

ALABAMA POWER COMPANY

JOSEPH M. FARLEY NUCLEAR PLANT

UNIT NUMBER 1, CYCLE 4

STARTUP TEST REPORT

PREPARED BY PLANT REACTOR ENGINEERING GROUP

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## 1.0 INTRODUCTION

The Joseph M. Farley Unit I Cycle 4 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 4 core loading consists of 157 17 x 17 fuel assemblies, including two optimized demonstration assemblies.

The Unit began commercial operation on December 1, 1977, and completed Cycle 1 on March 8, 1979 with an average core burnup of 15,450 MWD/MTU. Cycle 2 was completed on November 8, 1980, with an average core burnup of 10,177 MWD/MTU. Cycle 3 ended at 5180.8 MWD/MTU due to an extended forced outage caused by a major malfunction in the main turbine generator. Initial criticality for Cycle 4 was achieved on March 3, 1982, and full power operation was resumed on April 18, 1982.

## 2.0 Unit 1 CYCLE 4 CORE REFUELING

### REFERENCES

1. Westinghouse Refueling Procedure FP-ALA-R3
2. Westinghouse WCAP 10036 (The Nuclear Design and Core Management of the Joseph M. Farley Unit 1 Power Plant Cycle 4)

### 2.1 FUEL SHUFFLE OPERATIONS

Nuclear fuel (52 assemblies) for Cycle 4, Core Region 6, was received on site during the period from October 12, 1981, to October 27, 1981.

The refueling commenced on October 24, 1981. The refueling shuffle resulted in: (1) the removal of all assemblies to the spent fuel pit; (2) shuffling of the Region 1, 3, 4 and 5 assemblies into an approximate checkerboard pattern in the inner section of the core; (3) arranging the fresh (Region 6) assemblies into a ring surrounding the inner checkerboard; (4) relocation of inserts (such as control rods), and placement of thimble plugs in all fuel assemblies not containing other inserts.

During the fuel shuffle, the two optimized demonstration fuel assemblies (Region-4A), which had resided in the core since Cycle 2, were inspected by a team from the Westinghouse Nuclear Fuel Division. The optimized assemblies were found to be in good condition following two cycles of irradiation, and suitable for continued use.

