ALABAMA POWER COMPANY
JOSEPH M. FARLEY NUCLEAR PLANT
UNIT NUMBER 1, CYCLE 6
STARTUP TEST REPORT

PREPARED BY PLANT REACTOR ENGINEERING GROUP

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1.0 INTRODUCTION

The Joseph M. Farley Unit 1 Cycle 6 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 6 core loading consists of 77 new and 80 reused 17 x 17 fuel assemblies as tabulated in ¶ 2.2.

Unit 1 began commercial operations in December 1, 1977 and completed cycle 5 on February 10, 1984 with an average core burnup of 11096.8 MWD/MTU.

2.0 FUEL INSPECTION AND CORE REFUELING

References

1. Westinghouse Refueling Procedure FP-ALA-R5
2. Westinghouse WCAP-10525 (The Nuclear Design and Core Management of the Joseph M. Farley Unit 1 Power Plant Cycle 6)

2.1 Cycle 5 Fuel Inspection

All fuel assemblies were unloaded from the reactor core to permit Control Rod Drive Mechanism Split Pin modification. Each fuel assembly was visually inspected with binoculars during the core unload. No significant defects or damage were noted during the visual inspection.

The visual inspection was followed by vacuum sipping of each fuel assembly. General Electric gaseous vacuum sipping equipment was utilized for this testing. Final sipping results showed evidence that two assemblies (F-36 and E-09) leaked and another assembly (F-44) was suspected of leaking. Assembly E-09 was not scheduled for use in Cycle 6. Assemblies F-36 and F-44 had originally been scheduled for reuse in Cycle 6; therefore, two D assemblies which had not been used in Cycle 5 were substituted in baffle positions in Cycle 6.

Additionally, F-36, E-09, F-44 and ten other assemblies were tested for leakage using an ultrasonic method developed by Brown Boveri Reaktor of Germany. Failed rods were confirmed during ultrasonic testing of F-36 and E-09. F-44 and the other ten assemblies tested showed no evidence of leakage by the ultrasonic method.

TV visual inspections were also performed on baffle assemblies, F-36, E-09 and F-44. The only significant defect noted during the TV inspection was a clad crack on a peripheral rod in assembly E-09. This failure was a "T" shaped crack where the top end plug connects to the fuel rod.

2.2 Cycle 6 Core Refueling

The Cycle 6 core loading commenced on March 27, 1984 following the completion of Control Rod Drive Mechanism Split Pin modifications, and was completed on March 29, 1984. The as-loaded Cycle 6 core is depicted in Figures 2.1 - 2.3. The number of assemblies in the various regions of the Cycle 6 core is tabulated below:

Region	No. of Fuel Assemblies
4	2
6	38
7	40
8A	44
8B	33

Fuel assembly inserts consist of 48 full length control rods, 2 secondary sources, 49 burnable poison rod inserts, and 58 thimble plug inserts.

