ALABAMA POWER COMPANY

JOSEPH M. FARLEY NUCLEAR PLANT

UNIT NUMBER 1, CYCLE 7

STARTUP TEST REPORT

PREPARED BY PLANT REACTOR ENGINEERING GROUP

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DISK: CYCLES

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1.0 INTRODUCTION

The Joseph M. Farley Unit 1 Cycle 7 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 Westinghouse three loop pressurized water reactor rated at 2652 MWth. The Cycle 7 core loading consists of 157 17 x 17 fuel assemblies.

The Unit began commercial operations on December 1, 1977 and completed cycle 6 on April 6, 1985 with an average core burnup of 12,238.4 MWD/MTU.

2.0 UNIT 1, CYCLE 7 CORE REFUELING

REFERENCES

1. Westinghouse Refueling Procedure FP-ALA-R6
2. Westinghouse WCAP 10795 (The Nuclear Design and Core Management of the Joseph M. Farley Unit 1 Power Plant Cycle 7)
3. Westinghouse Letter 85AP*-G-0505, dated April 19, 1985

2.1 Cycle 6 Fuel Inspection

Each fuel assembly was visually inspected with binoculars during the core unload. No damage or significant defects were found during the fuel inspection. However, a white deposit was noted on Rod L1 between grids 5 and 7 on face 4 of assembly H63 (which was not present on the adjacent rods). Westinghouse confirmed that this was a common type of discoloration, and recommended re-use of assembly H63 with no restrictions.

Cycle 7 Core Refueling

The Cycle 6 core unload commenced on April 14, 1985 and was completed with all fuel assemblies in the Spent Fuel Pool on April 16th. The Cycle-7 core reload began on April 18th and was completed on April 20th. During the reload, grid no.6 on face 3 of assembly H11 was found to be bent. (The grid was bent outward and the tab at that location was folded behind.) After a detailed safety evaluation was performed on assembly H11 by Westinghouse and APCo, the assembly was determined to be acceptable for re-use with the bent tab and was reloaded.

On 8-8-85 the discovery was made that fuel assembly ZD3, which was required by core design to be in core location H4, was in fact located in the spent fuel pool. Upon further investigation it was determined that fuel assembly C03 was in the core location which should have been occupied by fuel assembly ZD3. An evaluation was performed and the determination was made that continued operation was justified.

The as-loaded Cycle-7 core is shown in Figures 2.1 - 2.4. The following table shows the number of assemblies in each Cycle-7 fuel region.

Region	3	4A	7	8A	8B	9A	9B
No. of Fuel Assemblies	1	1	2	44	33	52	24

Fuel assembly inserts consist of 48 full length control rods, 2 secondary sources, 52 burnable poison inserts and 55 thimble plug inserts. The nominal design lifetime of the Unit 1 Cycle 7 core is 17,400 MWD/MTU.