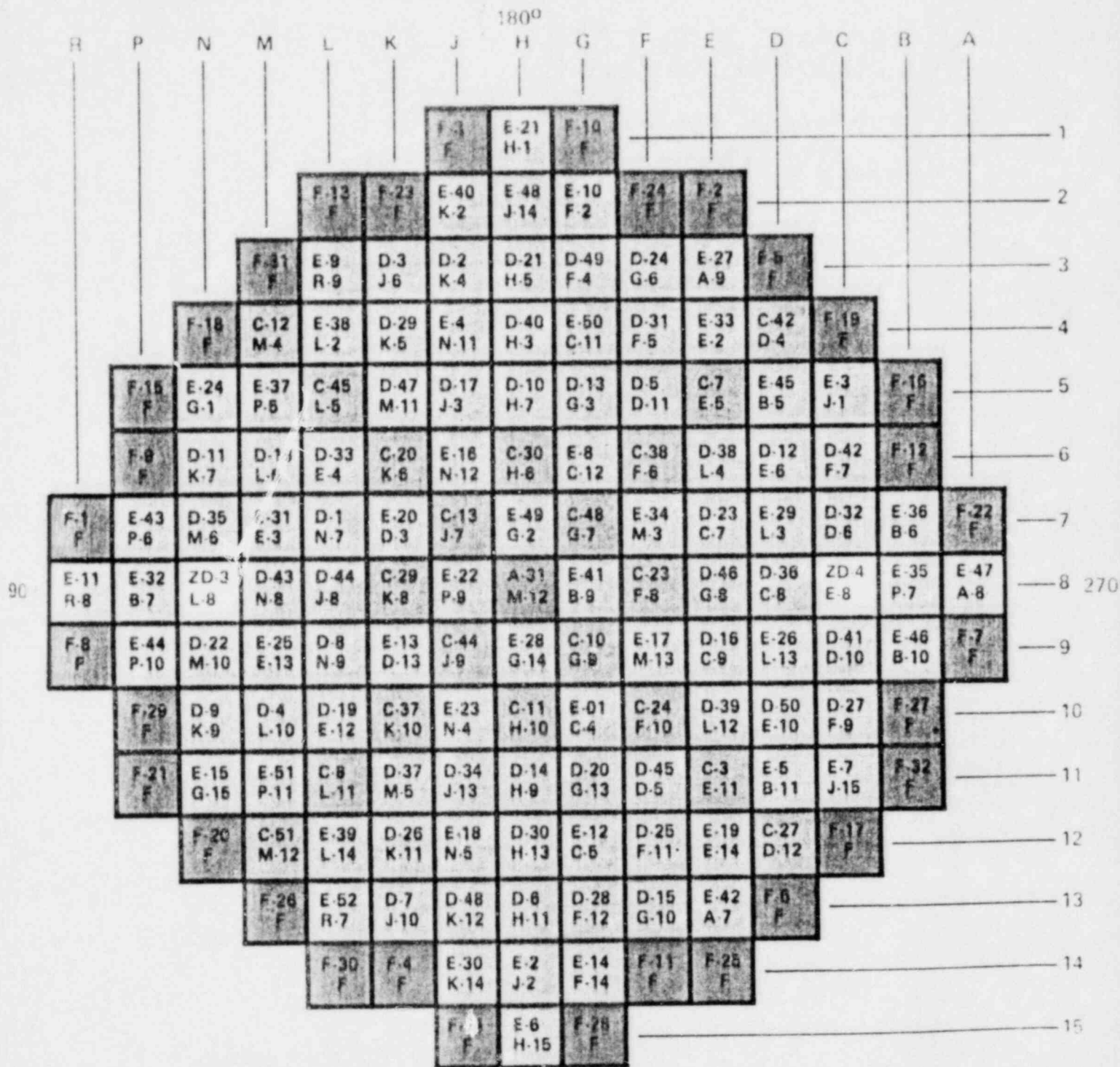
When the refueling operations were completed, the core was visually scanned and videotaped with an underwater TV camera to verify the correct location of each fuel assembly.

## 2.2 CYCLE 4 CORE DESCRIPTION

The as-loaded Cycle 4 core is depicted in Figures 2.1 through 2.3, which give the location of each fuel assembly and insert, and the assembly enrichment. The Cycle 4 core consists of 1 Region-1 fuel assembly, 20 Region-3 assemblies, 50 Region-4 assemblies, 2 Region-4A optimized test assemblies, 52 Region 5 assemblies and 32 Region 6 assemblies. Fuel assembly inserts consist of 48 full length rod cluster control assemblies, 107 thimble plug inserts, and 2 secondary sources.

