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Alabama Power
the southern electric system

April 22, 1991

Docket No. 50-364

U. S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, D. C. 20555

Gentlemen:

Joseph M. Farley Nuclear Plant
Unit 2 Cycle 8 - Startup Report

Enclosed is the Startup Report for Unit 2 Cycle 8 as referenced in our Cycle 8 Reload letter dated November 9, 1990.

If you have any questions, please advise.

Respectfully submitted,

J. D. Woodard

JDW/MDR:maf2969

Enclosure

cc: Mr. S. D. Ebneter
Mr. S. T. Hoffman
Mr. G. F. Maxwell

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ALABAMA POWER COMPANY
JOSEPH M. FARLEY NUCLEAR PLANT
UNIT NUMBER 2 CYCLE 8

STARTUP TEST REPORT
(Test Activities From 10-24-90 to 01-22-91)

PREPARED BY THE PLANT REACTOR ENGINEERING GROUP

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APPROVED:

C.D. [signature] 4-8-91 Technical Manager

[signature] General Manager - Nuclear Plant

MM/WSM/STARTRPT

INTRODUCTION

The Joseph M. Farley Unit 2 Cycle 8 Startup Test Report addresses the tests performed as required by plant procedures following core refueling. The report provides a brief synopsis of each test and gives a comparison of measured parameters with design predictions, Technical Specifications, or values in the FSAR safety analysis.

Unit 2 of the Joseph M. Farley Nuclear Plant is a Three Loop Westinghouse pressurized water reactor rated at 2652 MWth. The Unit began commercial operations on July 30, 1981. The Cycle 8 core loading consists of 157 17 x 17 fuel assemblies and has a design burnup capability of 15,900 MWD/MTU.

Cycle Completion Date and Average Burnup

Cycle Number	Completion Date	Avg. Burnup (MWD/MTU)
1	October 22, 1982	15,350.5
2	September 17, 1983	10,371.2
3	January 4, 1985	14,639.0
4	April 4, 1986	13,183.8
5	October 2, 1987	16,674.0
6	March 24, 1989	16,137.8
7	October 13, 1990	17,051.0

2.0 UNIT 2 CYCLE 8 CORE REFUELING

REFERENCES

1. Westinghouse Refueling Procedure FP-APR-R7

2. Westinghouse WCAP 12704 (The Nuclear Design and Management of the Joseph M. Farley Unit 2 Power Plant Cycle 8)

Unloading of the Cycle 7 core into the spent fuel pool commenced on 10-24-90 and was completed on 10-26-90 with no major problems. During the core unload, binocular and TV inspection disclosed that fuel assembly U41, scheduled for reload in the Cycle-8 core, had a gouge-like defect on one rod. As a result, assembly U41 was rejected for reload and the Cycle 8 core loading pattern was redesigned.

During Cycle-7 operation, radiochemistry data indicated that one or more fuel assemblies were leaking radioactivity into the reactor coolant. Therefore, all fuel assemblies unloaded from the Cycle-7 core were subjected to ultrasonic leak testing. One leaking fuel assembly, S44, was identified. However, S44 was not scheduled for reload, so the core design was not impacted. Following insert changeout, the Cycle 8 core reload began on 11-13-90 and was completed on 11-16-90.

The as-loaded Cycle 8 core is shown in Figures 2.1 through 2.4, which give the location of each fuel assembly and insert, including wet annular burnable absorber insert locations and configurations.

