PACIFIC GAS AND ELECTRIC COMPANY

DIABLO CANYON POWER PLANT

UNIT 1

SUPPLEMENT 1

TO THE

STARTUP REPORT

TO THE

UNITED STATES

NUCLEAR REGULATORY COMMISSION

LICENSE NUMBER DPR-80

FOR THE PERIOD NOVEMBER 1, 1984

THROUGH JANUARY 31, 1985

IE26
1/1

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DIABLO CANYON POWER PLANT
UNIT 1 STARTUP REPORT - SUPPLEMENT 1

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SUMMARY

The Diablo Canyon Power Plant Unit 1 Startup Program activities included in this report cover the period from November 1, 1984 to January 31, 1985.

The full power operating license was received on November 2, 1984. Preparations for power ascension began on November 3, 1984 and mode 1 was entered for the first time on November 9, 1984. The generator was synchronized on November 12, 1984 and testing at 15% power was completed on November 22, 1984. The power level was increased to 30% on November 22, 1984 and testing at 30% power level was completed on December 19, 1984. The major delay at this test plateau was the unsatisfactory steam generator level control system performance which resulted in design changes.

The power level was increased to 50% on December 20, 1984 and testing at this plateau was completed on January 5, 1985. After the rod group drop and plant trip from 50% power on January 5, 1985, capability to maintain hot standby conditions from outside control room was demonstrated and the plant was cooled down for a maintenance/inspection outage. As of January 31, 1985 the unit remained in cold shutdown conditions.

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

TEST PROCEDURE

The objective of this test was to collect statepoint data and verify core power level at various power ascension test plateaus. The data included temperatures, pressures, and steam generator water 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%, and 50% 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conditions at each plateau consisted of stable plant parameters, equilibrium xenon, and control rods at or near fully withdrawn positions.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 pressure, 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 determined by a secondary side heat balance on the steam generators. Parameters related to control and safety systems provided as 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 plateaus during the power ascension test program.

This test will also be performed at the 75%, 90%, and 100% test plateaus.

RESULTS

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

Measured steady state, equilibrium power distributions (i.e., relative assembly power, radial power shape, axial power shape, quadrant power tilt, peaking factors) were within Acceptance Criteria and very close to design predictions. Peaking factors were well below safety limits specified by the Technical Specifications. Results of the flux maps are summarized in Table 1 and Figures 2 and 3. At each power plateau, $F^N_{\Delta H}$ and F^T_Q obtained were compared to the limiting values at the next power plateau and found acceptable.

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

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

Table 1

Power Distribution Results

ITEM		30% POWER TEST PLATEAU	50% POWER TEST PLATEAU
CONDITIONS*	- temperature - boron concentration - burnup	~556 deg. F 1080 ppm 57 MWD/MT	~562 deg. F 1020 ppm 148 MWD/MT
DATE		11-24-84	12-22-84
$F^N_{\Delta H}$	- Measured value - location**	1.427 D04-IJ	1.391 M12-IH
F^T_Q	- Measured value - location**	2.103 D04-IJ @77"	2.033 M12-IH @74"
F_Z	- Measured value	1.369	1.359
Quadrant Tilt	- Measured value	1.008	1.006

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

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

FIGURE 2

CORE AVERAGE RADIAL POWER DISTRIBUTION - 30% TEST PLATEAU

Assembly Average Powers From Unrodded Flux Map

RELATIVE ASSEMBLY POWER (Pi)

$$\frac{\text{MEASURED Pi} - \text{EXPCTED Pi}}{\text{EXPECTED Pi}} \times 100$$

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

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DIABLO CANYON POWER PLANT - UNIT I

FIGURE 3

CORE AVERAGE RADIAL POWER DISTRIBUTION - 50% TEST PLATEAU

Assembly Average Powers From Unrodded Flux Map

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

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

NORTH ↓

DIABLO CANYON POWER PLANT - UNIT I

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 Power 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 reactor power levels and identify any location where shielding may be deficient.

TEST DESCRIPTION

Radiation measurement locations were selected such that locations with the highest uncertainty of proper shielding were measured. Common measurement locations were entrances to labyrinths, shield wall penetrations, and primary shield walls. Each measurement location was identified with a Radiation Base Point (RBP) number or penetration number.

Neutron and gamma radiation measurements were performed at each RBP for each reactor power level plateau. The radiation dose rates at each RBP were expected to increase linearly as a function of increasing reactor power. A linear regression by a least squares fit of the measurements was performed for each RBP. The resulting linear equation was extrapolated to 100% reactor power and the extrapolated dose rate compared to FSAR zone requirements.

A correlation coefficient was calculated for each RBP and used to measure the degree of linear relationship between reactor power and the measured dose rates. A larger correlation coefficient indicates a greater degree of linear relationship, with a correlation coefficient of 1.00 being an exact linear relationship.

TEST RESULTS

The maximum reactor power level achieved at the time of this report was 50%. Further radiation measurements will be performed as higher reactor power levels are achieved, the results of which will be included in a final report.

A preliminary review indicated that all surveyed areas, when extrapolated to their maximum value at 100% power, will meet the FSAR radiation zone requirements. Most RBPs exhibit a positive linear relationship between increasing reactor power and dose rate measurements. Special survey procedures have been initiated in those areas identified as possibly exceeding FSAR zone limits in the "Shielding Design Review". Close monitoring and review of survey data will continue throughout the remainder of the Bio-Shield Survey of all RBPs at the remaining test power plateaus.

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

TEST OBJECTIVE

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

TEST DESCRIPTION

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

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

After shutdown from power operations at the 50% power test plateau, the source range operating voltage was checked, the intermediate range detectors' compensation voltages were set and the current 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. The source range instrumentation data prior to core loading is listed in Table 2. The Nuclear Instrumentation data prior to Startup is shown in Table 3. Results of the nuclear instrumentation overlap data taken prior to criticality and at various power levels are shown in Table 4.

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

The following were performed shortly after shutdown with a core burnup of at least 1200 MWD using I&C procedures.

(1) Source range detector high voltage settings were determined to be 2200 Vdc for N31 and N32 using the reactor as neutron source. Figures 4 and 5 show the source range detector characteristics.

(2) Power range detector currents equivalent to 100% indicated power were determined. Results are shown in Table 5.

(3) With an overlap of about three decades existing with the source range, each intermediate channel was adjusted to provide proper compensation (6.5 Vdc for N35 and 25.8 Vdc for N36).

Power range detector high level trip setpoints were reset prior to power increase to the next power plateau as listed in Table 6.

This test will continue through the rest of the power ascension program.

Table 2

Source Range Instrumentation Data Prior to Core Loading

Parameter	Units	Detector	
		N31	N32
Attenuator Setting	db.	10	10
Discriminator Voltage	Vdc	-1.02	-1.013
Detector Voltage	Vdc	2350	2350
Detector Voltage bistable trip	Vdc	2240	2175
High Flux alarm	cps	18.7	18.5
High Flux trip	cps	8.578×10^4	8.33×10^4

Table 3

Nuclear Instrumentation Data Prior to Startup

Intermediate Range Channels

Parameter	Units	Detector	
		N35	N36
1. High Voltage Setting	Vdc	800.76	799
2. Compensating Voltage	Vdc	-40.287	-40.093
3. Compensating Voltage Bistable Trip	Vdc	-19.999	-20.0
4. Loss of Detector Trip	Vdc	703	700.2
5. P-6 Bistable Trip	amp	1.35×10^{-10}	1.33×10^{-10}
6. Rod Stop Bistable Trip	amp	7.4×10^{-5}	7.1×10^{-5}
7. Reactor Trip Bistable	amp	9.8×10^{-5}	9.8×10^{-5}

Power Range Channels

Parameter	Units	Detector			
		N41	N42	N43	N44
1. High Voltage Setting	Vdc	800	800	800	800
2. High Voltage Bistable Trip	Vdc	698.7	698.3	699.1	698.8
3. P10 Bistable Trip	%	10.3	10.26	10.21	10.27
4. P8 Bistable Trip	%	34.61	34.51	34.54	34.52
5. Overpower Rod Stop Bistable Trip	%	103.09	103.07	103.04	103.02
6. High Neutron Flux Rate Trip	%	4.704	4.752	4.728	4.776
7. Flux Rate Time Constant	sec	2.14	2.145	2.15	2.145