FIGURE 2.1

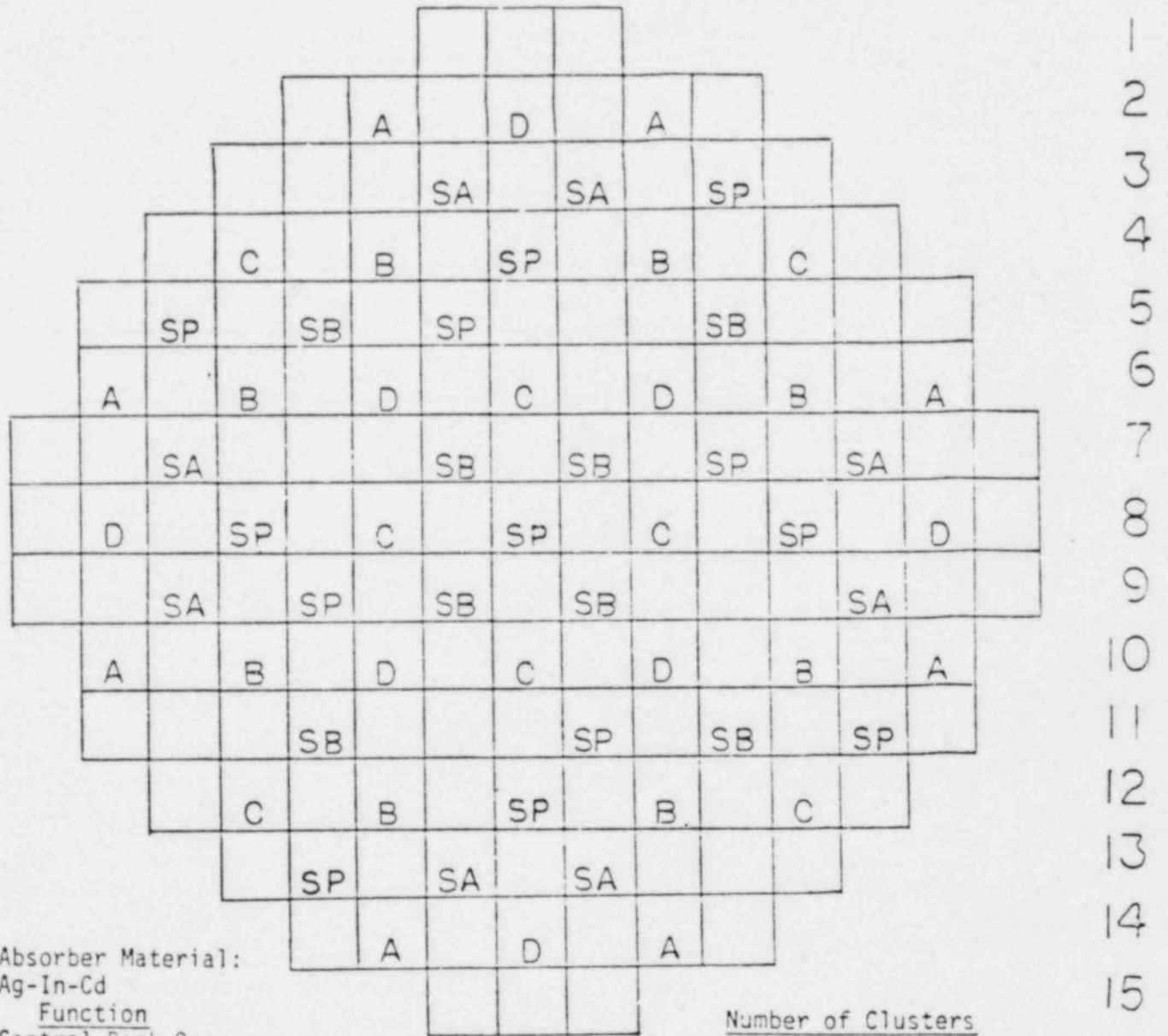
ALA Unit 1, Cycle 4 Reference Loading Pattern



COLOR CODE	ID XX	ID XX	ID XX	ID XX	ID XX	ID
REGION	1 (A)	3 (C)	4 (D)	4A (ZD)	5 (E)	6 (F)
FROM CYCLE	1	3	3	3	3	FEED
W/O U <sub>235</sub>	2.115	3.102	3.113	3.108	2.801	3.00

LEGEND:  
 ID: ASSEMBLY IDENTIFICATION  
 ZD: DEMONSTRATION ASSEMBLY  
 XX: PREVIOUS CYCLE LOCATION

R P N M L K J H G F E D C B A



Absorber Material:

Ag-In-Cd

Function

Control Bank D

Control Bank C

Control Bank B

Control Bank A

Shutdown Bank  $S_B$

Shutdown Bank  $S_A$

Number of Clusters

8

8

8

8

8

8

8

SP (Spare Rod Locations)