FIGURE 2.1: UNIT 2 CYCLE 8 REFERENCE LOADING PATTERN

A	B	C	D	E	F	G	H	J	K	L	M	N	P	R	
						205 U55	283 U51	291 U32							15
				244 U50	R113 W35	238 Y54	R112 W16	217 Y55	R107 W46	250 U45					14
			740 U12	233 Y34	16W970 Y41	R109 W04	16W870 Y42	R122 W48	16W840 Y51	222 Y04	226 U29				13
		274 U24	R103 W61	12W140 Y35	R127 W23	232 W57	SS04 W14	225 W63	R105 W06	12W137 Y31	R111 W50	242 U03			12
	278 U39	209 Y29	12W139 Y15	R116 W12	16W910 Y26	236 W27	16W940 Y12	258 W36	16W990 Y05	R132 W32	12W138 Y32	249 Y07	264 U46		11
	R115 W08	16W880 Y37	R145 W47	16W860 Y27	R148 U42	12W143 Y18	R104 U07	12W142 Y21	R140 U60	16W900 Y36	R106 W43	16W950 Y45	R123 W28		10
262 U23	206 Y43	R134 W38	285 W64	235 W24	12W130 Y22	R143 W05	227 W62	R102 W25	12W133 Y28	221 W07	234 W51	R120 W19	257 Y48	287 U64	9
237 U37	R121 W10	16W790 Y47	212 W09	16W100 Y30	R142 U21	273 W53	251 W39	270 W60	R144 U13	16W850 Y08	230 W29	16W101 Y50	R101 W41	201 U47	8
263 U63	208 Y49	R141 W33	228 W58	211 W49	12W135 Y09	R130 W21	207 W52	R118 W17	12W144 Y23	261 W42	216 W59	R139 W40	213 Y40	290 U22	7
	R128 W13	16W810 Y56	R124 W15	16W800 Y01	R136 U52	12W132 Y13	R146 U16	12W145 Y10	R129 U48	16W920 Y17	R117 W20	16W890 Y38	R137 W22		6
	272 U62	202 Y16	12W131 Y20	R133 W44	16W980 Y19	214 W02	16W780 Y06	210 W37	16W830 Y33	R110 W18	12W141 Y02	219 Y25	239 U40		5
		730 U34	R108 W55	12W136 Y24	R131 W26	256 W54	SS03 W31	231 W56	R119 W30	12W134 Y14	R125 W65	254 U31			4
			296 U25	220 Y11	16W960 Y53	R126 W34	16W930 Y52	R138 W01	16W820 Y46	215 Y03	204 U06				3
				223 U56	R114 W11	218 Y44	R135 W03	224 Y39	R147 W45	293 U44					2
						275 U20	203 U58	229 R21							1

xxx	= INSERT S/N
xxx	= FUEL ASSEMBLY S/N

NORTH

The original w/o U-235 enrichments were:

Region 5 (R) assemblies	3.402%
Region 8A (U) assemblies	3.598%
Region 8B (U) assemblies	3.994%
Region 9A (W) assemblies	3.792%
Region 9B (W) assemblies	4.202%
Region 10A (Y) assemblies ...	3.800%
Region 10B (Y) assemblies ...	4.200%

No. of Fuel Assemblies

Region 5	1
Region 8A	16
Region 8B	19
Region 9A	49
Region 9B	16
Region 10A	36
Region 10B	20
Total	157

FIGURE 2.2

CONTROL ROD LOCATIONS

	R	P	N	M	L	K	J	H	G	F	E	D	C	B	A
1															
2						A		D		A					
3							SA		SA		SP				
4				C		B		SP		B		C			
5			SP		SB		SP				SB				
6		A		B		D		C		D		B		A	
7			SA				SB		SB		SP		SA		
8 90°		D		SP		C		SP		C		SP		D	
9			SA		SP		SB		SB				SA		
10		A		B		D		C		D		B		A	
11					SB				SP		SB		SP		
12				C		B		SP		B		C			
13					SP		SA		SA						
14						A		D		A					
15															

0°

Absorber Material:
Ag-In-Cd

FUNCTION	NUMBER OF CLUSTERS
Control Bank D	8
Control Bank C	8
Control Bank B	8
Control Bank A	8
Shutdown Bank SB	8
Shutdown Bank SA	8
SP (Spare Rod Locations)	13

FIGURE 2.3

BURNABLE ABSORBER AND SOURCE ASSEMBLY LOCATIONS

	R	P	N	M	L	K	J	H	G	F	E	D	C	B	A
1															
2															
3						16		16		16					
4					12			SS			12				
5				12		16		16		16		12			
6			16		16		12		12		16		16		
7						12				12					
8			16		16						16		16		
9						12				12					
10			16		16		12		12		16		16		
11				12		16		16		16		12			
12					12			SS			12				
13						16		16		16					
14															
15															

