ALABAMA POWER COMPANY

JOSEPH M. FARLEY NUCLEAR PLANT

UNIT NUMBER 2, CYCLE 5

STARTUP TEST REPORT

PREPARED BY PLANT REACTOR ENGINEERING GROUP

APPROVED:

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DISK: CYCLE 2/7

IE26

1/1

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1.0 INTRODUCTION

The Joseph M. Farley Unit 2 Cycle 5 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 assumed 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 Cycle 5 core loading consists of 157 17 x 17 fuel assemblies.

The Unit began commercial operations on July 30, 1981, completed Cycle 1 on October 22, 1982 with an average core burnup of 15350.5 MWD/MTU, completed Cycle 2 on September 17, 1983 with an average core burnup of 10371.2 MWD/MTU, completed Cycle 3 on January 4, 1985 with an average core burnup of 14,639.0 MWD/MTU, and completed Cycle 4 on April 4, 1986 with average core burnup of 13,183.8 MWD/MTU.

2.0 UNIT 2 CYCLE 5 CORE REFUELING

REFERENCES

1. Westinghouse Refueling Procedure FP-APR-R4
2. Westinghouse WCAP 11150 (The Nuclear Design and Core Management of the Joseph M. Farley Unit 2 Power Plant Cycle 5)

The refueling commenced on 4/14/86 and was completed in 6 days on 4/20/86. The as-loaded Cycle 5 core is shown in Figures 2.1 through 2.4, which give the location of each fuel assembly and insert, including the burnable poison insert locations and configurations. The Cycle 5 core has a nominal design lifetime of 16600 MWD/MTU and consists of 1 region 4 assembly, 16 region 5 assemblies, 72 region 6 assemblies and 68 region 7 assemblies. Fuel assembly inserts include 48 full length control rod clusters, 68 burnable poison inserts, two secondary sources, and 39 thimble plug inserts.

FIGURE 2.1

APR Unit 2, Cycle 5 Reference Loading Pattern

	A	B	C	D	E	F	G	H	J	K	L	M	N	P	R
15							74D S29	203 S59	296 S44						
14					201 R35	R123 S30	4P35D T59	R105 S47	4P34D T44	R131 S55	218 R37				
13				249 R05	4P22D T61	12P153 T18 D	R128 S23	12P143 T31 1D	R146 S54	12P145 T35 D	4P21D T58	283 R61			
12			262 R53	R127 S21	16P103 T34 D	R147 S43	16P115 T60 1D	SS 4 S18	24P06 T22	R126 S34	16P112 T41 D	R124 S71	274 R63		
11		214 R55	4P25D T24	16P101 T21 D	R119 S20	16P109 T55 1D	242 S27	16P106 T47 1D	226 S50	12P152 T14 1D	R133 S39	16P117 T08 D	4P24D T10	217 R28	
10		R122 S49	12P158 T45 D	R125 S63	16P110 T56 1D	R136 S24	12P157 T25 1D	R148 S65	12P156 T23 1D	R102 S26	24P05 T20	R110 S45	12P148 T04 D	R113 S12	
9	256 S37	4P32D T29	R112 S53	16P116 T64 1D	278 S40	24P08 T33	R134 S38	12P142 T26 1D	R139 S08	24P04 T19	275 S32	24P03 T17	R104 S35	4P36D T67	270 S16
8	239 S57	R129 S02	12P147 T65 1D	207 S72	12P160 T30 1D	R132 S22	12P146 T32 1D	73D P42	20P30D T51	R101 S07	16P107 T49 1D	210 S09	20P32D T13	R137 S52	285 S17
7	216 S56	4P33D T36	R117 S41	12P161 T37 1D	237 S69	24P07 T27	R109 S70	20P29D T48	R103 S36	12P159 T28 1D	261 S25	16P108 T53 1D	R144 S58	4P29D T62	232 S60
6		R111 S42	12P144 T39 D	R114 S28	24P01 T07	R120 S01	16P105 T46 1D	R118 S61	16P102 T42 1D	R143 S48	16P100 T38 1D	R116 S64	12P154 T57 D	R145 S10	
5		223 R62	4P26D T16	16P114 T54 D	R108 S14	24P02 T15	260 S19	16P118 T68 1D	291 S04	12P149 T02 1D	R106 S15	16P111 T40 D	4P23D T01	202 R19	
4			244 R33	R130 S11	16P113 T52 D	R115 S67	12P150 T11 1D	SS 3 S03	16P104 T43 1D	R140 S13	16P119 T06 D	R142 S06	272 R56		
3				211 R03	4P27D T12	12P155 T03 D	R121 S31	20P31D T66	R107 S62	12P151 T05 D	4P28D T50	269 R40			
2					264 R44	R141 S05	4P31D T09	R135 S46	4P30D T63	R138 S33	233 R57				
1							250 S68	257 S51	235 S66						

