ALABAMA POWER COMPANY

JOSEPH M. FARLEY NUCLEAR PLANT

UNIT NUMBER 1, CYCLE 3

STARTUP TEST REPORT

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1.0 INTRODUCTION

The Joseph M. Farley Unit I Cycle 3 Startup Test Report addresses the tests identified as requirements 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.

The Joseph M. Farley Nuclear Plant consists of two pressurized water reactors, each rated at 2652 MWt. The 3-loop Nuclear Steam Supply Systems and Turbine Generator sets were supplied by Westinghouse, and provide a gross electrical output of 860 MW per unit. The initial core loading for each reactor consists of 157 fuel assemblies composed of 17 x 17 rod arrays.

Unit I began Cycle 1 commercial operation on December 1, 1977, following completion of the initial startup and power ascension test program. Cycle 1 operation ended on March 8, 1979, with an average core burnup of 15,450 MWD/MTU. Cycle 2 operation ended on November 8, 1980, with an average core burnup of 10,177 MWD/MTU. Initial criticality for Cycle 3 was achieved on March 25, 1981, and full power operation was resumed on April 20, 1981.

2.0 Unit 1 CYCLE 3 CORE REFUELING

REFERENCES

1. Westinghouse Refueling Procedure FP-ALA-R2

2. Westinghouse WCAP 9761 (The Nuclear Design and Core Management of the Joseph M. Farley Unit 1 Power Plant Cycle 3)

3. Westinghouse FP-AP-256 (Joseph M. Farley Nuclear Design Revisions for Unit 1 Cycle 3)

2.1 FUEL SHUFFLE OPERATIONS

Nuclear fuel (52 assemblies) was received on site for Cycle 3, Core Region 5, during the period from February 19, 1980, to March 13, 1980. No problems of significance were disclosed by fuel receipt inspections.

The core fuel shuffle commenced on November 30, 1980. The shuffle resulted in: (1) the removal of all assemblies to the spent fuel pit; (2) shuffling of the Region 1, 3, and 4 assemblies into an approximate checkerboard pattern in the inner section of the core; (3) arranging the fresh (Region 5) assemblies into a ring surrounding the inner checkerboard; and (4) relocation of inserts (such as control rods), and replacement of spent burnable poison inserts with thimble plug inserts.

During the fuel shuffle, two optimized fuel assemblies (Region-4A) were inspected by a team from the Westinghouse Nuclear Fuel Division. As detailed in the Westinghouse Preliminary Report,

the optimized assemblies were found to be in good condition following one cyle of irradiation, and suitable for continued use. (A copy of the Westinghouse Report is attached as Appendix A.)