FIGURE 2.1

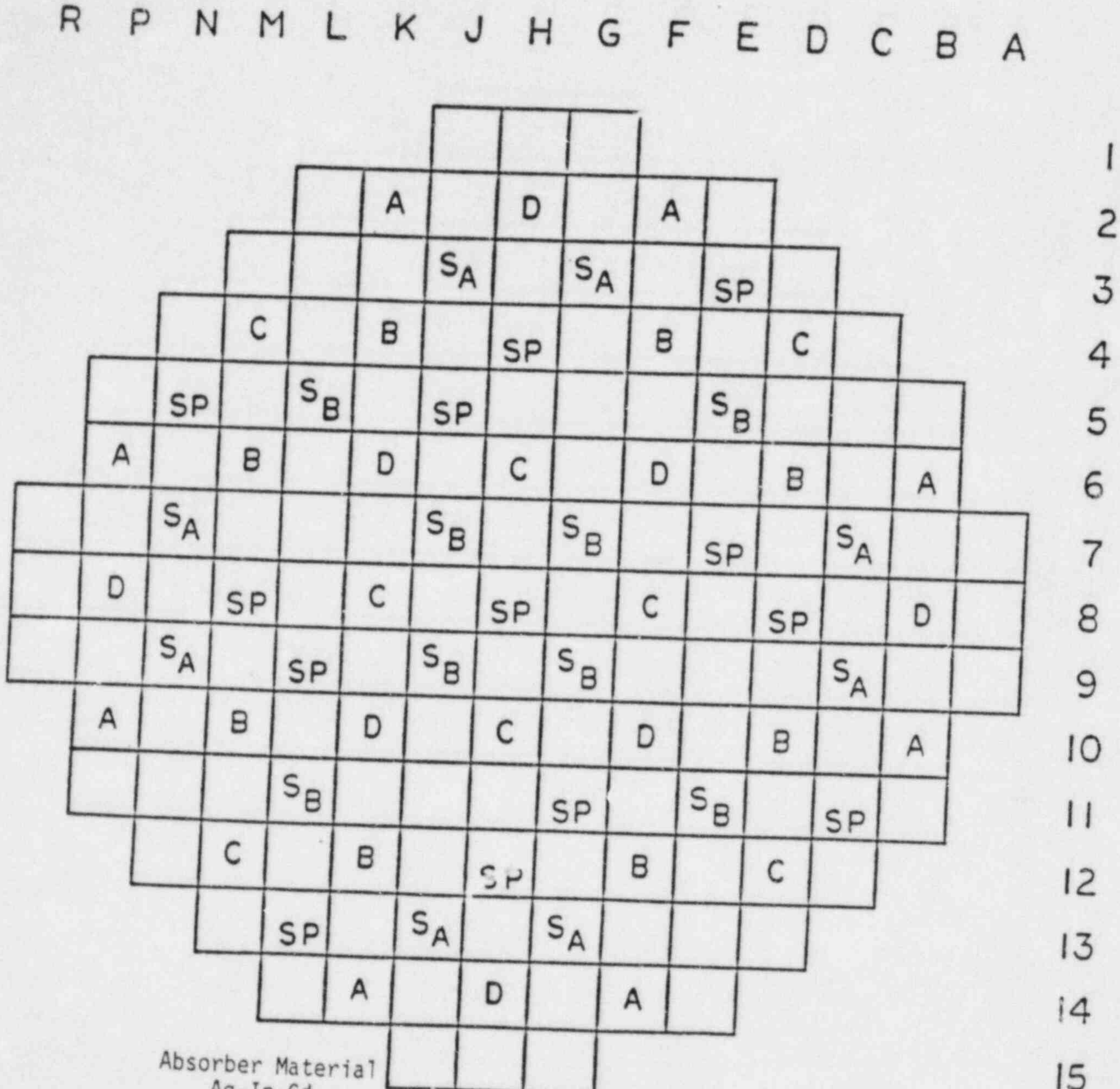
ALA CYCLE 6 LOADING PATTERN

	R	P	N	M	L	K	J	H	G	F	E	D	C	B	A
1							G-23 J-4	H-03 F	G-35 G-4						
2					F-04 D-8	H-16 F	H-47 F	G-24 F-7	H-54 F	H-19 F	F-24 H-4				
3				F-03 L-6	H-57 F	H-67 F	G-36 N-6	G-3 SS J-6	G-09 C-6	H-68 F	H-46 F	D-16 E-9+			
4			F-01 K-5	H-31 F	H-76 F	G-13 L-4	H-33 F	G-06 B-7	H-02 F	G-25 E-4	H-49 F	H-41 F	F-22 F-5		
5		F-26 G-8	H-62 F	H-63 F	F-13 H-2	H-08 F	F-48 L-3	H-26 F	F-49 E-3	H-27 F	F-16 B-8	H-59 F	H-60 F	F-09 H-7	
6		H-23 F	H-53 F	G-31 M-5	H-24 F	F-40 K-2	H-15 F	G-05 J-2	H-18 F	F-42 F-2	H-01 F	G-04 D-5	H-56 F	H-30 F	
7	G-28 M-7	H-45 F	G-07 K-3	H-14 F	F-34 N-5	H-10 F	G-26 D-7	F-46 R-8	F-45 B-6	H-38 F	F-52 C-5	H-36 F	G-19 F-3	H-66 F	F-10 E-6
8	H-29 F	G-29 K-7	G-10 J-10	G-32 P-7	H-44 F	G-02 J-14	F-35 H-1	H-52 F	F-50 H-15	G-11 G-2	H-28 F	G-21 B-9	G-39 G-6	G-01 F-9	H-25 F
9	F-14 L-10	H-51 F	G-30 K-13	H-35 F	F-41 N-11	H-09 F	F-39 B-10	F-43 A-8	G-18 M-9	H-40 F	F-33 C-11	H-07 F	G-27 F-13	H-73 F	G-34 D-9
10		H-17 F	H-71 F	G-15 M-11	H-34 F	F-37 K-14	H-22 F	G-22 G-14	H-37 F	F-51 F-14	H-39 F	G-16 D-11	H-65 F	H-43 F	
11		F-12 J-8	H-77 F	H-64 F	F-21 P-8	H-04 F	F-47 L-13	H-42 F	F-38 E-13	H-05 F	F-25 H-;4	H-61 F	H-70 F	F-29 H-9	
12			F-08 K-11	H-12 F	H-50 F	G-08 L-12	H-06 F	G-17 P-9	H-13 F	G-12 E-12	H-75 F	H-21 F	F-07 F-11		
13				D-01 L-7 +	H-48 F	H-55 F	G-14 N-10	G-38 SS G-10	G-37 C-10	H-69 F	H-58 F	F-28 E-10			
14					F-23 M-8	H-20 F	H-72 F	G-20 K-9	H-74 F	H-11 F	F-11 H-12				
15							G-33 J-12	H-32 F	G-40 G-12						

xx	Assembly ID
yy	Previous Cycle Location
ss	Secondary Source Location
+	Cycle 4 Location

