ALABAMA POWER COMPANY

JOSEPH M. FARLEY NUCLEAR PLANT

UNIT NUMBER 1, CYCLE 5

STARTUP TEST REPORT

PREPARED BY PLANT REACTOR ENGINEERING GROUP

APPROVED:

C. D. Nesbitt Technical Superintendent

W. G. Hairston III Plant Manager

DATE APPROVED 6/23/83

DISK: CYCLE 2

IE26
1/1

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1.0 INTRODUCTION

The Joseph M. Farley Unit 1 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 1 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.

Unit 1 began commercial operations on December 1, 1977, and completed Cycle 4 on January 14, 1983 with an average core burnup of 10621.7 MWD/MTU.

2.0 FUEL INSPECTION AND CORE REFUELING

REFERENCES

1. Westinghouse Refueling Procedure FP-ALA-R4

2. Westinghouse WCAP 10308 (The Nuclear Design and Core Management of the Joseph M. Farley Unit 1 Power Plant Cycle 5)

2.1 Cycle 4 Fuel Inspection

During cycle-3 operation reactor coolant activity levels indicated a small number of fuel rod failures in low burnup fuel assemblies. Since several other plants had recently found damaged fuel rods adjacent to baffle center injection joints, it was decided to perform TV Visual inspection of these assemblies and to peen these joints at the end-of-cycle-3 outage. This procedure had been used at other plants. Although TV examination showed no damage on the assemblies located at the center injection joints, peening was performed.

At the beginning of cycle-4 RCS activity levels increased significantly but remained below Tech. Spec. limits. Based upon experience at other plants which had peened at the center injection core baffle joints, it was assumed that fuel rod damage had occurred due to jetting through gaps at the corner injection baffle joints opened by the center injection joint peening process. Preparations were therefore made to conduct an extensive fuel inspection campaign to preclude the use of damaged assemblies in cycle-5. The inspection programs consisted of a binocular visual examination followed by vacuum sipping of all assemblies which showed no evidence of physical damage and which were to be returned to the cycle-5 core. TV visual inspection (high/low magnification) was used in cases where binocular inspection was inconclusive. General Electric gaseous vacuum sipping equipment was utilized. Additionally all assemblies that were located at baffle joints (center or corner injection) during cycle-4 that passed both sipping and TV visual inspection were examined for wear damage beneath the grid straps using Westinghouse SULO probe equipment (a differential strain gage technique).

Eleven one-cycle region F assemblies showed gross damage due to baffle jetting. All damage was at corner injection joints and was limited almost entirely to the top fuel rod span between grids 7 and 8.

Damage was limited generally to one or more of the first five fuel rods in from the corner.

The assemblies which suffered gross damage from baffle jetting were F02, F05, F06, F15, F17, F18, F19, F20, F30, F31, and F32. Assembly E50 was found to have a hydride blister at the upper end cap weld on one fuel rod and was rejected for use in cycle-5. E50 was later found to fail the sipping test. Three assemblies were determined to be leaking from the sipping and were eliminated from use in cycle-5. These were F27 located at a center injection joint, E-14 which was located at a center injection joint in cycle-3, and D34 which was located at a center injection joint in cycle-2.

2.2 Cycle-5 Core Refueling

The Cycle-5 core loading commenced on 3/2/83 following the completion of fuel inspection and debris cleanup activities, and was completed on 3/4/83. The as-loaded Cycle-5 core is depicted in Figures 2.1 - 2.3, which shows the location of each fuel assembly and insert, and gives the assembly enrichments. The number of assemblies in the various regions of the Cycle-5 core is tabulated below:

Region	No. of Fuel Assemblies
1	4
3	5
4	16
4A	2
5	50
6	40
7	40

Fuel assembly inserts consist of 48 full length control rods, 2 secondary sources, 51 burnable poison inserts, and 56 thimble plug inserts.