0
270 — 90
180

0 0 0 0

NORTH

Fuel assembly serial numbers are given in the lower (or lower left) portion of each location. The remaining numbers denote the assembly insert. The original w/o U-235 enrichments were:

Region 4 (P) assemblies 3.096%
Region 5 (R) assemblies 3.402%
Region 6 (S) assemblies 3.443%
Region 7 (T) assemblies 3.603%

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		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															

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

LOCATIONS N5 & C11 = CORE WATER LEVEL THERMOCOUPLE PROBES

FIGURE 2.3

BURNABLE ABSORBER AND SOURCE ASSEMBLY LOCATIONS

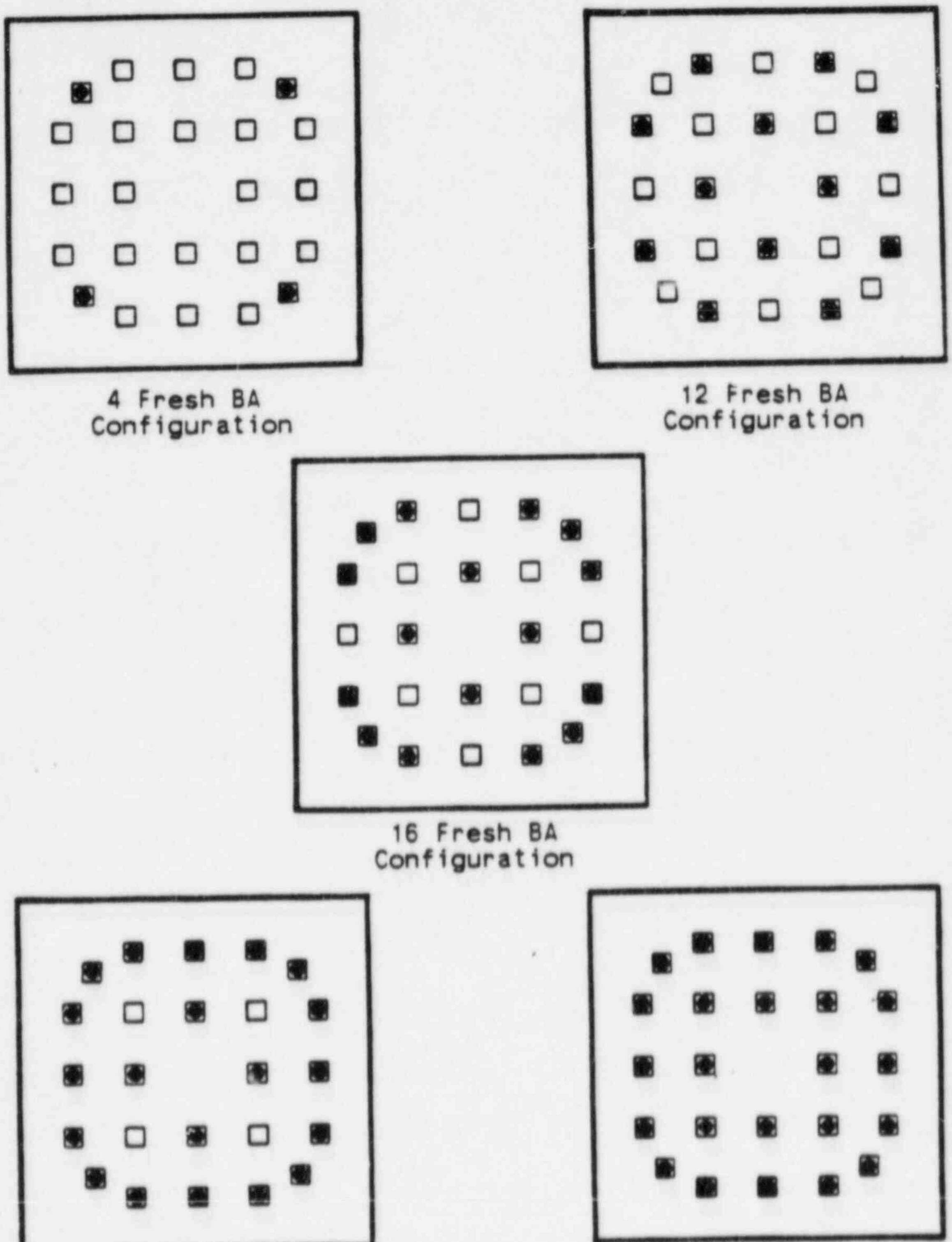
	R	P	N	M	L	K	J	H	G	F	E	D	C	B	A
1															
2							4		4						
3					4	12		20		12	4				
4					16		24	SS	24		16				
5			4	16		24		24		24		16	4		
6			12		24		24		24		24		12		
7		4		24		24		20		24		24		4	
8			20		24		20		20		24		20		
9		4		24		24		20		24		24		4	
10			12		24		24		24		24		12		
11			4	16		24		24		24		16	4		
12					16		24	SS	24		16				
13					4	12		20		12	4				
14							4		4						
15															