##	Number of WABAs
SS	Secondary Source

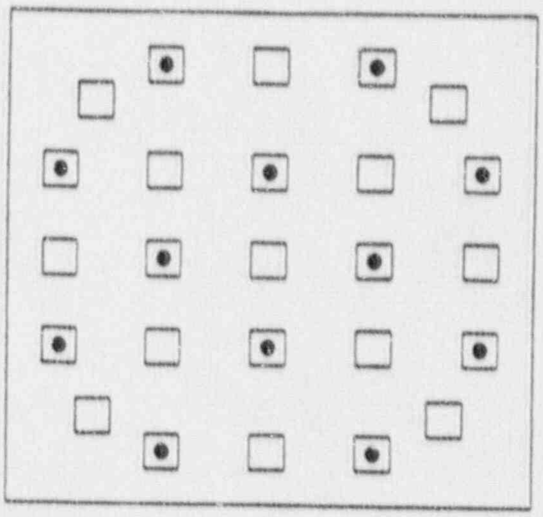
576 WABAs in
40 Clusters

Summary of Inserts

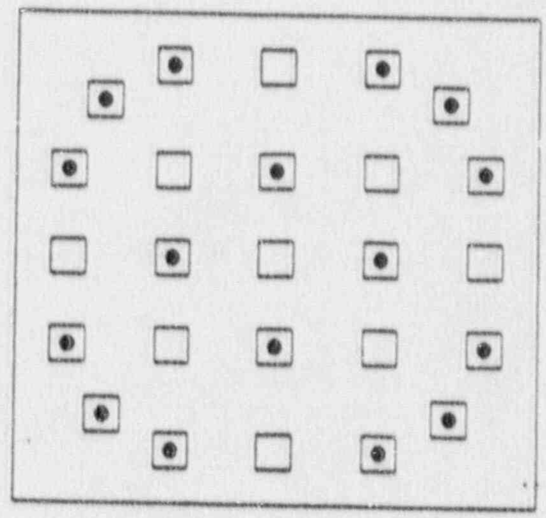
WABA Clusters	40
Control Rods	48
Thimble Plugs	67
Sec. Sources	2
Total	157

FIGURE 2.4

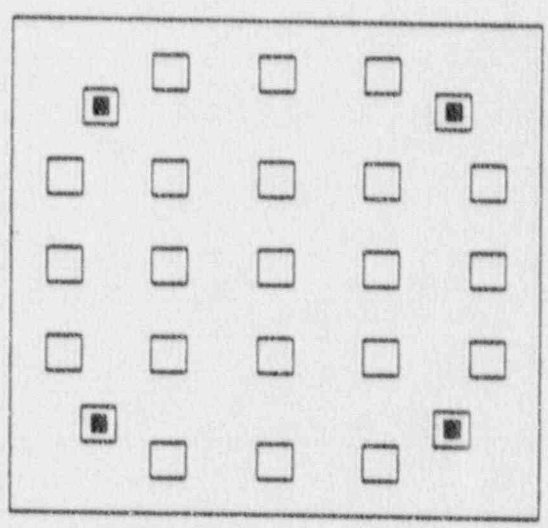
BURNABLE ABSORBER AND SECONDARY SOURCE ROD CONFIGURATIONS



12 BA CONFIGURATION



16 BA CONFIGURATION



SECONDARY SOURCE LOCATIONS

3.0 CONTROL ROD DROP TIME MEASUREMENT (FNP-2-STP-112)

PURPOSE

The purpose of this procedure was to measure the drop time of all full length control rods under hot-full flow conditions in the reactor coolant system to insure compliance with Technical Specification Requirements.

SUMMARY OF RESULTS

For the hot full-flow condition (Tavg $\geq$ 541 deg.F and all reactor coolant pumps operating) Technical Specification 3.1.3.4 requires that the drop time from the fully withdrawn position shall be $\leq$ 2.2 seconds from the beginning of stationary gripper coil voltage decay until dashpot entry. All full length rod drop times were measured to be less than 2.2 seconds. The longest drop time recorded was 1.48 seconds for rod B-6. The rod drop time results for both dashpot entry and dashpot bottom are presented in Figure 3.1. Mean drop times are summarized below:

TEST CONDITIONS	MEAN TIME TO DASHPOT ENTRY	MEAN TIME TO DASHPOT BOTTOM
Hot full-flow	1.374 sec	1.829 sec

To confirm normal rod mechanism operation prior to conducting the rod drop test, the Verification of Rod Control System Operability (FNP-0-ETP-3643) was performed. In this test, the stepping waveforms of the stationary, lift and movable gripper coils were examined, rod speed was measured, and the functioning of the Digital Rod Position Indicator (DRPI) and bank overlap unit were checked.

As a part of the RCCA wear reduction program, the Cycle 8 control rod fully-withdrawn position was changed from 228 steps to 231 steps by increasing bank overlap from 100 steps to 103 steps. During the Rod Control System Operability test, the bank overlap unit switch settings and functions were verified correct for 103 steps of bank overlap. In addition, the actual fully-withdrawn position of each RCCA was measured using stationary gripper coil traces to provide data for planning RCCA repositioning for future fuel cycles.