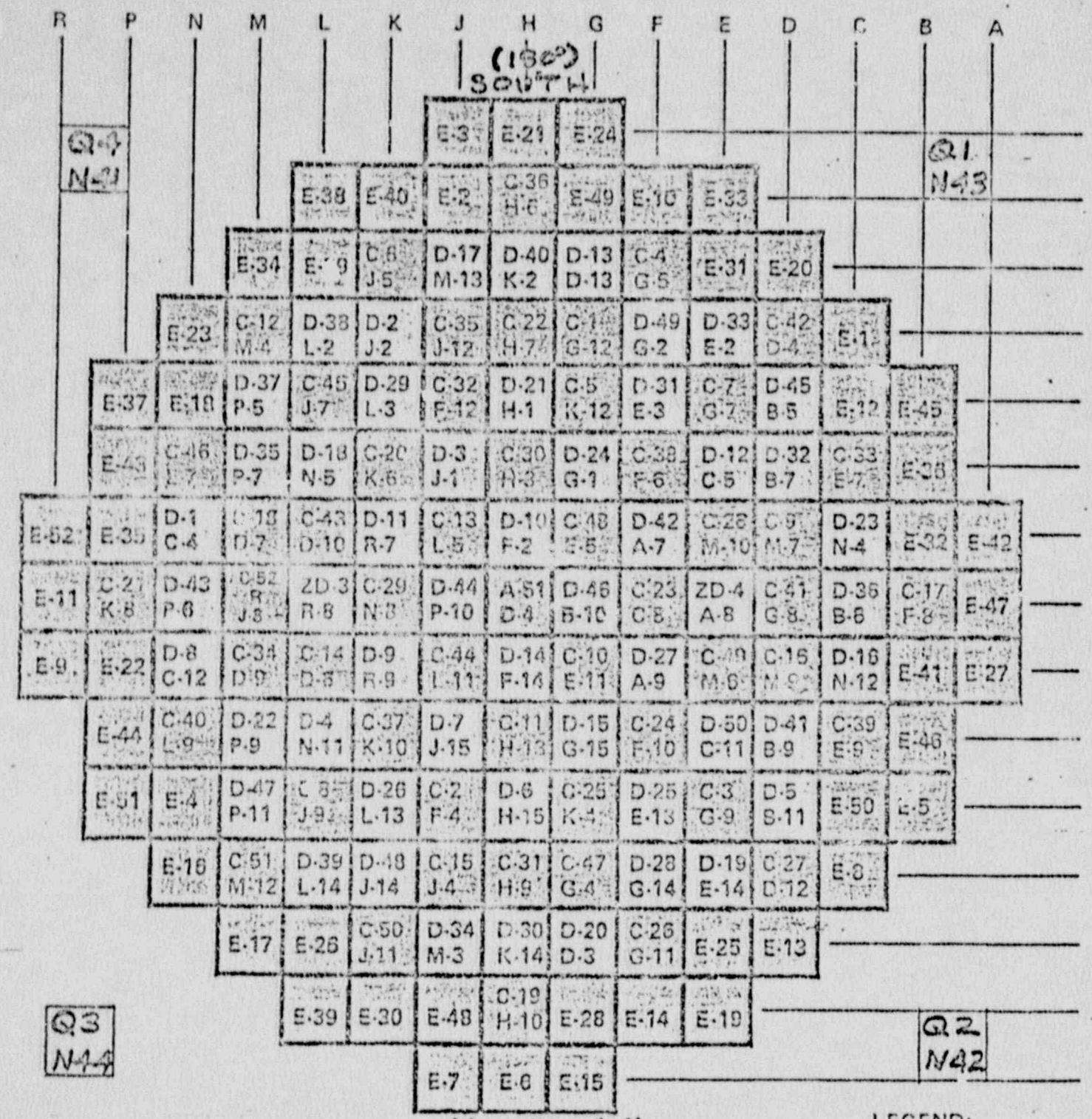
When the refueling operations were completed, the core was scanned and videotaped with an underwater TV camera to verify the correct location of each fuel assembly.

2.2 CYCLE 2 CORE DESCRIPTION

The as-loaded Cycle 3 core is depicted in Figures 2.1 through 2.3, which give the location of each fuel assembly and component, and provide information on assembly enrichment and burnup. The core is composed of 1 Region-1 fuel assembly, 52 Region-3 assemblies, 50 Region-4 assemblies, two Region-4A optimized test assemblies, and 52 Region 5 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 3 Reference Loading Pattern