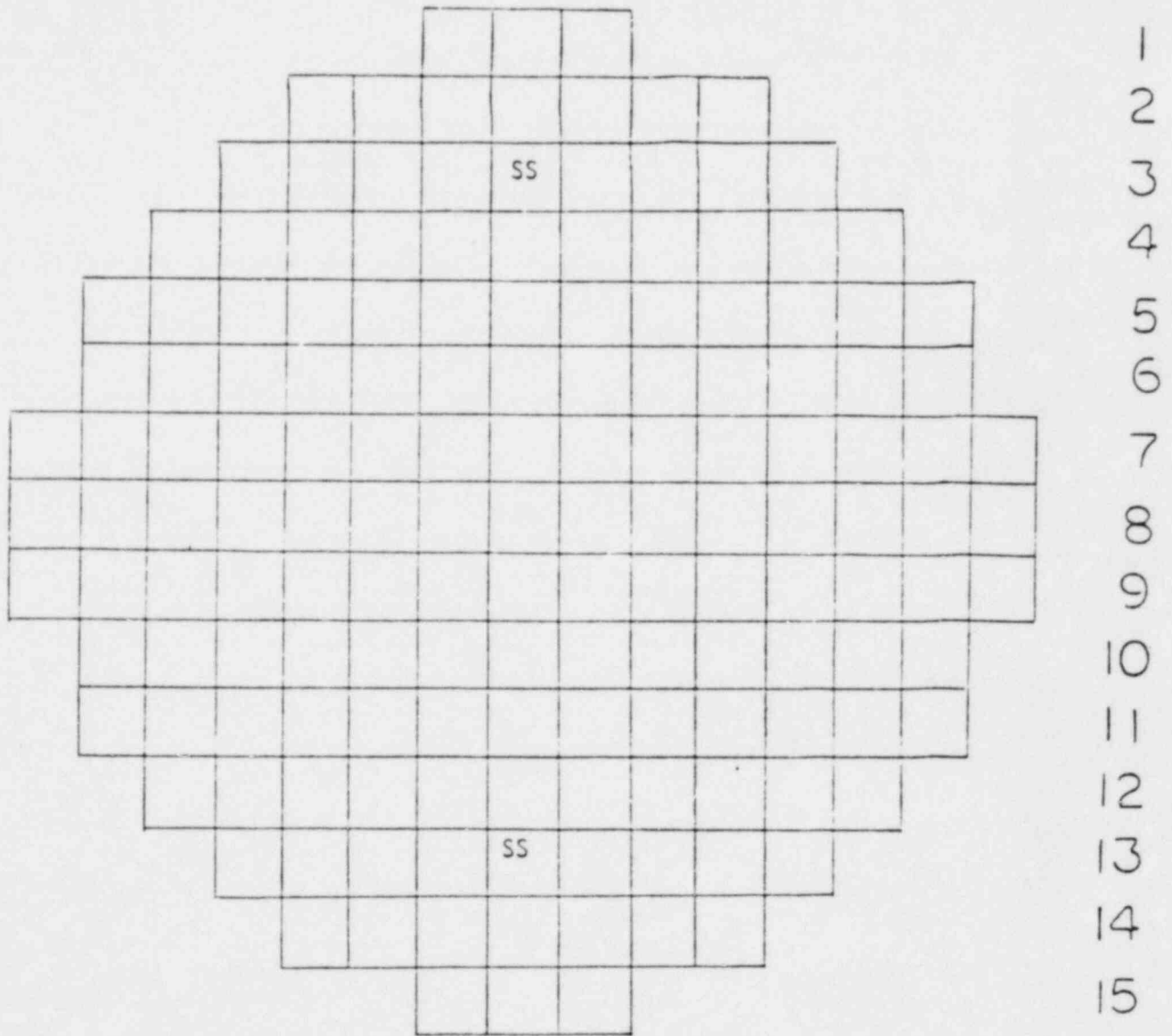
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Figure 2.2 CONTROL ROD LOCATIONS

FIGURE 2.3

Source Assembly Locations

R P N M L K J H G F E D C B A



SS Secondary Source



### 3.0 CONTROL ROD DROP TIME MEASUREMENT

(Procedure 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 rod drop time is in compliance with Technical Specification requirements.