FIGURE 2.1

ALA Cycle 7 Loading Pattern

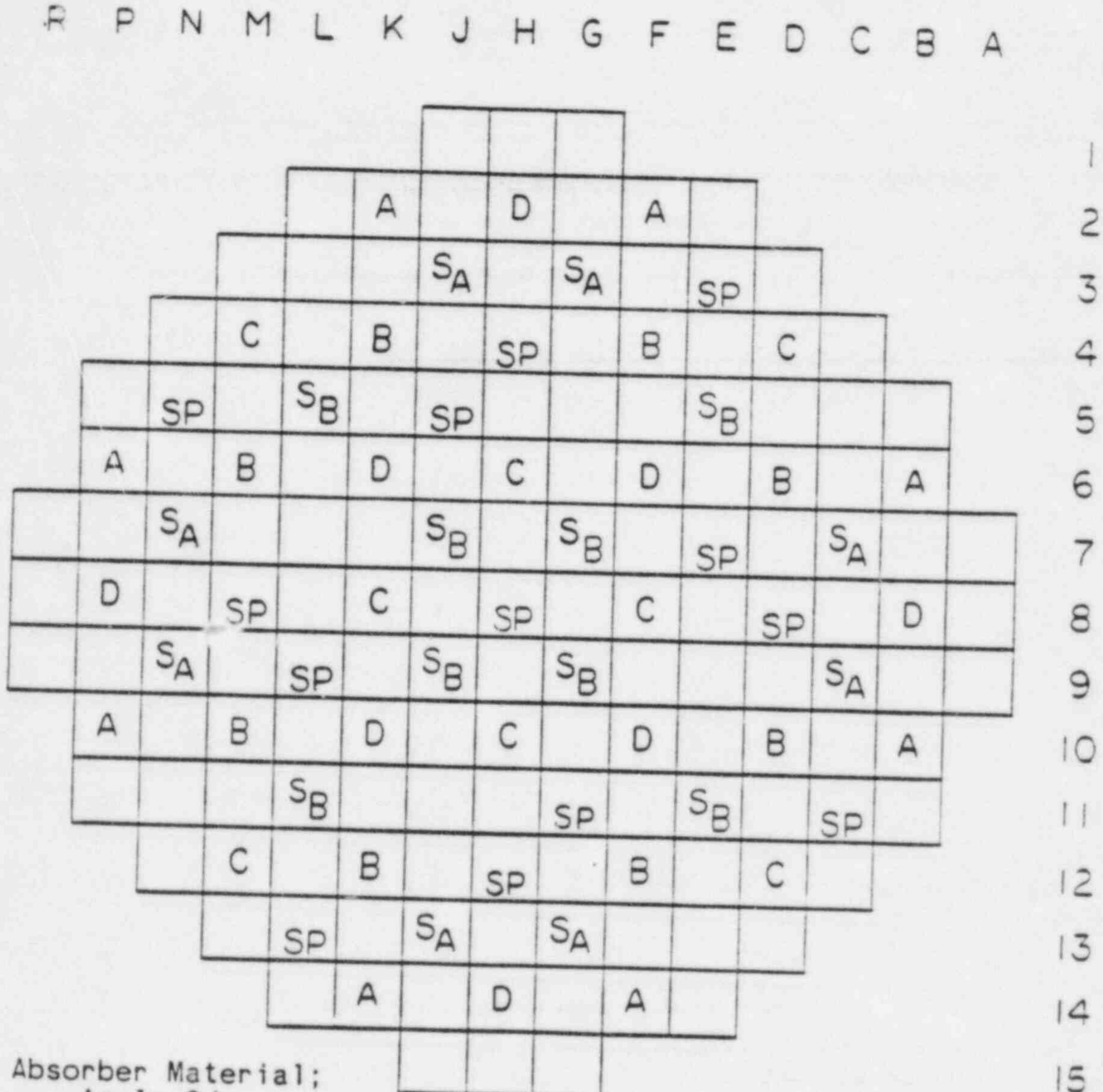
	R	P	N	M	L	K	J	H	G	F	E	D	C	B	A
1							H14 / M7	J55 / F	H36 / D7						
2					H38 / F7	J03 / F	J60 / F	H72 / J14	J62 / F	J21 / F	H10 / K7				
3				H01 / E6	J24 / F	J71 / F	H48 / L13	H54 / G2	H58 / E13	J72 / F	J38 / F	H24 / L6			
4			H05 / F11	J40 / F	J02 / F	H77 / N11	J54 / F	C03 / E11	J75 / F	H70 / C11	J49 / F	J10 / F	H04 / K11		
5		H15 / J6	J33 / F	J34 / F	H45 / P7	J29 / F	H43 / B10	J31 / F	H17 / P10	J20 / F	H66 / B7	J23 / F	J01 / F	H18 / G6	
6		J37 / F	J56 / F	H75 / E12	J18 / F	H26 / H5	J12 / F	H31 / M4	J45 / F	H28 / E8	J43 / F	H50 / L12	J66 / F	J36 / F	
7	H33 / J4	J67 / F	H64 / M11	J52 / F	H20 / K14	J35 / F	H68 / F3	H03 / H1	H67 / K3	J32 / F	H11 / F14	J42 / F	H61 / D11	J63 / F	H02 / G4
8	J64 / F	H71 / N10	H56 / C6	G28 / R7	J11 / F	H41 / D4	H29 / R8	H52 / H8	H25 / A8	H12 / M12	J47 / F	G34 / A9	H53 / N6	H65 / C10	J59 / F
9	H06 / J12	J61 / F	H63 / M5	J09 / F	H16 / K2	J22 / F	H69 / F13	H32 / H15	H55 / K13	J41 / F	H19 / F2	J27 / F	H59 / D5	J53 / F	H13 / G12
10		J44 / F	J70 / F	H49 / E4	J46 / F	H44 / L8	J51 / F	H21 / D12	J04 / F	H42 / H11	J50 / F	H76 / L4	J58 / F	J05 / F	
11		H22 / J10	J16 / F	J08 / F	H51 / P9	J17 / F	H30 / B6	J26 / F	H23 / P6	J25 / F	H73 / B9	J13 / F	J39 / F	H37 / G10	
12			H27 / F5	J30 / F	J48 / F	H62 / N5	J57 / F	ZD4 / E8	J74 / F	H60 / C5	J28 / F	J14 / F	H08 / K5		
13				H39 / E10	J19 / F	J73 / F	H57 / L3	H74 / G14	H46 / E3	J69 / F	J07 / F	H34 / L10			
14					H40 / F9	J06 / F	J76 / F	H47 / J2	J68 / F	J15 / F	H09 / K9				
15							H35 / M9	J65 / F	H07 / D9						