Figure 2.1

ALA Unit 1 Cycle 5 Reference Loading Pattern

180°

	R	P	N	M	L	K	J	H	G	F	E	D	C	B	A
1							A-34 * R-8	F-35 F	A-18 * H-2						
2					D-19 L-10	F-40 F	G-05 F	F-13 L-2	G-11 F	F-42 F	D-39 E-10				
3				D-15 F-13	F-48 F	G-07 F	E-09 L-3	E-21 SS H-1	E-27 E-3	G-19 F	F-49 F	D-07 K-13			
4			D-27 C-10	E-04 J-4	G-13 F	E-37 M-5	G-23 F	F-24 F-2	G-35 F	E-45 D-5	G-25 F	C-32 ++ G-11	D-09 N-10		
5		D-05 F-5	F-34 F	G-31 F	E-31 M-7	F-01 R-7	E-16 J-6	E-40 J-2	E-08 G-6	F-22 A-7	E-29 D-7	G-04 F	F-52 F	D-47 K-5	
6		F-38 F	G-36 F	E-38 L-4	F-03 J-1	E-02 H-14	G-03 F	E-49 H-7	G-39 F	E-48 H-2	F-10 G-1	E-33 E-4	G-09 F	F-45 F	
7	C-49 ++ E-9	G-32 F	E-24 N-5	G-28 F	E-20 K-7	G-29 F	E-43 P-7	F-09 P-6	E-36 B-7	G-24 F	E-34 F-7	G-26 F	E-03 C-5	G-06 F	C-14 ++ L-9
8	F-46 F	F-21 P-11	E-11 R-8	F-23 K-2	ZD-3 N-8	E-22 J-8	F-12 B-6	E-10 G-2	F-26 M-13	E-41 G-8	ZD-4 C-8	F-04 K-14	E-47 A-8	F-16 B-5	F-43 F
9	C-28 ++ E-7	G-17 F	E-15 N-11	G-18 F	E-13 K-9	G-20 F	E-44 P-9	F-29 P-10	E-46 B-9	G-01 F	E-17 F-9	G-34 F	E-07 C-11	G-21 F	C-43 ++ L-7
10		F-44 F	G-14 F	E-39 L-12	F-14 J-15	E-32 P-8	G-10 F	E-28 H-9	G-38 F	E-35 B-8	F-28 G-15	E-19 E-12	G-37 F	F-39 F	
11		D-45 F-11	F-41 F	G-15 F	E-25 M-9	F-08 R-9	E-23 J-10	E-30 J-14	E-01 G-10	F-07 A-9	E-26 D-9	G-16 F	F-33 F	D-37 K-11	
12			D-42 C-6	E-18 J-12	G-08 F	E-51 M-11	G-33 F	F-11 F-14	G-40 F	E-05 D-11	G-12 F	E-12 G-12	D-11 N-6		
13				D-24 F-3	F-47 F	G-30 F	E-52 L-13	E-06 SS H-15	E-42 E-13	G-27 F	F-38 F	D-03 K-3			
14					D-33 L-6	F-37 F	G-02 F	F-25 E-14	G-22 F	F-51 F	D-38 E-6				
15							A-43 * P-8	F-50 F	A-35 * H-14						

270° (at row 8)

0°

* XX Represents Cycle 1 Location

++ XX Represents Cycle 3 Location

Color Code	ID XX	ID XX	ID XX	ID XX	ID XX	ID XX	ID XX
Region	1	3	4	4A	5	6	7
From Cycle	1	3	4A	4A	4A	4A/FEED	FEED
W/O U_{235}	2.115	3.102	3.113	3.108	2.801	2.995	3.00

Legend:

ID: Assembly Identification

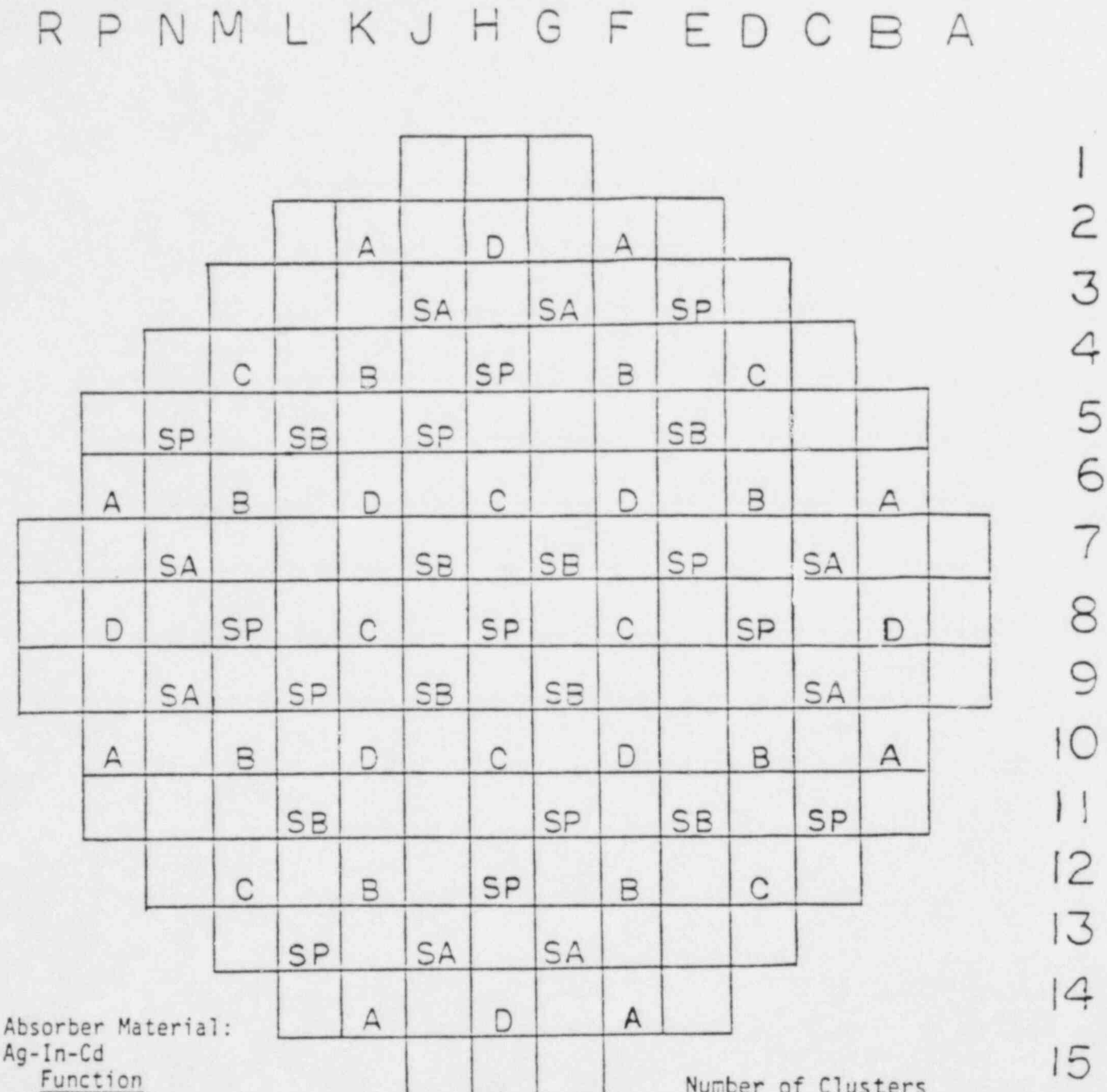
ZD: Demonstration Assembly

XX: Previous Cycle Location

SS: Secondary Source

Figure 2.2

CONTROL ROD LOCATIONS



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 S_B	8
Shutdown Bank S_A	8
SP (Spare Rod Locations)	13

Figure CONTROL ROD LOCATIONS

Figure 2.3

Source Assembly Locations

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

SP Spent BP Assembly
SS Secondary Source

424 Fresh Standard BPs
12 Spent Standard BPs

3.0 CONTROL ROD DROP TIME MEASUREMENT (FNP-1-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 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 1.8 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	1.628 sec.	2.182 sec

To confirm normal rod mechanism operation prior to conducting the rod drops, a Control Rod Drive Test (FNP-0-IMP-230.3) was performed. In the test, the stepping waveforms of the stationary, lift and moveable gripper coils were examined and rod stepping speed measurements were conducted. All results were satisfactory.

