Duke Power Company
Catawba Nuclear Generation Department
4800 Concord Road
York, SC 29745

WILLIAM R. MCCOLLUM, JR.
Vice President
(803)831-3200 Office
(803)831-3426 Fax

DUKE POWER

December 18, 1996

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555

Subject: Catawba Nuclear Station, Unit 1
Docket No. 50-413
Startup Report, Unit 1 Cycle 10

In accordance with Section 6.9.1 of the Catawba Nuclear Station Technical Specifications, find attached the Unit 1 Startup Report for Cycle 10 core design.

Any questions concerning this report may be directed to Kay Nicholson at (803) 831-3237.

Sincerely,

W. R. McCollum, Jr.

KEN/U1C10.SR

Attachment

xc: S. D. Ebneter
Regional Administrator, Region II

P. S. Tam, ONRR

R. J. Freudenberger, SRI

IE26 1/1

300041

Duke Power Company
Catawba Nuclear Station
Unit 1 Cycle 10

STARTUP REPORT

December 1996

TABLE OF CONTENTS

LIST OF TABLES

LIST OF FIGURES

1.0 INTRODUCTION

Catawba Unit One Cycle 10 includes a feed batch of 72 MkBW fuel assemblies manufactured by Framatome Cogerna Fuels (FCF). The feed batch enrichments are 32 F/A's at 3.8% (w/o), and 40 F/A's at 4.09% (w/o). Burnable poison rod assemblies used in the feed batch were also manufactured by FCF.

Catawba Unit One Cycle 10 core loading began at 0345 on September 7, 1996 and ended at 0950 on September 11, 1996. Initial criticality for Cycle 10 occurred at 0707 on October 2, 1996. Zero Power Physics Testing was completed at 1422 on October 3, 1996. The unit reached full power at 2109 on October 10, 1996. Power Escalation testing, including testing at full power, was completed by November 11, 1996.

Table 1 contains some important characteristics of the Catawba 1 Cycle 10 core design.

TABLE 1
C1C10 CORE DESIGN DATA

1. C1C9 end of cycle burnup: 426 EFPD

2. C1C10 design length: 410 ± 10 EFPD

Region	Fuel Type	Number of Assemblies	Enrichment, w/o U^{235}	Loading, MTU**	Cycles Burned
8C	MkBW	8	3.55	3.6496	3
9C	MkBW	5	3.45	2.2810	2
9D	MkBW	4	3.45	1.8248	3
10B	MkBW	28	3.65	12.7736	2
11A	MkBW	76	3.86	34.6712	1
12B	MkBW	32	3.80/2.00*	14.5984	0
12A	MkBW	40	4.09/2.00*	18.2480	0
Totals		193		88.0466	

* 2.00 w/o enriched U blanketed fuel assemblies (6 inches top and bottom)
** Design MTU loadings which were used in all design calculations.

2.0 PRECRITICAL TESTING

Precritical testing includes:

- Core Loading
- Preliminary Calibration of Nuclear Instrumentation
- Dilution of Reactor Coolant System to Estimated Critical Boron concentration
- Rod Drop Timing Test
- Control Rod Drag Test

Sections 2.1 through 2.5 describe results of precritical testing for Catawba 1 Cycle 10.

2.1 Total Core Reloading

The Cycle 10 core was loaded under the direction of PT/0/A/4150/22, Total Core Reloading. Plots of Inverse Count Rate Ratio (ICRR) versus number of fuel assemblies loaded were maintained for each applicable source range and boron dilution mitigation system (BDMS) channel.

Core loading commenced at 0345 on September 7, 1996 and concluded at 0950 on September 11, 1996. Core loading was verified by PT/0/A/4550/03C, Core Verification, which was completed by 0300 on September 12, 1996.

Figure 1 shows the core loading pattern for Catawba 1 Cycle 10.

2.2 Preliminary NIS Calibration

Periodic test procedure PT/0/A/4600/05E, Preliminary NIS Calibration, is performed before initial criticality for each new fuel cycle. Intermediate range reactor trip and rod stop setpoints are adjusted using measured power distribution from the previous fuel cycle and predicted power distribution for the upcoming fuel cycle. Power Range NIS full power currents are similarly adjusted. Intermediate Range NIS Rod Stop and Rx Trip setpoints are checked and revised as necessary for initial power ascension. An added conservatism of 5% was applied to I/R setpoints to account for any uncertainties that may have been introduced by the T-AVG reduction resulting from replacement of the Steam Generators during the outage. The effectiveness of this measure is discussed in Section 4.10.

Table 4 shows the calibration data calculated by PT/0/A/4600/05E. Calculations were performed on September 17, 1996. Calibrations were complete by September 29, 1996.

2.3 Reactor Coolant System Dilution

The reactor coolant system boron concentration was diluted from the refueling boron concentration to the estimated critical boron concentration per PT/0/A/4150/19, 1/M Approach to Criticality. Inverse Count Rate Ratio (ICRR) was plotted versus gallons of demineralized water added.

Initial reactor coolant boron concentration was 2598 ppmB. The estimated critical boron concentration was calculated to be 1828 ppmB. The calculated volume of demineralized water required was 23658 gallons. This change in boron concentration was expected to decrease ICRR from 1.0 to 0.35.

Reactor coolant system dilution at 87 GPM began at 1213 on September 30, 1996 and concluded at 1647 on September 30, 1996. The final reactor coolant system boron concentration, after allowing system to mix, was 1851 ppmB. Figure 2 shows ICRR versus volume of water used.

2.4 Control Rod Drag Test

This testing was performed to assess mechanical binding between RCCA's and their resident Fuel Assemblies in accordance with NRC Bulletin 96-01. EOC9 Drag Testing was satisfactorily performed on all 53 RCCAs in the Spent Fuel Pool following total core offload, however, C1C10 Core Design utilizes 13 reinsert Fuel Assemblies from the SFP (11 of which are in rodded core locations. This necessitated additional Drag Testing to assure the reinserted F/As afforded smooth movement of the RCCAs within them. The test methodology involved the withdrawal and insertion of the 11 affected RCCAs via the Reactor Building Manipulator Crane's auxiliary hoist as load cell indications were monitored to ensure specified weight limits were not exceeded. This testing was performed following C1C10 Core Loading, with the Upper Internals inplace, and all RCCAs latched to their Drive Rods.

The first three selected RCCAs were raised and lowered (by their Drive Rods) to determine average weights at 2 feet (to establish Insertion Drag Load criteria) and at 10 feet (to establish Withdrawal Drag Load criteria). The results of these measurements established acceptance criteria for RCCA withdrawal at 958 ± 40 lbs; and for RCCA insertion at 915 ± 100 lbs.

The C1C10 RCCA Drag Testing performed on September 12, 1996 satisfied all acceptance criteria. Results are summarized in Table 2 below:

TABLE 2
RCCA DRAG TEST RESULTS

RCCA CORE LOCATION	WITHDRAWAL DATA Maximum Dashpot Weight	INSERTION DATA Minimum Dashpot Weight
D-4	960 lbs	916 lbs
D-12	960 lbs	912 lbs
F-6	960 lbs	914 lbs
F-10	960 lbs	914 lbs
H-6	958 lbs	914 lbs
H-8	958 lbs	914 lbs
H-10	958 lbs	910 lbs
K-6	958 lbs	918 lbs
K-10	960 lbs	914 lbs
M-4	960 lbs	916 lbs
M-12	956 lbs	914 lbs

2.5 Control Rod Drop Timing Test

This testing is performed prior to each post-refueling startup to verify that, when dropped from the fully withdrawn position at Hot, No-load conditions, each RCCA completely inserts and that it's drop time is ≤ 2.2 seconds (pursuant to Tech Spec 3.1.3.4). The 2.2 second criterion applies to the time measured from beginning of decay of Stationary Gripper coil voltage to Dashpot entry.

All BOC10 RCCA drop times satisfied the acceptance criterion. Table 3 summarizes not only the BOC10 data, but, for comparison purposes, the BOC9 and EOC9 drop times as well. It should be noted that "Time to DP" is the data to be compared to the 2.2 second criterion. "Time in DP" is a parameter that is measured for the purposes of assessing resistance to the RCCA in the Dash Pot region, which was at one time poatulated to be the culprit in increasing drop times industry wide. "Total Time" is merely the sum of these two measures. The burnup of the fuel assembly into which the RCCA was dropped is recorded for the purpose of assessing of the impact on drop time of potential exposure related mechanical deformation of the fuel assembly.

TABLE 3
CYCLE 9 AND CYCLE 10 ROD DROP TIMING RESULTS

Catawba 1 Rod Drop Test Results															
	BOC-9 3/21/95				EOC9 6/14/96				EOC-BOC			BOC-10 9/29/96			
Core Loc	Time to DP	Time in DP	Total Time	Burnup MWD/MTU	Time to DP	Time in DP	Total Time	Burnup MWD/MTU	Time to DP	Time in DP	Total Time	Time to DP	Time in DP	Total Time	Burnup MWD/MTU
H08	1.600	0.541	2.141	26000	1.743	0.639	2.382	42710	0.143	0.098	0.241	1.570	0.600	2.170	30017
H10	1.529	0.501	2.030	25900	1.626	0.542	2.168	42260	0.097	0.041	0.138	1.560	0.610	2.170	30076
F08	1.640	0.631	2.271	25800	1.699	0.656	2.355	42280	0.059	0.025	0.084	1.580	0.630	2.210	29654
K08	1.562	0.542	2.104	25800	1.611	0.563	2.174	42490	0.049	0.021	0.070	1.560	0.630	2.220	30011
H02	1.559	0.544	2.103	19100	1.538	0.568	2.106	35080	-0.021	0.024	0.003	1.570	0.650	2.220	12241
B08	1.527	0.506	2.033	19500	1.547	0.483	2.030	35270	0.020	-0.023	-0.003	1.560	0.630	2.190	12255
H14	1.529	0.565	2.094	19800	1.556	0.539	2.095	35350	0.027	-0.026	0.001	1.580	0.610	2.190	12410
P08	1.603	0.604	2.207	19400	1.566	0.564	2.130	35220	-0.037	-0.040	-0.077	1.570	0.620	2.190	12227
F06	1.529	0.532	2.061	20100	1.632	0.559	2.191	39450	0.103	0.027	0.130	1.560	0.610	2.170	30943
F10	1.525	0.528	2.053	20000	1.570	0.542	2.112	38080	0.045	0.014	0.059	1.580	0.630	2.210	30252
K10	1.632	0.579	2.211	19900	1.600	0.577	2.177	38880	-0.032	-0.002	-0.034	1.570	0.630	2.200	30423
K06	1.516	0.531	2.047	19800	1.597	0.556	2.153	39360	0.081	0.025	0.106	1.580	0.630	2.210	30495
D02	1.714	0.603	2.317	23800	1.641	0.577	2.218	32810	-0.073	-0.026	-0.099	1.630	0.640	2.270	18464
B12	1.646	0.593	2.239	23400	1.585	0.574	2.159	32440	-0.061	-0.019	-0.080	1.570	0.610	2.180	18125
M14	1.735	0.621	2.356	23600	1.652	0.592	2.244	32690	-0.083	-0.029	-0.112	1.660	0.640	2.300	18304
P04	1.639	0.604	2.243	23200	1.562	0.559	2.121	32290	-0.077	-0.045	-0.122	1.580	0.610	2.190	18381
B04	1.589	0.520	2.109	23000	1.599	0.469	2.068	32040	0.010	-0.051	-0.041	1.600	0.640	2.240	18395
D14	1.666	0.536	2.202	23700	1.693	0.557	2.250	32640	0.027	0.021	0.048	1.660	0.650	2.310	18332
P12	1.600	0.563	2.163	23300	1.562	0.542	2.104	32230	-0.038	-0.021	-0.059	1.610	0.630	2.240	18398
M02	1.600	0.560	2.160	23500	1.617	0.542	2.159	32550	0.017	-0.018	-0.001	1.630	0.630	2.260	18261
E03	1.557	0.526	2.083	17000	1.579	0.542	2.121	35220	0.022	0.016	0.038	1.590	0.630	2.220	21550
C11	1.538	0.535	2.073	16500	1.553	0.510	2.063	34820	0.015	-0.025	-0.010	1.550	0.620	2.170	21311
L13	1.567	0.508	2.075	16800	1.553	0.510	2.063	35050	-0.014	0.002	-0.012	1.590	0.600	2.190	21832
N05	1.533	0.519	2.052	17100	1.541	0.542	2.083	35470	0.008	0.023	0.031	1.590	0.610	2.200	21549
C05	1.557	0.533	2.090	17100	1.547	0.542	2.089	35410	-0.010	0.009	-0.001	1.570	0.640	2.210	21748
E13	1.568	0.534	2.102	17100	1.576	0.560	2.136	35440	0.008	0.026	0.034	1.580	0.610	2.190	21544
N11	1.546	0.517	2.063	17000	1.535	0.525	2.060	35090	-0.011	0.008	-0.003	1.580	0.590	2.170	21443
L03	1.639	0.595	2.234	16900	1.526	0.642	2.168	35360	-0.113	0.047	-0.066	1.580	0.610	2.190	21300
H04	1.507	0.533	2.040	27100	1.559	0.556	2.115	44430	0.052	0.023	0.075	1.570	0.680	2.250	21606
D08	1.546	0.524	2.070	26700	1.626	0.560	2.186	43910	0.080	0.036	0.116	1.590	0.630	2.220	21566
H12	1.542	0.516	2.058	27100	1.614	0.560	2.174	44030	0.072	0.044	0.116	1.580	0.620	2.200	21127
M08	1.542	0.515	2.057	26800	1.562	0.509	2.071	43870	0.020	-0.006	0.014	1.610	0.630	2.240	21872
F02	1.635	0.614	2.249	18100	1.579	0.592	2.171	32720	-0.056	-0.022	-0.078	1.640	0.680	2.320	21340
B10	1.514	0.555	2.069	16500	1.544	0.524	2.068	33180	0.030	-0.031	-0.001	1.580	0.610	2.190	21079
K14	1.576	0.514	2.090	18100	1.597	0.489	2.086	32830	0.021	-0.025	-0.004	1.640	0.640	2.280	20853
P06	1.541	0.554	2.095	16200	1.553	0.559	2.112	32910	0.012	0.005	0.017	1.590	0.630	2.220	21446
B06	1.520	0.533	2.053	15900	1.559	0.512	2.071	32520	0.039	-0.021	0.018	1.600	0.670	2.270	21090
F14	1.600	0.561	2.161	16100	1.629	0.574	2.203	32950	0.029	0.013	0.042	1.640	0.630	2.270	21076
P10	1.554	0.518	2.072	16200	1.567	0.510	2.077	32790	0.013	-0.008	0.005	1.590	0.630	2.220	21013
K02	1.602	0.548	2.150	16100	1.579	0.527	2.106	33000	-0.023	-0.021	-0.044	1.640	0.680	2.320	21448
D04	1.540	0.531	2.071	19900	1.550	0.542	2.092	37650	0.010	0.011	0.021	1.590	0.630	2.220	30332
M12	1.552	0.534	2.086	19900	1.547	0.542	2.089	37530	-0.005	0.008	0.003	1.560	0.620	2.180	30693
D12	1.562	0.545	2.107	20000	1.582	0.542	2.124	37830	0.020	-0.003	0.017	1.580	0.630	2.210	30645
M04	1.540	0.490	2.030	19700	1.564	0.490	2.054	37540	0.024	0.000	0.024	1.570	0.620	2.190	30019
H06	1.649	0.596	2.245	30200	1.702	0.592	2.294	47940	0.053	-0.004	0.049	1.680	0.610	2.290	31040
G03	1.628	0.542	2.170	16800	1.564	0.542	2.106	36310	-0.064	0.000	-0.064	1.580	0.630	2.210	21160
C09	1.530	0.504	2.034	16700	1.544	0.492	2.036	36020	0.014	-0.012	0.002	1.610	0.640	2.250	20934
J13	1.511	0.535	2.046	16600	1.518	0.521	2.039	35900	0.007	-0.014	-0.007	1.580	0.610	2.190	21049
N07	1.530	0.504	2.034	16600	1.503	0.521	2.024	36130	-0.027	0.017	-0.010	1.580	0.590	2.170	20950
C07	1.550	0.525	2.075	16300	1.582	0.539	2.121	35670	0.032	0.014	0.046	1.560	0.590	2.150	20616
G13	1.580	0.516	2.096	16300	1.576	0.528	2.104	35590	-0.004	0.012	0.008	1.610	0.620	2.230	20806
N09	1.556	0.508	2.064	16900	1.588	0.539	2.127	36210	0.032	0.031	0.063	1.580	0.630	2.210	21170
J03	1.510	0.482	1.992	16600	1.526	0.490	2.016	36160	0.016	0.008	0.024	1.550	0.610	2.160	21036
Average	1.573	0.543	2.116	20062	1.583	0.546	2.129	36445	0.011	0.003	0.013	1.59	0.63	2.22	22387