Initially, during the Rod Control System Operability test, none of the Group-2 rods would move. Since half of the Group-2 rods are driven by the 1AC power cabinet, and the other half by the 2BD power cabinet, this suggested that a problem existed in both cabinets. Upon investigation, one of the power fuses on top of the 2AC power cabinet was discovered to be blown and was replaced. Although the specific problem with the 2BD power cabinet was never identified, the cabinet became operational following the removal and reinsertion of several circuit cards during the course of troubleshooting. Thus, the problem was attributed to a loose circuit card connection.

UNIT 2 CYCLE 8

NORTH

90°

0° 180°

270°

	15	14	13	12	11	10	9	8	7	6	5	4	3	2	1
R															
P						1.38 1.81		1.37 1.83		1.41 1.86					
N							1.38 1.83		1.37 1.8						
M				1.37 1.8		1.37 1.83				1.35 1.8		1.39 1.83			
L					1.38 1.83						1.35 1.79				
K		1.47 1.91		1.35 1.8		1.33 1.85		1.32 1.85		1.34 1.83		1.36 1.78		1.40 1.85	
J			1.37 1.79				1.34 1.77		1.33 1.78				1.38 1.81		
H		1.41 1.86				1.35 1.87				1.35 1.87				1.38 1.81	
G			1.37 1.79				1.32 1.77		1.37 1.78				1.40 1.85		
F		1.41 1.83		1.40 1.85		1.36 1.85		1.33 1.81		1.37 1.87		1.38 1.81		1.37 1.83	
E					1.35 1.8						1.38 1.87				
D				1.38 1.8		1.36 1.81				1.36 1.79		1.37 1.83			
C							1.40 1.86		1.38 1.83						
B						1.40 1.88		1.43 1.88		1.48 1.97					
A															

DRIVE LINE "DROP TIME" TABULATION

TEMPERATURE - 548.27 PRESSURE - 2291.021 % FLOW - 100

X.XX BREAKER "OPENING" TO DASHPOT ENTRY - IN SECONDS
X.XX BREAKER "OPENING" TO DASHPOT BOTTOM - IN SECONDS

DATE - 1-1-91

FIG. 3.1

During the Rod Drop Time Measurement, the same reversal was noted in the rod D12 rod drop trace that had been observed previously during the Cycle-7 startup . Although no anomalies were observed in the stepping traces for rod D12 during the Rod Operability Test, the traces were re-examined and were confirmed to be normal. It is believed that the magnetic polarity of the rod D12 drive shaft was reversed, causing its rod drop trace to be reversed with respect to the remaining rod drop traces. The drop time of rod D12 was normal (1.38 seconds to dashpot entry).

4.0 INITIAL CRITICALITY (FNP-2-ETP-3601)

PURPOSE

The purpose of this procedure was to achieve initial criticality under carefully controlled conditions, establish the upper flux limit for the conduct of zero power physics tests, and operationally verify the calibration of the reactivity computer.

SUMMARY OF RESULTS

Initial Reactor Criticality for Cycle 8 was achieved during dilution mixing at 1816 hours on January 3, 1991. The reactor was allowed to stabilize at the following conditions:

RCS Pressure	2230 psig
RCS Temperature	547.0°F
Intermediate Range Power	1.5×10^{-8} Amp
RCS Boron Concentration	1893.5 ppm
Bank D Position	186.5 steps

Following stabilization, the point of adding nuclear heat was determined and a checkout of the reactivity computer using positive and negative flux periods was performed. In addition, NIS source and intermediate range overlap data were taken during the flux increase prior to criticality to demonstrate that adequate overlap existed.

Initial criticality for Cycle 8 was achieved with the flux in the source range as a result of withdrawing the source-range (SR) detectors to the maintenance position to reduce the SR count rate. During previous cycles, the high SR count rate resulted in criticality being achieved in the intermediate range. Testing performed prior to criticality (FNP-0-ETP-3652) demonstrated that withdrawal of the SR detectors reduced the count rate to one-third of the value with the detectors in their normal, inserted position.

5.0 ALL-RODS-OUT ISOTHERMAL TEMPERATURE COEFFICIENT AND BORON ENDPOINT (FNP-2-ETP-3601)

PURPOSE

The objectives of these measurements were to determine the hot, zero power isothermal and moderator temperature coefficients for the all-rods-out (ARO) configuration and to measure the ARO boron endpoint concentration.