Table 4

Nuclear Instrumentation Overlap Data

DETECTOR	PRECRITICAL READINGS	0% POWER	~3% POWER	~15% POWER	~30% POWER	~50% POWER
SOURCE RANGE (cps)						
N31 - Control Board	45	$3.8x10^{4}$	Blocked	Blocked	Blocked	Blocked
N31 - NI Drawer	45	$3.1x10^{4}$	Blocked	Blocked	Blocked	Blocked
N32 - Control Board	55	$4.8x10^{4}$	Blocked	Blocked	Blocked	Blocked
N32 - NI Drawer	55	$4.8x10^{4}$	Blocked	Blocked	Blocked	Blocked
INTERMEDIATE RANGE (amps)						
N35 - Control Board	$1.0x10^{-11}$	$2x10^{-10}$	$1.2x10^{-5}$	$8x10^{-5}$	$1.0x10^{-4}$	$2.5x10^{-4}$
N35 - NI Drawer	$1.0x10^{-11}$	$2x10^{-10}$	$9.9x10^{-6}$	$6x10^{-5}$	$1.0x10^{-4}$	$1.4x10^{-4}$
N36 - Control Board	$1.0x10^{-11}$	$2x10^{-10}$	$1.3x10^{-5}$	$8x10^{-5}$	$1.0x10^{-4}$	$2.5x10^{-4}$
N36 - NI Drawer	$1.0x10^{-11}$	$2x10^{-10}$	$1.0x10^{-5}$	$6x10^{-5}$	$1.0x10^{-4}$	$1.4x10^{-4}$
POWER RANGE (%)						
N41 - Control Board	0	0	3.0	15.0	32.0	51
N41 - NI Drawer	0	0	2.9	14.5	31.0	50
N42 - Control Board	0	0	3.0	14.0	31.0	46
N42 - NI Drawer	0	0	3.0	14.0	31.5	46.5
N43 - Control Board	0	0	1.0	13.0	30.0	50
N43 - NI Drawer	0	0	2.9	14.0	31.0	50.5
N44 - Control Board	0	0	2.1	14.0	31.0	47
N44 - NI Drawer	0	0	2.9	14.5	32.5	47.5

Table 5

Power Range Detector Currents to Give 100% Reading

Detector	Upper Detector Current (μa)	Lower Detector Current (μa)
N41	432.7	432.2
N42	432.3	467.4
N43	434.4	450.5
N44	441.1	439.6

Table 6

Power Range High Level Trip Set Points

Power Plateaus (% RTP)	Desired Setpoint (%)	Actual Set Point (%)			
		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

Figure 4

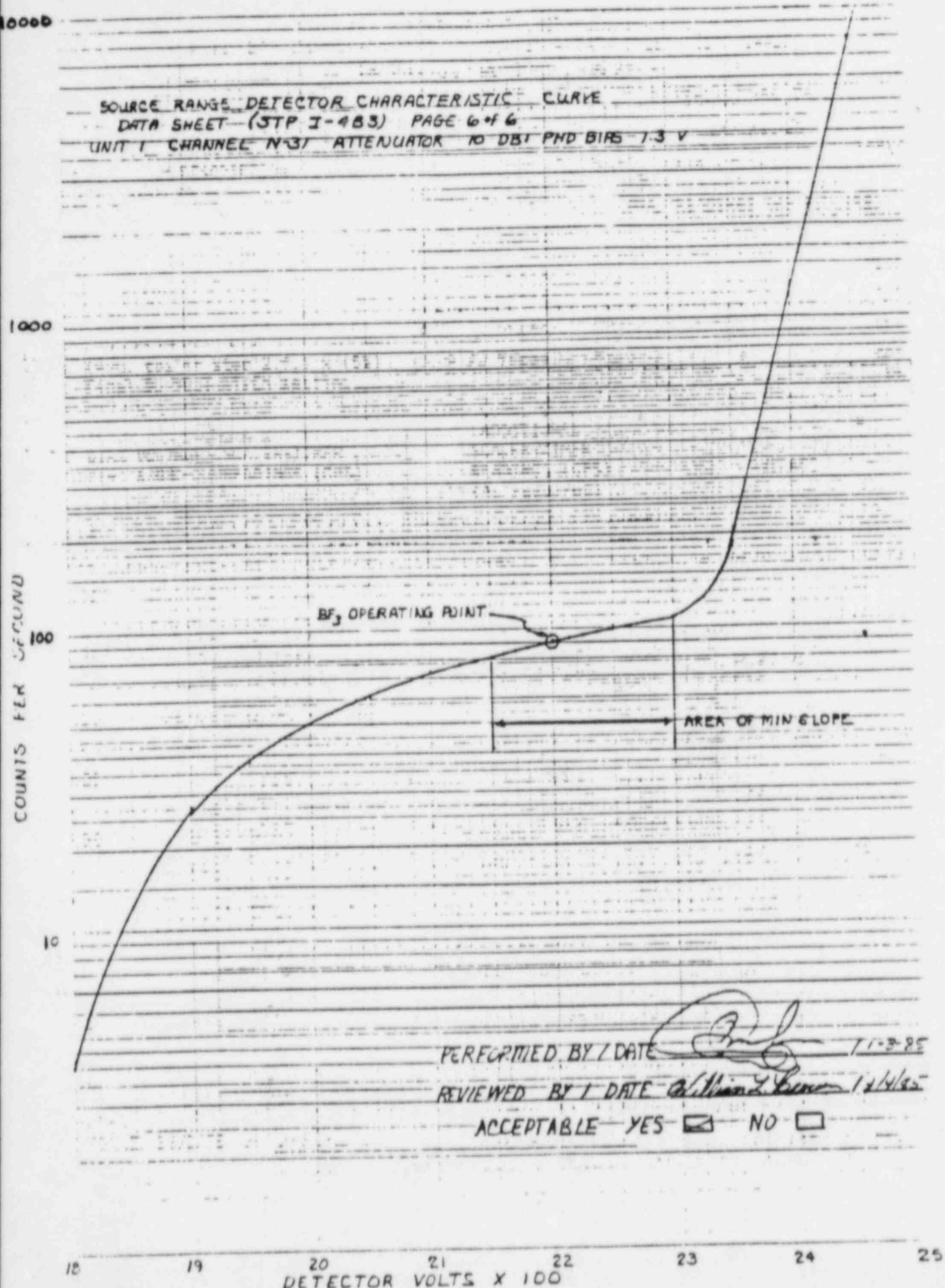
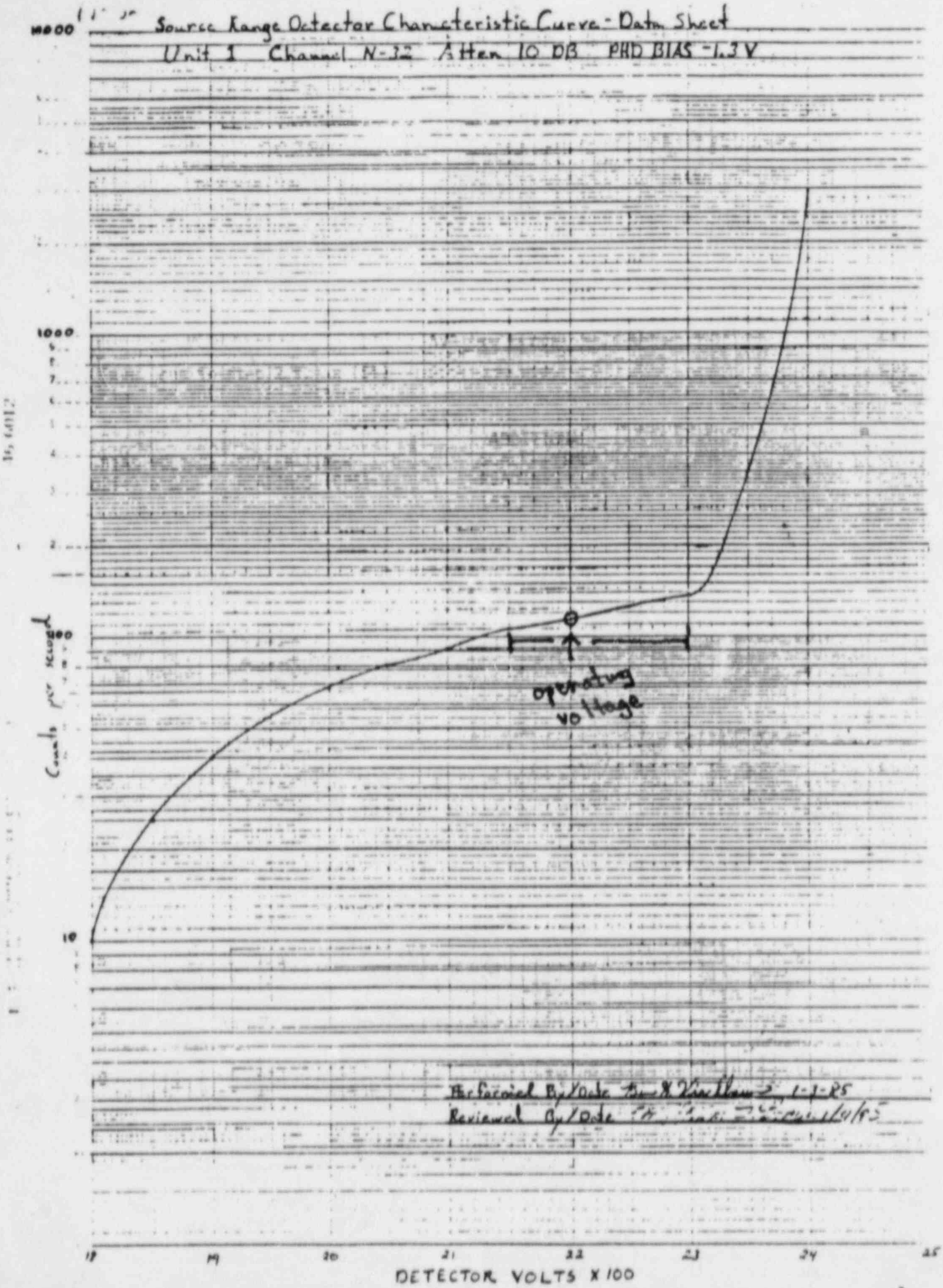