FIGURE 1
CORE LOADING PATTERN, CATAWBA UNIT 1 CYCLE 10

North
^
|

180 — FUEL TRANSFER CANAL ------>

90 (left side, row 8) — 270 (right side, row 8) — 0 (bottom, column H)

	R	P	N	M	L	K	J	H	G	F	E	D	C	B	A
1					H22 PD	AA21 PD	AA70 PD	J63 PD	AA34 PD	AA42 PD	H18 PD				
2			K70 PD	AA35 RCCA	AC03 BP	AA66 RCCA	AC15 BP	AA58 RCCA	AC1A BP	AA12 RCCA	AC2A BP	AA43 RCCA	K34 PD		
3		K69 PD	AA39 PD	AC2C BP	AA72 RCCA	AC02 BP	AA50 RCCA	K58 PD	AA13 RCCA	AC39 BP	AA49 RCCA	AC0A BP	AA25 PD	K20 PD	
4		AA28 RCCA	AC23 BP	K53 RCCA	AC24 BP	K46 PD	AC43 BP	AA32 RCCA	AC50 BP	K01 PD	AC20 BP	K05 RCCA	AC21 BP	AA46 RCCA	
5	H04 PD	AC14 BP	AA76 RCCA	AC19 BP	AA14 PD	AC53 BP	AA65 PD	AC63 BP	AA74 PD	AC64 BP	AA52 PD	AC01 BP	AA64 RCCA	AC29 BP	H16 PD
6	AA08 PD	AA75 RCCA	AC13 BP	K15 PD	AC42 BP	J58 RCCA	AC91 BP	K06 RCCA	AC45 BP	J09 RCCA	AC5A BP	K52 PD	AC10 BP	AA07 RCCA	AA63 PD
7	AA73 PD	AC34 BP	AA40 RCCA	AC5C BP	AA33 PD	AC55 BP	AA22 PD	AC56 BP	AA48 PD	AC4C BP	AA01 PD	AC52 BP	AA19 RCCA	AC32 BP	AA68 PD
8	J46 PD	AA53 RCCA	K68 SS	AA62 RCCA	AC41 BP	K40 RCCA	AC62 BP	J20 RCCA	AC4A BP	K59 RCCA	AC49 BP	AA02 RCCA	K51 SS	AA47 RCCA	J47 PD
9	AA51 PD	AC26 BP	AA60 RCCA	AC6C BP	AA26 PD	AC69 BP	AA05 PD	AC61 BP	AA11 PD	AC[illegible]2 BP	AA24 PD	AC60 BP	AA59 RCCA	AC40 BP	AA15 PD
10	AA44 PD	AA37 RCCA	AC25 BP	K04 PD	AC59 BP	J72 RCCA	AC66 BP	K25 RCCA	AC46 BP	J53 RCCA	AC90 BP	K73 PD	AC22 BP	AA16 RCCA	AA23 PD
11	H17 PD	AC31 BP	AA27 RCCA	AC0C BP	AA04 PD	AC44 BP	AA38 PD	AC51 BP	AA56 PD	AC65 BP	AA09 PD	AC36 BP	AA20 RCCA	AC09 BP	H05 PD
12		AA45 RCCA	AC16 BP	K48 RCCA	AC30 BP	K26 PD	AC54 BP	AA17 RCCA	AC6A BP	K45 PD	AC1C BP	K47 RCCA	AC33 BP	AA69 RCCA	
13		K76 PD	AA06 PD	AC35 BP	AA30 RCCA	AC3C BP	AA61 RCCA	K31 PD	AA18 RCCA	AC04 BP	AA54 RCCA	AC3A BP	AA67 PD	K28 PD	
14			K03 PD	AA71 RCCA	AC05 BP	AA29 RCCA	AC11 BP	AA31 RCCA	AC06 BP	AA10 RCCA	AC12 BP	AA41 RCCA	K36 PD		
15					H19 PD	AA55 PD	AA03 PD	J64 PD	AA36 PD	AA57 PD	H20 PD				

X##	Fuel Assembly Region Reference Number
XX	Fuel component (PD = plugging device, BP = burnable poison rod assembly, SS = secondary source)

TABLE 4
PRELIMINARY NIS CALIBRATION DATA

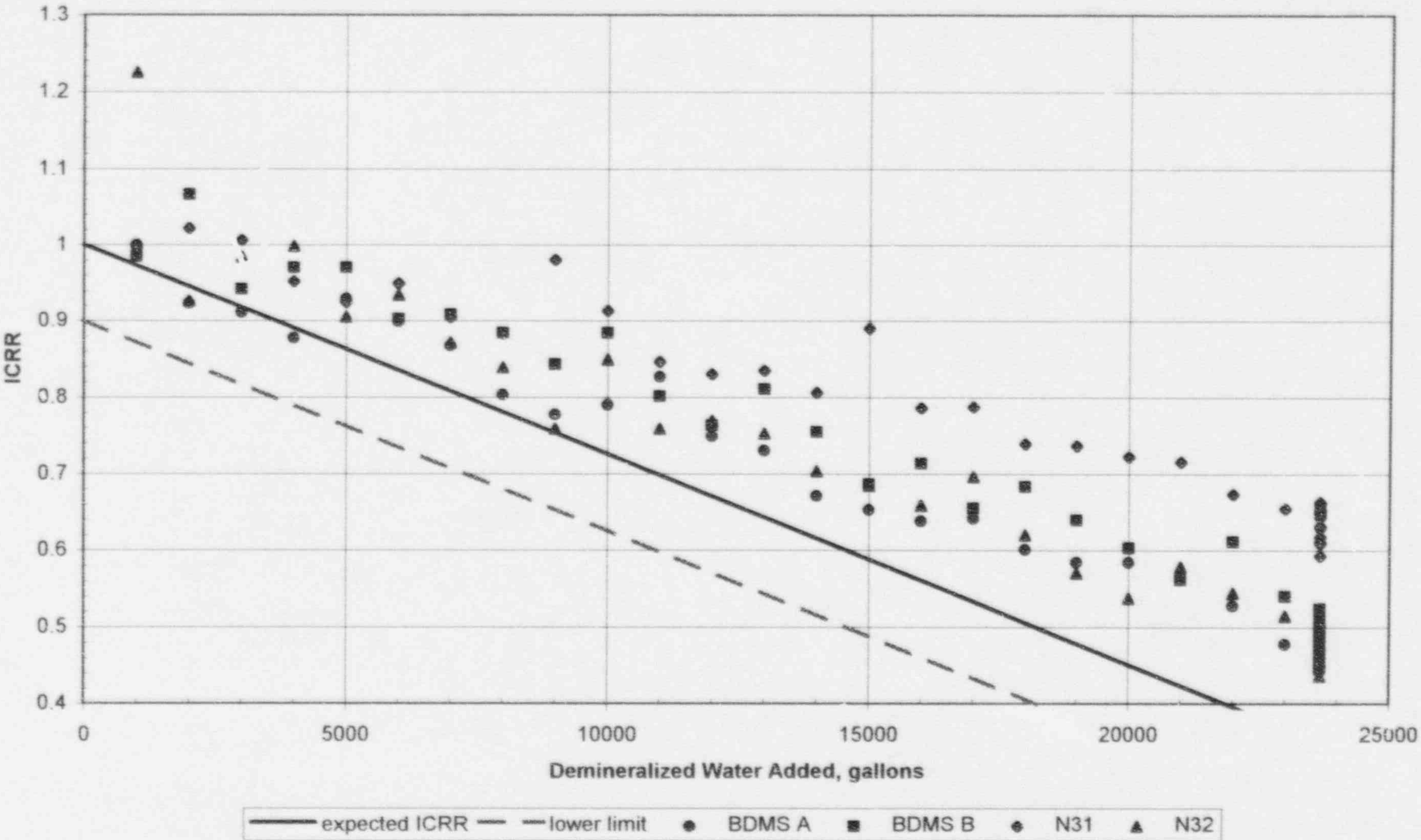
Intermediate Range

Channel	Ratio (BOC 10 ÷ Cycle 9)	Cycle 9 Reactor Trip Setpoint, μAmps	BOC 10 Reactor Trip Setpoint, μAmps	BOC 10 Rod Stop Setpoint, μAmps
N35	1.010	78.1	74.9	59.9
N36	1.025	72.0	70.1	56.1

Power Range

Channel	Ratio (BOC 10 ÷ Cycle 9)	Axial Offset, %	Cycle 9 Full Power Current, μAmps		BOC 10 Full Power Current, μAmps	
			Upper	Lower	Upper	Lower
N41	1.064	+20	303.7	238.4	323.1	253.7
		0	261.9	276.7	278.7	294.4
		-20	220.1	315.0	234.2	335.2
N42	1.065	+20	291.1	214.0	310.0	227.9
		0	250.7	248.8	267.0	265.0
		-20	210.3	283.4	224.0	301.8
N43	1.058	+20	254.8	196.6	269.6	208.0
		0	221.3	228.7	234.1	242.0
		-20	187.7	261.0	198.6	276.1
N44	1.066	+20	247.4	197.9	263.7	211.0
		0	214.4	231.1	228.6	246.4
		-20	181.6	264.4	193.6	281.9

FIGURE 2
ICRR vs. DEMIN WATER ADDED DURING REACTOR COOLANT SYSTEM DILUTION



3.0 ZERO POWER PHYSICS TESTING

Zero Power Physics Testing (ZPPT) is performed at the beginning of each cycle and is controlled by PT/0/A/4150/01, Controlling Procedure for Startup Physics Testing. Test measurements are made below the Point of Nuclear Heat using the output of one Power Range NIS detector connected to a reactivity computer. Measurements are compared to predicted data to verify core design. The following tests/measurements are included in the ZPPT program:

- 1/M Approach to Criticality
- Measurement of Point of Adding Heat
- Reactivity Computer checkout
- All Rods Out Critical Boron Concentration measurement
- All Rods Out Isothermal Temperature Coefficient measurement
- Measurement of Reference Bank worth by dilution
- Reference Bank in Critical Boron Concentration measurement
- Differential Boron Worth determination
- Control Rod Worth Measurement by Rod Swap

Zero power physics testing for Catawba 1 Cycle 10 began at 0232 on October 2, 1996 commencing with rod withdrawal for approach to criticality. ZPPT ended at 1422 on October 3, 1996 following analysis of Rod Swap data. Table 5 summarizes results from ZPPT. All acceptance criteria were met.

Sections 3.1 through 3.10 describe ZPPT measurements and results.

3.1 1/M Approach to Criticality

Initial criticality for Catawba 1 Cycle 10 was achieved per PT/0/A/4150/19, 1/M Approach to Criticality. In this procedure, Estimated Critical Rod Position (ECP) is calculated based on latest available Reactor Coolant boron concentration. Control rods are withdrawn until Boron Dilution Mitigation System (BDMS) or Source Range count rate doubles. Inverse Count Rate Ratio (ICRR) is plotted for each source range and BDMS channel. ICRR data is used to project critical rod position. If projected critical rod position is acceptable, rod withdrawal may continue.

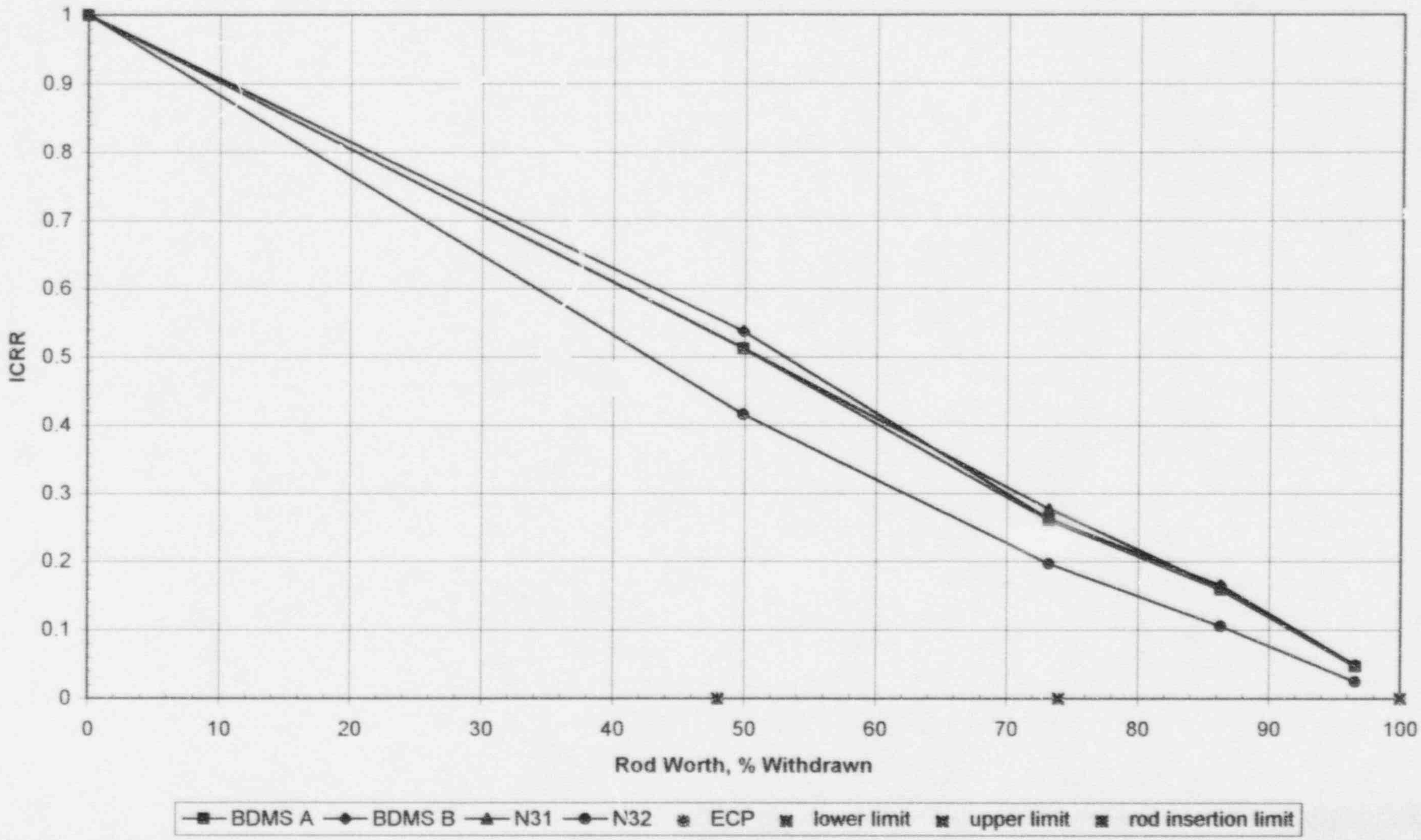
Rod withdrawal for the approach to criticality began at 0232 on October 2, 1996. Criticality was achieved at 0707 on October 2, 1996 with Control Bank D at 207 steps withdrawn.

Figure 3 shows the ICRR behavior during the approach to criticality. All acceptance criteria of PT/0/A/4150/19 were met.

TABLE 5
SUMMARY OF ZPPT RESULTS

PARAMETER	MEASURED VALUE	PREDICTED VALUE OR ACCEPTANCE CRITERIA
Nuclear Heat	7.7×10^{-7} amps (N43)	N/A
ZPPT Test Band	10^{-8} to 2.0×10^{-7} amps (N43)	N/A
ARO Critical Boron	1840 ppmB	1833 ± 50 ppmB
ARO ITC	-3.26 pcm/°F	-3.05 ± 2 pcm/°F
ARO MTC	-1.59 pcm/°F	-1.58 pcm/°F
Reference Bank (Shutdown Bank B) Worth	893.5 pcm	857 ± 128.6 pcm
Ref. Bank in Critical Boron	1725 ppmB	1712 ppmB
Differential Boron Worth	-7.77 pcm/ppmB	-7.09 ± 1.06 pcm/ppmB
Control Bank D Worth	599.0 pcm	562 ± 200 pcm
Control Bank C Worth	778.8 pcm	807 ± 242 pcm
Control Bank B Worth	692.2 pcm	634 ± 200 pcm
Control Bank A Worth	310.0 pcm	346 ± 200 pcm
Shutdown Bank E Worth	455.2 pcm	466 ± 200 pcm
Shutdown Bank D Worth	496.7 pcm	443 ± 200 pcm
Shutdown Bank C Worth	486.2 pcm	438 ± 200 pcm
Shutdown Bank A Worth	346.0 pcm	296 ± 200 pcm
Total Rod Worth	5057.6 pcm	4849 ± 485 pcm

FIGURE 3
ICRR vs. CONTROL ROD WORTH DURING APPROACH TO CRITICALITY



3.2 Source Range/Intermediate Range Overlap Data

During the initial approach to criticality, Source Range and Intermediate Range NIS data was obtained to verify the existance of at least one decade of overlap. If one decade of overlap did not exist, intermediate range compensation voltage would have been adjusted to provide the overlap.