SS Secondary Source

1120 Fresh Standard BA's

FIGURE 2.4

BURNABLE ABSORBER CONFIGURATIONS



4 Fresh BA
Configuration

12 Fresh BA
Configuration

16 Fresh BA
Configuration

20 Fresh BA
Configuration

24 Fresh BA
Configuration

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

PURPOSE

The purpose of this test 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 ($T_{avg} \geq 541°F$ and all reactor coolant pumps operating) Technical Specification 3.1.3.4 requires that the rod drop time from the fully withdrawn position shall be < 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 1.49 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	1.37 sec	1.86 sec

To confirm normal rod mechanism operation prior to conducting the rod drops, a Control Rod Drive Test (FNP-0-ETP-3643) was performed. In the test, the stepping waveforms of the stationary, lift and moveable gripper coils were examined, and the functioning of the Digital Rod position indicator and the bank overlap unit was checked. Rod stepping speed measurements were also conducted. All results were satisfactory.

NORTH

UNIT 2 CYCLE 5

	15	14	13	12	11	10	9	8	7	6	5	4	3	2	1
R								90°							
P						1.36 1.85		1.38 1.88		1.41 1.90					
N							1.37 1.87		1.36 1.82						
M				1.37 1.85		1.37 1.85				1.36 1.82		1.38 1.85			
L					1.38 1.88						1.34 1.83				
K		1.48 1.99		1.35 1.83		1.35 1.84		1.35 1.84		1.35 1.83		1.36 1.85		1.40 1.90	
J			1.36 1.86				1.33 1.81		1.30 1.80				1.36 1.82		
H	0°	1.41 1.91				1.38 1.85				1.37 1.85				1.38 1.87	180°
G			1.34 1.84				1.32 1.82		1.34 1.78				1.39 1.88		
F		1.40 1.88		1.38 1.87		1.36 1.82		1.43 1.95		1.36 1.87		1.38 1.88		1.37 1.83	
E					1.36 1.85						1.40 1.91				
D				1.36 1.85		1.33 1.82				1.33 1.82		1.36 1.83			
C							1.37 1.79		1.38 1.86						
B						1.39 1.88		1.43 1.93		1.49 1.95					
A								270°							

DRIVE LINE "DROP TIME" TABULATION

TEMPERATURE - 549.7 F PRESSURE - 2235 psig % FLOW - 100

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

DATE - 5-11-86

FIGURE 3.1

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

PURPOSE

The purpose of this procedure was to achieve initial reactor 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 5 was achieved during dilution mixing at 1211 hours on May 11, 1986. The reactor was allowed to stabilize at the following critical conditions: RCS pressure- 2235 psig, RCS temperature 548.0°F, intermediate range power 8×10^{-9} amp, RCS boron concentration 1730 ppm, and Control Bank D position- 190.5 steps. Following stabilization, the point of adding nuclear heat was determined and a checkout of the reactivity computer using both positive and negative flux periods was successfully accomplished. In addition, source and intermediate range neutron channel overlap data were taken during the flux increase preceding initial criticality to demonstrate that adequate overlap existed.