SOURCE RANGE DETECTOR CHARACTERISTIC CURVE
DATA SHEET (STP I-483) PAGE 6 of 6
UNIT 1 CHANNEL N-31 ATTENUATOR 10 DB PHD BIAS 1.3 V
10000
1000
100
10
COUNTS PER SECOND
BF3 OPERATING POINT
AREA OF MIN SLOPE
PERFORMED BY / DATE 11-3-85
REVIEWED BY / DATE 11/4/85
ACCEPTABLE YES NO
18
19
20
21
22
23
24
25
DETECTOR VOLTS X 100

Figure 5


Source Range Detector Characteristic Curve - Data Sheet
Unit 1 Channel N-32 Atten 10 DB PHD BIAS -1.3V
10000
1000
100
10
Counts per second
operating voltage
18
19
20
21
22
23
24
25
DETECTOR VOLTS X 100
Performed By/Date
Reviewed By/Date

4.0 Test Procedure 42.8 - Operational Alignment of Reactor Coolant System Temperature Instrumentation

TEST OBJECTIVE

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

TEST DESCRIPTION

ΔT and Tavg data collected in Test Procedure 42.5 - Thermal Power Measurment and State Point Data collection, was transcribed to this test at each power level. At isothermal conditions, ΔT and Tavg values were also determined from Thot and Tcold values. Linear regression analysis will be performed at the 75% and 100% plateaus to determine the extrapolated ΔT and Tavg at 100% power and the channels calibrated, if required.

TEST RESULTS

At isothermal conditions, ΔT and Tavg values agreed with the values calculated from Thot and Tcold values within the specified tolerance as shown in Table 7. Further analysis and calibration, if any, will be performed at the 75% and 100% power plateaus.

Table 7

<u>ΔT and Tavg at Isothermal Conditions</u>

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
ΔT (calculated)	-0.25	-0.31	-0.16	-0.46
ΔT (measured)	-0.19	-0.227	-0.14	-0.37
Tavg (calculated)	546.3	546.3	546.28	546.31
Tavg (measured)	546.45	546.43	546.45	546.6

<u>Acceptance Criteria</u>

ΔT (measured) = ΔT (calculated) ±0.3 deg. F
Tavg (measured) = Tavg (calculated) ±1 deg. F

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

TEST OBJECTIVES

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

TEST DESCRIPTION

The feedwater and steam flow instrumentation output signals were checked against the reference feedwater flow at steady state power levels of 30% and 50% Rated Thermal Power. Test data collected as part of Startup T.P. 42.5, Thermal Power Measurement and Statepoint Data Collection, were analyzed by this procedure to determine deviation of steam and feedwater flow compared to the reference feedwater flow.

TEST RESULTS

On November 24, 1984, data were analyzed for the 30% test plateau. Five Steam Flow Transmitters and two Feedwater Flow Transmitters were outside the tolerance of ≤2% and ≤1.5% respectively. As allowed by the procedure at 30% plateau only, it was decided to recalibrate all Steam Flow Transmitters but none of the Feedwater Flow Transmitters.

On December 27, 1984, data was analyzed for the 50% test plateau. Two Steam Flow Transmitters and one Feedwater Flow Transmitter were outside of tolerance. These will be calibrated during the outage and reverified to be within tolerances at 50% power prior to escalation to 75% power.

The test will be repeated at the 75%, 90% and 100% power plateaus.

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

TEST OBJECTIVE

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

TEST DESCRIPTION

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

TEST RESULTS

M-21B was performed satisfactorily and the results are shown in Table 8.

Table 8

Turbine Overspeed Setpoints

	Trip Setting	Actual	Acceptance Criteria
DEH	103%	1855 rpm	1854 ± 5 rpm
Mechanical	111%	1966 rpm	1998 + 2, -36 rpm
DEH*	111.5%	1928 rpm	1926 ± 5 rpm

*NOTE: The setpoints are automatically reduced by 4.5.% while testing the 11.5% trip setting from the digitial electrohydraulic (DEH) Unit.

7.0 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 30% and 50% power plateaus. The flux maps were also used to fulfill the routine surveillance requirements.

TEST DESCRIPTION

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

Below is a chronological summary of the flux maps taken during this portion of the power ascension test program to 50% power:

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

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

-30% power, Control Bank D (except RCCA B-6) at ~177 steps, RCCA B-6 fully withdrawn to simulate a rod ejection; provided post-ejection data for T.P. 42.2 - RCCA Pseudo Ejection and RCCA above bank position measurements.

-50% power, ARO, equilibrium xenon; provided baseline data for T.P. 42.5 - Thermal Power Measurment and State Point Data Collection and provided reference data for T.P. 42.3 - Static Rod Drops and RCCA below bank position measurements.

-50% power, ARO (except RCCA H-4), RCCA H-4 fully inserted to simulate a rod drop; provided data for T.P. 42.3 - Static Rod Drop and RCCA below bank position measurements.

-50% power, ARO equilibrium xenon; provide reference data for STP R-13 - Nuclear Power Range Incore-Excore Detector Calibration.

-7 quarter-core maps at 50% power provided data for STP R-13 Nuclear Power Range Incore-Excore Detector Calibration.

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.

The incore power distribution will also be determined at 75%, 90% and 100% power levels.

8.0 Test Procedure No. 1.16 - Effluents and Effluent Monitoring

TEST OBJECTIVE

The main objective of this procedure was to document the existence of an adequate program to verify the level of 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 Power 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 this verification 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 50% power test plateau, the activity levels at each monitor had not been large enough to verify the calibration of the monitors. The test will continue through the remainder of the Power Ascension Program.

9.0 Test Procedure No. 1.17 - Chemical and Radiochemical Analysis

TEST OBJECTIVE

The objectives of this procedure were to document the ability to perform reactor plant chemical and radiochemical analysis and to document the ability to control water chemistry.

TEST DESCRIPTION

Plant systems sampling, analysis, and chemistry control are part of an ongoing program by Nuclear Plant Operations (NPO) involving the use of approved procedures in the Plant Manual. One specific area of interest was to evaluate the performance of the Boron Concentration Measurement System (BCMS).