CODE	A XX	C XX	D XX	ZD XX	E
REGION	1	3	4	4A	5
FROM CYCLE	1	2	2	2	FEED
W/O U_{235}	2.115	3.102	3.113	3.108	2.801

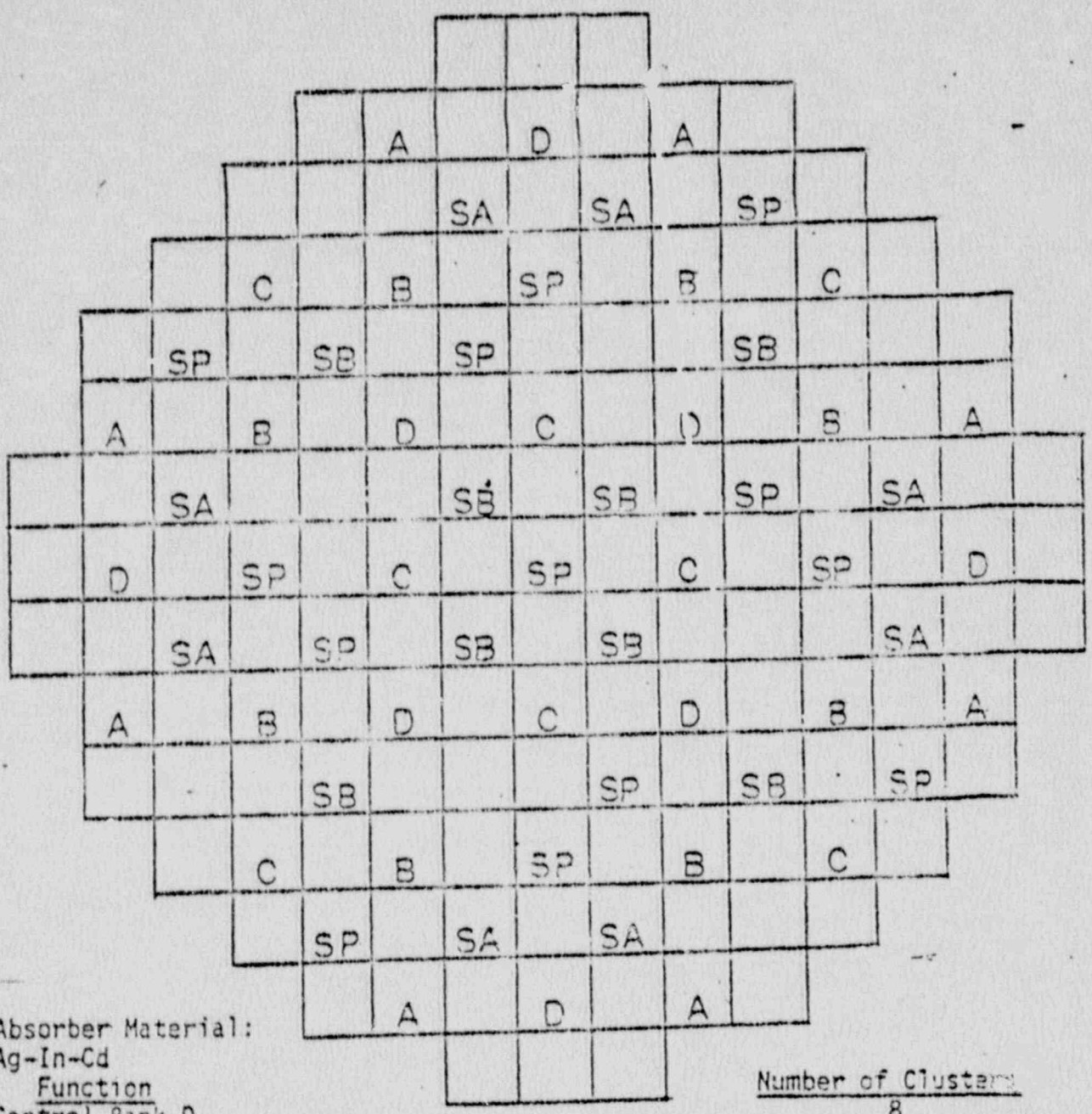
LEGEND:

ID: ASSEMBLY IDENTIFICATION

ZD: DEMONSTRATION ASSEMBLY

R: REMOVABLE FUEL RODS

XX: PREVIOUS CYCLE



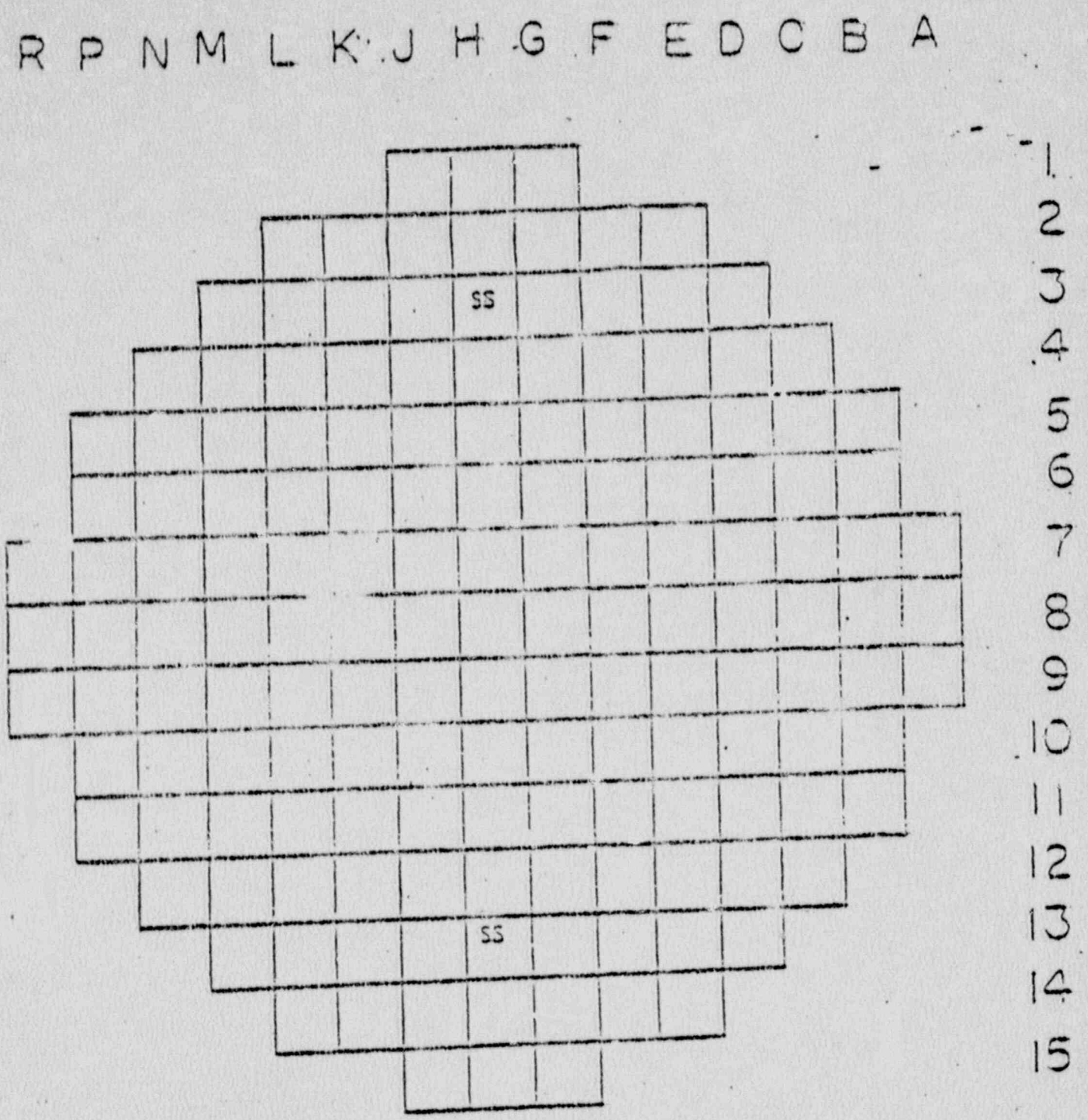
Absorber Material:
Ag-In-Cd

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 2.2 CONTROL ROD LOCATIONS

Figure 2.3

Source Assembly Locations



SS Secondary Source

3.0 CONTROL ROD DROP TIME MEASUREMENT

(Procedure FNP-1-STP-112)

PURPOSE

The purpose of this procedure was to measure the drop time of all full length control rods under hot-full flow conditions in the reactor coolant system- to insure system performance was in compliance with Technical Specification requirements.

