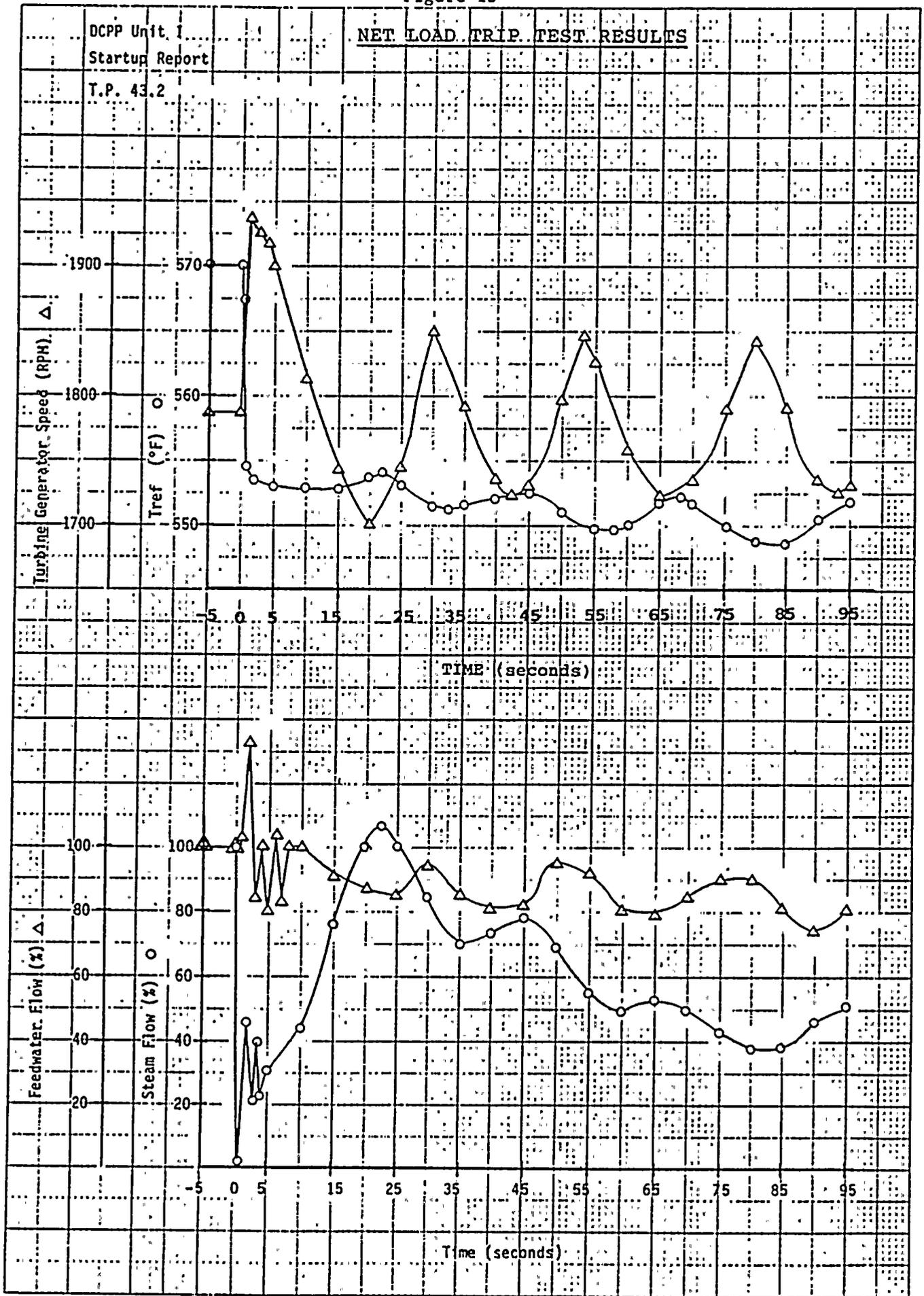