TEST RESULTS

Reactor plant chemistry has been maintained within Technical Specifications Limits up to and including the 50% power test plateau. Main Control Room indication of RCS boron concentration has agreed with chemical samples to within 10 ppm when the plant is operating under steady state conditions.

10.0 Control Systems Checkout

10.1 Test Procedure No. 38.1 - Automatic Reactor Control

TEST OBJECTIVES

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

TEST DESCRIPTION

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

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

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

TEST RESULTS

The Automatic Reactor Control System responded properly to a ±6 deg. F temperature mismatch between Tavg and Tref. The Test Acceptance Criteria were met and the plant stabilized properly, within acceptable limits, after the Rod Control System automatically compensated for the temperature mismatch and brought Tavg back to Tref. Table No. 9 compares the actual data obtained with acceptable data limits.

Table 9

<u>Automatic Reactor Control System Response</u>

Initial Condi- tion	Description	Acceptable Data Limits	Actual Data
	Tref - Tavg (Initial Condition)	±1.5 deg. F	0.5 deg. F
Tavg > Tref	Maximum - Initial Pressurizer Pressure Initial - Minimum Pressurizer Pressure Peak-to-Peak Amplitude of Tavg Oscillation Minimum Period of Tavg Oscillation |Tref - Tavg| (After Transient)	≤65 psig ≤65 psig ≤5 deg. F ≥60 sec. ±1.5 deg. F	15 psig 60 psig 3.5 deg.F 99.6 sec. +0.4 deg.F
Tavg < Tref	Maximum - Initial Pressurizer Pressure Initial - Minimum Pressurizer Pressure Peak-to-Peak Amplitude of Tavg Oscillation Minimum Period of Tavg Oscillation |Tref - Tavg| (After Transient)	≤65 psig ≤65 psig ≤5 deg. F ≥60 sec ±1.5 deg. F	5.7 psig 12 psig 2.6 deg. F 138 sec +0.63 deg.F

10.0 Control Systems Checkout

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

TEST OBJECTIVE

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

TEST DESCRIPTION

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

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

TEST RESULTS

The test was started with initial settings suggested by Westinghouse. This resulted in unacceptable oscillations in level. The problem was traced to too fast a response of the level control valves. These valves had volume boosters to meet closing time requirements. Changing the control system gains and resets did not resolve the problem. After discussions with Westinghouse, a design change was implemented to delete the volume boosters, modify the pneumatic tubing and install proper solenoid valves to meet the required closing times. After the change was made, the test was successfully completed with the settings initially suggested by Westinghouse.

The Feedwater Pump Speed Controllers worked satisfactorily without any adjustment to controller settings.

10.0 Control Systems Checkout

10.3 Test Procedure 41.8 - Dynamic Automatic Steam Dump Control

TEST OBJECTIVE

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

TEST DESCRIPTION

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

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

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

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

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

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

TEST RESULTS

During the conduct of the Turbine Trip Controller Test, a controller module (TM500) in the Tavg control circuit was discovered to require re-calibration. The out-of-tolerance controller module created too much of a temperature mismatch which caused the Reactor Coolant System to cooldown below the expected Tavg value of 548 deg. F. After it was calibrated, this portion of the test was repeated satisfactorily. On the power increase transient from 1% to 6%, the Steam Dump System responded satisfactorily and Tavg stabilized at 550.1 deg. F (within the acceptance criteria of 549.4 deg. F to 554.6 deg. F).

The testing of the Load Rejection Controller was performed without any problems. During the transient, the Steam Dump System responded satisfactorily and Tavg stabilized at 547.2 deg. F (within the acceptance criteria of 546.4 deg. F to 551.6 deg. F).

The Steam Dump System responded satisfactorily during the Steam Pressure Controller Test. The transient steam pressure stabilized at 987 psig (within the acceptance criteria of 986.2 to 1023.8 psig).

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

10.0 Control Systems Checkout

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

TEST OBJECTIVE

The objective of this test procedure was to determine the reactor coolant average temperature program required to maintain the design full load Turbine Impulse Chamber pressure.

TEST DESCRIPTION

Reactor Coolant Tavg, Steam Generator pressure and Turbine Impulse Chamber pressure were recorded at 0%, 30% and 50% Rated Thermal Power (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 Tavg. This correction was applied to the design temperature program generated by the Reactor Control System, Steam Dump Control System and plant computer. With Tavg controlled at the new Tref, Turbine Impulse Chamber pressure is compared to the 50% load design value and agreement verified. This entire process is repeated at 75% RTP to obtain a further refinement in the temperature program. Upon reaching 100% RTP, the temperature program is adjusted (if necessary) to obtain the design value of Turbine Impulse Chamber pressure. Throughout this procedure, changes in the temperature program are constrained to design limitations on Tavg and Turbine Inlet pressure.

TEST RESULTS

Test data were taken at 0%, 30% and 50% RTP and the results were plotted and extrapolated to 100% RTP. A Tref correction of -3.46 deg. F was subtracted from the max design Tref of 576.6 to yield the new projected 100% RTP Tref of 573.14 deg. F. This value correlates 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 Controllers TC-505 and TC-505A. The calibration will be done during the outage and the readings reverified at 50% power.

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.

This test will be performed at 75% and 100% power plateaus also.

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

TEST OBJECTIVE

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

TEST DESCRIPTION

The test was performed at the 30% power plateau with the plant stable and Control Bank D at the hot full power rod insertion limit of 177 steps.

After verifying these conditions, the movable incore detectors were used to perform a flux map to determine the "pre-ejection" power distribution.

While maintaining constant turbine power and boron concentration, RCCA B-6 was withdrawn from 177 to 200 steps. At 200 steps a partial flux map (i.e., data from 6 of the 58 flux thimble locations) was taken, after which the rod was fully withdrawn. With RCCA B-6 withdrawn, a full core flux map was taken for the post-ejection power distribution. Finally, RCCA B-6 was returned to its initial position.

TEST RESULTS

Power distribution results are summarized in Table 10 and core average radial power distributions are shown in Figure 6. All Acceptance Criteria were met and no significant problems were encountered during the performance of this test.

The post-ejection value of F_Q^T (i.e., heat flux hot channel factor) was 2.294, well below the Acceptable Criteria (FSAR) limit of 7.07. Ejected rod worth was 14.3 pcm, well below the Acceptance Criteria limit of 200 pcm.

Table 10

Power Distribution Results - Pre and Post Pseudo Rod Ejection

ITEM	PRE-EJECTED FLUX MAP	POST-EJECTED FLUX MAP
Conditions - power - temperature (RCS) - boron concentration - burnup	30% ~553 deg. F 1060 ppm 200 mwd/mt	30% ~556 deg. F 1060 ppm 200 mwd/mt
Date	12-11-84	12-11-84
Rod Configuration	Bank D @ 177 steps	RCCA B-6 @ 228 steps
$F^N_{\Delta H}$ - measured value - location *	1.398 M12-IH	1.490 D04-HJ
F^T_Q - measured value - location *	2.218 L02-QQ @ 58 in	2.294 P11-QM @ 58 in
F_Z - measured value	1.447	1.414
QUADRANT TILT - measured value	1.006	1.044