#### SUMMARY OF RESULTS

For the Hot-full flow condition ( $T_{avg} \geq 541^{\circ}F$  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  $\leq 2.2$  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 1.78 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.

<u>TEST</u> <u>CONDITIONS</u>	<u>MEAN TIME TO</u> <u>DASHPOT ENTRY</u>	<u>MEAN TIME TO</u> <u>DASHPOT BOTTOM</u>
Hot-Full Flow	1.604 sec	2.163 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 movable gripper coils were examined and rod stepping speed measurements were conducted. All results were satisfactory.



#### 4.0 INITIAL CRITICALITY

(Procedure FN<sup>2</sup>-1-ETP-80)

##### 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 was achieved during dilution mixing at 0224 hours on March 3, 1982. The reactor was allowed to stabilize at the following critical conditions: RCS pressure- 2235 psig, RCS temperature- 547°F, intermediate range power-  $1.2 \times 10^{-8}$  amp, RCS boron concentration- 1345 ppm, and Control Bank D position- 179.5 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 and immediately following initial criticality to demonstrate that adequate overlap existed.

## 5.0 CONTROL ROD AND BORON WORTH MEASUREMENTS

(Procedures FNP-1-ETP-81, -83, -84, -85, -86, and -88)

### PURPOSE

The objectives of these procedures were: (1) to measure the differential and integral reactivity worth of each control rod bank, both individually and when moving in overlap, (2) to determine the differential boron worth over the range of control bank movement, and (3) compare results with the design calculations.

### SUMMARY OF RESULTS

The results of the control bank worth measurements both for banks moving individually and in overlap mode, together with boron worth determinations are summarized in Table 5.1. All measurements satisfied their respective review criteria.

TABLE 5.1

## SUMMARY OF CONTROL ROD AND BORON WORTH MEASUREMENTS

<u>Rod Configuration</u>	<u>Avg. Boron Conc. (ppm)</u>	<u>Predicted Bank Worth &amp; Review Criteria (pcm)</u>	<u>Measured Bank Worth (pcm)</u>	<u>Percent Difference</u>	<u>Design Boron Worth (pcm/ppm)</u>	<u>Measured Boron Worth (pcm/ppm)</u>
D	1324	1062 ± 159	1055.6	0.60%	-9.43	-8.29
D + C	1226	901 ± 135	890.0	1.22%	-9.48	-8.98
D + C + B	1095	1649 ± 247	1640.2	0.53%	-----	-9.98
D + C + B + A	945	1216 ± 182	1234.0	-1.48%	-----	-9.65
12 Cumulative Data from Control Banks moving individually during dilution		4828 ± 483	4819.8	0.17%	-----	-9.70
Cumulative Data from Control Banks moving in overlap during boration		4828 ± 193	4741.4	-1.80%	-----	-9.11

Conditions of Measurement: Hot Zero Power (547°F; 2235 psig)

6.0 ARO HZP FLUX DISTRIBUTION, MODERATOR TEMPERATURE  
COEFFICIENT, AND BORON ENDPOINTS

(Procedures FNP-1-ETP-82, -83, -84, -85 and -86)

PURPOSE

The objectives of these procedures were to:

- (1) determine the core flux distribution for the HZP all-rods-out configuration; (2) determine the hot zero power isothermal and moderator temperature coefficients in the all-rods-out and other configurations; and
- (3) measure the boron end point concentrations for the ARO, D-in, D + C-in, D + C + B-in and the D + C + B + A-in rod configurations.