Figure 14

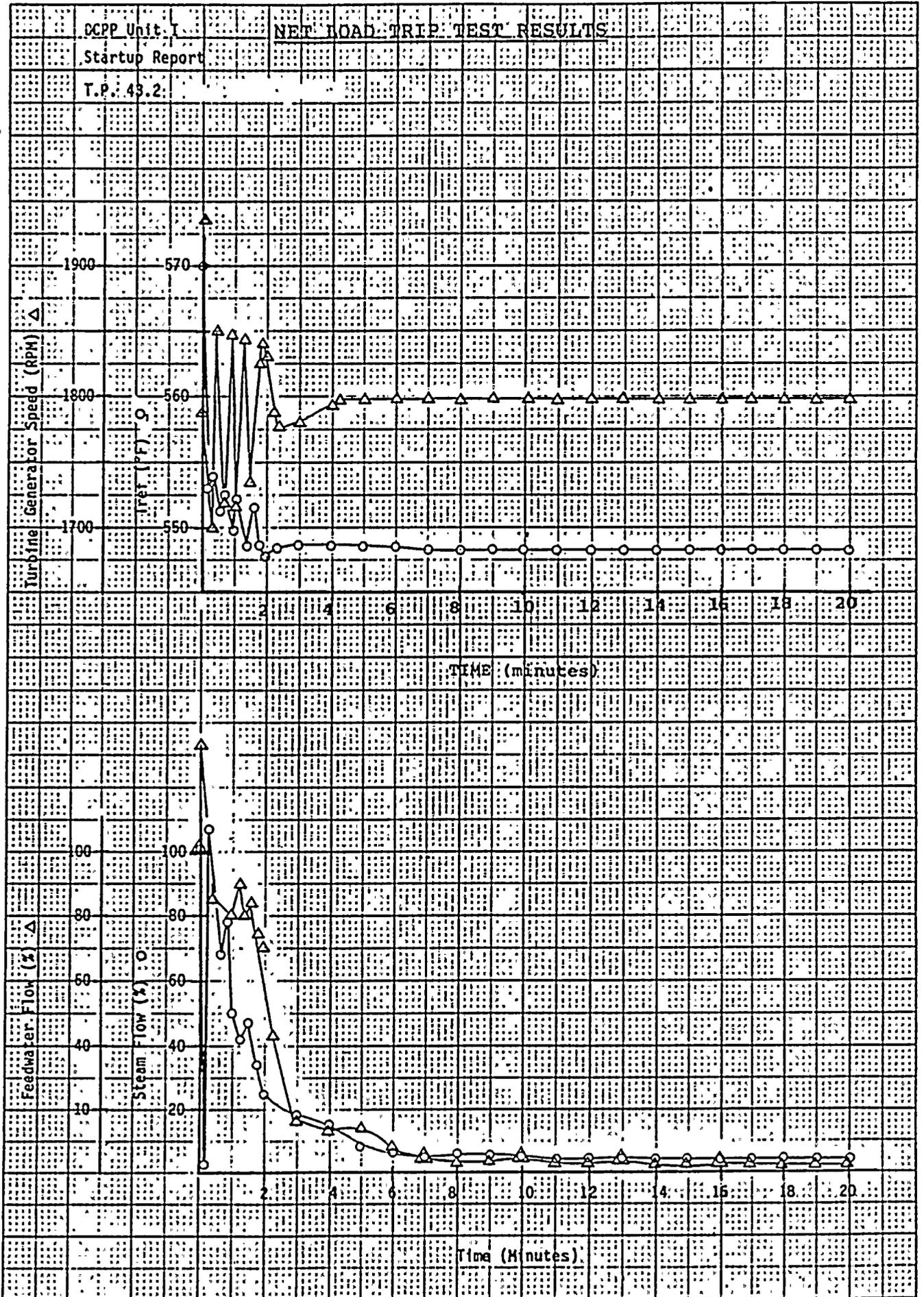