Figure 3.1

UNIT 1 CYCLE 5

NORTH

	15	14	13	12	11	10	9	8	7	6	5	4	3	2	1
R								90°							
P						1.64 2.20		1.63 2.17		1.60 2.15					
N							1.63 2.20		1.65 2.20						
M				1.59 2.15		1.60 2.17				1.63 2.22		1.65 2.20			
L					1.59 2.13						1.60 2.17				
K		1.70 2.25		1.58 2.15		1.67 2.22		1.60 2.15		1.66 2.22		1.65 2.22		1.60 2.15	
J			1.60 2.17				1.55 2.10		1.57 2.12				1.60 2.17		
H	0°	1.65 2.20				1.64 2.20				1.60 2.15				1.63 2.18	180°
G			1.66 2.22				1.57 2.12		1.58 2.12				1.64 2.19		
F		1.68 2.22		1.63 2.20		1.66 2.21		1.58 2.10		1.59 2.14		1.63 2.20		1.59 2.15	
E					1.61 2.17						1.64 2.21				
D				1.66 2.20		1.63 2.20				1.60 2.16		1.60 2.16			
C							1.61 2.17		1.65 2.21						
B						1.70 2.22		1.72 2.19		1.80 2.38					
A								270°							

DRIVE LINE "DROP TIME" TABULATION

TEMPERATURE - 547°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 - 3-27-83

4.0 INITIAL CRITICALITY (FNP-1-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 test, and operationally verify the calibration of the reactivity computer.

SUMMARY OF RESULTS

Initial Reactor Criticality for Cycle 5 was achieved during dilution mixing at 0614 hours on March 28, 1983. The reactor was allowed to stabilize at the following critical conditions: RCS pressure - 2229 psig, RCS temperature 547°F, intermediate range power 8×10^{-9} amp, RCS boron concentration 1333 ppm, and Control Bank D position 170 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 sucessfully accomplished. In addition, source and intermediate range neutron channel overlap data were taken during the flux increase preceding and immediately following initial criticality to demonstrate that adequate overlap existed.

5.0 ALL-RODS-OUT ISOTHERMAL TEMPERATURE COEFFICIENT, BORON ENDPOINT AND FLUX DISTRIBUTION (FNP-1-ETP-3601)

PURPOSE

The objectives of these measurements were to: (1) determine the hot, zero power isothermal and moderator temperature coefficients for the all-rods-out (ARO) configuration; (2) measure the ARO boron endpoint concentration; and (3) determine the hot, zero power ARO flux distribution in the reactor core.

SUMMARY OF RESULTS

The measured ARO, hot zero power temperature coefficients and the ARO boron endpoint concentration are shown in Table 5.1. The moderator temperature coefficient was found to be slightly positive (+0.38 pcm/°F). The NRC was notified by special report and control rod withdrawal limits were established in accordance with Technical Specifications to maintain a negative MTC during normal plant operation. The design acceptance criterion for the ARO critical boron concentration was satisfactorily met.

Following the control and shutdown bank worth measurements (Section 6.0) a flux distribution map was obtained at the ARO configuration. As summarized in Table 5.2, the differences between measured and design-predicted relative assembly powers satisfied the design criteria, but incore tilt was in excess of 1.02. Westinghouse was immediately notified in accordance with their recommended policy and the plant commenced power escalation. The next two flux distribution maps, taken at 48% and 78% power, demonstrated that incore tilt had decreased to below 1.02.

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	1347.5	-2.32	+0.38	-3.2 ± 3

α_T - Isothermal temperature coefficient, includes -2.7 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	1357.5	1325 ± 50

TABLE 5.2

RESULTS OF HZP, ARO FLUX DISTRIBUTION MAP

A. $F\Delta H$ percent error between measured and design - predicted values versus relative assembly power Pi of assembly i.

Item	Value	Pi	Design Criterion
Maximum positive percent error	+10.2%	0.783	± 15% for Pi < 0.9
Maximum negative percent error	-7.6%	1.062	± 10% for Pi $\geq$ 0.9

B. Incore Quadrant Tilt:

Maximum Incore Tilt	Design Criterion
1.037*	$\leq$ 1.02

*Corrected value obtained by subtracting design asymmetry (1.0127) from the measured core tilt (1.0497). The (uncorrected) measured core tilts at 48% and 78% power were 1.0168 and 1.0135, respectively.

6.0 CONTROL AND SHUTDOWN BANK WORTH MEASUREMENTS (FNP-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; and (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	630 ± 94	658.7	+4.6
Control B (Ref.)	1301 ± 130	1292.3*	-0.7
Control C	732 ± 110	715.3	-2.3
Control D	1237 ± 185	1196.4	-3.3
Shutdown A	1147 ± 172	1129.1	-1.6
Shutdown B	973 ± 146	943.6	-3.0
All Banks Combined	6020 ± 602	5935.4	-1.4

*Measured by dilution method

7.0 POWER ASCENSION PROCEDURE (FNP-1-ETP-3605)

PURPOSE

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

1. Ramp rate and control rod movement limitations
2. Incore movable detector system final alignment
3. Flux map at less than 50% power
4. Adhering to the delta flux band during ascension to 75% power
5. Incore/Excore calibration at 75% power.