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

FIGURE 6

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

PRE-EJECTION ASSEMBLY POWER
POST-EJECTION ASSEMBLY POWER

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

b) Ejected location: B-6

	R	P	N	M	L	K	J	H	G	F	E	D	C	B	A
1					.603 .571	.671 .656	.807 .785	.730 .721	.799 .826	.649 .725	.598 .624				
2			.521 .556	.894 .940	1.062 1.021	.974 .934	1.035 1.016	1.022 .967	1.012 1.032	.925 .962	.972 1.034	.856 .869	.515 .555		
3		.521 .536	1.084 1.091	.970 1.039	1.177 1.111	1.157 1.130	1.180 1.113	1.130 1.100	1.150 1.104	1.045 1.074	1.072 1.085	.944 1.002	1.073 1.135	.531 .599	
4		.896 .890	.963 1.018	1.288 1.246	1.071 1.021	1.174 1.086	1.159 1.126	1.200 1.151	1.140 1.145	1.151 1.141	1.052 1.102	1.293 1.301	.978 1.067	.902 .965	
5	.576 .597	1.000 1.031	1.113 1.106	1.066 1.076	1.181 1.012	1.145 1.109	1.209 1.151	1.154 1.150	1.195 1.165	1.124 1.146	1.167 1.152	1.080 1.122	1.123 1.136	1.013 1.067	.573 .596
6	.646 .701	.935 .948	1.072 1.072	1.170 1.124	1.143 1.121	1.136 1.086	1.027 1.028	1.091 1.066	1.021 1.039	1.125 1.117	1.121 1.152	1.184 1.181	1.096 1.146	.960 1.013	.667 .702
7	.785 .845	1.016 1.056	1.140 1.095	1.142 1.135	1.170 1.125	1.012 1.008	1.045 1.004	.946 .952	1.054 1.012	1.018 1.025	1.181 1.151	1.156 1.166	1.163 1.174	1.033 1.098	.813
8	.721 .753	1.020 1.003	1.108 1.101	1.192 1.141	1.119 1.116	1.075 1.033	.931 .934	.951 .928	.951 .951	1.103 1.056	1.142 1.129	1.210 1.166	1.130 1.158	1.034 1.041	.7 .794
9	.783 .809	1.016 1.038	1.138 1.085	1.146 1.126	1.170 1.130	1.018 1.016	1.036 .992	.948 .948	1.065 1.014	1.042 1.021	1.196 1.133	1.161 1.143	1.165 1.144	1.031 1.070	.805 .835
10	.676 .700	.972 .988	1.066 1.083	1.165 1.130	1.133 1.141	1.133 1.105	1.004 1.008	1.087 1.053	1.024 1.134	1.161 1.104	1.153 1.129	1.187 1.127	1.093 1.075	.961 .961	.666 .681
11	.605 .647	1.066 1.158	1.184 1.262	1.060 1.065	1.178 1.124	1.138 1.122	1.169 1.109	1.107 1.109	1.147 1.121	1.129 1.129	1.185 1.134	1.077 1.067	1.103 1.044	1.024 1.057	.581 .592
12		.912 .936	.975 1.011	1.294 1.279	1.076 1.067	1.182 1.130	1.139 1.106	1.167 1.126	1.117 1.115	1.160 1.117	1.073 1.056	1.294 1.222	.972 1.023	.898 .942	
13		.516 .535	1.075 1.088	.976 1.022	1.119 1.061	1.061 1.074	1.126 1.075	1.076 1.081	1.139 1.102	1.093 1.097	1.121 1.111	.965 .990	1.071 1.089	.522 .579	
14			.516 .537	.888 .888	1.012 1.019	.945 .932	1.018 1.051	1.025 1.019	1.052 1.077	1.001 .967	1.023 1.058	.881 .919	.518 .567		
15					.578 .572	.652 .654	.803 .833	.742 .779	.839 .859	.692 .680	.581 .590				

NORTH

DIABLO CANYON POWER PLANT - UNIT 1

12.0 Test Procedure No. 42.3 - Static Rod Drop and RCCA Below Bank Position Measurements

TEST OBJECTIVE

The objective of this test was to determine the power distribution associated with a dropped rod configuration.

TEST DESCRIPTION

Initial conditions were established as follows:

The test was performed at the 50% power plateau with the plant stable and all RCCAs at or nearly at full withdrawn position.

After verifying these conditions, the movable incore detectors were used to perform a full-core flux map to determine the reference, or "pre-drop", power distribution.

RCCA H-4 was inserted in steps (H-4 was chosen because this RCCA was predicted to cause the most severe change in power distribution) maintaining reactor power and temperature nearly constant through boron dilution. Rod motion was suspended for RCCA H-4 at positions 190, 150, 100, and 50 steps while partial flux maps (i.e., data from 6 of 58 incore flux thimble locations) were taken.

Rod insertion was completed in approximately 45 minutes, leaving a configuration with RCCA H-4 fully inserted and all other rods withdrawn. A full-core post-drop flux map was immediately performed, after which RCCA H-4 was returned to the top of the core. During the rod withdrawal, reactivity changes were balanced with boron addition. The entire RCCA H-4 insertion/withdrawal process was completed in less than four hours.

TEST RESULTS

All Acceptance Criteria were met and no significant problems were encountered during the performance of this test.

The pre- and post-drop nuclear enthalpy rise hot channel factors ($F^N_{\Delta H}$) occurred in the same assembly (M-12). The value increased from 1.391 to 1.652, below the Acceptance Criteria of 1.660. The results are summarized in Table 11 and the core-average radial power distributions are shown in Figure 7. The negative reactivity associated with the rod created a flux tilt across the core, as shown by the quadrant tilt map in Table 11. Due to the symmetric location of RCCA H-4 (see Figure 1), the upper two quadrants had almost identical decreases in relative flux and power levels; the lower quadrants indicated corresponding increases in flux and power.

"Dropped" rod worth was 97 PCM, a result based on reactivity computer data during the rod insertion. However, this value was a rough estimate because strong doppler feedback at 50% power led to poor resolution of the reactivity computer data.

Table 11

<u>Power Distribution Results - Pre and Post Rod Drop</u>

ITEM	PRE-DROP FLUX MAP	POST-DROP FLUX MAP
Conditions - power - temperature (RCS) - boron concentration - burnup	50 % ~562 deg. F 1020 ppm 220 MWD/MT	50% ~562 deg. F 997 ppm 220 MWD/MT
Date	12-22-84	12-23-84
Rod Configuration	ARO	RCCA H-4 @ 0 steps
$F^N_{\Delta H}$ - measured value - location *	1.391 M12-IH	1.652 M12-JH
F^T_Q - measured value - location *	2.033 M12-IH @ 74 in	2.409 M12-JH @ 74 in
F_Z - measured value	1.359	1.356
QUADRANT TILT - measured value	1.006	1.137

*Assembly location (i.e., M12) as shown in Figure <u>1</u>.
Pin location within assembly (i.e., IH) based on 17x17 matrix ranging from AA to QQ.

FIGURE 7

POWER DISTRIBUTIONS FOR ROD DROP (T.P. 42.3)

PRE-DROP ASSEMBLY POWER
POST-DROP ASSEMBLY POWER

NOTE: a) Box entries are relative assembly powers.
b) Drop location: B-4

	R	P	N	M	L	K	J	H	G	F	E	D	C	B	A
1					.586 .464	.686 .511	.812 .579	.741 .519	.812 .616	.683 .567	.561 .502				
2			.517 .493	.692 .810	1.020 .836	1.013 .740	1.036 .719	1.026 .653	1.026 .727	.993 .754	.997 .824	.859 .711	.503 .453		
3		.511 .490	1.056 .970	.962 .897	1.114 .908	1.105 .850	1.154 .725	1.110 .623	1.143 .708	1.063 .775	1.081 .854	.940 .826	1.040 .977	.509 .535	
4		.879 .836	.960 .926	1.271 1.106	1.059 .873	1.167 .842	1.132 .669	1.195 .408	1.136 .674	1.164 .838	1.047 .879	1.269 1.101	.952 .935	.875 .870	
5	.589 .621	1.014 1.040	1.108 1.068	1.059 1.000	1.173 .985	1.138 .911	1.198 .819	1.154 .716	1.198 .810	1.136 .898	1.158 .970	1.061 .984	1.103 1.032	1.010 .990	.576 .558
6	.681 .765	1.003 1.045	1.077 1.107	1.156 1.110	1.125 1.057	1.149 .981	1.044 .866	1.122 .862	1.038 .852	1.138 .963	1.120 1.034	1.167 1.098	1.091 1.067	1.004 .971	.679 .671
7	.798 .934	1.035 1.152	1.142 1.163	1.133 1.171	1.161 1.128	1.031 .991	1.076 .961	.987 .902	1.066 .951	1.016 .982	1.172 1.118	1.139 1.156	1.155 1.151	1.038 1.074	.815 .828
8	.721 .846	1.022 1.119	1.099 1.200	1.183 1.219	1.107 1.169	1.103 1.084	.979 .986	1.037 1.013	.965 .983	1.094 1.081	1.127 1.151	1.191 1.204	1.118 1.176	1.034 1.065	.746 .798
9	.790 .930	1.027 1.186	1.142 1.232	1.136 1.225	1.153 1.210	1.015 1.096	1.057 1.092	.968 1.041	1.059 1.094	1.023 1.090	1.179 1.210	1.148 1.212	1.165 1.212	1.042 1.136	.817 .861
10	.683 .789	1.007 1.140	1.071 1.247	1.160 1.263	1.119 1.258	1.127 1.229	1.015 1.136	1.092 1.178	1.023 1.141	1.146 1.223	1.143 1.253	1.184 1.259	1.114 1.218	1.022 1.129	.692 .780
11	.594 .696	1.034 1.232	1.129 1.322	1.054 1.199	1.158 1.290	1.120 1.285	1.160 1.290	1.113 1.265	1.156 1.277	1.126 1.267	1.179 1.292	1.074 1.212	1.112 1.199	1.039 1.210	.597 .6[illegible]4
12		.880 1.027	.954 1.159	1.290 1.445	1.071 1.235	1.179 .1321	1.131 1.315	1.176 1.318	1.126 1.298	1.167 1.302	1.070 1.225	1.182 1.404	.96[illegible] 1.153	.899 1.057	
13		.507 .624	1.047 1.259	.965 1.171	1.117 1.243	1.094 1.279	1.130 1.304	1.088 1.283	1.137 1.305	1.088 1.286	1.112 1.279	.960 1.118	1.060 1.214	.522 .646	
14			.507 .630	.877 1.030	1.023 1.184	1.007 1.145	1.022 1.253	1.017 1.203	1.034 1.272	1.013 1.184	1.024 1.226	.887 1.026	.518 .625		
15					.578 .687	.673 .601	.796 .983	.727 .910	.809 1.003	.686 .820	.588 .696				