Figure 15

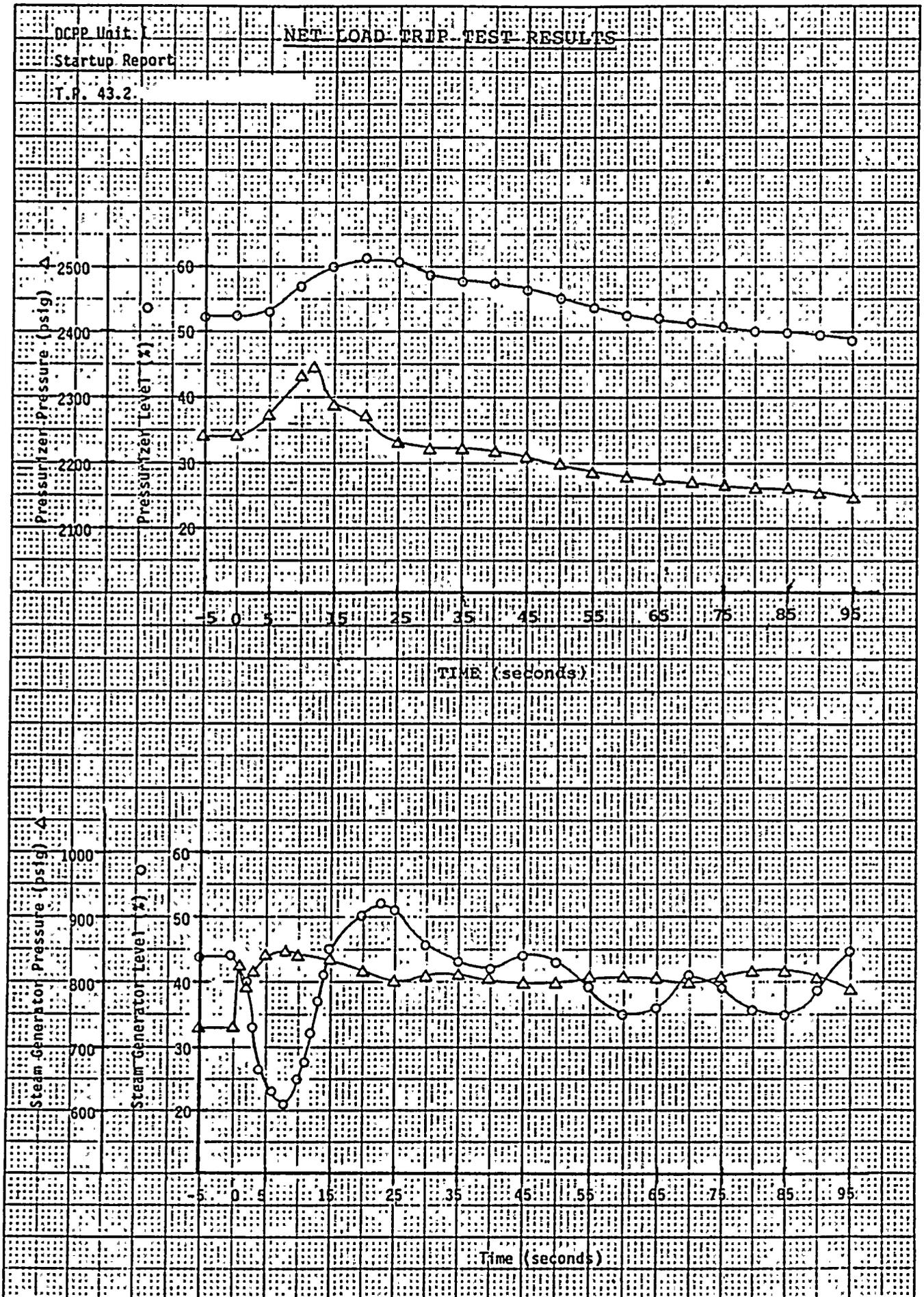