Overlap data for Cycle 10 was obtained per PT/0/A/4150/01, Controlling procedure for Startup Physics Testing, on October 2, 1996. Table 6 contains the overlap data. The acceptance criterion was met.

TABLE 6
SOURCE RANGE/ INTERMEDIATE RANGE OVERLAP DATA

	SOURCE RANGE		INTERMEDIATE RANGE	
	N31, cps	N32, cps	N35, amps	N36, amps
INITIAL DATA: NIS Meters	500	400	1×10^{-11}	1×10^{-11}
OAC	570	390	1.10×10^{-11}	1.00×10^{-11}
FINAL DATA: NIS Meters	15,000	15,000	1.50×10^{-10}	2.00×10^{-10}
OAC	23,950	12,200	1.442×10^{-10}	1.788×10^{-10}

3.3 Point of Nuclear Heat Addition

The Point of Nuclear Heat Addition is measured by trending Reactor Coolant System temperature, Pressurizer level, flux level, and reactivity while slowly increasing reactor power. A slow, constant startup rate is initiated by rod withdrawal. An increase in Reactor Coolant System temperature and/or Pressurizer level accompanied by a change in reactivity and/or rate of flux increase indicates the addition of Nuclear Heat.

For Cycle 10, the Point of Nuclear Heat Addition was determined per PT/0/A/4150/01, Controlling Procedure for Startup Physics Testing, on October 2, 1996. Table 7 summarizes the data obtained.

The Zero Power Physics Test Band was set at 10^{-8} to 2.0×10^{-7} amps on Power Range channel N43 (connected to reactivity computer). This test band provided more than a factor of two margin to nuclear heat for zero power physics testing. Acceptance criterion was satisfied.

TABLE 7
NUCLEAR HEAT DETERMINATION

	Reactivity Computer (N43), amps	Intermediate Range Channel N35, amps	Intermediate Range Channel N36, amps
RUN #1	7.70×10^{-7}	4.50×10^{-7}	5.70×10^{-7}
RUN #2	8.10×10^{-7}	5.30×10^{-7}	6.58×10^{-7}

3.4 Reactivity Computer Checkout

The reactivity computer checkout was performed per PT/0/A/4150/01, Controlling Procedure for Startup Physics Testing, to verify that the Power Range channel connected to the reactivity computer can provide reliable reactivity data. Reactivity Insertions of approximately +25 and -25pcm (but no greater than +45 and -45 pcm) are made. The resulting Periods are measured and used to determine the corresponding theoretical reactivities. The measured reactivity is compared to the theoretical reactivity and verified to be within 4.0%.

The checkout was performed for Cycle 10 on October 2, 1996. Table 8 lists the results of the reactivity insertion. The acceptance criterion was met.

TABLE 8
REACTIVITY COMPUTER CHECKOUT

Period, seconds	Theoretical Reactivity, pcm	Measured Reactivity, pcm	Absolute Error, pcm	Percent Error,%
220.99	29.12	28.87	0.25	-0.84
-259.56	-33.43	-32.74	0.69	-2.07

3.5 ARO Boron Endpoint Measurement

This test is performed at the beginning of each cycle to verify that measured and predicted total core reactivity are consistent. The test is performed near the all rods out (ARO) configuration. Reactor Coolant System boron samples are obtained while Control Bank D is pulled to the fully withdrawn position. The reactivity difference from criticality to the ARO configuration is measured and converted to an equivalent boron worth using the predicted differential boron worth. The average measured boron concentration is adjusted accordingly to obtain the ARO critical boron concentration.

The Cycle 10 beginning of cycle, hot zero power, all rods out, critical boron concentration was measured on October 2, 1996 per PT/0/A/4150/10, Boron Endpoint Measurement. The ARO, HZP boron concentration was measured to be 1840 ppmB. Predicted ARO critical boron concentration was 1833 ppmB. The acceptance criterion, measured boron within 50 ppmB of predicted, was met.

3.6 ARO Isothermal Temperature Coefficient Measurement

The all rods out (ARO) Isothermal Temperature Coefficient (ITC) is measured at the beginning of each cycle to verify consistency with predicted value. In addition, the Moderator Temperature Coefficient (MTC) is obtained by subtracting the Doppler Temperature Coefficient from the ITC. The MTC is used to ensure compliance with Technical Specification limits.

To measure the ITC, statepoint data is obtained prior to cooldown. A Reactor Coolant System cooldown is initiated, within administrative cooldown limits. When sufficient data (at least 5 °F) is obtained, statepoint data is again obtained. A heatup is performed while again maintaining administrative limits. The Delta Reactivity divided by the Delta Temperature (for each cooldown/heatup) are used to determine the ITC. The cooldown/heatup cycle is repeated if additional data is required.

The Beginning of Cycle 10 ITC was measured per PT/0/A/4150/12A, Isothermal Coefficient of Reactivity Measurement, on October 2, 1996. No additional cooldown/heatup cycles were required because of good agreement between initial heatup and cooldown results. Table 9 summarizes the data obtained during the measurement.

Average ITC was determined to be -3.26 pcm/°F. Predicted ITC was -3.05 pcm/°F. Measured ITC was therefore within acceptance criterion of predicted ITC ± 2 pcm/°F.

The MTC was determined to be -1.59 pcm/°F. This value was used with procedure PT/0/A/4150/21, Temporary Rod Withdrawal Limits Determination, to ensure that MTC would remain within Technical Specification limits at all power levels. No rod withdrawal limits were required.

TABLE 9
ITC MEASUREMENT RESULTS

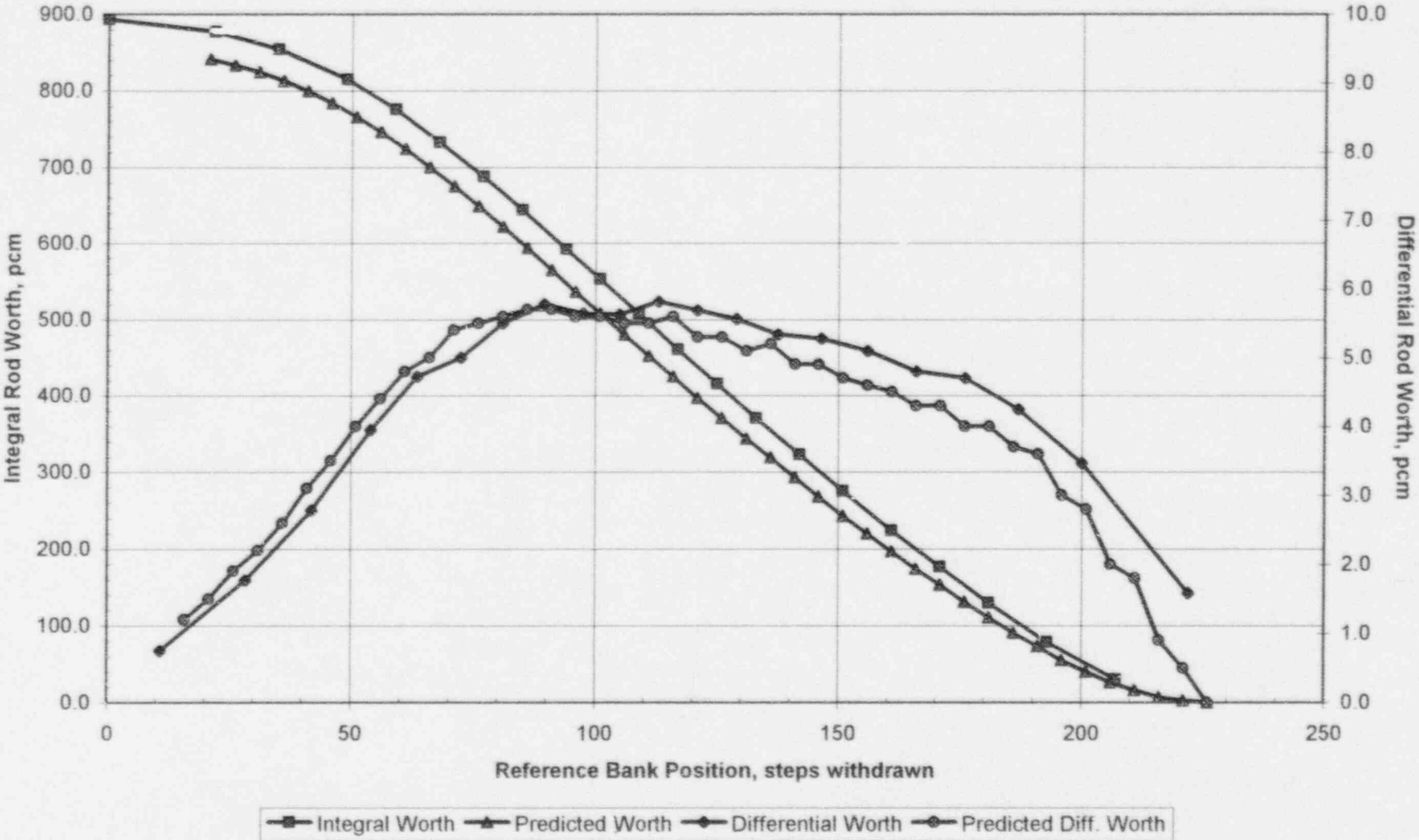
	ΔT, °F	$\Delta\rho$, pcm	T_{avg}, °F	ITC, pcm/°F
Cooldown	-5.95	+19.0	554.58	-3.19
Heatup	+6.00	-20.0	554.60	-3.33
				Average: -3.26

3.7 Reference Bank Worth Measurement by Dilution

The control rod bank predicted to have the highest worth is designated the Reference Bank. This RCCA bank is measured by inserting the bank (with all other rod banks fully withdrawn) in discrete steps while slowly diluting the Reactor Coolant System (at rate < 500 pcm/hr). The reactivity worths of the discrete steps of rod insertion are measured using the Reactivity Computer and summed to obtain the integral worth of the Reference Bank.

The Beginning of Cycle 10 Reference Bank (Shutdown Bank B) worth was measured on October 3, 1996 per PT/0/A/4150/11A, Control Rod Worth Measurement by Boration/Dilution. Figure 4 shows integral worth of Reference Bank versus bank position. The Reference Bank was measured to be worth 893.5 pcm; predicted worth was 857 pcm. The acceptance criterion, measured worth within ± 15% of predicted, was met.

FIGURE 4
INTEGRAL AND DIFFERENTIAL WORTH OF REFERENCE BANK



3.8 Reference Bank in Boron Endpoint Measurement

This test is performed at the beginning of each cycle to measure the critical boron concentration with the Reference Bank fully inserted and all other control rod banks fully withdrawn. The measured boron concentration is used with the measured ARO critical boron concentration and the measured worth of the reference bank to calculate the differential boron worth. Reactor Coolant System boron samples are obtained while control rods are inserted or withdrawn to the "Reference Bank In" configuration. The reactivity difference from criticality to the "Reference Bank In" configuration is measured and converted to an equivalent boron worth using the predicted differential boron worth. The average measured boron concentration is adjusted accordingly to obtain the "Reference Bank In" critical boron concentration.

The Cycle 10 Beginning of Cycle, Hot Zero Power, Reference Bank In, critical boron concentration was measured on October 3, 1996 per PT/0/A/4150/10, Boron Endpoint Measurement. This boron concentration was measured to be 1725 ppmB. Predicted "Reference Bank In" critical boron concentration was 1712 ppmB. There is no quantitative acceptance criterion associated with this test.

3.9 Differential Boron Worth Determination

The differential boron worth is calculated from the measured ARO critical boron concentration, "Reference Bank In" critical boron concentration, and total measured reactivity worth of Reference Bank. The calculated value is compared to predicted value to verify consistency. This calculation also provides an indirect check of measured Reference Bank worth and of the Boron Endpoint measurements.

The Beginning of Cycle 10, Hot Zero Power differential boron worth was calculated to be -7.77 pcm/ppmB per PT/0/A/4150/01, Controlling Procedure for Startup Physics Testing. The predicted value was -7.09 pcm/ppmB. The acceptance criterion (measured within ± 15% of predicted), was met.

3.10 Control Rod Worth Measurement by Rod Swap

The worths of all control rod banks except the Reference Bank are measured by inserting each bank while withdrawing the Reference Bank and/or previously measured bank to maintain near critical conditions. When the bank being measured is fully inserted, the Reference Bank is positioned to achieve critical conditions with all other rod banks fully withdrawn. The worth of the fully inserted bank is determined from the critical position of the Reference Bank. The measured worth is compared to predicted worth to verify consistency. The sum of the worths of all banks, including the reference bank, is also compared to predicted total.

The Beginning of Cycle 10 rod worth measurement by Rod Swap was performed on October 3, 1996 per PT/0/A/4150/11B, Control Rod Worth Measurement by Rod Swap. Table 10 summarizes the results. All acceptance criteria were met.

TABLE 10
CONTROL ROD WORTH MEASUREMENT DATA

Bank	Adjusted Reference Bank Worth	Critical Position of Ref. Bank	Remaining Worth of Ref. Bank	Alpha	Measured Worth, pcm	Predicted Worth, pcm	Difference (Predicted - Measured)	% Diff. (Pred - Meas)/Pred × 100
Shutdown B (Ref. Bank)	N/A	N/A	N/A	N/A	893.5	857	-36.5	-4.3
Shutdown A	912.5	100	558	1.015	346.0	296	-50.0	-16.9
Control A	912.8	101	553	1.091	310.0	346	+36.0	+10.4
Shutdown C	913.1	121	438	0.974	486.2	438	-48.2	-11.0
Shutdown D	913.4	123	427	0.976	496.7	443	-53.7	-12.1
Shutdown E	913.6	105	530	0.865	455.2	466	+10.8	+2.3
Control D	913.9	149	286	1.101	599.0	562	-37.0	-6.6
Control B	914.2	148	291	0.762	692.2	634	-58.2	-9.2
Control C	914.5	177	148	0.915	778.0	807	+28.2	+3.5
				Total	5057.6	4849	-208.6	-4.3

4.0 POWER ESCALATION TESTING

Power Escalation Testing is performed during the initial power ascension to full power for each cycle and is controlled by PT/0/A/4150/01, Controlling Procedure for Startup Physics Testing. Tests are performed from 0% through 100% power with major testing plateaus at ~30%,~75%, and 100% power.

Significant tests performed during C1C10 Power Escalation were:

- Core Power Distribution (at ~30%, ~71%, and 100% power)
- One-Point Incore/Excore Calibration (at 30% power)
- Reactor Coolant Delta Temperature Measurement (at 71% and 100% power)
- Hot Full Power Critical Boron Concentration Measurement (at 100% power)
- Incore/Excore Calibration (at 100% power)
- Calorimetric Reactor Coolant Flow Measurement (at 100% power, This test is not under the control of PT/0/A/4150/01)
- Unit Load Steady State - at 30%, 50%, 75%, and 100% (Steam Generator Replacement Post-Mod testing)
- Unit Load Transient Test - at 50% and 71% (Steam Generator Replacement Post-Mod testing)
- Replacement S/G Functional Tuning and Testing of DFCS - at 10%, 30%, 50%, and 71% (Steam Generator Replacement Post-Mod testing)
- Evaluation of Intermediate Range NIS Rod Stop and Rx Trip Setpoints

In addition to the tests listed above, PT/0/A/4150/01 performs evaluations of the Movable Incore Detector System, and on-line (OAC) Thermal Power program. The results of these are not included in this report.

Power Escalation Testing for Catawba 1 Cycle 10 began on October 4, 1996. Full power was reached on October 10, 1996. Full power testing was completed on October 18, 1996. Sections 4.1 through 4.10 describe the significant tests performed during power escalation and their results.

4.1 Core Power Distribution

Core power distribution measurements are performed during power escalation at low power (approximately 30%), intermediate power (approximately 75%), and full power. Measurements are made to verify flux symmetry and to verify core peaking factors are within limits. Data obtained during this test are also used to check calibration of Power Range NIS channels and to calibrate them if required (see sections 4.2 and 4.6). Measurements are made using the Moveable Incore Detector System and analyzed using Duke Power's CORE and MONITOR codes (adapted from Shangstrom Nuclear Associates' CORE package and FCF's MONITOR code, respectively).

The Catawba 1 Cycle 10 Core Power Distribution measurements were performed on October 5, 1996 (30% power), October 8, 1996 (71% power), and October 14, 1996 (100% power). Tables 11 through 13 summarize the results. All acceptance criteria were met.