SUMMARY OF RESULTS

Table 6.1 gives a tabulation of the measured boron end point concentrations compared with the design values for each rod configuration considered. The design acceptance criterion for the all-rods-out critical boron concentration was satisfactorily met.

Table 6.2 is a tabulation of measured isothermal and moderator temperature coefficients for the all-rods-out, Bank D-inserted, and Banks C + D-inserted configurations. Although the design acceptance criterion for the ARO isothermal temperature coefficient was met, the moderator temperature coefficient was determined to be positive. A Special Report describing the operating limits established for Control Bank D withdrawal was submitted to the NRC as required by Section 3.1.1.4 of the Technical Specifications.

TABLE 6.1

## HZP BORON ENDPOINT CONCENTRATIONS

Rod Configuration	Measured $C_B$ (ppm)	Design-Predicted $C_B$ (ppm)
ARO	1374.5	1367 $\pm$ 50 ppm*
D in	1274.0	1254
D+C in	1178.5	1158
D+C+B in	1012.1	
D+C+B+A in	878.6	

\*Design Acceptance Criterion.



TABLE 6.2

## HZP ISOTHERMAL AND MODERATOR TEMPERATURE COEFFICIENT

Rod Configuration	Boron Concentration	Measured $\alpha_T$	Calculated $\alpha_{mod}$	$\alpha_T$ Design Acceptance Criterion
	ppm	pcm/°F	pcm/°F	pcm/°F
All Rods Out	1374.5	-1.22	+0.78	-1.8 ± 3
Bank D In	1274.0	-4.31	-2.31	
Banks C & D In	1178.5	-6.78	-4.78	

$\alpha_T$  - Isothermal temperature coefficient, includes -2.0 pcm/°F doppler coefficient

$\alpha_{mod}$  - Moderator only temperature coefficient

## 7.0 POWER ASCENSION PROCEDURE

(FNP-1-ETP-100)

### 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 map at 44% power.

Flux maps were taken at 44%, and 74% power. The results for these maps were within Technical Specification limits.

A preliminary incore/excore calibration verification was performed at 44%, and the final calibration was performed at approximately 74% power. Results of the final incore/excore calibration are given in Section 8.0.

## 8.0 INCORE-EXCORE DETECTOR CALIBRATION

(Procedure 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 obtaining data for calibrating the delta flux penalty to the overtemperature  $\Delta T$  protection system, and for calibrating the control board and plant computer axial flux difference (AFD) channels.

### SUMMARY OF RESULTS

Preliminary and final verifications of excore AFD channel calibration was performed at 44% and 74% power, respectively. The flux maps for the final verification were run at average percent core axial offsets of +17.555, +0.22, -13.084 and -22.322, as determined from the incore results.

The 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  $\Delta T$  setpoint. (See Figure 8.1)

FIGURE 8.1

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

CHANNEL N41:

$$\begin{aligned} \text{I-Top} &= 1.057*AO + 254.08 \mu\text{a} \\ \text{I-Bottom} &= -1.825*AO + 262.27 \mu\text{a} \end{aligned}$$

CHANNEL 42:

$$\begin{aligned} \text{I-Top} &= 1.174*AO + 255.63 \mu\text{a} \\ \text{I-Bottom} &= -1.918*AO + 259.00 \mu\text{a} \end{aligned}$$

CHANNEL N43:

$$\begin{aligned} \text{I-Top} &= 1.055*AO + 250.42 \mu\text{a} \\ \text{I-Bottom} &= -1.989*AO + 277.80 \mu\text{a} \end{aligned}$$

CHANNEL N44:

$$\begin{aligned} \text{I-Top} &= 1.154*AO + 244.91 \mu\text{a} \\ \text{I-Bottom} &= -1.819*AO + 255.68 \mu\text{a} \end{aligned}$$

## 9.0 REACTOR COOLANT SYSTEM FLOW MEASUREMENT

(Procedure 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 287,128 gpm, which meets the above criterion.