NORTH

DIABLO CANYON POWER PLANT - UNIT I

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

TEST OBJECTIVE

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

TEST DESCRIPTION

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

TEST RESULTS

The two rods dropping simultaneously caused a negative rate trip which caused a Reactor trip and a Turbine trip. The acceptance criteria for the test was met as the transient did not cause (i) a safety injection (ii) reactor coolant pump tripping (iii) steam line safety valve lifting or (iv) pressurizer safety valve lifting. The plant responded as expected to the plant trip with the exception of pressurizer level and Tavg both of which went below expected values due to the Auxiliary Steam demand. A summary of selected parameter response to the transient is shown in Table 12.

Table 12

Rod Group Drop and Plant Trip

Parameter	Units	Initial	TRANSIENT		Final
			Max.	Min.	
Reactor Power	%	49	-	-	0
Electrical Output (Gross)	MW	491	-	-	0
Tref	deg. F	560	-	-	549
Tavg	deg. F	559	-	-	535
Pressurizer Pressure	psig	2250	2231	2047	2080
Pressurizer Level	%	39	35	15.2	18
Steam Header Pressure	psig	831	906	831	856
(Loop1) Steam Generator Level	%	45	44.5	1.3	35
Core Exit Thermocouple (F5)	deg. F	580	–	–	528

Pressurizer level and Tavg did not meet expected values of ≥20% and 547 deg. F respectively. The reason for this was the continued use of auxiliary steam. Pressurizer level dropped with Tavg and went below 20% about 80 sec. after trip. Tavg dropped below 547 deg. F about 17 secs. after trip and settled around 535 deg. F and MSIVs had to be closed to raise Tavg.

Feedwater isolation occurred about 8 secs. after trip.

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

TEST OBJECTIVE

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

TEST DESCRIPTION

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

TEST RESULTS

The test was performed on January 5, 1985, with the unit operating at approximately 50% power. Following a reactor trip from outside the Main Control Room (ref. T.P. 43.5), a minimum Operations Test crew consisting of six members evacuated the Main Control Room, assumed control of the plant from the HSDP and manned other stations to monitor and control plant parameters in accordance with Operating Procedure OP AP-8 "Control Room Inaccessibility."

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

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

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

(1) MSIVs were closed to isolate the steam losses on the secondary side thereby, preventing any further decreases in Tavg. The plant staff will update Operating Procedure AP-8 to instruct the operating crew to close the MSIVs prior to leaving the Control Room.

(2) The test crew operator could not initiate letdown by manually actuating a relay because the instrument rack for the relay was locked. The plant staff will update AP-8 to instruct the operating crew to take the keys with them prior to leaving the Control Room.

(3) Containment High Pressure Alarm actuated. Control Room Operators placed the Containment Fan Cooler (CFCU) in high speed. Due to a thermal overload problem, the CFCUs will not automatically restart in high speed after an auto transfer. A Design Change has been initiated to resolve this problem.

(4) A safety valve lifted on Main Steam Lead 1-2. The Control Room Operators lowered the pot settings on the 10% steam dumps to reseat the safety valve. Manipulation of the 10% steam dumps from the Control Room was performed so that Operations could determine the correct pot setting to prevent the safety from lifting. Operations will update AP-8 to instruct the operating crew to set the correct pot settings on the 10% Steam Dump Controllers prior to leaving the Control Room.

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

15.0 Test Procedure No. 43.1 - Load Swing Tests

TEST OBJECTIVE

The objective of this test was to verify nuclear plant dynamic response, including automatic control system performance, to 10% step load changes introduced at the Turbine Generator at each of the power test plateaus.

TEST DESCRIPTION

The plant was stabilized and on automatic control at the specified power. The plant output was reduced by 10% at the maximum rate (2200 MWe/min.). During the load decrease and until plant conditions stabilize, the following parameters were recorded on Chart Recorders:

Nuclear power
Controlling Bank position
Reactor Coolant temperatures
Steam flow and pressure
Feedwater flow
Steam Generator level
Pressurizer pressure and level

The plant output was then increased by 10% at the maximum rate (2200 MWe/min.) to the previous power, again recording the above parameters. The plant parameters are evaluated for acceptable dynamic response.

TEST RESULTS

The test was performed at the 30% power plateau on December 19, 1984 and at the 50% power plateau on December 31, 1984. On both occasions, all Acceptance Criteria were met satisfactorily as follows:

Reactor and Turbine did not trip
Safety Injection did not initiate
Main Steam Safety Valves did not lift
Pressurizer Relief Valves or Safety Valves did not lift
Manual intervention was not necessary to bring the plant to steady state conditions
Nuclear power under and overshoot was less than 3%

For the load swing from 40 to 50% power it was necessary to run both feedwater pumps in order to meet the "no manual intervention" criteria. In each run, Steam Generator level variation was within the expected value of ±10%, however the expected Pressurizer pressure swings of less than 50 psi was not met for the following load swings:

30% - 20% (56.4 psi)
20% - 30% (54.0 psi)
50% - 40% (58.5 psi)

The expected steam pressure overshoot or undershoot of $\pm$25 psi from the final value was not met for the following load swings:

30% - 20% (32.5 psi: overshoot)
50% - 40% (32.5 psi: overshoot)
40% - 50% (36.2 psi: overshoot)

These deviations from expected values were evaluated by Westinghouse both at 30% and 50% plateaus. Westinghouse identified Control System setting changes for fine tuning the systems to be implemented prior to load swing tests at 75% power.

The test will also be performed at 75% and 100% (only -10% swing) power levels.

16.0 Test Procedure 42.1 - Doppler Power Reactivity Coefficient Measurement

OBJECTIVE

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

TEST DESCRIPTION

At each test plateau, 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 (2%) at a rate of 2200 MWe/min (200% per 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 in Figure 8. The period between each load swing was long enough to allow stabilization of Tavg and ΔT. Boron concentration and control rod position were maintained constant throughout the test.

For each increase in load, ΔT increases and Tavg decreases. The increased ΔT causes a negative reactivity effect due to the fuel's doppler coefficient, which is offset by the positive reactivity due to isothermal temperature coefficient (ITC) effect on decreased Tavg.

In a similar manner, load decreases involve a decrease in ΔT and an associated increase in Tavg.

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

TEST RESULTS

At the 30% and 50% power test plateaus, the ratios of doppler coefficient to ITC were within acceptance criteria. The inferred doppler coefficients were -11.84 pcm/% power at 30% power and -10.94 pcm/% power at 50% 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 absorption peaks.

This test will be repeated at the 75% and 90% test plateaus.