SUMMARY OF RESULTS

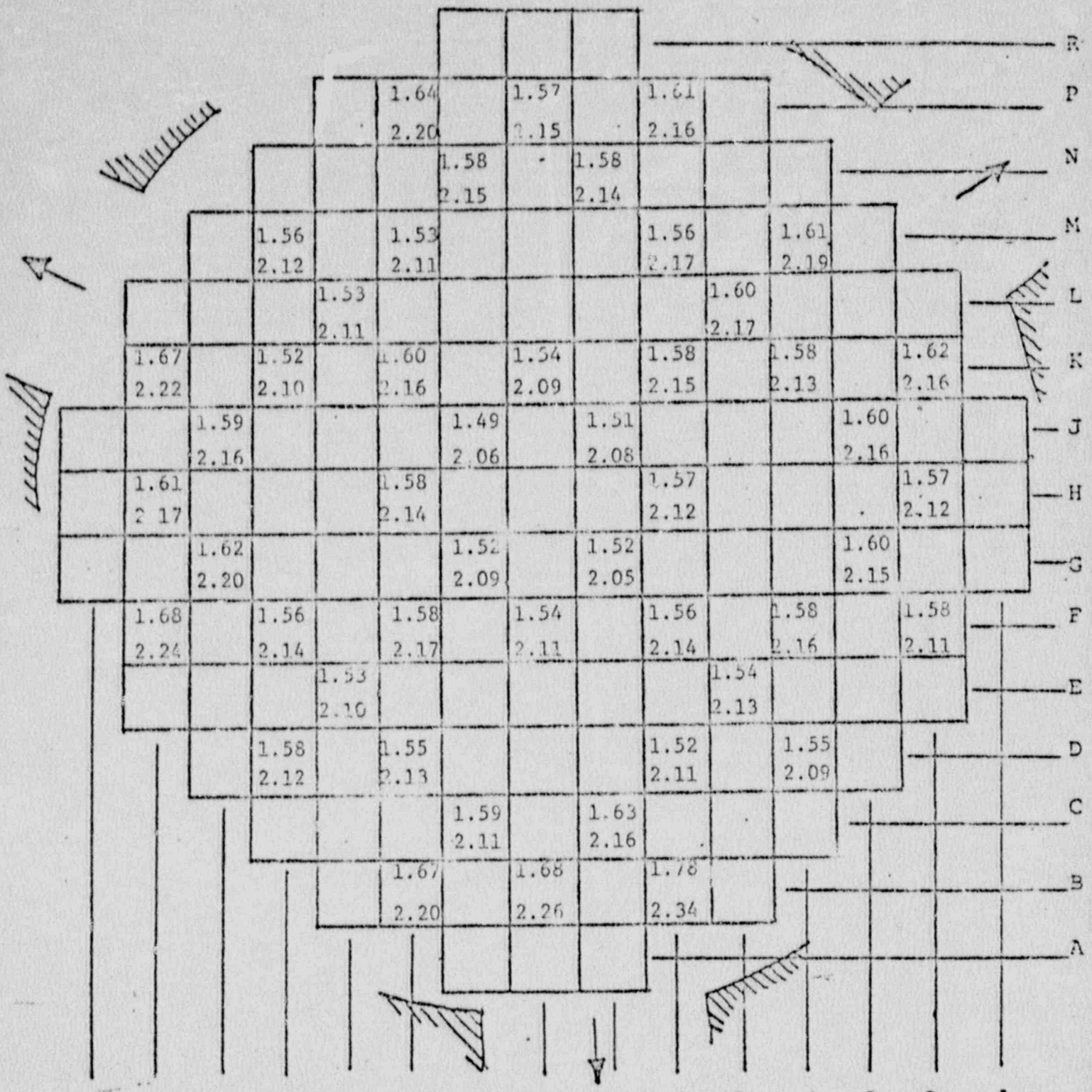
For the Hot-full flow condition ($T_{avg} \geq 541°F$ and all reactor coolant pumps operating) Technical Specification 3.1.3.4 requires that the rod drop time from the fully withdrawn position shall be $\leq$ 2.2 seconds from the beginning of stationary gripper coil voltage decay until dashpot entry. All full length rod drop times were measured to be less than 2.2 seconds. The longest drop time recorded was 1.780 seconds for rod B-6. A tabulation of the drop times for the remaining full length control rods is presented in core diagram form in Figure 3.1 together with drop times to dashpot bottom. Mean drop times are summarized below.

TEST CONDITIONS	MEAN TIME TO DASHPOT ENTRY	MEAN TIME TO DASHPOT BOTTOM
Hot-Full Flow	1.58 sec	2.15 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 speed measurements were conducted. All results were satisfactory.