TABLE 11
CORE POWER DISTRIBUTION RESULTS
30% POWER

Plant Data

Map ID:	FCM/1/10/001
Date of Map:	October 5, 1996
Cycle Burnup:	0.30 EFPD
Power Level:	29.95% F.P.
Control Rod Position:	Control Bank D at 215 Steps Wd
Reactor Coolant System Boron Concentration:	1667 ppmB

CORE Results

Core Average Axial Offset:	12.421%
Tilt Ratios for Entire Core Height: Quadrant 1:	0.97997
Quadrant 2:	1.03602
Quadrant 3:	1.01995
Quadrant 4:	0.96407
Maximum F_Q (nuclear):	2.083
Maximum $F_{\Delta H}$ (nuclear):	1.555
Maximum Error between Pred. and Meas $F_{\Delta H}$:	9.0%
Average Error between Pred. and Meas. $F_{\Delta H}$:	3.62%
Maximum Error between Expected and Measured Detector Response:	9.50%
RMS of Errors between Expected and Measured Detector Response:	4.5%

MONITOR Results

Minimum F_Q Operational Margin:	29.38%
Minimum F_Q RPS Margin:	9.79%
Minimum F_Q LCO Margin:	54.95%
Minimum $F_{\Delta H}$ Surveillance Margin:	35.43%
Minimum $F_{\Delta H}$ LCO Margin:	19.25%

TABLE 12
CORE POWER DISTRIBUTION RESULTS
71% POWER

Plant Data

Map ID:	FCM/1/10/003
Date of Map:	October 8, 1996
Cycle Burnup:	1.77 EFPD
Power Level:	70.897% F.P.
Control Rod Position:	Control Bank D at 215 Steps Wd
Reactor Coolant System Boron Concentration:	1425 ppmB

CORE Results

Core Average Axial Offset:	3.319%
Tilt Ratios for Entire Core Height: Quadrant 1:	0.99037
Quadrant 2:	1.01704
Quadrant 3:	1.01166
Quadrant 4:	0.98092
Maximum F_Q (nuclear):	1.822
Maximum $F_{\Delta H}$ (nuclear):	1.466
Maximum Error between Pred. and Meas $F_{\Delta H}$:	6.77%
Average Error between Pred. and Meas. $F_{\Delta H}$:	2.31%
Maximum Error between Expected and Measured Detector Response:	6.80%
RMS of Errors between Expected and Measured Detector Response:	3.10%

MONITOR Results

Minimum F_Q Operational Margin:	23.16%
Minimum F_Q RPS Margin:	15.58%
Minimum F_Q LCO Margin:	43.79%
Minimum $F_{\Delta H}$ Surveillance Margin:	21.95%
Minimum $F_{\Delta H}$ LCO Margin:	18.27%

TABLE 13
CORE POWER DISTRIBUTION RESULTS
100% POWER

Plant Data

Map ID:	FCM/1/10/004
Date of Map:	October 14, 1996
Cycle Burnup:	6.70 EFPD
Power Level:	99.834% F.P.
Control Rod Position:	Control Bank D at 215 Steps Wd
Reactor Coolant System Boron Concentration:	1234 ppmB

CORE Results

Core Average Axial Offset:	-1.596%
Tilt Ratios for Entire Core Height: Quadrant 1:	0.99172
Quadrant 2:	1.01714
Quadrant 3:	1.00269
Quadrant 4:	0.98845
Maximum F_Q (nuclear):	1.784
Maximum $F_{\Delta H}$ (nuclear):	1.449
Maximum Error between Pred. and Meas $F_{\Delta H}$:	7.06%
Average Error between Pred. and Meas. $F_{\Delta H}$:	1.58%
Maximum Error between Expected and Measured Detector Response:	7.0%
RMS of Errors between Expected and Measured Detector Response:	2.2%

MONITOR Results

Minimum F_Q Operational Margin:	6.74%
Minimum F_Q RPS Margin:	13.84%
Minimum F_Q LCO Margin:	23.01%
Minimum $F_{\Delta H}$ Surveillance Margin:	1.51%
Minimum $F_{\Delta H}$ LCO Margin:	12.64%

4.2 One-Point Incore/Excore Calibration

PT/0/A/4600/05D, One-Point Incore/Excore Calibration, is performed using results of Power Range NIS data taken at 30% power and the incore axial offset measured at 30%. Power Range channels are calibrated before exceeding 50% in order to have valid indications of Axial Flux Difference and Quadrant Power Tilt Ratio for subsequent power ascension. The calibration is checked using the intermediate power level flux map (71% F.P. for C1C10). If necessary, Power Range NIS is recalibrated per PT/0/A/4600/05D or PT/0/A/4600/05A, Incore/Excore Calibration.

Data for Catawba 1 Cycle 10 was obtained on October 5, 1996 and all Power Range NIS calibrations were completed on October 6, 1996. Results are presented in Table 14. All acceptance criteria were met.

TABLE 14
ONE-POINT INCORE/EXCORE CALIBRATION RESULTS

Reactor Power = 30.00% Axial Offset = 12.421%

Measured Power Range Currents, μAmps

	N41	N42	N43	N44
Upper	75.0	77.0	72.0	62.0
Lower	69.0	67.0	64.0	58.0

Ratio, Extrapolated (from measured) Currents to "Expected" (from last calibration) Currents

	N41	N42	N43	N44
Upper	0.9604	1.0363	1.1236	1.0243
Lower	0.9991	1.0903	1.1468	1.0534

New Calibration Currents, μAmps

Axial Offset, %	N41		N42		N43		N44	
	Upper	Lower	Upper	Lower	Upper	Lower	Upper	Lower
+20	263.7	216.8	271.0	210.3	252.6	200.9	217.7	181.7
0	227.4	251.6	233.4	244.4	219.3	233.7	188.7	212.2
-20	191.1	286.4	195.8	278.5	186.1	266.6	159.8	242.7

4.3 Reactor Coolant Loop Delta Temperature Measurement

Reactor Coolant System (NC) Hot Leg and Cold Leg temperature data is normally obtained between 50% and 80% power and at 100% power per PT/0/A/4600/26, NC Temperature Calibration, to ensure that full power delta temperature constants (ΔT_0) are valid. ΔT_0 is used in the Overpower and Overtemperature Delta Temperature reactor protection functions.

In the case of C1C10, the four loop ΔT_0's were each preliminarily established at 57.9°F per Steam Generator Replacement Project. Due to the fact that NC Loop 1B has always been significantly cooler than the other loops, PT/0/A/4600/26 was required to be performed (and Loop B ΔT_0 adjusted accordingly) at 30.0% F.P. on October 6, 1996, because of the resulting discrepency this created between Loop 1B's and the other Loops' %F.P.ΔT's. PT/0/A/4600/26 was subsequently repeated at 71% F.P. on October 9, 1996 and at 100% F.P. on October 11, 1996. All four NC Loop ΔT_0's were adjusted per results obtained at 71% power. Loops A, B, and C were ultimately recalibrated using full power results. Table 15 summarizes the test results.

TABLE 15
REACTOR COOLANT DELTA TEMPERATURE DATA

Reactor Power = 71.0685%

	Loop A	Loop B	Loop C	Loop D
Meas. T_{HOT}, °F	597.7	593.3	597.0	596.2
Meas. T_{COLD}, °F	552.1	551.9	551.2	552.5
Calc. Δh, BTU/lb	60.225	54.265	60.298	57.671
Calc. Δh_0, BTU/lb	84.742	76.356	84.845	81.148
Calc. ΔT_0, °F	61.6	56.1	61.7	59.3
Current ΔT_0, °F	57.90	51.44	57.90	54.94
Difference, °F	3.7	4.7	3.8	4.3

Reactor Power = 99.5611%

	Loop A	Loop B	Loop C	Loop D
Meas. T_{HOT}, °F	615.6	610.5	615.0	613.7
Meas. T_{COLD}, °F	553.7	553.5	553.0	554.3
Calc. Δh, BTU/lb	84.414	76.952	84.408	80.672
Calc. Δh_0, BTU/lb	84.786	77.291	84.780	81.028
Calc. ΔT_0, °F	62.1	57.2	62.2	59.6
Current ΔT_0, °F	61.6	56.1	61.7	59.3
Difference, °F	0.5	1.1	0.5	0.3

4.4 Hot Full Power Critical Boron Concentration Measurement

The Hot Full Power critical boron concentration is measured using PT/0/A/4150/04, Reactivity Anomaly Calculation. Reactor Coolant boron concentration is measured (average of three samples) with reactor at essentially all rods out, Hot Full Power, equilibrium xenon conditions. The measured boron is corrected for any off-reference condition (e.g. inserted rod worth, temperature error, difference from equilibrium xenon) and compared to predicted value.

For the purposes of Startup Physics testing, the predicted critical boron concentration is adjusted for the difference between predicted and measured critical boron concentration measured at Zero Power. The difference between measured boron concentration and adjusted predicted value is used to compare to acceptance criterion (±50 ppmB).

For Catawba 1 Cycle 10, the Hot Full Power critical boron concentration was measured on October 14, 1996. The measured critical boron concentration was 1221 ppmB. Predicted critical boron concentration was 1250 ppmB; when adjusted for difference at zero power, the adjusted predicted critical boron concentration was 1257 ppmB. The difference between measured and adjusted predicted critical boron concentration was -36 ppmB, which met the acceptance criterion.

4.5 Incore/Excore Calibration

Excore NIS Power Range channels are calibrated at full power per PT/0/A/4600/05A, Incore/Excore Calibration. Incore data (flux maps) and Power Range NIS currents are obtained at various axial power distributions. A least squares fit of the output of each detector (upper and lower chambers) as a function of measured incore axial offset is determined. The slopes and intercepts of the fit for the upper and lower chamber for each channel are used to determine calibration data for that channel.

This test was performed for Catawba 1 Cycle 10 on October 14 and 15, 1996. All Power Range NIS calibrations were completed on October 17. Eight flux maps, with axial offset ranging from -8.443% to +4.054% were used. Table 16 summarizes the results. All acceptance criteria were met.

TABLE 16
INCORE/EXCORE CALIBRATION RESULTS

Full Power Currents, Microamps

Axial Offset, %	N41		N42		N43		N44	
	Upper	Lower	Upper	Lower	Upper	Lower	Upper	Lower
+20%	294.6	230.6	287.8	215.4	275.1	211.0	228.5	184.8
0%	252.0	269.4	247.5	250.9	236.9	247.3	196.8	216.2
-20%	209.5	308.3	207.3	286.3	198.7	283.7	165.1	247.6

Correction (M_J) Factors

N41	N42	N43	N44
1.278	1.316	1.297	1.305

4.6 Calorimetric Reactor Coolant Flow Measurement

With clean venturis, PT/2/A/4150/13B, Calorimetric Reactor Coolant Flow Measurement, is performed to establish a Primary Loop Delta T correction value to correct Primary power to Secondary Power. This is needed due to the fact that the NC Loop Elbow Tap Correction Factors are now fixed constants per Tech Specifications.

This test was performed on October 21, 1996. The Primary Loop D/T Correction was calculated to be 0.972. Table 17 summarizes the results. All acceptance criteria were met.

TABLE 17
CALORIMETRIC REACTOR COOLANT FLOW MEASUREMENT

Run Number	Secondary Thermal Output %	Primary Thermal Output %
1	99.775	100.288
2	99.670	100.252
3	99.633	100.245
Average	99.693	100.262
	Primary Loop D/T Correction (Secondary Power/ Primary)	0.972

4.7 Unit Load Steady State Test

In order to verify satisfactory steady state plant operation with Replacement Steam Generators (NSM CN-19815, TT/1/A/9200/085, Unit Load Steady State Test for NSM CN-19815 was performed at approximately 15%, 30%, 50%, 75%, and 100%. With the plant at steady power level data on the following parameters was obtained.

- Reactor Power
- NC Loop Cold Leg Temperature
- NC Loop Hot Leg Temperature
- NC Loop Average Temperature
- NC Loop Delta Temperature
- Pressurizer Level
- Turbine Control Valve Positon
- Turbine Impulse Pressure
- NIS Power Range Indication
- NIS Intermediate Range Indication
- Main Steam Pressure

The test was performed at ~15% power on October 4, 1996, ~30% on October 5, 1996, ~50% on October 6, 1996, ~75% on October 8, 1996, and ~100% on October 11, 1996. All acceptance criteria were met. Tables 18 through 25 document the results.

Table 18
NC Loop Cold Leg Temperatures

Power Level	NC Loop A	NC Loop B	NC Loop C	NC Loop D
15.58	556.9	556.8	556.3	557.1
30.66	554.0	554.0	553.3	554.4
50.36	553.5	553.4	552.6	553.8
71.03	552.0	551.9	551.1	552.5
99.53	553.7	553.5	552.9	554.3

Table 19
NC Loop Hot Leg Temperatures

Power Level	NC Loop A	NC Loop B	NC Loop C	NC Loop D
15.58	567.2	564.7	566.9	566.3
30.66	573.9	571.8	573.6	572.8
50.36	585.9	582.9	585.6	584.7
71.03	596.3	592.6	596.1	595.6
99.53	613.8	609.6	613.9	612.9

Table 20
NC Loop Average Temperatures

Power Level	NC Loop A	NC Loop B	NC Loop C	NC Loop D
15.58	562.2	561.0	561.5	561.8
30.66	564.2	562.9	563.3	563.6
50.36	570.2	568.3	569.2	569.5
71.03	574.8	572.5	573.7	574.3
99.53	584.7	581.9	583.7	583.9

Table 21
NC Loop Delta Temperatures

Power Level	NC Loop A	NC Loop B	NC Loop C	NC Loop D
15.58	10.3	7.8	10.5	9.1
30.66	19.7	17.7	20.1	18.3
50.36	32.5	29.5	33.0	30.8
71.03	44.3	40.7	45.0	43.1
99.53	60.1	56.1	61.0	58.7

Table 22
Pressurizer Level Data

Power Level	PZR Level Setpoint	PZR Level Channel 1	PZR Level Channel 2	PZR Level Channel 3
15.58	30.5	30.5	N/A	N/A
30.66	32.7	32.8	33.3	33.4
50.36	39.0	39.0	39.6	39.9
71.03	44.0	44.0	44.9	44.9
99.53	54.4	54.4	55.1	55.4

Table 23
Turbine Control Valve Positions

Power Level	CV 1	CV 2	CV 3	CV 4
15.58	11.3%	11.5%	0.0%	0.0%
30.66	21.0%	20.0%	0.0%	0.0%
50.36	34.7%	34.4%	0.0%	0.0%
71.03	91.1%	91.0%	7.3%	0.0%
99.53	100.0%	100.0%	88.1%	25.9%

Table 24
Turbine Impulse Pressure
(PSIG)

Power Level	Ch 1	Ch 2
15.72	70.4	71.2
30.67	167.5	168.0
48.49	297.2	297.8
71.01	440.0	440.4
99.83	692.1	691.4

Table 25
NIS Power Range Data
(μAmps)

Power Level	N-41	N-42	N-43	N-44
15.72	74	77	72	63
30.67	146	146	138	120
48.49	240	238	227	197
71.01	350	340	327	284
99.83	518	495	481	411

Table 26
NIS Intermediate Range Data
(μAmps)

Power Level	N-35	N-36
15.72	43.1	45.6
30.67	78.9	82.7
48.49	138.0	140.0
71.01	196.0	194.0
99.83	286.0	283.0

Table 27
Main Steam Pressure
(PSIG)

Power Level	S/G A	S/G B	S/G C	S/G D
15.58	1055.5	1051.8	1065.8	1063.1
30.66	1022.0	1017.9	1032.0	1029.7
50.36	1004.2	999.9	1014.0	1011.7
71.03	980.5	976.1	990.2	988.2
99.53	975.7	971.1	985.6	983.9

4.8 Unit Load Transient Test

TT/1/A/9200/86, Unit Load Transient Test for NSM CN-19815, was performed to verify proper operation of the modifications performed on various control systems per NSM CN-19815, Replacement Steam Generator Instrumentation and Control. The purpose of the test was to demonstrate proper plant response, including automatic control system performance, to a ~10% step load change (initiated via Turbine/Generator Control). The test verifies that the control systems work as designed to prevent the following plant transients (in response to a ~10% step load change):

- Reactor Trip
- Turbine Trip
- Actuation of Safety Injection
- Pressurizer and Steam Safeties or PORVs Lifting

This test satisfies the transient retest as required for the Post Modification Testing for Replacement Steam Generator Instrumentation and Control.