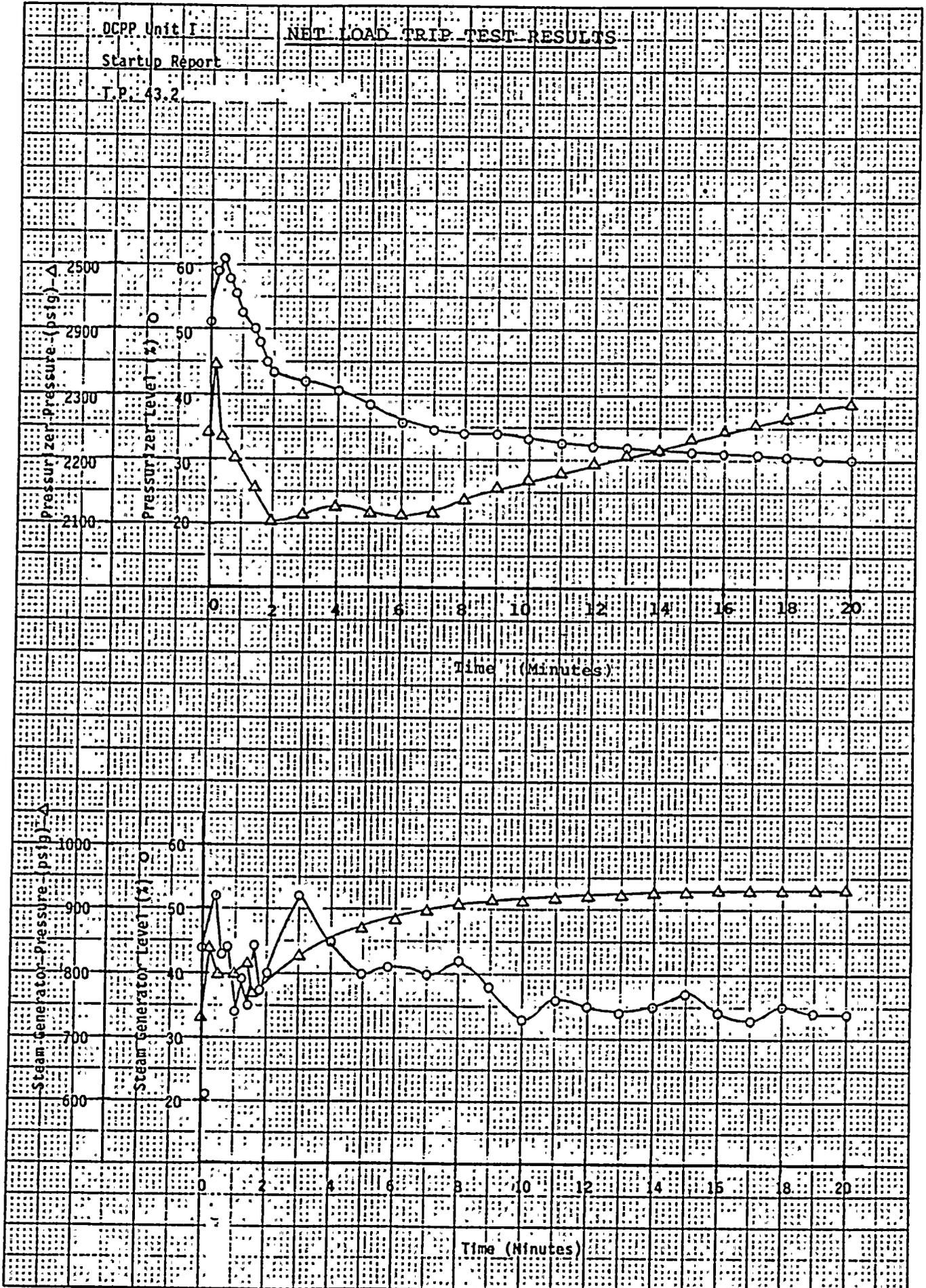