NORTH

UNIT I CYCLE 3



DRIVE LINE "DROP TIME" TABULATION

TEMPERATURE - 547°F PRESSURE- 2236 PSIG % FLOW - 100

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

DATE- 3/22/81

FIGURE 3.1

4.0 INITIAL CRITICALITY

(Procedure FNP-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 0300 hours on March 25, 1981. Conditions were allowed to stabilize at $1X10^{-8}$ Amps in the intermediate range at a Control Bank D position of 145.5 steps and an RCS Boron Concentration of 1333 ppm. The approach to criticality proceeded without incident. Following stabilization, the point of adding nuclear heat was determined and a checkout of the reactivity computer 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 adequate overlap existed.

5.0 CONTROL ROD AND BORON WORTH MEASUREMENTS

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

PURPOSE

The purpose of these procedures was to measure the differential and integral reactivity worth of each control rod bank, both individually and when moving in overlap, and to determine the differential boron worth over the range of control bank movement.

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

Rod Configuration	Avg. Boron Conc. (ppm)	Predicted Bank Worth & Review Criteria (pcm)	Measured Bank Worth (pcm)	Percent Difference	Design Boron Worth (pcm/ppm)	Measured Boron Worth (pcm/ppm)
D	1325	1056 ± 158	1034.0	-2.1%	-9.39	-10.27
D + C	1200	1084 ± 162	1085.5	-.14%	-9.51	-11.32
D + C + B	1057	1646 ± 247	1599.5	-2.1%	-----	-9.17
D + C + B + A	887.4	1306 ± 196	1355.5	-3.79%	-----	-9.43
Cumulative Data from Control Banks moving individually during dilution		5092 ± 509	5074.5	-.34%	-----	-9.32
Cumulative Data from Control Banks moving in overlap during boration		5092 ± 203.7	4949.2	-2.8%	-----	-9.22

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) measure the hot zero power moderator temperature coefficient in the all-rods-out configuration; (2) Determine the core flux distribution for the all-rods-out configuration; 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 vendor's recommended 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 HZP all-rods-out configuration, together with the isothermal temperature coefficient obtained from core design curves. The design acceptance criterion for the isothermal temperature coefficient was satisfactorily met, and the moderator temperature coefficient was found to be negative, as required by plant Technical Specifications.

TABLE 6.1

HZP BORON ENDPOINT CONCENTRATIONS

Rod Configuration	Measured C_B (ppm)	Design-Predicted C_B (ppm)
ARO	1357.8	1374 $\pm$ 50 ppm*
D in	1252.4	1260
D+C in	1143.4	1145
D+C+B in	971.4	
D+C+B+A in	822.5	

*Design Acceptance Criterion.

TABLE 6.2

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	1357.8	-2.31	-0.31	-2.10 ± 3

α_T - Isothermal moderator temperature coefficient, includes -2.0 pcm/°F doppler coefficient

α_{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 the following:

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

SUMMARY OF RESULTS

To comply with the fuel warranty, 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.

The final, detailed alignment of the incore movable detector system was performed during power ascension (at power levels above 5%) prior to taking the 48% power flux map.

Flux maps were taken at 48%, and 74% power. The results, including power tilts, were well within Technical Specification Limits.

A preliminary incore/excore calibration verification was performed at 48%, and a final calibration was performed at 75% Power. Results of the final incore/excore calibration are given under Section 8.0.

8.0 INCORE-EXCORE DETECTOR CALIBRATION

(Procedure FNP-1-STP-121)

PURPOSE

The purpose of this procedure was to determine the relationship between power range upper and lower excore detector currents, and incore axial offset -- to provide data for 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

Preliminary and final verifications of excore AFD channel calibration were performed at 48% and 75% power, respectively. The flux maps for the final verification were run at average core axial offsets of +3.635, -12.601, -21.717, +10.17 and +15.29, 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 equations for each detector's top and bottom current.

Using these equations, the detector current calibration data was generated and utilized to recalibrate the delta flux instrumentation.