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 measured ARO, hot zero power temperature coefficients and the ARO boron endpoint concentration are shown in Table 5.1. The isothermal temperature coefficient was measured to be -3.10 pcm/°F which meets the design acceptance criteria. This gives a calculated moderator temperature coefficient of -0.77 pcm/°F which is within the Technical Specification limit of +5.0 pcm/°F. Thus, no rod withdrawal limits were needed to ensure the +5.0 pcm/°F limit was met. The design acceptance criterion for the ARO critical boron concentration was satisfactorily met.

TABLE 5.1

ARO, HZP ISOTHERMAL AND MODERATOR TEMPERATURE COEFFICIENT

Rod Configuration	Boron Concentration ppm	Measured α_T pcm/°F	Calculated α_{mod} pcm/°F	α_T Design Acceptance Criterion pcm/°F
All Rods Out	1723.8	-3.10	-0.77	-3.07 ± 3

α_T - Isothermal temperature coefficient, includes -2.33 pcm/°F doppler coefficient

α_{mod} - Moderator only temperature coefficient

ARO, HZP BORON ENDPOINT CONCENTRATION

Rod Configuration	Measured C_B (ppm)	Design - predicted C_B (ppm)
All Rods Out	1730.2	1751 ± 50

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 (designated as 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 control and shutdown bank worth measurement results are given in Table 6.1. The measured worths satisfied the review criteria both for the banks measured individually and for the combined worth of all banks.

TABLE 6.1

SUMMARY OF CONTROL AND SHUTDOWN BANK WORTH MEASUREMENTS

Bank	Predicted Bank Worth & Review Criteria (pcm)	Measured Bank Worth (pcm)	Percent Difference
Control A	472 ± 100	465.8	-1.3
Control B (Ref.)	1357 ± 136	1311.8*	-3.3
Control C	859 ± 129	795.6	-7.4
Control D	1092 ± 164	1023.6	-6.3
Shutdown A	1142 ± 171	1106.0	-3.2
Shutdown B	1017 ± 152	930.9	-8.5
All Banks Combined	5939 ± 594	5633.7	-5.1

*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. HZP reactor physics tests (FNP-2-ETP-3601).

   b. incore movable detector system alignment (FNP-2-ETP-3636).

   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 core physics test exceptions, preliminary trip setpoints of less than or equal to 25% power were used for the NIS intermediate and power range channels. When physics tests were completed, the power range setpoint was increased to 80% to enable power escalation (above 25%) for calorimetric recalibration of the power range channels. (The 80% setpoint was used instead of 109% in case the uncalibrated power range channels were indicating non-conservatively.) At approximately 35% power, the power range channels were recalibrated, the high-range trip setpoint was restored to 109%, and setpoint currents were determined for the intermediate range channels.

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 100% power.

The startup test program is addressed elsewhere in this report except for the following notes:

Determination of the incore movable detector system core limit settings (FNP-2-ETP-3606) was accomplished for all modes of system operation during the ascension to 35% power.

In previous startups, the incore-excore recalibration was performed at approximately 75% power. During the Cycle 5 startup, however, a preliminary recalibration was performed at 35% power and was refined using additional data collected at higher power levels. No excessive quadrant power tilt ratios or delta flux channel calibration problems were encountered during the ascension above 35% power.

The revised incore-excore recalibration program resulted in seven quarter-core, and four full-core flux maps being taken between 35% and 100% power. The results from the full-core maps were within Technical Specification Limits, and are summarized in Table 7.1.

TABLE 7.1
SUMMARY OF POWER ASCENSION FLUX MAP DATA

Parameters	Map 118	Map 124	Map 125	Map 128
Date	5/13/86	5/24/86	5/25/86	06/02/86
Time	23:11	17:53	22:47	12:45
Avg. % Power	34.19	31.75	48.32	98.42
Max $F\Delta H$	1.4812	1.4926	1.4951	1.4645
Max Power Tilt*	1.0059	1.0072	1.0042	1.0052
Avg. Core % A.O.	-3.441	-0.164	-0.328	-3.918
Maximum FQ(Z)	2.1608	2.0643	2.0488	1.9171
FQ Limit	4.6084	4.5969	4.5969	2.3400
Xenon Conditions	Non-Equilibrium	Equilibrium	Non-Equilibrium	Equilibrium

*Calculated power tilts based on assembly $F\Delta HN$ from all assemblies.