SUMMARY OF RESULTS

The ARO, hot zero power temperature coefficients and the ARO boron endpoint concentration are tabulated below. As described in Par. 6.0, these data have been adjusted to correct for an error in the delayed neutron data provided for reactivity computer calibration.

ARO, HZP ISOTHERMAL AND MODERATOR TEMPERATURE COEFFICIENT

Rod Configuration	Boron Conc. ppm	Measured ITC pcm/°F	ITC Design Acc. Criterion pcm/°F	Calculated MTC pcm/°F
All Rods Out	1913.7	-1.13	-0.38 ± 2	+.694

ITC = Isothermal temperature coefficient, includes -1.72 pcm/°F doppler coefficient

MTC = Moderator only temperature coefficient, normalized to the ARO Condition

ARO, HZP BORON ENDPOINT CONCENTRATION

Rod Configuration	Measured C_B (ppm)	Design-predicted C_B (ppm)
All Rods Out	1913.7	1955 ± 50

Since the measured MTC (0.694 pcm/°F) was well within the Technical Specification limit of +5.0 pcm/°F, no rod withdrawal limits were required. The design acceptance criterion for the ARO boron concentration also was satisfied.

6.0 CONTROL AND SHUTDOWN BANK WORTH MEASUREMENTS (FNP-2-ETP-3601)

PURPOSE

The objective of the bank worth measurements was to determine the integral reactivity worth of each control and shutdown bank for comparison with the values predicted by design.

SUMMARY OF RESULTS

The rod worth measurements were performed using the bank interchange method in which: (1) the worth of the bank having the highest design worth (the "Reference Bank") is carefully measured using the standard dilution method; then (2) the worths of the remaining control and shutdown banks are derived from the change in reference bank reactivity needed to offset full insertion of the bank being measured.

The initial results of the reference bank measurement were approximately 10% high. During the review and evaluation process, an error was discovered in the delayed neutron data provided in Ref. 2 for calibration of the reactivity computer. In order to correct this

error, Westinghouse indicated that all reactivity computer measurements (including the boron endpoint and isothermal temperature coefficient measurements outlined in Par. 5.0) should be reduced by a factor of 0.9055. The adjusted control and shutdown bank worths (tabulated below) satisfied the review criteria both for the banks measured individually and for the combined worth of all banks.

SUMMARY OF CONTROL AND SHUTDOWN BANK WORTH MEASURFMENTS

Control or Shutdown Bank	Predicted Bank Worth & Review Criteria (pcm)	Measured Bank Worth (pcm)	Percent Difference
A	335 ± 100	326.84	-2.44
B (Ref.)	1168 ± 117	1169.7*	+0.15
C	848 ± 127	849.65	+0.19
D	940 ± 141	923.06	-1.80
SDA	908 ± 136	870.16	-4.17
SDB	1070 ± 160	1085.17	+1.42
Combined	5269 ± 526.9	5224.58	-0.84

*Measured by dilution method

7.0 STARTUP AND POWER ASCENSION PROCEDURE (FNP-2-ETP-3605)

PURPOSE

The purpose of this procedure was to provide controlling instructions for:

1. NIS intermediate and power range setpoint changes, as required prior to startup and during power ascension.

2. Ramp rate limitation and control rod movement recommendations.

3. Conduct of startup and power ascension testing, to include:

   a. Hot zero power (HZP) physics tests (FNP-2-ETP-3601).

   b. Incore movable detector system alignment (FNP-2-ETP-3606).

   c. Incore-excore AFD channel recalibration (FNP-2-STP-121).

   d. Core hot channel factor surveillance (FNP-2-STP-110).

   e. Reactor coolant system flow measurement (FNP-2-STP-115.1).

SUMMARY OF RESULTS

In order to satisfy Technical Specification requirements for invoking special HZP physics test exceptions, preliminary trip setpoints of less than or equal to 25% power were used for the NIS intermediate and power range channels. In addition, the intermediate range

channel setpoint currents were reduced to address the effects of SR/IR detector repositioning (described in Section 4.0) and projected changes in core neutron leakage from the previous core cycle. When physics tests were completed,the power range setpoint was increased to 80% to allow power escalation above 25% for calorimetric recalibration of the power range channels. (The 80% setpoint was administratively imposed to address the possibility that the uncalibrated power range channels could be indicating non-conservatively.) At approximately 34% power, the power range channels were recalibrated and setpoint currents were determined for the intermediate range channels. Following recalibration of the power range channels, the NIS PR high-range trip setpoint was restored to 109%.