FIGURE 2.2

CONTROL ROD LOCATIONS

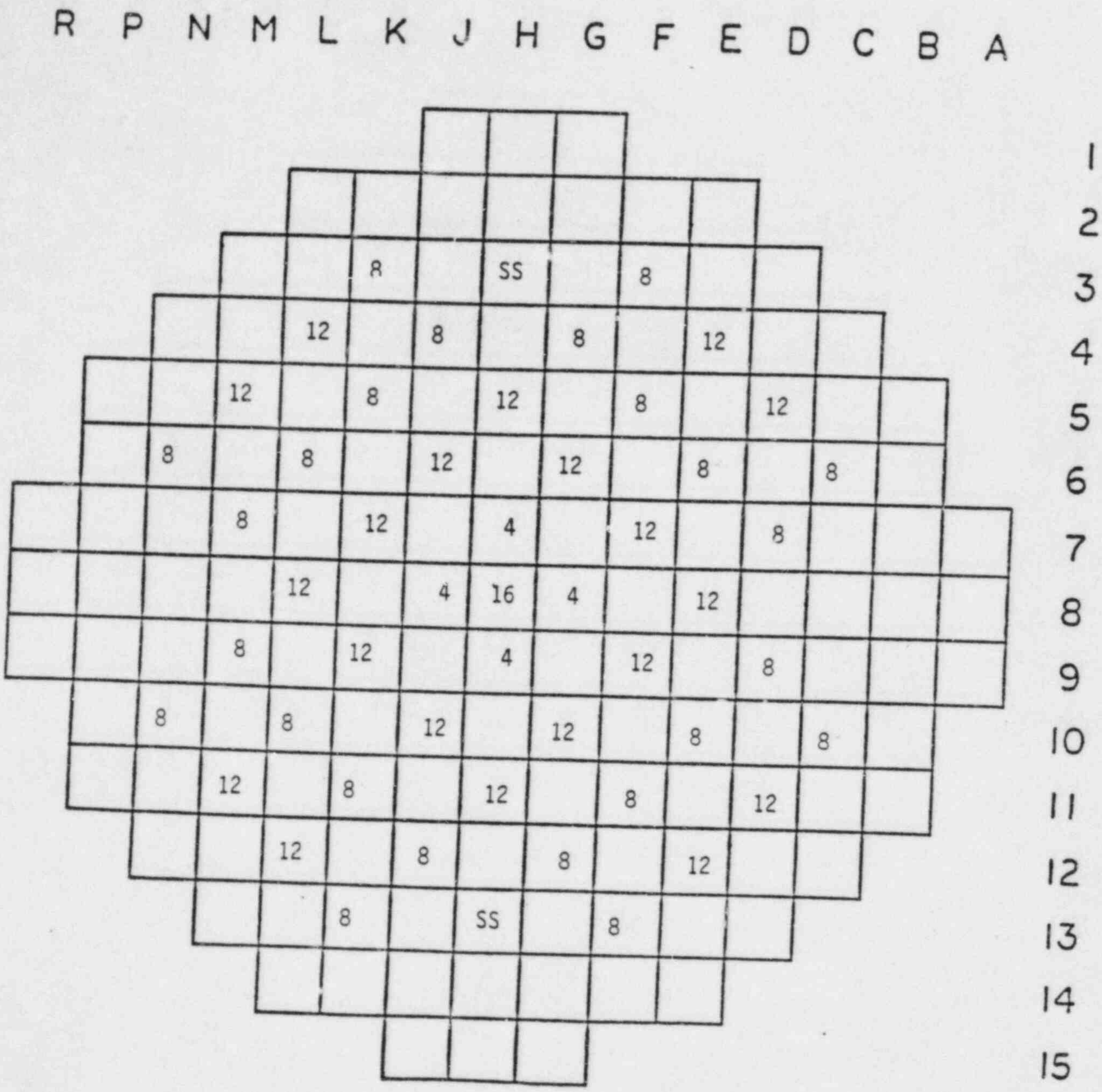


Absorber Material
Ag-In-Cd

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 2.3

BURNABLE POISON AND SOURCE ASSEMBLY LOCATIONS



SS Secondary Source

464 Fresh 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-flow