Figure 16





## 17.0 Test Procedure No. 43.6 - Nuclear Steam Supply System Acceptance Test

### TEST OBJECTIVE

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

### TEST DESCRIPTION

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

### TEST RESULTS

The 100-hour NSSS Acceptance Test began on March 24, 1985, and was completed without problems. As shown in Table 11, key plant parameters remained steady over the full 100 hours. Average NSSS output was 3351.6 MWt, well within the Acceptance Criteria of 3350 MWt (+2%, -5%). During the 4-hour performance test, NSSS output averaged 3350.4 MWt, easily meeting the 3350 MWt +2% Acceptance Criteria. In addition, steam generator pressures and feedwater temperatures were very close to their respective design values of 805 psia and 432 deg. F, while the averaged reactor power level was almost exactly 100% RTP.



Table 11

## Nuclear Steam Supply System Acceptance Test Data

Date	Time	Reactor Power (%)	NSSS Power (%)	NSSS Output (MWt)	Gen. Output (MWe)	Steam Generator Pressure (psia)				Feedwater Temperature (deg. F)				
						Loop 1	Loop 2	Loop 3	Loop 4	Loop 1	Loop 2	Loop 3	Loop 4	
1	3/24/85	1130	99.52	99.52	3333.8	1120	805	810	807	806	428	428	428	428
2	3/24/85	2300	99.95	99.95	3348.2	1120	805	804	807	806	428	428	428	428
3	3/25/85	1130	99.89	99.88	3346.0	1122	805	806	807	803	428	428	428	428
4	3/26/85	0030	100.06	100.05	3351.8	1122	812	809	813	812	428	428	428	428
5	3/26/85	1000	99.79	99.78	3342.7	1123	805	803	807	807	428	428	428	428
6	3/26/85	1030	99.93	99.92	3347.2	1123	805	804	808	807	438	428	428	428
7	3/26/85	1100	99.97	99.96	3348.7	1123	806	804	808	805	428	428	428	428
8	3/26/85	1130	100.34	100.33	3361.2	1123	807	805	809	809	428	428	428	428
9	3/26/85	1200	100.07	100.07	3352.2	1123	804	802	806	806	428	428	428	428
10	3/26/85	1230	100.11	100.10	3353.3	1123	805	803	807	807	428	428	428	428
11	3/26/85	1300	99.98	99.97	3349.0	1123	804	802	806	806	428	428	428	428
12	3/26/85	1330	99.96	99.96	3348.5	1123	804	802	807	806	428	428	428	428
13	3/26/85	2300	100.25	100.24	3358.0	1123	807	805	810	810	428	428	429	429
14	3/27/85	1030	100.41	100.40	3363.4	1122	813	811	815	815	428	428	428	428
15	3/27/85	2200	100.26	100.25	3358.5	1123	806	807	808	804	429	429	429	429
16	3/28/85	0630	100.40	100.39	3363.1	1120	810	808	812	812	429	428	429	429
AVERAGE			100.06	100.02	3351.6	1122	806	805	809	808	428	428	428	428



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

### TEST OBJECTIVE

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

### TEST DESCRIPTION

With the plant at 100% power, stable, and on automatic control, the event was initiated by tripping the turbine from the control console. During the transient, the following parameters were among those monitored:

- Reactor Coolant Temperature
- Reactor Trip and 500 KV breakers (opening)
- Control Rod Bank Position
- Pressurizer Pressure and Level
- Steam Flow and Pressure (Loop 1 only)
- Steam Generator Level (Loop 1 only)
- Feedwater Pump Speed and Control Variables
- Turbine Speed
- Feedwater Flow and Pressure (Loop 1 only)

### TEST RESULTS

The following acceptance criteria were met satisfactorily:

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

The acceptance criterion for overall RTD response time was not met. The actual response time is defined as the interval of time measured between the point where the neutron flux has decreased by 50% of its initial value to the point where the hot leg temperature has decreased by 33-1/2% of the initial loop  $\Delta T$  value. The actual time was 7.6 seconds and the acceptance criterion specified this time must be no greater than 7.3 seconds. This was reviewed by Westinghouse and determined to be satisfactory provided previous RTD Bypass response time testing had not violated Technical Specification requirements. A review by PGandE NPO Nuclear Engineering verified the adequacy of previous RTD Bypass response time testing.



Responses of key plant parameters are given in Table 12.

Actual plant response for pressurizer level, pressurizer pressure and time delay between turbine trip and generator trip did not meet expectations. Minimum pressurizer level was 19% despite being expected to remain greater than or equal to 20%. Pressurizer pressure varied between 1975 - 2263 psig; anticipated variation was between 2000 and 2240 psig. The time delay between turbine trip and generator trip was 44.4 seconds as opposed to an expected time of 30 seconds. These deviations from expected values were reviewed by Westinghouse and PGandE Engineering. Both organizations deemed the results acceptable and no setpoint changes were recommended.



Table 12

## Key Plant Parameters During Plant Trip from 100% Power

Pressurizer Level: Minimum Level (%)	19
Pressurizer Pressure: Initial Pressure (psig) Minimum Pressure (psig) Maximum Pressure (psig) Maximum - Initial (psid) Initial - Minimum (psid)	2240 1975 2263 23 265
Tave: At Feedwater Isolation (deg. F) Need for intervention	553.7 none
Steam Generator Levels: Minimum S/G 1-1 Level (%) Minimum S/G 1-2 Level (%) Minimum S/G 1-3 Level (%) Minimum S/G 1-4 Level (%)	<0 (narrow range) <0 (narrow range) <0 (narrow range) <0 (narrow range)
Steam Dump Valves:	modulated closed
Time Delay Between Turbine Trip and Generator Trip (sec.)	44.4

Expected Responses

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



## 19.0 Test Procedure 42.7 - Natural Circulation Boron Mixing Test

### TEST OBJECTIVE

The main objectives of the test were to accomplish the following:

- a) Establish and maintain natural circulation conditions using core decay heat,
- b) Confirm that adequate mixing of borated water added to the Reactor Coolant System (RCS) prior to cooldown can be achieved under natural circulation conditions,
- c) Verify that the RCS can be borated to the Cold Shutdown concentration,
- d) Determine if cooldown and depressurization of the RCS from normal Hot Standby to Cold Shutdown conditions can be accomplished using only safety grade equipment.