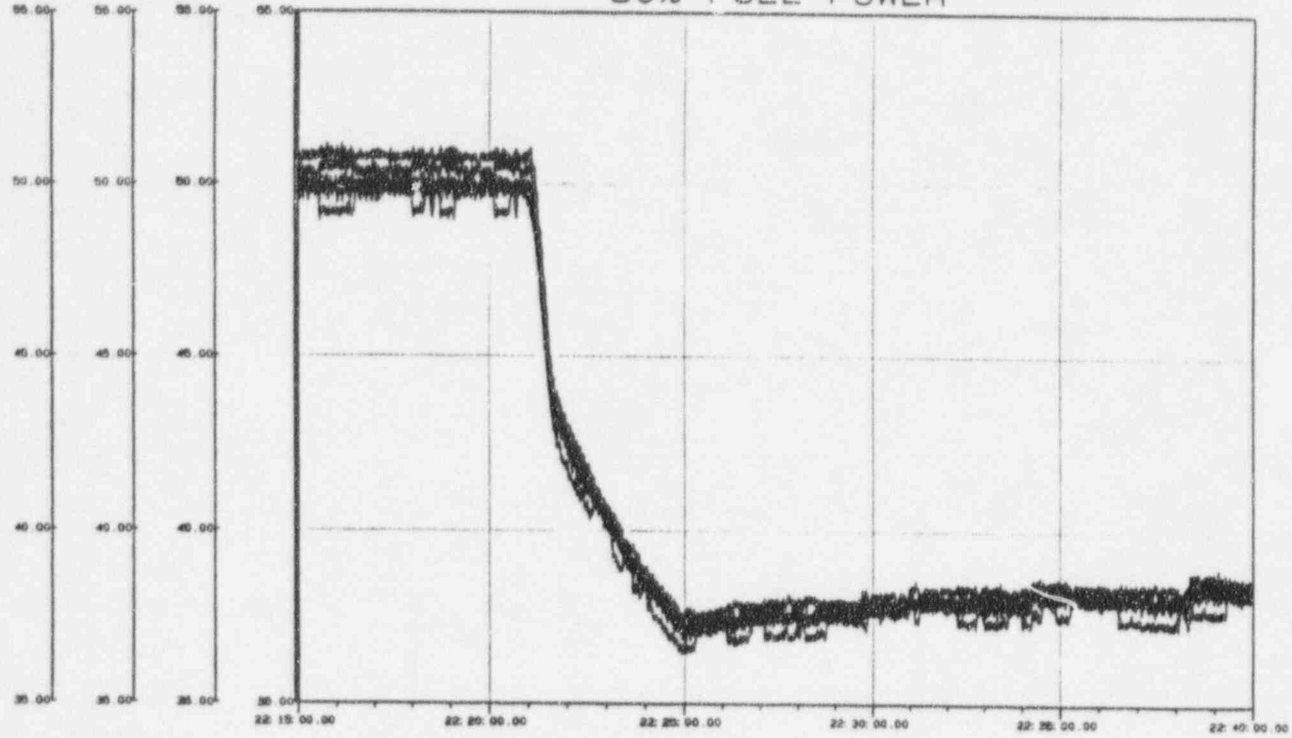
This test was performed from 50% F.P. on October 7, 1996 and from 70% F.P. on October 8, 1996. All acceptance criteria for the test were met as follows:

1) Reactor and Turbine did not trip
2) Safety Injection was not initiated
3) No Manual Operator Intervention was required to stabilize the Unit
4) Pressurizer PORV's did not lift
5) Pressurizer Code Safety Valves did not lift
6) Monitored plant parameters did not indicate sustained or diverging oscillations
7) Nuclear Power undershoot/overshoot was < 3%
8) Initial step load change was ≥ 80 MWe

Figures 5 through 20 illustrate response of plant parameters during the tests.

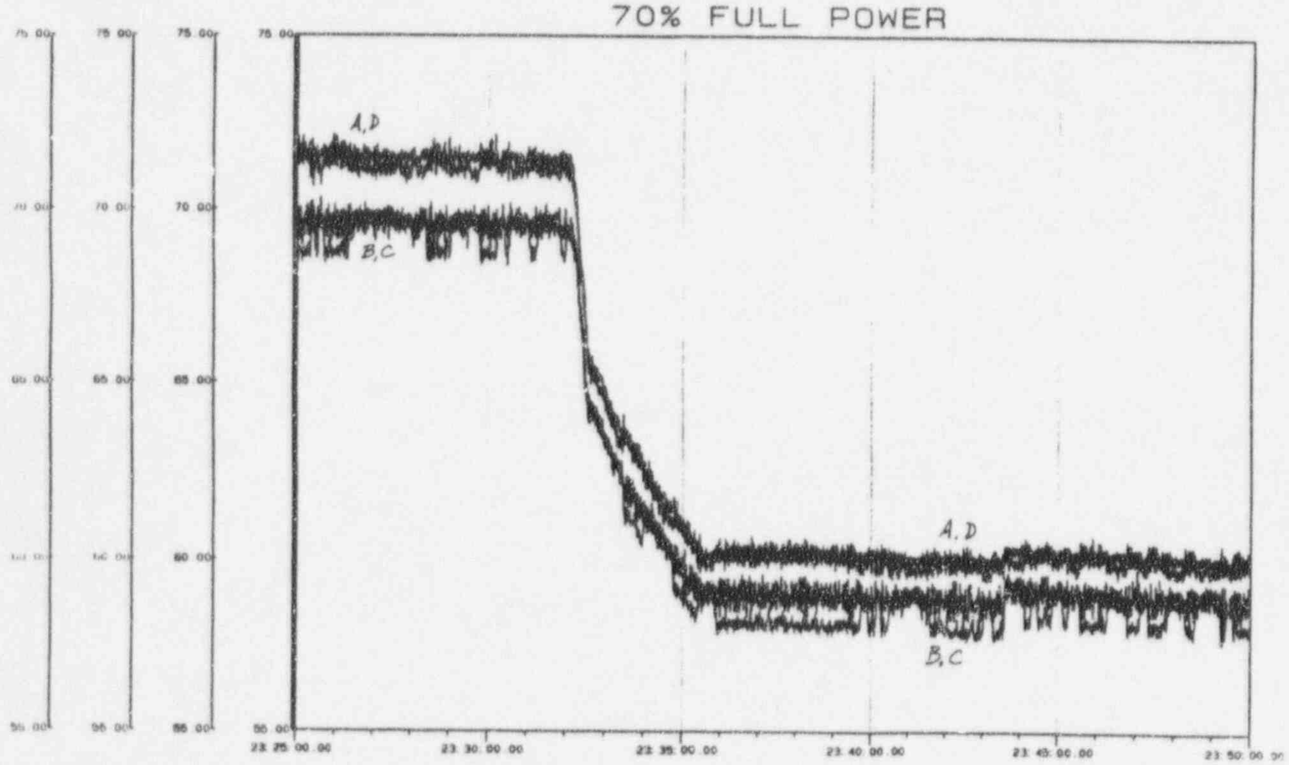
FIGURE 5
UNIT LOAD TRANSIENT TEST POWER RANGE LEVEL

50% FULL POWER



PID	Description	Value at trip time	Units/ Set Mag	Max Value	Min Value
C1A0758	POWER RANGE AVG LEVEL QUADRANT 1 (N-43)	38.6000	%	51.1600	37.2900
C1A0759	POWER RANGE AVG LEVEL QUADRANT 2 (N-42)	38.4200	%	50.3600	36.7100
C1A0760	POWER RANGE AVG LEVEL QUADRANT 3 (N-44)	38.1000	%	50.1600	36.4600
C1A0761	POWER RANGE AVG LEVEL QUADRANT 4 (N-41)	38.2900	%	50.8000	37.0100

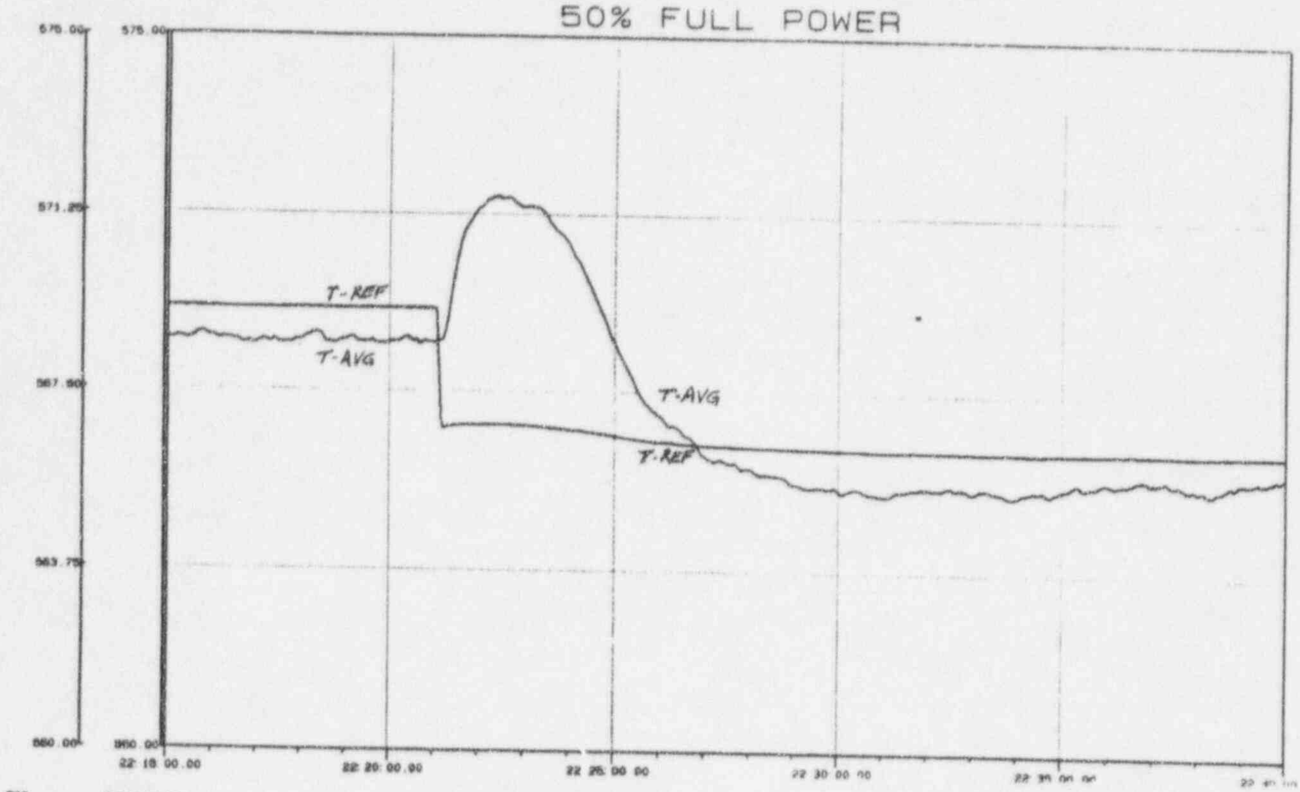
70% FULL POWER



PID	Description	Value at trip time	Units/ Set Mag	Max Value	Min Value
C1A0758	POWER RANGE AVG LEVEL QUADRANT 1 (N-43)	58.5900	%	72.1300	58.4700
C1A0759	POWER RANGE AVG LEVEL QUADRANT 2 (N-42)	58.8000	%	70.3100	58.1000
C1A0760	POWER RANGE AVG LEVEL QUADRANT 3 (N-44)	58.1200	%	70.1600	57.6900
C1A0761	POWER RANGE AVG LEVEL QUADRANT 4 (N-41)	59.6500	%	71.8100	58.4100

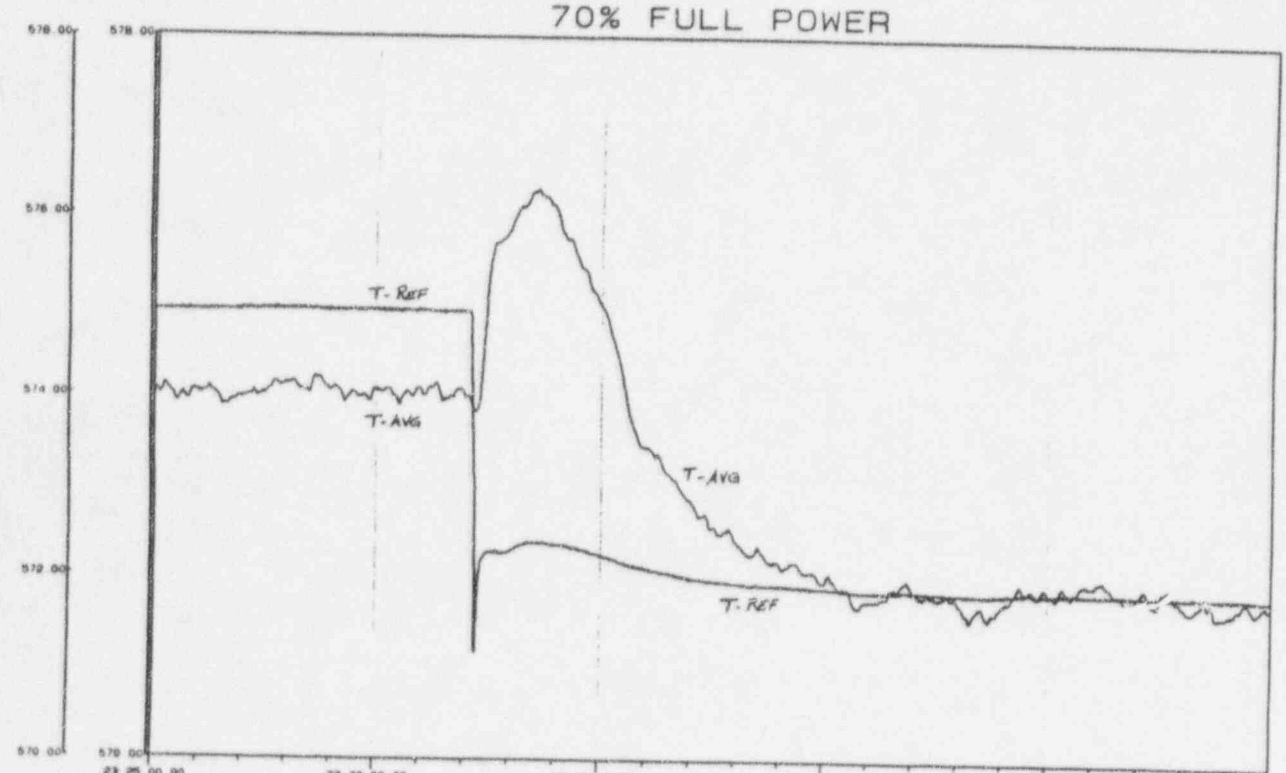
FIGURE 6
UNIT LOAD TRANSIENT TEST NC LOOP HIGHEST T-AVG / T-REF

50% FULL POWER



PID	Description	Value at trip time	Units/ Set Msg	Max Value	Min Value
C1A0701	NC LOOP HIGHEST T-AVG	566.8700	DEG F	571.6400	565.3700
C1A0725	T-REF	566.3000	DEG F	569.2600	566.2700