condition (Tavg $\geq$ 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 1.81 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:

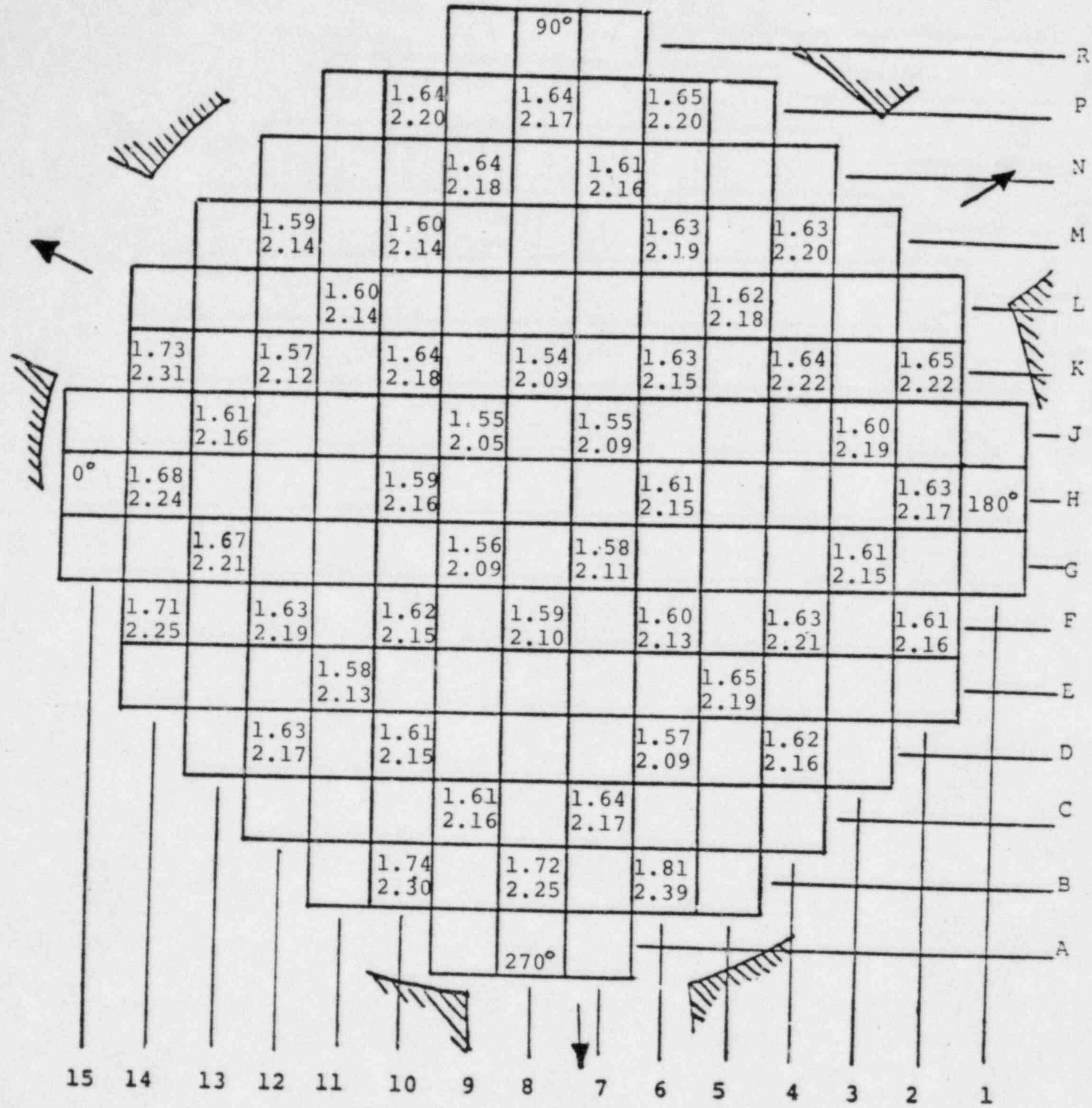
Test Conditions	Mean Time To Dashpot Entry	Mean Time to Dashpot Bottom
Hot Full-flow	1.626 sec.	2.173 sec.