### TEST DESCRIPTION

The natural circulation, boron mixing, and cooldown test was performed at Diablo Canyon Unit 1 on March 28 and 29, 1985. The test was structured to collect and evaluate data on the various phenomena associated with a natural circulation cooldown. The sequence of events is outlined in the following paragraph.

Plant trip was initiated by manually initiating a turbine trip. The plant was then stabilized in the Hot Standby mode. Natural circulation was established by tripping the reactor coolant pumps at 0028 hours on March 29. The Boron Mixing/Maintain Hot Standby Conditions phase of the test commenced when natural circulation conditions were declared established at 0048 hours. During this portion of the test, a charging pump was aligned to deliver the contents (900 gallons of 21000 ppm borated water) of the Boron Injection Tank (BIT) to each of the four RCS cold legs. Boron injection was initiated at 0052 hours, terminated at 0113 hours, and boron mixing verified during the remainder of the Maintaining Hot Standby Conditions phase of the test. The Cooldown/Depressurization test phase was initiated at 0450 hours. Cooldown/Depressurization to Residual Heat Removal (RHR) system initiation conditions was achieved and RHR system operation initiated at 1805 hours. The cooldown/depressurization phase was continued until Cold Shutdown conditions were achieved at 2245 hours.

### TEST RESULTS

Stable natural circulation flow rates to ensure core decay heat removal, boron mixing and plant cooldown were established and maintained.  $\Delta T$  was approximately 15-20 deg. F throughout the test.

Upon injection from the BIT, RCS boron concentration increased to approximately 1224 ppm (340 ppm increase) and mixed very quickly to a final value of approximately 1186 ppm (300 ppm increase). Approximately 12 minutes after the delivery of the BIT's contents, the RCS boron concentration was within 20 ppm of the final RCS boron concentration.



Plant cooldown was achieved without significant problems and within Technical Specification limits. All active portions of the RCS remained within 100 deg. F of the average core exit temperature and both the steam generators and reactor vessel upper head were cooled to below 450 deg. F when core exit temperature was at 350 deg. F.

Subsequent actions involved RCS depressurization to approximately 375 deg. F, initiation of the RHR system, and finally the successful establishment of Cold Shutdown conditions.

A more detailed description of the test sequence of events and the test results is being submitted to the Nuclear Regulatory Commission in a separate report.



20.0 Unscheduled Reactor Trips

Trip	Date	Time	Power (%)	Cause	Comments
1	5/6/84	2200	0	OPΔT, OTΔT on Loops 2 and 3	Loop 3 was out of service
2	5/8/84	1300	0	High steam flow	Related to malfunction in in Steam Dump Control System
3	11/24/84	1614	~31	P-14; High S/G level	Related to DEH problem
4	11/28/84	2207	~20	Overfeeding S/G 1-1 caused P-14 Turbine trip	During transfer from Bypass Valves to Main Feedwater Reg. Valves when decreasing Turbine load
5	12/5/84	1636	~10	P-13; Turbine trip	P-13 failed to reset
6	12/6/84	0656	0	High Source Range Count	P-6 permissive and High Source Range Trip set-points were set too close
7	1/2/85	1945	~50	Reactor Coolant Loop Low Flow	Solenoid valve failure during Test Procedure 43.7 prevented Main Turbine governor valve operation which caused a drop in turbine speed, generator frequency and hence RCS flow
8	1/4/85	1355	~10	P-10; Power >10% with Turbine tripped	Main Feedwater Turbine tripped, causing reactor power to exceed 10% power
9	2/13/85	0045	~75	High Steam Line ΔP	Radio transmission near the steam line pressure transmitters caused an erroneous transmitter signal
10	2/17/85	0415	~47	Manual trip after both Main Feedwater Pumps tripped	Feedwater Pumps tripped on thrust bearing wear
11	3/21/85	1648	100	Power leads shorted	During RWST troubleshooting in Hagen rack room