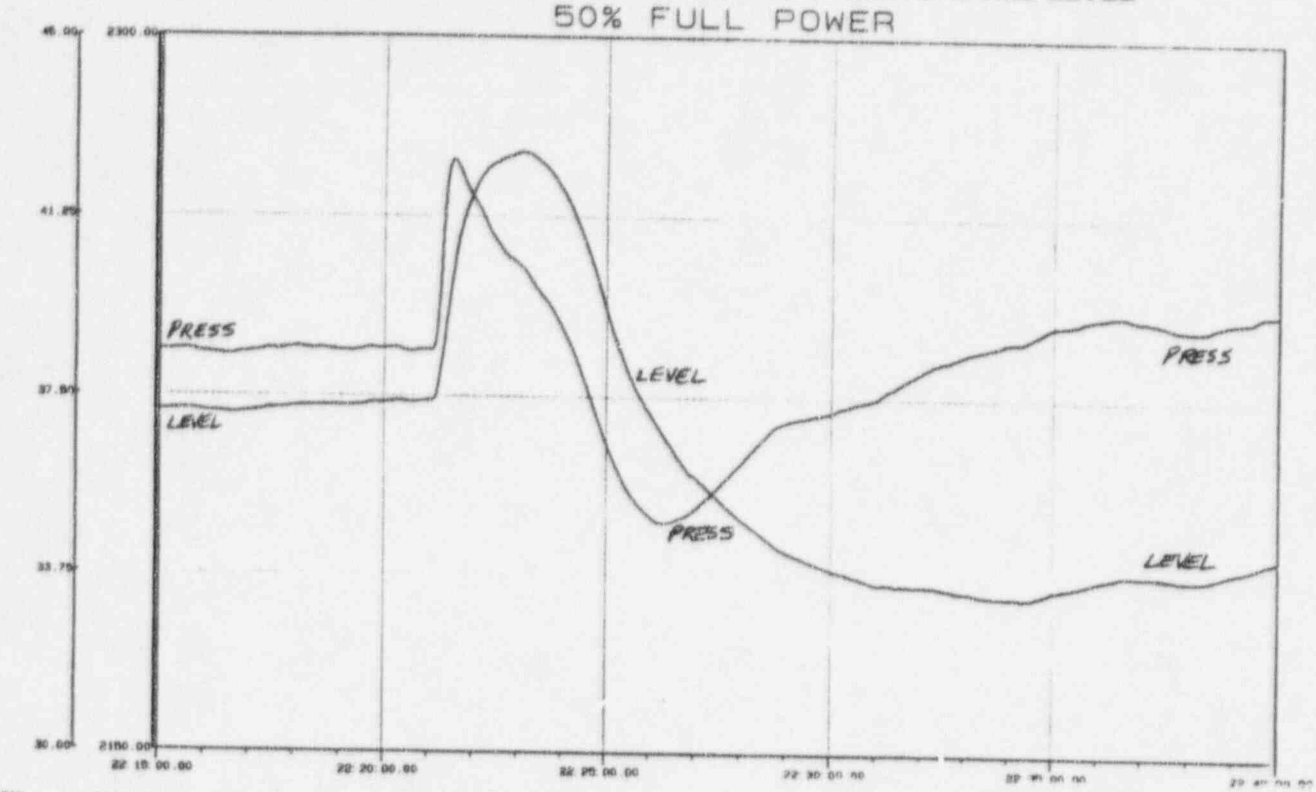
70% FULL POWER



PID	Description	Value at trip time	Units/ Set Msg	Max Value	Min Value
C1A0701	NC LOOP HIGHEST T-AVG	571.7700	DEG F	576.3100	571.5400
C1A0725	T-REF	571.8300	DEG F	574.0800	571.1800

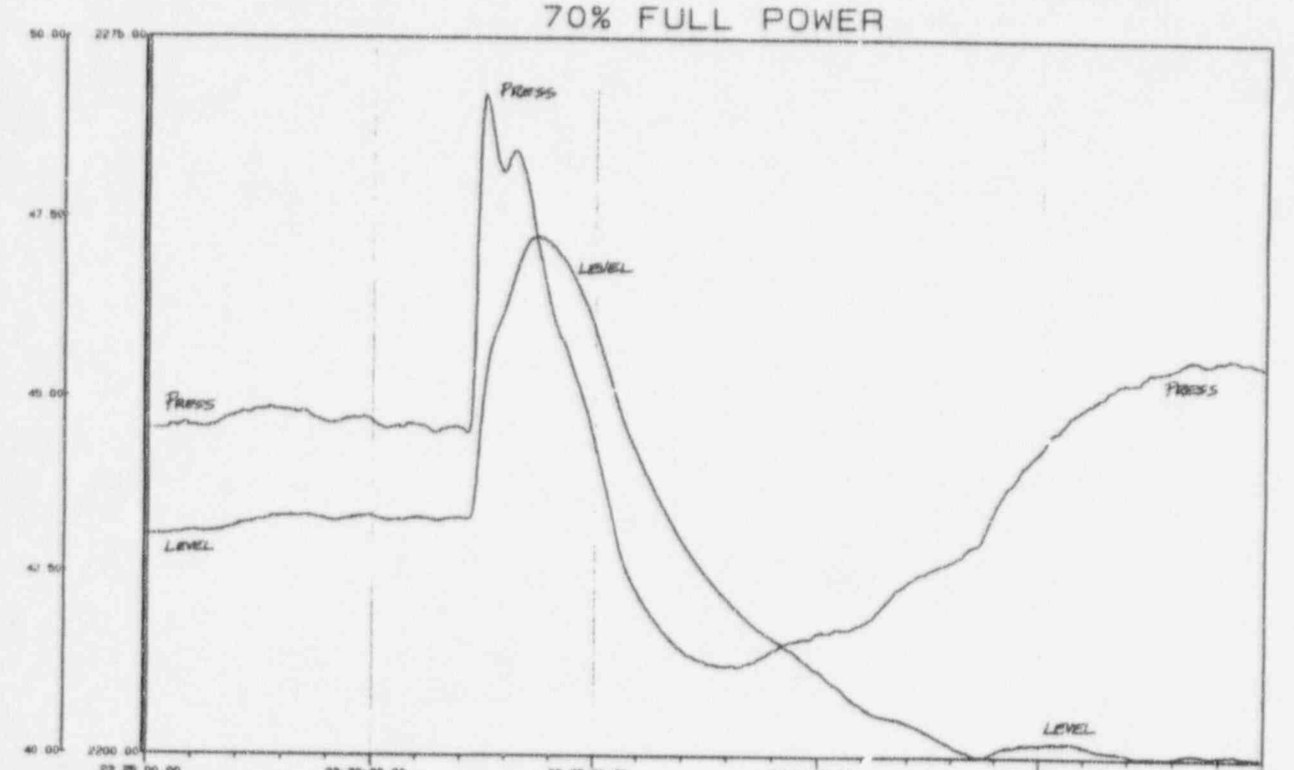
FIGURE 7
UNIT LOAD TRANSIENT TEST PRESSURIZER PRESSURE AND LEVEL

50% FULL POWER



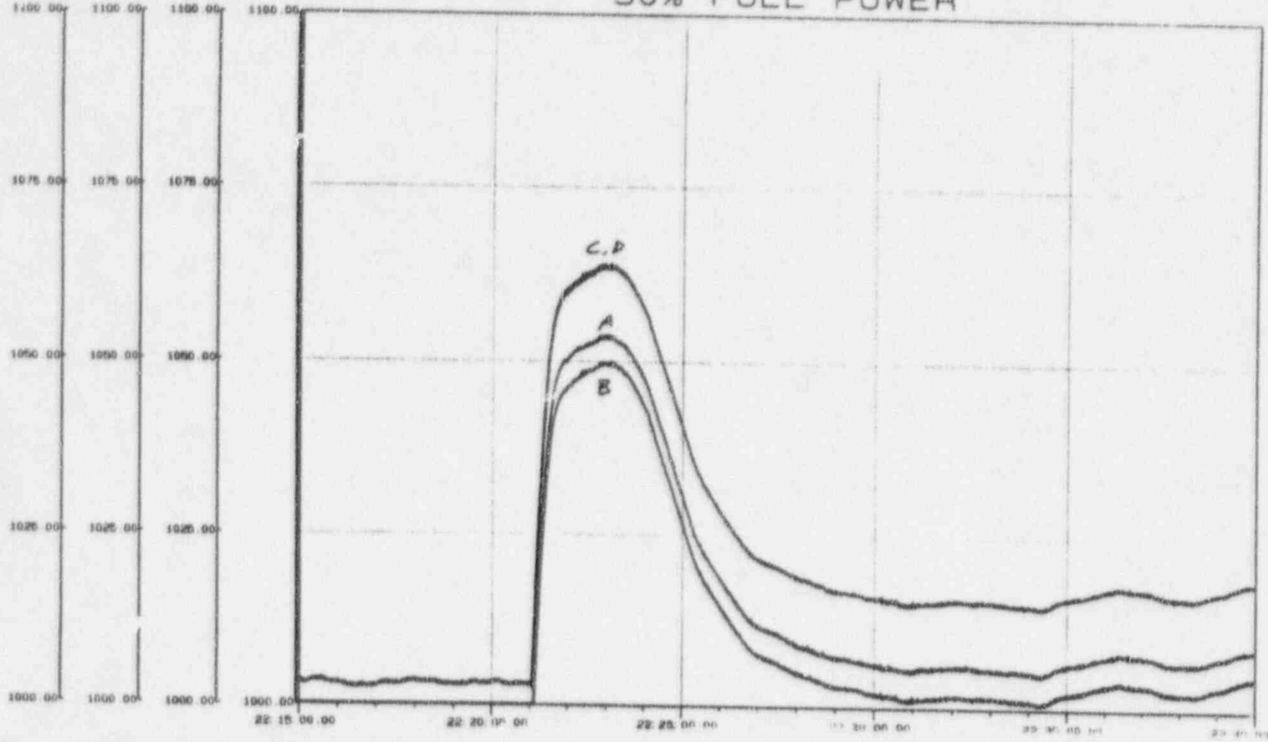
PID	Description	Value at trip time	Units/ Set Meg	Max Value	Min Value
C1A0713	PZR PRESS CH 1	2243.8700	PSIG	2274.4700	2198.8200
C1A0707	PZR LEVEL CH 1	34.2000	%	42.8200	33.2900

70% FULL POWER

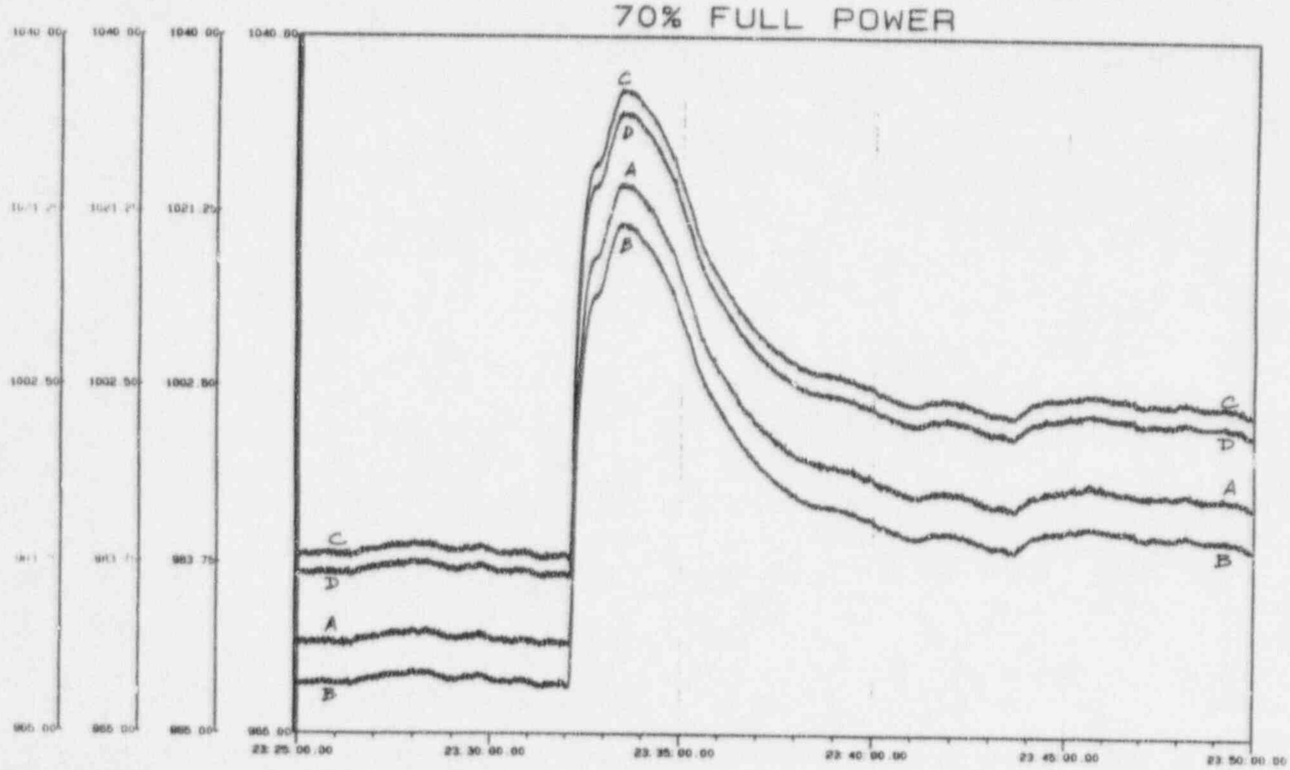


PID	Description	Value at trip time	Units/ Set Meg	Max Value	Min Value
C1A0713	PZR PRESS CH 1	2241.2700	PSIG	2269.2400	2208.2800
C1A0707	PZR LEVEL CH 1	40.0000	%	47.2500	38.9800

FIGURE 8
UNIT LOAD TRANSIENT TEST S/G STEAM PRESSURE



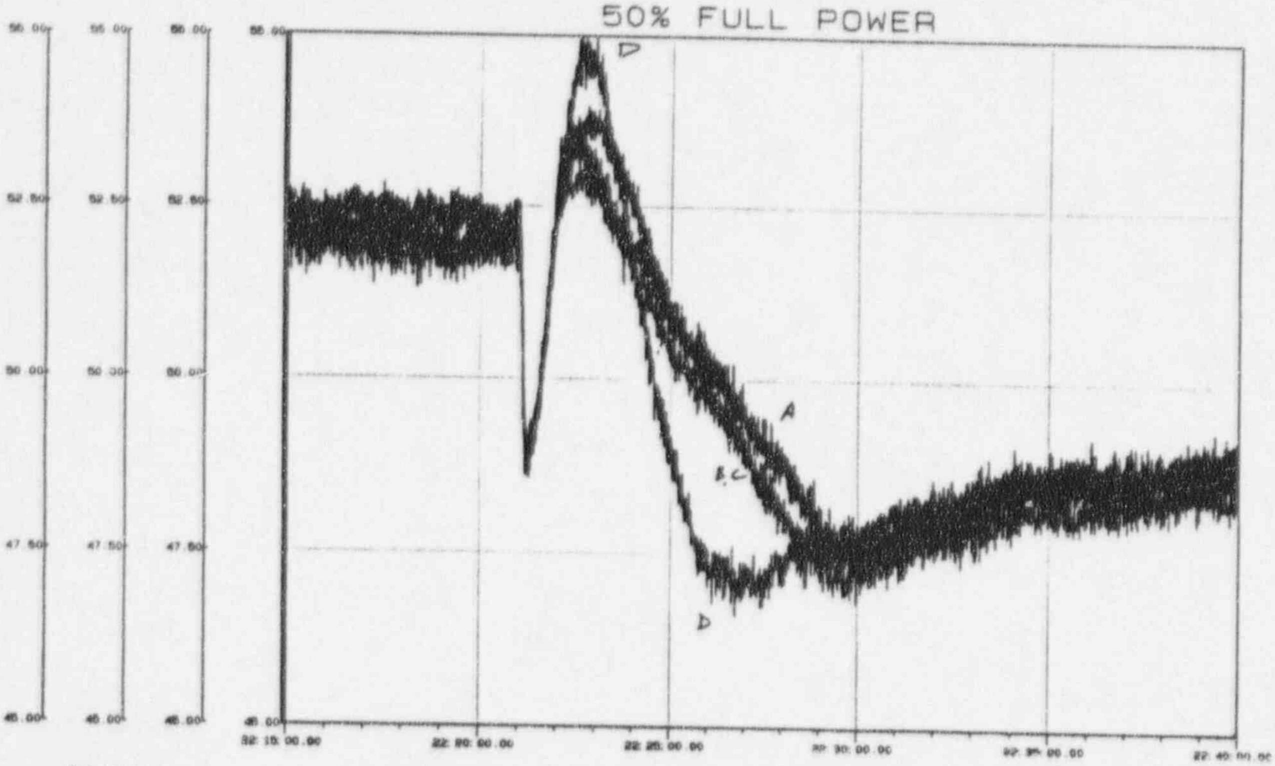
PID	Description	Value at trip time	Units/ Set Msg	Max value	Min value
C1A0723	S/G A STEAM PRESS CH 1	1000.1200	PSIG	1053.8100	992.7100
C1A1286	S/G B STEAM PRESS CH 2	1004.8600	PSIG	1050.0400	988.4500
C1A1304	S/G C STEAM PRESS CH 3	1018.6800	PSIG	1064.1800	1002.4300
C1A1310	S/G D STEAM PRESS CH 2	1018.6800	PSIG	1063.5300	1002.5900



PID	Description	Value at trip time	Units/ Set Msg	Max value	Min value
C1A0723	S/G A STEAM PRESS CH 1	989.9500	PSIG	1024.3700	974.4700
C1A1286	S/G B STEAM PRESS CH 2	985.7200	PSIG	1020.1400	969.8200
C1A1304	S/G C STEAM PRESS CH 3	995.8600	PSIG	1034.4400	984.0600
C1A1310	S/G D STEAM PRESS CH 4	997.4200	PSIG	1032.0000	981.9900