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.

NORTH

FIGURE 3.1

UNIT 1 CYCLE 6



DRIVE LINE "DROP TIME" TABULATION

TEMPERATURE - 547 F PRESSURE- 2100 psig % FLOW - 100

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

DATE- 4/20/84

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 6 was achieved during dilution mixing at 0616 hours on April 22, 1984. The reactor was allowed to stabilize at the following critical conditions: RCS pressure - 2235 psig, RCS temperature - 545 °F, intermediate range power - 1.3×10^{-8} amp, RCS boron concentration - 1807 ppm, and Control Bank D position - 185 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

Purpose

The objective of these measurements was 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 (MTC) was found to be positive (+ 4.55 pcm/°F) as expected from the core design. Technical Specification 3.1.1.3.a was changed before the outage to allow a maximum MTC of +5.0 pcm/°F. The design acceptance criterion for the ARO critical boron concentration was satisfactorily met. (See Table 5.1.)

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 power satisfied the design criteria for the maximum positive percent error, but not for the maximum negative percent error.

The design criteria states that the percent error between measured and expected relative fuel assembly powers should be within ± 10% for assemblies with relative powers $\geq$ 0.9, and within ±15% for assemblies with relative powers $<$ 0.9. Two assemblies failed to meet these criteria: The assembly in core position L-11 (relative power = 0.955) showed an error of -10.7%, and the error for the assembly in location L-12 (relative power = 1.123) was also -10.7%. In addition the HZP, ARO flux map indicated that the incore tilt exceeded the design criterion of 1.02. (See Table 5.2.)

Westinghouse was notified of the assemblies that failed the relative power criteria and of the incore tilt exceeding 1.02. Westinghouse agreed that power escalation could continue up to 75% power. In addition, a rod insertion limit of D at 150 steps was recommended as long as the incore tilt exceeded 1.02.

Subsequent flux maps at 35% and 44% full power indicated that the percent difference between measured and expected assembly power decreased to well within the design acceptance criteria in both assemblies. These two full core flux maps also indicated 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_I (pcm/°F)	Calculated α_{Mod} (pcm/°F)	Design α_{Mod} (pcm/°F)
All Rods Out	1799	+1.94	+4.55	+4.05

α_I - Isothermal temperature 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	1804	1792 ± 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 P_i of assembly i.