In each core location the assembly identification is given at the top and the previous cycle core location is given below (except for assemblies C-03 and ZD-4, for which the Cycle 4/5 locations are given).

Assembly letter designator	C	ZD	G	H	H	J	J
Fuel enrichment region	3	4a	7	8A	8B	9a	9b
From core cycle	4	5	6	6	6	Feed	Feed
w/o U-235 (original):	3.102	3.108	3.002	2.999	3.443	3.6	3.9

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 SB	8
Shutdown Bank SA	8
SP (Spare Rod Locations)	8
	13

FIGURE 2.3

BURNABLE ABSORBER AND SOURCE ASSEMBLY LOCATIONS

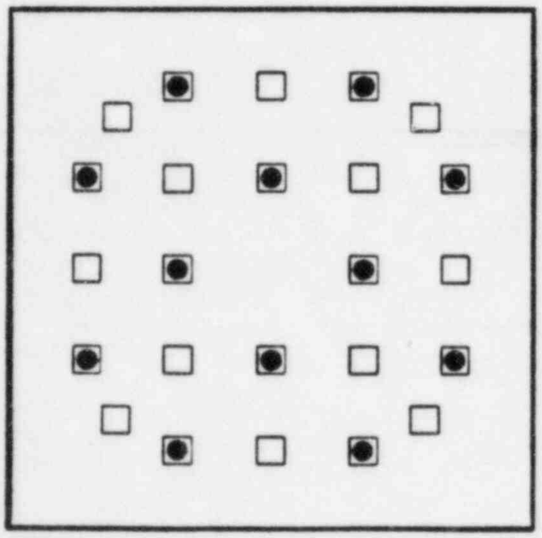
R	P	N	M	L	K	J	H	G	F	E	D	C	B	A	
															1
															2
				4	16		SS		16	4					3
				16		16		16		16					4
		4	16		12		16		12		16	4			5
		16		12		12		12		12		16			6
			16		12				12		16				7
				16						16					8
			16		12				12		16				9
		16		12		12		12		12		16			10
		4	16		12		16		12		16	4		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