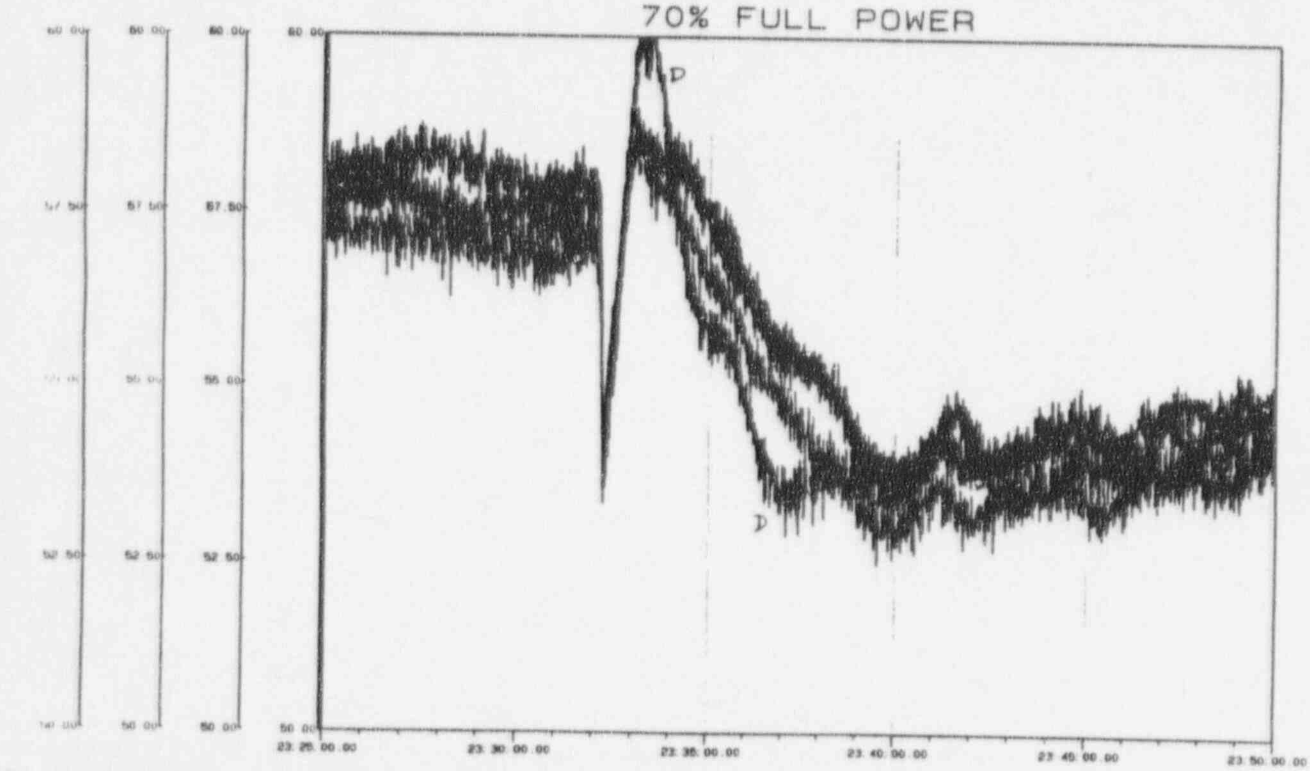
FIGURE 9
UNIT LOAD TRANSIENT TEST S/G NARROW RANGE LEVEL

50% FULL POWER



PID	Description	Value at trip time	Units/ Set Msg	Max Value	Min Value
C1A0531	S/G A NARROW RANGE LEVEL CH 1	49.1200	%	53.8200	47.0700
C1A0630	S/G B NARROW RANGE LEVEL CH 1	48.8500	%	53.9600	47.1800
C1A0633	S/G C NARROW RANGE LEVEL CH 1	48.3300	%	53.2200	46.8000
C1A0549	S/G D NARROW RANGE LEVEL CH 1	48.5200	%	55.1900	46.6200

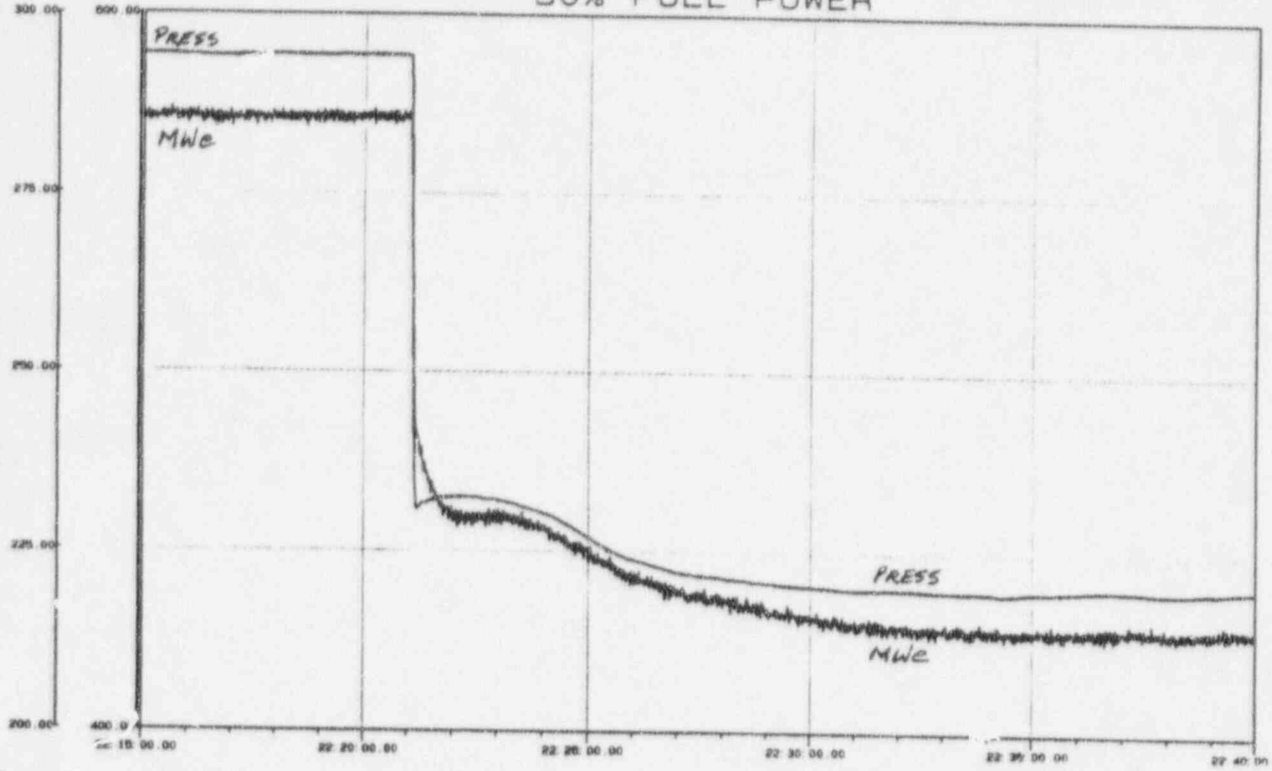
70% FULL POWER



PID	Description	Value at trip time	Units/ Set Msg	Max Value	Min Value
C1A0531	S/G A NARROW RANGE LEVEL CH 1	55.0800	%	59.2300	53.2800
C1A0638	S/G B NARROW RANGE LEVEL CH 1	54.8400	%	58.7400	53.3700
C1A0633	S/G C NARROW RANGE LEVEL CH 1	54.1000	%	58.5800	52.5000
C1A0549	S/G D NARROW RANGE LEVEL CH 1	54.3900	%	60.5000	52.9600

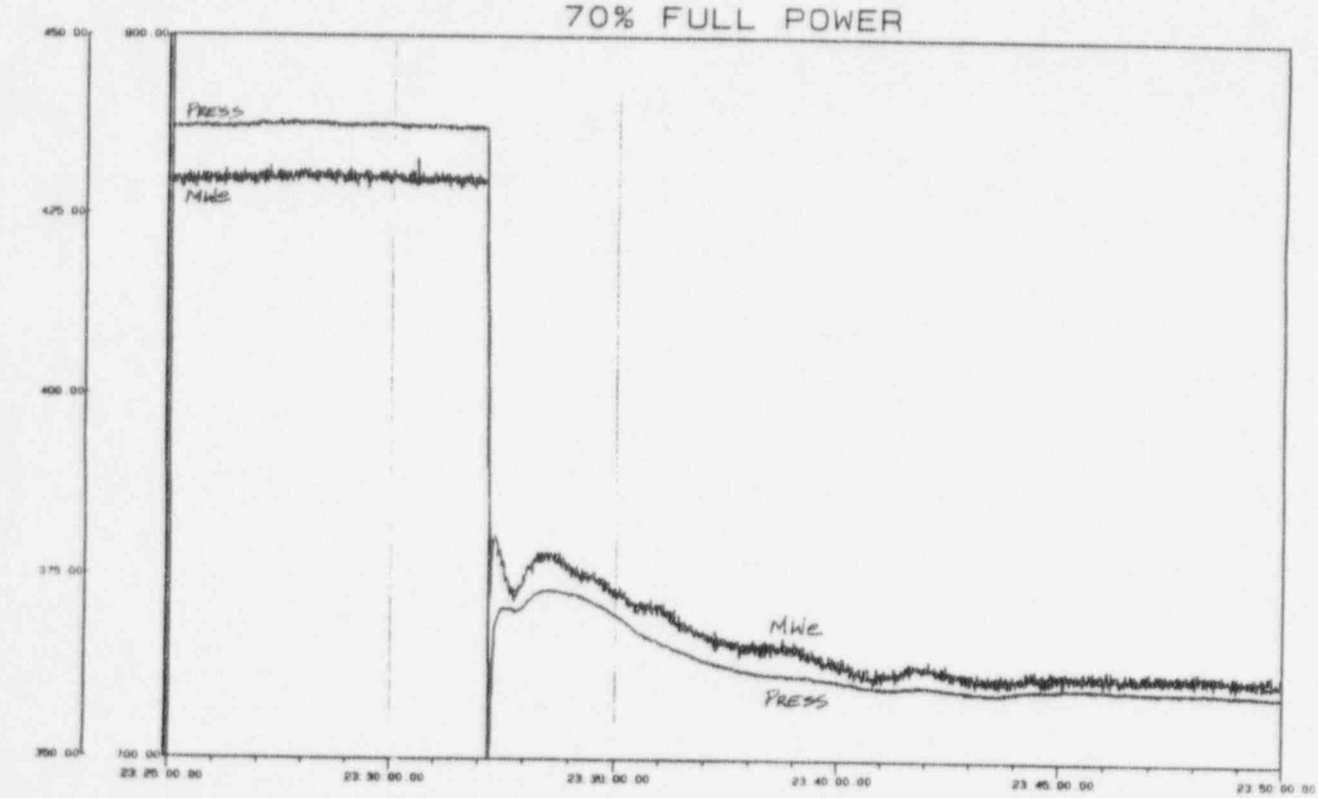
FIGURE 10
UNIT LOAD TRANSIENT TEST GENERATOR MEGAWATTS AND 1ST STAGE PRESSURE

50% FULL POWER



PID	Description	Value at trip time	Units/ Set Msg	Max Value	Min Value
C1A1632	UNIT 1 GENERATOR MEGAWATTS	427.3700	MW	673.9900	426.8400
C1A0737	TURBINE FIRST STAGE PRESS 1	220.1300	PSIG	294.4700	219.3300

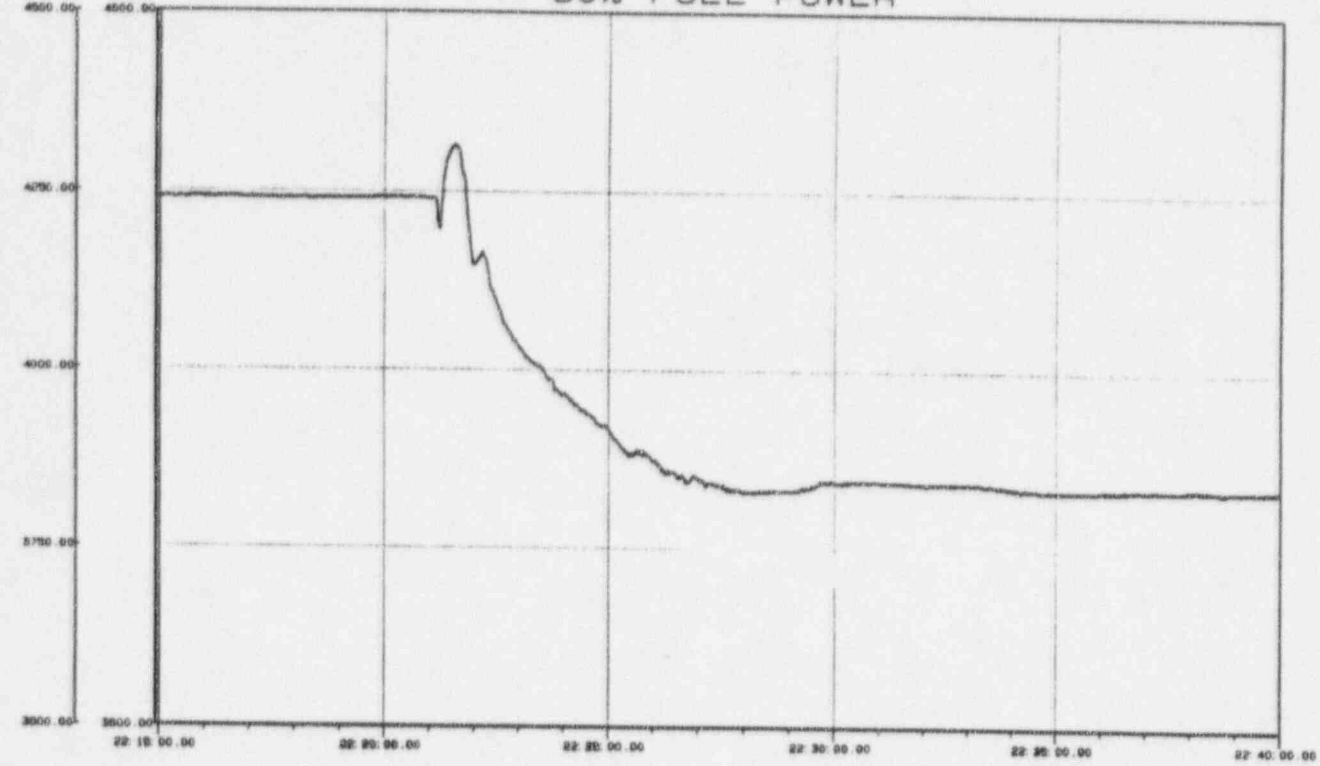
70% FULL POWER



PID	Description	Value at trip time	Units/ Set Msg	Max Value	Min Value
C1A1632	UNIT 1 GENERATOR MEGAWATTS	721.9000	MW	865.9100	720.0100
C1A0737	TURBINE FIRST STAGE PRESS 1	359.1400	PSIG	436.3500	342.9400

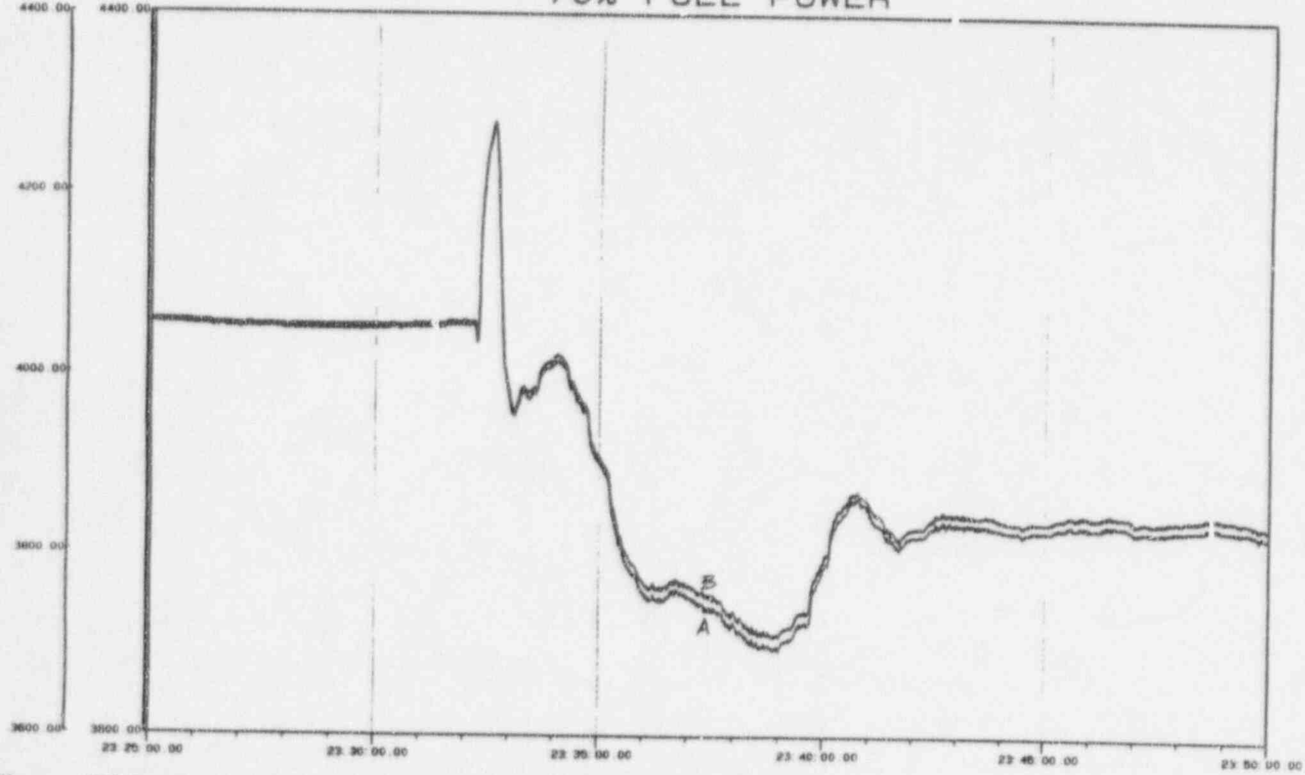
FIGURE 11
UNIT LOAD TRANSIENT TEST CF PUMP TURBINE SPEED

50% FULL POWER



PID	Description	Value at trip time	Units/ Set Mag	Max Value	Min Value
C1A1136	CF PUMP TURB A SPEED	3838.4400	RPM	4318.2400	3829.0300
C1A1137	CF PUMP TURB B SPEED	0.7400	RPM	3.4800	-7.8000