Table 13

Doppler Coefficient Measurements

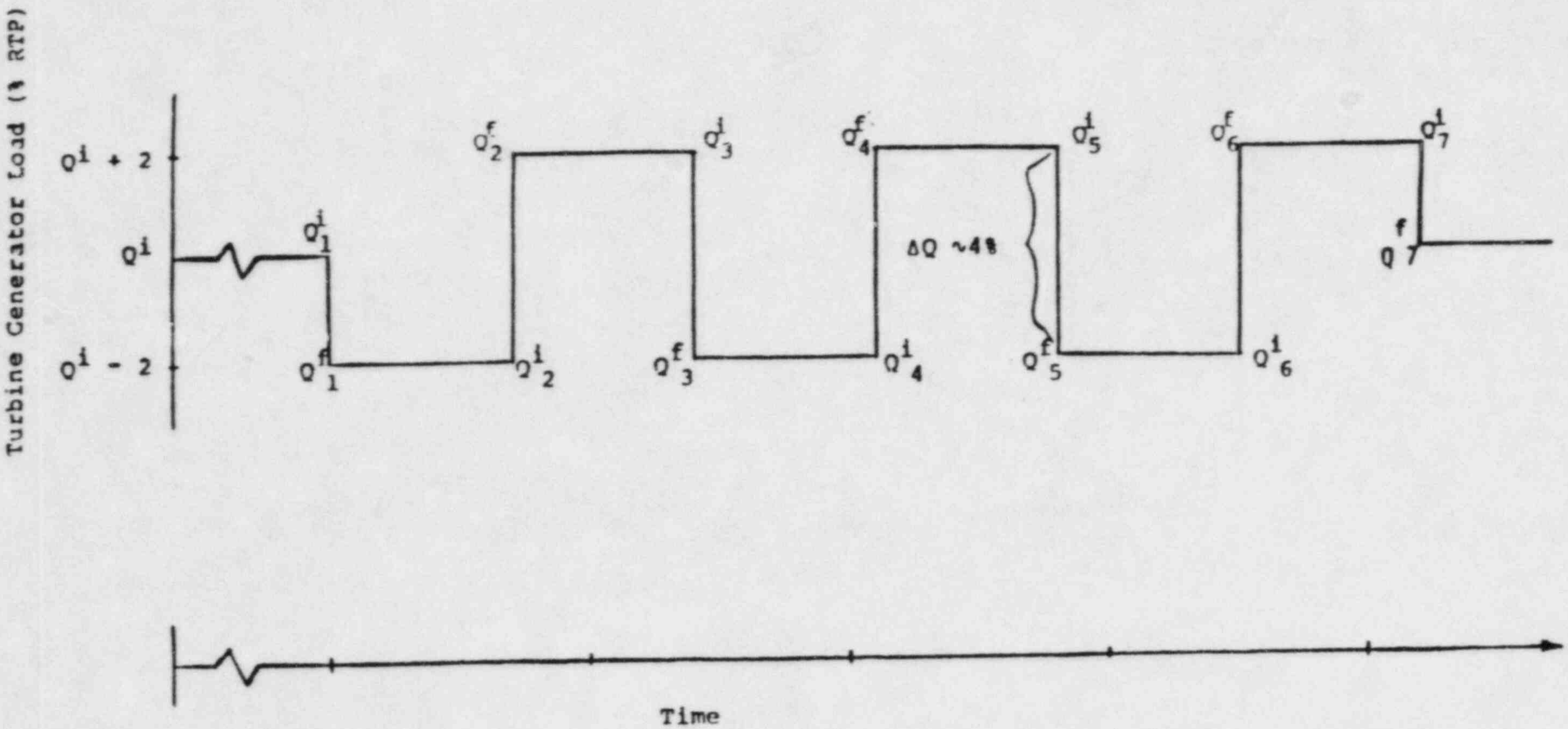
TEST PLATEAU (% RTP) (NOMINAL)	MEASURED RATIO* (deg.F/% power)	DESIGN RATIO* (deg.F/% power)	INFERRED DOPPLER COEFFICIENT (pcm/% power)	DESIGN DOPPLER COEFFICIENT (pcm/% power)
30	2.83	3.22	-11.84	-13.5
50	2.03	2.40	-10.94	-12.8

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

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



17.0 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 will be 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.

TEST RESULTS (50% Power Plateau)

Control Bank D rods were inserted in two (2) increments of approximately twenty (20) steps each. Incore AO shifted from about -3% initially (Bank D at 208 steps to about -9% (188 steps) and then to about -17% (168 steps). Upon holding the latter configuration for approximately two hours, rods were returned to their original position. During the subsequent axial xenon oscillation, five more quarter-core flux maps were produced at incore AO's ranging from about -7.8% to +2%. Thus the total span of minimum to maximum AO was from about -17% at about +2%

The entire test required approximately 27 hours to complete.

Slopes (gains) of the incore vs. excore AO plots were approximately 1.6 for all channels. Offsets ranged from about -0.3% for channel N44 to +6.4% 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).

Completion of the I&C calibrations per STP I-2D took approximately four (4) days. Rather than performing only the applicable gain adjustments a full calibration was performed on all power range channels.

This test will be performed again at the 75% power level.