Item	Value	P_i	Design Criterion
Maximum positive percent error	+15.0%	0.564	± 15% for $P_i < 0.9$
Maximum negative percent error	-10.7%	1.123	± 10% for $P_i \geq 0.9$

B. Incore Quadrant Tilt:

Maximum Incore Tilt	Design Criterion
1.0528*	$\leq$ 1.02

*The measured incore tilts at 35% and 44% power were 1.0183 and 1.0136, respectively.

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 comparision 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	513 ± 77	517.4	+0.9
Control B (Ref.)	1401 ± 140	1353.5*	-3.4
Control C	1018 ± 153	956.0	-6.1
Control D	1056 ± 158	1006.4	-4.7
Shutdown A	959 ± 144	949.0	-1.0
Shutdown B	1079 ± 162	991.9	-8.1
All Banks Combined	6026 ± 603	5774.2	-4.2

*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 operating 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.

In accordance with Westinghouse recommendations a rod insertion limit of 150 steps was established on Control Bank D. This was necessary due to the 1.0528 incore tilt indicated by the HZP flux map. The 35% power flux map incore tilt was below 1.02 and the insertion limit was discontinued.

Design-predicted NIS detector currents equal to 80% of the Cycle - 5 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 35%, 44% and 78% power. The results for the first two flux maps met all Technical Specification Limits. The 78% power map was performed while under the exception of Technical Specification 3.2.1.a.2.b for Incore-Excore recalibration. The results of these maps are summarized in Table 7.1

An incore/excore calibration check at 35% power indicated that a preliminary redetermination of the incore/excore intercept currents was necessary. This calculation was performed and new current values were issued to calculate quadrant power tilts. A full recalibration of the excore AFD channels was performed at approximately 78% power to comply with Technical Specification requirements. When 100% power was reached, the excore ambient tilts had to be rezeroed due to a shift in core axial tilt. The Incore-Excore recalibration is described in section 8.0.

TABLE 7.1

SUMMARY OF POWER ASCENSION FLUX MAP DATA

Parameter	Map 137	Map 139	Map 140
Date	4/27/84	5/3/84	5/5/84
Time	18:00	05:51	04:30
Avg % Power	34.97	43.86	77.89
Max. F_Q (Z)	2.1459	2.0725	2.0255
Max. FΔH	1.5722	1.5181	1.5329
Max. Power Tilt*	1.0183	1.0136	1.0201
Avg. Core % A.O.	+4.711	+7.148	+6.184

*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

Quadrant power tilt calculations performed at 35% power indicated new 100% normalized zero axial offset currents needed to be calculated. These calculations were completed according to Appendix C of FNP-1-STP-121. Subsequent quadrant power tilt calculations were performed using the new detector current values with satisfactory results. The Power Range Axial offset calibration check STP-121, was performed at 35% power. This procedure verified indicated axial offset was within three percent of the actual incore axial offset. Therefore, an interim incore-excore calibration was not required and power was increased to 78% for the complete incore-excore recalibration. Flux maps for incore-excore recalibration were run at approximately 78% power at average percent core axial offsets of +6.184, -12.002, -19.743, and +16.807, 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 for the overtemperature ΔT setpoint.

During power ascension, a channel deviation alarm occured. Investigation revealed that the ambient core axial tilt that was present when performing the incore-excore calibration had shifted. A full core flux map was performed at 100% power and a new core axial offset value was obtained. Using this new value, the excore detector equations derived at 78% power were normalized to the new core axial offset using the method prescribed in Appendix C of FNP-1-STP-121. The refinements made to the original recalibration equations are presented in Figure 8.1.

FIGURE 8.1

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

<u>CHANNEL N41</u>:

I - Top   = 1.0127 * A.O. + 192.73
I - Bottom = 1.0556 * A.O. + 190.14

<u>CHANNEL N42</u>:

I - Top   = 1.0263 * A.O. + 186.60
I - Bottom = 1.0718 * A.O. + 182.01

<u>CHANNEL N43</u>:

I - Top   = 0.9892 * A.O. + 184.15
I - Bottom = -1.1010 * A.O. + 197.91

<u>CHANNEL N44</u>:

I - Top   = 0.9619 * A.O. + 174.73
I - Bottom = 1.0645 * A.O. + 174.79

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 requirements 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 284,074.8 gpm, which meets the above criterion.