SS Secondary Source

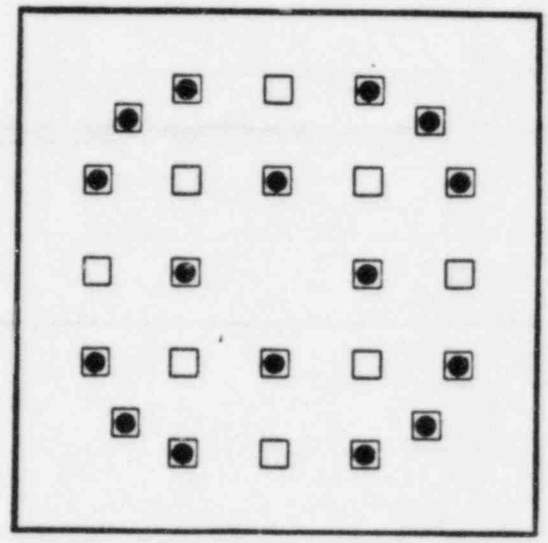
672 Fresh Standard BA's

FIGURE 2.4

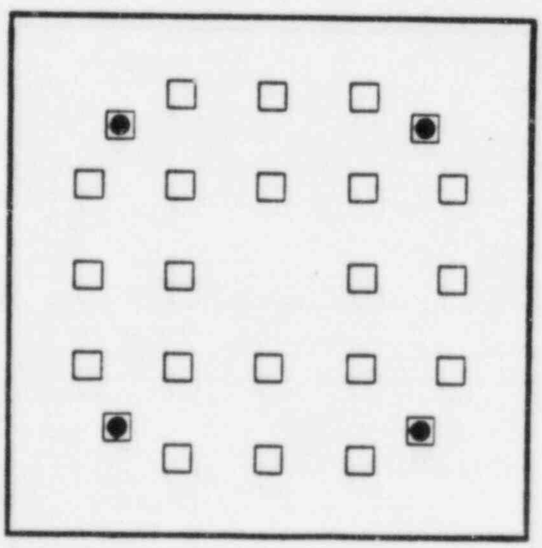
BURNABLE ABSORBER CONFIGURATIONS



12 Fresh BA
Configuration



16 Fresh BA
Configuration



4 Fresh BA
Configuration

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 ensure 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 1.79 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	1.62 sec	2.17 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 the functioning of the Digital Rod position indicator and the bank overlap unit were checked. Rod stepping speed measurements were also conducted. All results were satisfactory.

NORTH

UNIT 1 CYCLE 7

	15	14	13	12	11	10	9	8	7	6	5	4	3	2	1
R								90°							
P						1.63 2.17		1.62 2.20		1.59 2.10					
N							1.61 2.19		1.60 2.17						
M				1.58 2.10		1.60 2.16				1.62 2.20		1.63 2.16			
L					1.61 2.16						1.61 2.18				
K		1.71 2.22		1.57 2.10		1.63 2.21		1.56 2.12		1.61 2.19		1.63 2.22		1.61 2.14	
J			1.61 2.20				1.55 2.09		1.56 2.13				1.60 2.18		
H	0°	1.65 2.24				1.60 2.15				1.58 2.13				1.61 2.20	180°
G			1.65 2.23				1.58 2.10		1.57 2.13				1.62 2.19		
F		1.69 2.23		1.64 2.21		1.62 2.19		1.55 2.12		1.59 2.16		1.62 2.19		1.61 2.12	
E					1.59 2.16						1.58 2.16				
D				1.60 2.14		1.59 2.15				1.58 2.11		1.61 2.14			
C							1.62 2.17		1.63 2.20						
B						1.67 2.20		1.74 2.31		1.79 2.35					
A								270°							

DRIVE LINE "DROP TIME" TABULATION

TEMPERATURE - 547°F PRESSURE - 2235°F % FLOW - 100

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

DATE - 5-19-85

FIGURE 3.1

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 tests, and operationally verify the calibration of the reactivity computer.

SUMMARY OF RESULTS

Initial Reactor Criticality for Cycle 7 was achieved during dilution mixing at 0627 hours on May 26, 1985. The reactor was allowed to stabilize at the following critical conditions: RCS pressure- 2230 psig, RCS temperature 547.5°F, intermediate range power 1×10^{-8} amp, RCS boron concentration 2060 ppm, and Control Bank D position- 198 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 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 AND FNP-1-ETP-3605)

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 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 isothermal temperature coefficient was measured to be +2.2 pcm/°F which meets the design acceptance criteria. This gives a calculated moderator temperature coefficient of +4.5 pcm/°F which is within the Technical Specification limit of +5.0 pcm/°F. Thus, no rod withdrawal limits are needed to ensure the +5.0 pcm/°F limit is met. The design acceptance criterion for the ARO critical boron concentration was also satisfactorily met.