SUMMARY OF RESULTS

In compliance with Westinghouse recommendations and fuel warranty provisions, the power ramp rate was limited to 3% of full power per hour between 20% and 100% power until full power was achieved for 72 cumulative hours out of any seven-day operation period. Control rod motion during the initial return to power was minimized, and the startup was conducted with the rods withdrawn as far as possible. The rod withdrawal rate was limited to 3 steps per hour above 50% power.

Final alignment of the incore movable detector system was completed during power ascension (at power levels above 5%) prior to performing the flux max at 48% power.

Due to the lower neutron leakage of the cycle 5 core, design-predicted NIS detector currents equal to 80% of the Cycle-4 values were used for initial reactor trip and rod stop setpoints. At 30% power, detector current readings and calorimetric data were obtained to verify the adequacy of the initial settings and to provide data for rescaling the NIS intermediate range setpoints.

Full core flux maps were taken at 48% and 76% power. The results were within Technical Specification Limits and are summarized in Table 7.1.

An incore/excore calibration check was performed at 48% power with satisfactory results. A full recalibration of the excore AFD channels was performed at approximately 75% power to comply with Technical Specification requirements. The incore/excore recalibration is described in section 8.0.

TABLE 7.1
SUMMARY OF POWER ASCENSION FLUX MAP DATA

Parameter	Map 119	Map 120
Date	4/1/83	4/2/83
Time	05:00	11:47
Avg. % Power	48.82	75.87
Max. F_Q (Z)	2.0322	1.9416
Max. $F\Delta H$	1.4863	1.4634
Max. Power Tilt*	+1.0168	+1.0135
Avg. Core % A.O.	+2.922	-0.030

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

8.0 INCORE-EXCORE DETECTOR CALIBRATION (FNP-1-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

A preliminary verification of excore AFD channel calibration was performed at 48% power to insure AFD could be kept within the target band during the ascension to 76% power. Flux maps for incore-excore recalibration were run at approximately 75% power at average percent core axial offsets of + 15.097, -0.030, -18.181, and -25.916, as determined from the incore printouts.

The measured detector currents were normalized to 100% power, and a least squares fit was performed to obtain the linear equation for each top and bottom detector current versus core axial offset.

Using these equations, detector current data was generated and utilized to recalibrate the AFD channels and the delta flux penalty to the overtemperature ΔT setpoint. (See Figure 8.1)

FIGURE 8.1

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

<u>CHANNEL N41:</u>

I-Top	=	1.0393*AO	+	194.3945	μa
I-Bottom	=	-1.0822*AO	+	194.2360	μa

<u>CHANNEL 42:</u>

I-Top	=	1.0609*AO	+	189.6089	μa
I-Bottom	=	-1.1447*AO	+	183.9130	μa

<u>CHANNEL N43:</u>

I-Top	=	0.9731*AO	+	180.6990	μa
I-Bottom	=	-1.1589*AO	+	193.4700	μa

<u>CHANNEL N44:</u>

I-Top	=	1.0932*AO	+	184.6970	μa
I-Bottom	=	-1.1680*AO	+	190.1275	μa

9.0 REACTOR COOLANT SYSTEM FLOW MEASUREMENT (FNP-1-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 1 Technical Specifications.

SUMMARY OF RESULTS

To comply with the Unit 1 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 283,178.5 gpm, which meets the above criterion.

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F. L. CLAYTON, JR.
Senior Vice President

Alabama Power
the southern electric system

June 28, 1983

Docket No. 50-348

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

Attention: Mr. S. A. Varga

J. M. Farley Unit 1 Startup Report

Gentlemen:

Enclosed is the Startup Report for Unit 1 Cycle 5 as required by the March 17, 1983 letter from F. L. Clayton, Jr. to Mr. S. A. Varga.

If you have any questions, please advise.

Yours very truly,

F. L. Clayton, Jr.

F. L. Clayton, Jr.

FLC,Jr/MDR:cl
Enclosure

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Mr. G. F. Trowbridge
Mr. J. P. O'Reilly
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