The Westinghouse fuel warranty limits the power ramp rate to 3% of full power per hour between 20% and 100% power until full power has been sustained for 72 cumulative hours out of any seven-day operating period. This ramp rate was observed during the ascension to full power.

The determination of incore movable detector system core limit settings (FNP-2-ETP-3606) was accomplished during the ascension to 34% power. This was followed by the incore-excore recalibration test (FNP-2-STP-121) at 34% power and the reactor coolant flow measurement (FNP-2-STP-115.1) at 100% power, which are described in Sections 8.0 and 9.0 of this report. As summarized in Table 7.1, core hot channel factor surveillance was performed on the incore-excore full-core base case flux map taken under non-equilibrium conditions at 34% power, and on the full-core flux maps performed at equilibrium xenon at 33.6% and 100% power.

TABLE 7.1

SUMMARY OF POWER ASCENSION FLUX MAP DATA

Parameter	Map 196	Map 202	Map 203
Avg. % power	34.0%	33.6%	100.2%
Max FDH	1.5577	1.5873	1.4783
Max power tilt*	1.0007	1.0007	1.0005
Core avg. % A.O.	10.468	10.885	1.742
Limiting FQ(Z)**	2.1390	2.2195	1.8508
FQ Limit	4.5240	4.5240	2.3084
Xenon conditions	Non-Equilibrium	Equilibrium	Equilibrium

*Calculated power tilts based on assembly FDHN from all assemblies.

**The most limiting FQ(Z), based on margin to the FQ limit.

8.0 INCORE-EXCORE DETECTOR CALIBRATION (FNP-2-STP-121)

PURPOSE

The objective of this procedure was to determine the relationship between power range upper and lower excore detector currents and axial offset for the purpose of calibrating the control board and the plant computer axial flux difference (AFD) channels, and for calibrating the delta flux penalty to the overtemperature delta-T protection system.

SUMMARY OF RESULTS

At an indicated power of approximately 34%, a full-core base case flux map and five quarter-core flux maps were performed at various positive and negative axial offsets to develop equations relating detector current to core axial offset. To reduce error, an effort was made to perform all flux maps at the same reactor coolant system temperature. The power range NIS channels were adjusted to incorporate the revised calibration data.

Following the recalibration, escalation to full power proceeded without incident. At 100% power, under equilibrium xenon conditions, a full-core flux map was performed to correct the incore-excore calibration for the effects of the rise in RCS average temperature as power is increased. Table 8.1 gives the final detector current vs axial offset equations obtained following the calibration correction at 100& power.

TABLE 8.1

DETECTOR CURRENT VERSUS AXIAL OFFSET EQUATIONS
OBTAIN FROM INCORE-EXCORE CALIBRATION TEST

CHANNEL N41:

I-Top	=	0.8136*AO	+	156.91 uA
I-Bottom	=	-0.9486*AO	+	152.87 uA

CHANNEL N42:

I-Top	=	0.8266*AO	+	157.63 uA
I-Bottom	=	-1.0001*AO	+	152.59 uA

CHANNEL N43:

I-Top	=	0.8258*AO	+	160.81 uA
I-Bottom	=	-1.0066*AO	+	155.46 uA

CHANNEL N44:

I-Top	=	0.9208*AO	+	173.93 uA
I-Bottom	=	-1.1521*AO	+	176.06 uA

9.0 REACTOR COOLANT SYSTEM FLOW MEASUREMENT (FNP-2-STP-115.1)

PURPOSE

The purpose of this procedure was to measure the flow rate in each reactor coolant loop in order to confirm that the total core flow met the minimum flow requirement given in the Unit 2 Technical Specifications.

SUMMARY OF RESULTS

To comply with the Unit 2 Technical Specification, the total reactor coolant system flow rate measured at normal operating temperature and pressure must equal or exceed 261,600 gpm for three loop operation. From the average of twelve calorimetric heat balance measurements, the total core flow was determined to be 279,780 gpm, which meets above the criterion.

The RCS flow test data (which include direct RTD measurements of RCS T-hot, T-cold and percent thermal power) were also used to determine delta-T for each RCS loop in order to evaluate the need to rescale the delta-T protection and control systems. Based on the results of this measurement, Loops 2 and 3 were rescaled.