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

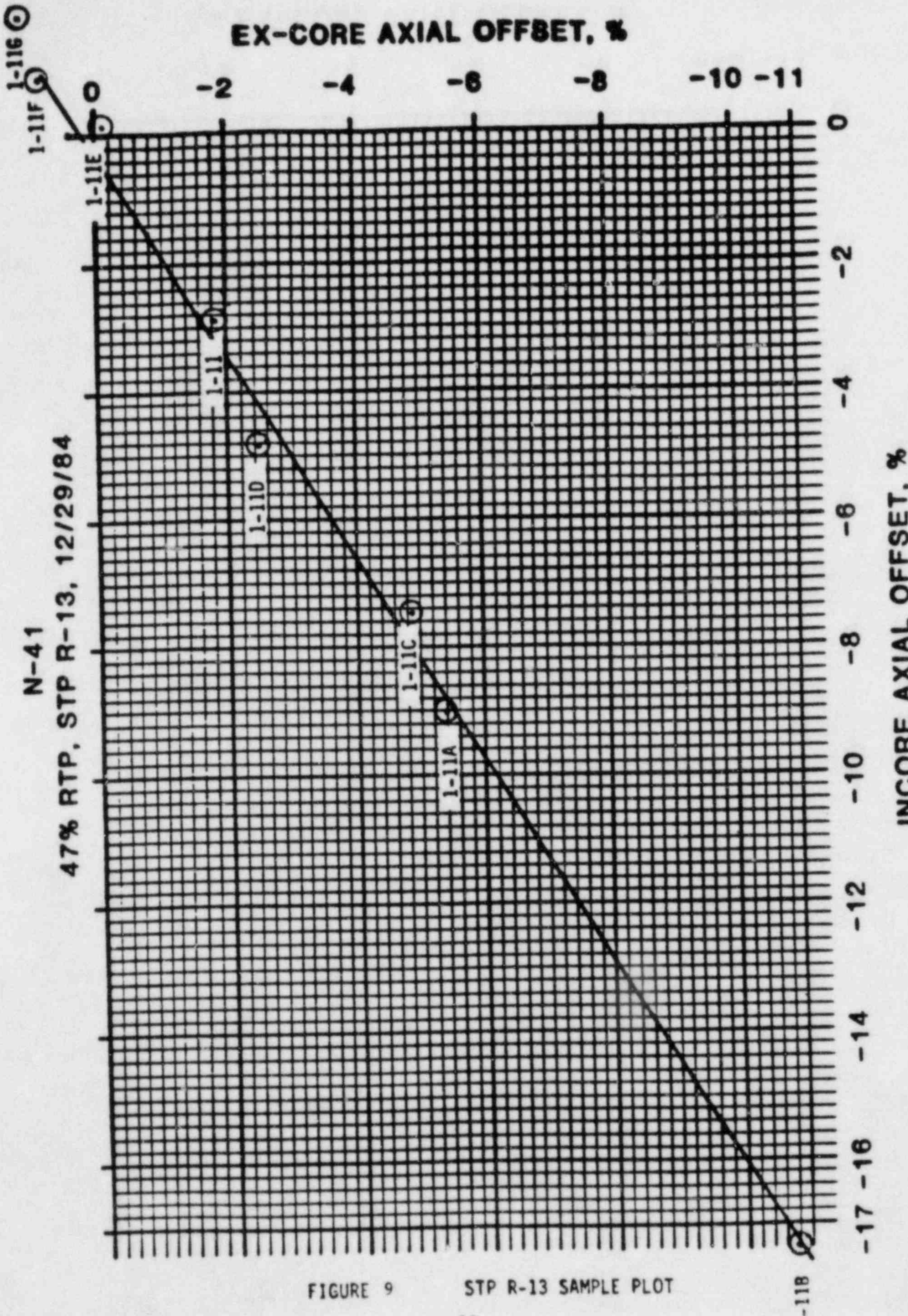


FIGURE 9 STP R-13 SAMPLE PLOT

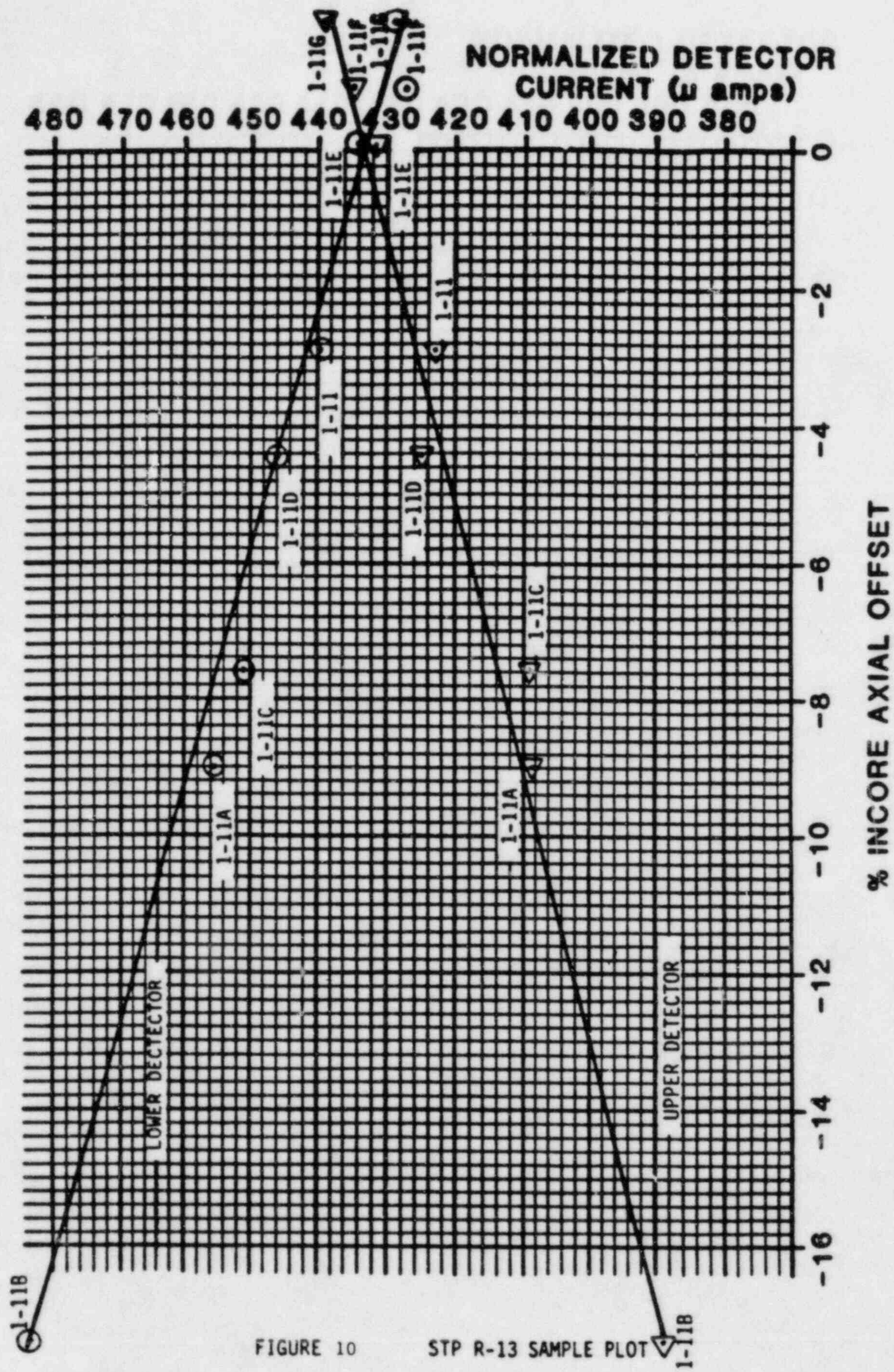


FIGURE 10 STP R-13 SAMPLE PLOT

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

TEST OBJECTIVE

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

TEST DESCRIPTION

The plant was stable at nominal 50% RTP and on automatic control. The load rejection was initiated by opening the main transformer high side breakers. The plant was then stabilized using Nuclear Plant Operations (NPO) Emergency Operating Procedure OP-AP-2 - Full Load Rejection. During the transient, various plant parameters were monitored for analysis of the plant response to the transient.

TEST RESULTS

The first attempt of the test resulted in a reactor trip caused by low RCS flow. This trip was caused by a solenoid valve in the Turbine Control System that stuck in the open position not allowing the Turbine Governor Valves to reopen. This resulted in a drop in turbine speed, generator frequency and hence in RCS flow. The solenoid valve was replaced and tested successfully before the test was repeated.

The second attempt was successful with the unit sustaining the transient. The rod control was taken on manual at about 20% so that the Steam Generator levels could be controlled by the Main Feedwater Regulating Valves. (The bypass valves do not have automatic controls.) Response of selected parameters is tabulated in Table 14. The acceptance criteria were met and the plant response was reviewed by engineering and Westinghouse. No additional setpoint changes were recommended. (Changes were recommended by Westinghouse after reviewing Load Swing Tests at 30% and 50% power and the Rod Group Drop and Plant Trip Test at 50% power. These changes will be incorpor ated prior to performing the Load Swing Test at 75% power).

Table 14

Plant Response to Net Load Trip From 50% Power

Parameter	Units	INITIAL	DURING TRANSIENT		FINAL
			Minimum	Maximum	
Reactor Power	%	47.8	23	50	27
Electrical Output	MWe	478	-	-	47
Tref	deg. F	561	-	-	550
Tavg	deg. F	559	552	562.5	556
ΔT	deg. F	30	14	30	14
Pressurizer Pressure	psig	2250	2184	2268	2250
Pressurizer Level	%	38	33.5	40	30
Steam Header Pressure	psig	839.2	837.5	931.2	932
S.G. Level	%	43	27.5	53	31
FW Heater 1-1A Outlet Temperature	deg. F	363.6	233.6	363.6	237
Control Bank D Position	steps	182	-	-	120

19.0 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 30% and 50% 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 was then obtained during a nominal 30 minute period with plant conditions stabilized and plant load constant.

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

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

TEST RESULTS

30% Full Power Test

Total RCS flow was measured as 374,500 gpm which is approximately 3% more than required by Technical Specifications. No recalibrations of loop flow meters were required; however loop flow constants (in STP I-1A Surveillance Test Procedure done each shift by the operators) were updated.

50% Full Power Test

Total RCS flow was measured as 376,100 gpm which is 3 1/2% more than required by Technical Specifications. One loop flow meter indicator required recalibration. Since the flow change was in the conservative direction, no changes were required for the Surveillance Test Procedure flow constants mentioned above.

This test will be repeated at 75, 90 and 100% power levels.

PACIFIC GAS AND ELECTRIC COMPANY

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JAMES D. SHIFFER
VICE PRESIDENT
NUCLEAR POWER GENERATION

April 29, 1985

PGandE Letter No.: DCL-85-174

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 1 to Startup Report

Dear Mr. Knighton:

As required by the Operating License for Unit 1 (Section 6.9 of the Technical Specifications), Supplement 1 to the Startup Report for the period from November 1, 1984, to January 31, 1985, 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,

for J. D. Shiffer

Enclosure

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

IE26
1/1

ENCLOSURE