Core flux distribution was determined by the performance of a flux map at 33% power (see section 7.0 for results).

TABLE 5.1

ARO, HZP ISOTHERMAL AND MODERATOR TEMPERATURE COEFFICIENT

Rod Configuration	Boron Concentration	Measured α_T	Calculated α_{mod}	α_T Design Acceptance Criterion
	ppm	pcm/°F	pcm/°F	pcm/°F
All Rods Out	2060.5	+2.2	+4.5	+1.70 ± 3

α_T - Isothermal temperature coefficient, includes -2.3 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	2066.4	2072 ± 207.2

6.0 CONTROL AND SHUTDOWN BANK WORTH MEASUREMENTS (FNP-1-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	496 ± 74	501.9	+1.2
Control B (Ref.)	1240 ± 124	1225.5*	-1.2
Control C	1035 ± 155	1006.2	-2.8
Control D	1107 ± 166	1078.4	-2.6
Shutdown A	835 ± 125	828.0	-0.8
Shutdown B	1205 ± 181	1137.8	-5.6
All Banks Combined	5918 ± 592	5777.8	-2.4

*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

Westinghouse fuel warranty provisions recommend that the power ramp rate be limited to 3% of full power per hour between 20% and 100% power until full power is achieved for 72 cumulative hours out of any seven-day operation period. This ramp rate was followed throughout the ascension to 100% power except for one hour when the power was increased by 3.3%.

Alignment of the incore movable detector system normal, calibrate and emergency paths was accomplished during power ascension (at power levels above 5%) prior to performing the flux map at 33% power.

Full flux maps were taken at 33%, and 78% power. As summarized in Table 7.1, all results were within Technical Specification Limits.

An incore/excore calibration check was performed at 33% power and all channels were found to be out of calibration. Therefore, a correction to the incore-excore calibration data was performed per Appendix B of FNP-1-STP-121 and revised currents were issued for calculating AFD (STP-7.1) and QPTR (STP-7.0). At approximately 75% power, a complete incore-excore recalibration was performed as described in Section 8.0.

TABLE 7.1
SUMMARY OF POWER ASCENSION FLUX MAP DATA

Parameters	Map 156	Map 157
Date	5/30/85	6/2/85
Time	0950	0100
Avg. % Power	33%	78%
Max FΔH	1.4553	1.4471
Max. Power Tilt*	1.0080	1.0056
Avg. Core % A. O.	+5.471	+2.706

*Calculated power tilts based on assembly FΔHN 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 correction to the excore AFD channel calibration was performed at 33% power to insure that an AFD target band could be defined for ascension to 78% power. Flux maps for incore-excore recalibration were run at approximately 78% power at average core percent axial offsets of +2.706, -7.717, -13.518 and +8.376 as determined from the INCORE code 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 were generated and utilized to recalibrate the AFD channels and the delta flux penalty to the overtemperature ΔT setpoint. (See Table 8.1)

TABLE 8.1

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

CHANNEL N41:

I-Top	=	0.9891*AO +	192.8114	μa
I-Bottom	=	-1.2469*AO +	191.4415	μa

CHANNEL 42:

I-Top	=	1.0681*AO +	188.3704	μa
I-Bottom	=	-1.3229*AO +	186.2551	μa

CHANNEL N43:

I-Top	=	0.9945*AO +	188.3289	μa
I-Bottom	=	-1.3461*AO +	200.6703	μa

CHANNEL N44:

I-Top	=	1.0361*AO +	181.7421	μa
I-Bottom	=	-1.2834*AO +	187.0081	μa

NOTE: Asterisk (*) denotes multiplication in the above equations.

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 283,881.0 gpm, which meets the above criterion.