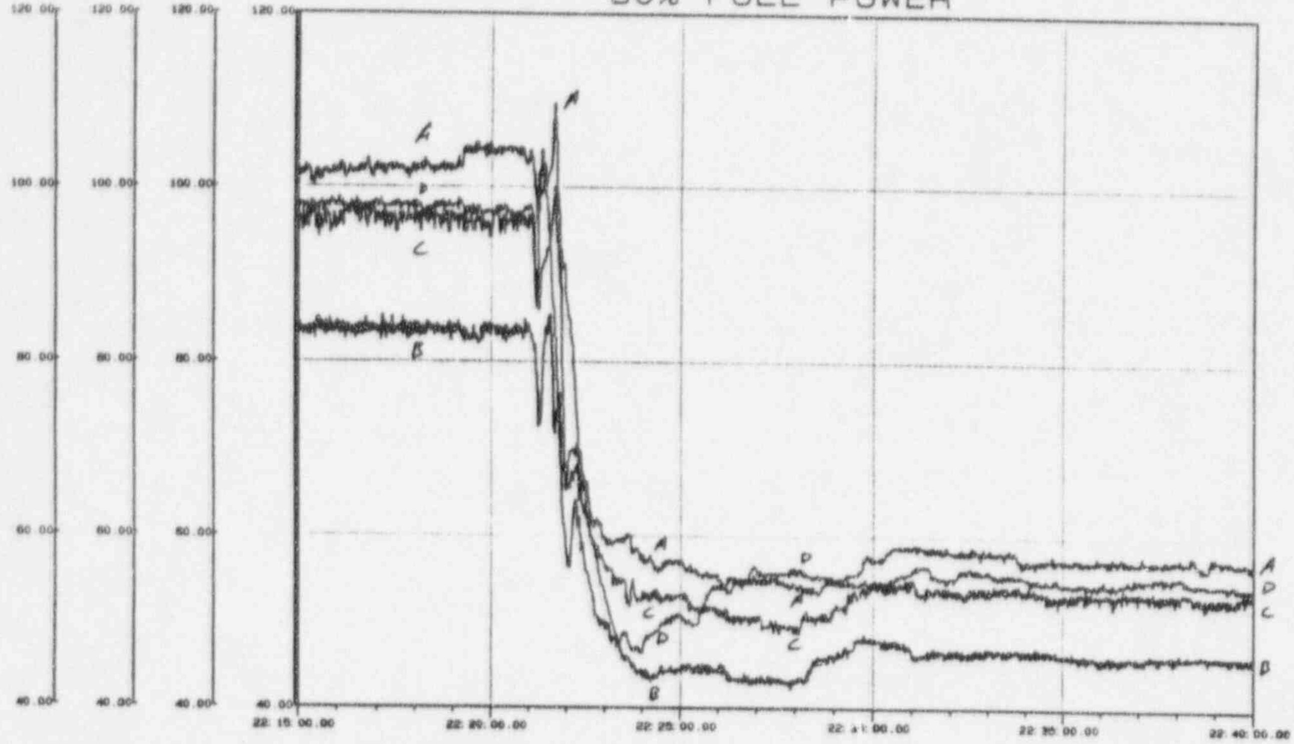
70% FULL POWER



PID	Description	Value at trip time	Units/ Set Mag	Max Value	Min Value
C1A1136	CF PUMP TURB A SPEED	3824.4600	RPM	4281.8500	3694.4400
C1A1137	CF PUMP TURB B SPEED	3832.0300	RPM	4280.4500	3708.1700

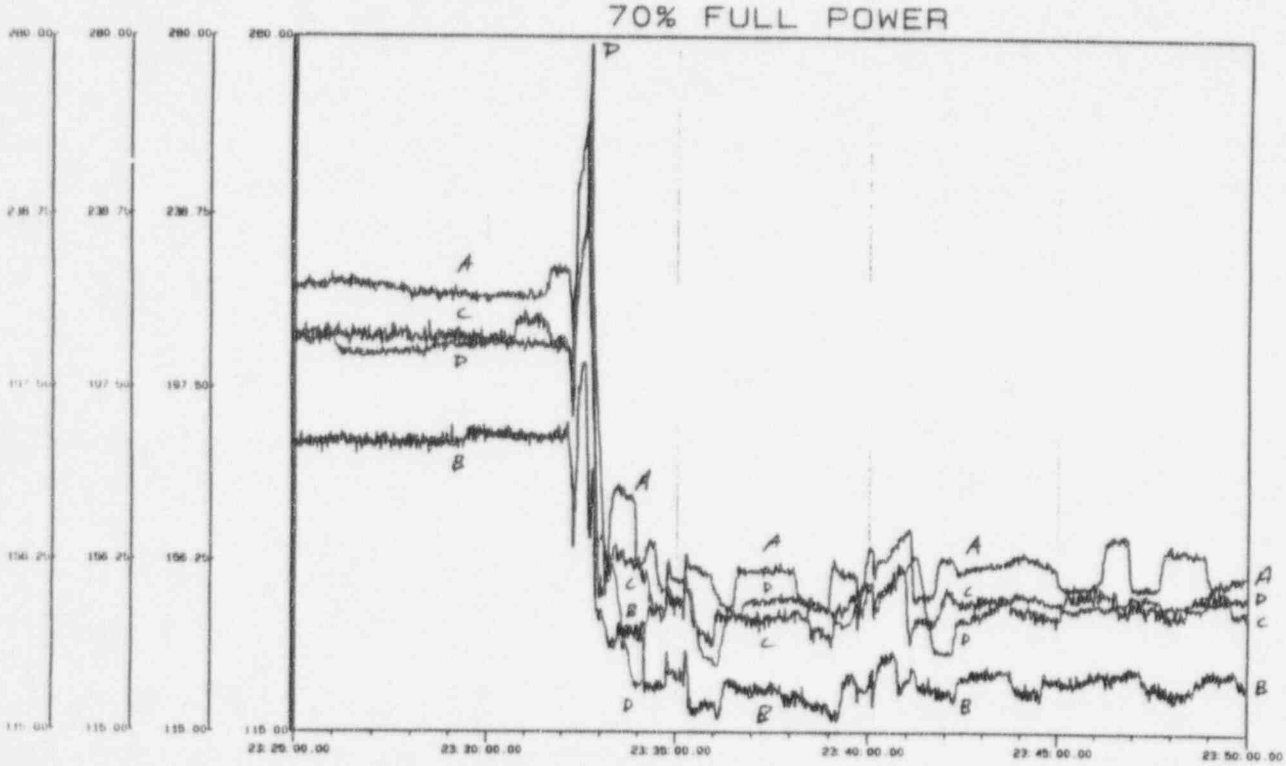
FIGURE 12
UNIT LOAD TRANSIENT TEST S/G FEEDWATER FLOW

50% FULL POWER



PID	Description	Value at trip time	Units/ Set Msg	Max Value	Min Value
C1A0634	S/G A FEEDWATER FLOW CH 1	57.0300	IN H2O	106.6200	53.1700
C1A0646	S/G B FEEDWATER FLOW CH 1	45.7700	IN H2O	85.4300	42.3300
C1A0635	S/G C FEEDWATER FLOW CH 1	53.3200	IN H2O	104.3200	48.6800
C1A0650	S/G D FEEDWATER FLOW CH 1	53.8200	IN H2O	100.1000	46.3700

70% FULL POWER



PID	Description	Value at trip time	Units/ Set Msg	Max Value	Min Value
C140634	S/G A FEEDWATER FLOW CH 1	152.4000	IN H2O	159.8400	143.6000
C140646	S/G B FEEDWATER FLOW CH 1	127.1300	IN H2O	203.5500	117.7400
C140635	S/G C FEEDWATER FLOW CH 1	143.6700	IN H2O	264.7900	136.4400
C140650	S/G D FEEDWATER FLOW CH 1	146.3800	IN H2O	278.1900	125.4700

4.9 Replacement S/G Tuning and Testing of DFCS

TT/1/A/9200/96, Replacement Steam Generator Functional Tuning and Testing of the Feedwater Control System, was performed to record the behavior of S/G Level Controls and the course of action taken to optimally tune the system following replacement of the Westinghouse D3 Steam Generators. Testing was performed prior to Startup (as DFCS was initially placed in AUTO) and at various power level during initial power ascension to allow monitoring of control algorithms of S/G Level Controls and the Feedwater Pump Speed Control System. The Main and Bypass Feedwater Control Valves along with the Feed Pump D/P Program received the dominant scrutiny during this testing. The following discussion summarizes the results of this testing:

Monitoring System Being Placed in Automatic Control

The DFCS was placed in automatic control on September 27, 1996. While the Unit was in mode 3, 'B' Main Feedpump was placed in service. Due to the plant conditions, the pump speed had to be brought up to minimum speed setting before the pump could be placed in auto. When pump discharge pressure was approximately 1070 PSID and FW/STM Hdr D/P at approximately 160 PSID, the pump was placed in automatic control. The Pump went immediately to minimum speed (approx. 2950 RPM) and remained constant. The CF Bypass valves momentarily opened from 8% to 20% to adjust to new D/P and stabilized immediately with no oscillations observed.

Tuning Tests at 10% Power

Steam Generators A through D were subjected to five (5) percent level perturbations on October 3, 1996. These tests were conducted on one generator at a time. A decreasing step change was applied to the narrow range level setpoint followed by an increasing step change of the same magnitude to return the setpoint to its programmed position. The S/G levels responded adequately for the decreasing step change and a slight overshoot of 2% was noticed for the increasing step change. The overshoot was expected due to the magnitude of the change which required the Main Reg. valves to momentarily open to approximately 12%. This test was to validate the response of the system while utilizing the 'Low-Power' controllers. No problems encountered or tuning adjustments were required for this phase of testing.

Monitoring the Placement of Main Turbine On-Line

Placement of the Main Turbine On-Line occurred on October 4, 1996. All S/G levels, valve demands, and Feedwater Pump parameters being monitored for this event. This event caused the S/G levels to decrease approximately 2-3%. The S/G levels leveled off to setpoint evenly with no oscillations occurring. The pump responded accordingly with a slight decrease in the Fw/Stm Hdr D/P and returned to program level with no oscillations. No problems encountered or tuning adjustments were required for this phase of testing.

Tuning Tests at 30% Power

The identical level perturbation tests performed at 10% power were repeated for the 30% plateau testing on October 5, 1996. A sight amount of undershoot and overshoot was observed for these tests which settled out within one cycle. This test was performed to validate the response of the system while utilizing the 'High-Power' controllers. No problems encountered or t

Tuning Tests at 50% Power

Steam Generator A was subjected to five (5) percent level perturbations and the Feedpump was subjected to a 20 PSID increasing/decreasing step change to the Fw/Stm D/P program on October 6, 1996. The level tests were conducted on S/G 'A' and then followed by the Feedpump test.

Prior to the test, it was noticed that the Steam Flows for all generators indicated approximately 20% higher than expected. This error effects the level controls by inducing a steam flow/feed flow mismatch to the S/G flow controllers and offsetting the Fw/Stm D/P program of approximately 20%. The effects of these errors and their implications were discussed to insure valid testing results could be achieved for this plateau. It was decided that the errors would have minimal impact to the test results and testing would continue.

All tests were performed satisfactory with no problems encountered or tuning adjustments required for this power level.

10% Load Rejection Test at 50% Power

The unit was run back approximately 10% on October 7, 1996. All systems performed as expected. It was noticed that 'D' Steam Generator lagged the other three slightly but after the initial swell in the S/G's, all levels returned to program setpoint with no oscillations being noted. No adjustments were required for this test.

10% Load Rejection Test at 70% Power

Prior to the start of testing at this power level, it was determined by the test coordinator that final tuning constants are adequate and will not have to be adjusted. Further level perturbation testing will not be required. The only plant differences for this test was an increase of 20% reactor power and both Main Feed pumps were now in service.

The unit was run back approximately 10% on October 8, 1996. All systems performed as expected. It was noticed that 'D' Steam Generator continued to lag the other three S/G's slightly but after the initial swell, all S/G levels returned to program setpoint with no oscillations being noted. No adjustments were required for this test.

Observations & Conclusions

1) Steam Flows were corrected/normalized at 50% power.

2) Nozzle Swap from CA to CF did not operate properly. It was determined that the stoke time for the CA valves has increased, thus timing out the DFCS timers (60 sec.). Engineering is to pursue the reasons for increased stroke time and/or adjust DFCS internal timers to resolve this problem.

3) The original tuning constants for the D3 Steam Generators were adequate for the Replacement Steam Generators and did not require any adjustment. The level controls responded as good as or better than the old S/G's. Due to this fact, further testing beyond 70% power was unnecessary.

4) S/G 'D' lags the other S/Gs during transients. This is due to piping differences and not considered a problem with the DFCS level control. Adjustments have been made to Unit 2 to resolve these differences and similar changes could be applied to Unit 1 if desired. The differences are small and not recommended at this time.

5) The final position of the Main Reg. valves are approximately 55% open. Adjustment could be made to the Fw/Stm D/P program to allow them to go further open thus reducing speed on the Feedpumps.

4.10 Intermediate Range NIS Setpoint Evaluation

PT/0/A/4150/01, Controlling Procedure for Startup Physics Testing, performs an evaluation of Intermediate Range NIS response in comparison to 20% (Rod Stop) and 25% F.P. (Rx Trip) setpoints preliminarily instated per PT/0/A/4600/05E prior to Unit Startup. This evaluation acquires N35 and N36 indication data as close to 20% and 25% Thermal Power as possible, and then uses it to perform a linear extrapolation to derive expected I/R NIS Channel responses at 31% F.P. The extrapolated channel responses are then compared to the instated 25% F.P. I/R NIS Rx Trip setpoints to ensure that each channel's indication exceeds the corresponding setpoint value at 31% F.P. This verifies that the Rx Trip setpoints have been established conservatively enough to ensure compliance with allowable Tech Spec tolerance on this Reactor Protection function.

In the case of C1C10 Startup, this evaluation indicated that I/R NIS Channel N35 was extrapolated to indicate 68.0 μamps at 31% Thermal Power. This indication was noted to be less than the 78.1 μamps preliminarily incorporated as the 25% F.P. Rx Trip setpoint, resulting in the channel's declared inoperabilty and entry into the Tech Spec Action Item Log with the required Tech Spec action of recalibration of the setpoint at the next available opportunity (Rx Power < 10% F.P., P10 enabled) noted.

The fact that the setpoint was unacceptably non-conservative despite having been derived with an additional 5% of conservatism applied to account for lower Downcomer temperature, was attributable to unanticated assymetry of the radial core power distribution. The 30% F.P. flux map indicated excessive Incore Tilt in Quadrants 2 and 3, meaning that the flux levels in Quadrants 1 and 4 (adjacent to N35) were low enough to cause the channel's response to be deficient with respect to the I/R NIS Rx Trip setpoint.

A Past Operability evaluation concluded that N35's indication at an actual Thermal Power level of 31% did, in fact, exceed the 25% F.P. Rx Trip setpoint. The channel was therefore actually operable at the time the Protection Function was enable (< 10% F.P., P10 enabled) during initial power ascension. Inherent inaccuracy associated with the I/R NIS response extrapolation ultimately lead to N35's inoperabilty.