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 incore axial offset for the purpose of calibrating the delta flux penalty to the overtemperature ΔT protection system, and for calibrating the control board and plant computer axial flux difference (AFD) channels.

SUMMARY OF RESULTS

During previous startups, incore-excore recalibration was performed at approximately 75% power. However, during the Cycle 5 startup, the following modified sequence was used:

(a) At approximately 35% power, a full-core base case flux map and five quarter-core flux maps were run to perform the basic incore-excore recalibration. The power range NIS channels were adjusted to incorporate the revised calibration data.

(b) At a later time, a full-core flux map was performed at approximately 32% power under more stable xenon conditions than the original base case map. The evaluation of this map included a verification of the effectiveness of the 35% power NIS recalibartion.

(c) During power escalation, a full-core map was taken at 48% power and quarter core maps were taken at 51% and 55% power to develop additional data for comparison with the 35% power results. Problems with the incore movable detector system prevented five quarter-core maps from being obtained as was planned.

(d) When xenon equilibrium was achieved at approximately 100% power, a full-core flux map was taken to renormalize the calibration data to compensate for temperature decalibration (the change in reactor core neutron leakage caused by the changes in coolant temperature associated with changes in power). The power range NIS channels were recalibrated to incorporate this correction.

At 35% power, the six flux maps were performed at axial offsets of approximately + 18%, + 7%, -3%, -15%, -22%, and -27%. the detector currents measured during the flux maps were normalized to 100% power, and a least squares fit was performed to derive the output current vs. axial offset equation for each top and bottom detector. Calibration values obtained from these equations were used to recalibrate the NIS channels.

Evaluation of the flux map taken at 32% power showed a small error in the recalibration of NIS channel N44, but confirmed that the calibration equations derived at 35% power were satisfactory. The calibration error on channel N44 was corrected.

Between 48% and 55% power, three flux maps were performed at axial offsets of approximately +6%, -0.3% and -5%. By subtracting the current contributions due to temperature decalibration, the currents measured during these maps were, in effect, normalized to 35% power so that they could be merged with the 35% power data. Using the combined 35% and 48% - 55% power data, the incore-excore equations were recalculated to obtain improved equation slopes. (The resulting changes in slope were small.) No recalibration of the NIS channels was performed at this time.

At approximately 100% power, revised zero percent axial offset (I-zero) currents were determined from flux map data and were combined with the merged 35% and 48% - 55% equation slopes to yield the finalized incore - excore equations given in Table 8.1. Using these results, the NIS channels were recalibrated.

TABLE 8.1

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

CHANNEL N41:

I-Top	=	0.7889*AO	+	185.20 μa
I-Bottom	=	-1.1085*AO	+	182.84 μa

CHANNEL 42:

I-Top	=	0.8766*AO	+	180.55 μa
I-Bottom	=	-1.1619*AO	+	179.08 μa

CHANNEL N43:

I-Top	=	0.8250*AO	+	189.63 μa
I-Bottom	=	-1.1836*AO	+	184.58 μa

CHANNEL N44:

I-Top	=	1.0003*AO	+	202.10 μa
I-Bottom	=	-1.3355*AO	+	202.23 μa

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 Specifications, the total reactor coolant system flow rate measured at normal operating temperature and pressure must equal or exceed 265,500 gpm for three loop operation. From the average of six calorimetric heat balance measurements, the total core flow was determined to be 285,767.6 gpm, which meets the above criterion.

NT-86-0392

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Senior Vice President
Flintridge Building

AlabamaPower
the southern electric system

August 28, 1986

Docket No. 50-364

Director, Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Attention: Mr. L. S. Rubenstein

Joseph M. Farley Nuclear Plant - Unit 2
Cycle 5 - Startup Report

Gentlemen:

Enclosed is the Startup Report for Unit 2 Cycle 5 as required by the April 29, 1986 letter from Mr. R. P. McDonald to Mr. L. S. Rubenstein.

If you have any questions, please advise.

Yours very truly,

RPMcDonald

R. P. McDonald

RPM/MDR:emb

Enclosure

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Dr. J. N. Grace
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Mr. W. H. Bradford

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