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J. D. Woodard
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February 22, 1993

Docket No. 50-348

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555

Joseph M. Farley Nuclear Plant
Unit 1 Cycle 12 - Startup Report

Gentlemen:

Enclosed is the Startup Report for Unit 1 Cycle 12. If you have any questions, please advise.

Respectfully submitted,

J. D. Woodard

EFB:cht-Ulcycl12.efb
NEL-93-0076
Enclosure

cc: Mr. S. D. Ebneter
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SOUTHERN NUCLEAR OPERATING COMPANY
JOSEPH M. FARLEY NUCLEAR PLANT

STARTUP TEST REPORT
UNIT 1 CYCLE 12

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APPROVED:

CL Buck ______________________ Technical Manager

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1.0 INTRODUCTION

The Joseph M. Farley Unit 1 Cycle 12 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 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 unit began commercial operations on December 1, 1977. The Cycle 12 core loading consists of 157 17 x 17 fuel assemblies, of which 105 are Westinghouse Low Parasitic (LOPAR) assemblies and the remaining 52 assemblies represent the first phase of transition to Westinghouse Vantage 5 fuel.

All thimble plug and burnable poison inserts have been removed from the Cycle 12 core and two new double encapsulated secondary source inserts were added to be activated for use in Cycle 13. The new sources provide compatability with Vantage 5 fuel and the double encapsulation gives an additional margin of protection against source material leakage into the reactor coolant. The burnable absorber inserts have been supplanted by Westinghouse Integral Fuel Burnable Absorbers (IFBAs) incorporated in the Vantage 5 assemblies. The design burnup capability of the Cycle 12 core is 16000 MWD/MTU.

Previous Cycle Completion Dates and Average Burnups

Cycle	Date Critical	Start of Cycle	EOL Date	EOL Burnup (MWD/MTU)	EOL Burnup (EFPD)	Total EFPY
1	08-09-77	08-18-77	03-08-79	15450	420.60	1.152
2	10-31-79	11-04-79	11-07-80	10177	276.70	1.910
3	03-25-81	04-03-81	09-10-81	5180	140.70	2.296
4	03-03-82	03-07-82	01-14-83	10622	288.10	3.085
5	03-28-83	03-30-83	02-10-84	11096	301.30	3.911
6	04-22-84	04-24-84	04-06-85	12238	333.58	4.825
7	05-26-85	05-27-85	10-03-86	17231	470.04	6.112
8	11-30-86	12-02-86	03-25-88	16190	443.26	7.326
9	05-20-88	05-21-88	09-23-89	17456	479.29	8.639
10	11-08-89	11-10-89	03-08-91	16910	464.17	9.911
11	05-18-91	05-21-91	09-25-92	17513	480.26	11.227

2.0 UNIT 1 CYCLE 12 CORE REFUELING

REFERENCES

1. Westinghouse Refueling Procedure FP-ALA-R11.
2. Westinghouse WCAP 13434 (The Nuclear Design and Core Management of the Joseph M. Farley Unit 1 Power Plant Cycle 12)

Unloading of the Cycle 11 core into the spent fuel pool commenced on 10-05-92 and was completed on 10-06-92 with no significant problems or delays. During the offload, each fuel assembly was inspected with binoculars for indications of damage or other problems. No visual defects were noted.

Radiochemistry detection of an iodine peak following a reactor trip approximately a year before the end of cycle shutdown indicated the probability that the Cycle 11 core contained leaking fuel assembles. Therefore, following core unload, each fuel assembly removed from the

Cycle 11 core was subjected to ultrasonic leak testing (UT). The UT program identified three leaking fuel assemblies (2B48, 2B30 and 2A47). In addition, high magnification TV examinations of the leaking assemblies disclosed that rod Q5 of assembly 2B30 had a hydride blister on the side and that the top end plug was displaced upward and to the side, leaving a visible opening. No visible defects could be found during the TV inspection of assemblies 2B48 and 2A47. Assemblies 2B30 and 2A47 were scheduled for discharge into the spent fuel pool for storage, but leaking assembly 2B48 was scheduled for reload into the Cycle 12 core. Therefore, 2B48 was replaced with assembly 2A14 in the Cycle 12 core design.

The Cycle 12 Core reload commenced on 11-3-92 and was completed on 11-5-92. The as-loaded, redesigned Cycle 12 core is shown in Figures 2.1 through 2.5

FIGURE 2.1: UNIT 1 CYCLE 12 REFERENCE LOADING PATTERN

	R	P	N	M	L	K	J	H	G	F	E	D	C	B	A
1							2A58	2B32	2A68						
2					2B55	2C04 R14	2D31	2D05 R44	2D34	2C11 R13	2B54				
3				2B57	2C39	2D36	2C05 R05	2D14	2C15 R01	2D42	2C61	2B56			
4			2B58	2D01 R47	2D46	2C27 R03	2C60	2C28 SS05	2C46	2C21 R29	2D48	2D02 R35	2B46		
5		2B52	2C47	2D49	2B41 R43	2D16	2B31	2D09	2B37	2D17	2B22 R26	2D50	2C51	2B50	
6		2C01 R21	2D43	2C19 R42	2D22	2C10 R02	2C34	2C50 R45	2C45	2C31 R07	2D23	2C18 R36	2D30	2C12 R19	
7	2A63	2D44	2C13 R12	2C59	2B18	2C58	2D25 R41	2C55	2D26 R09	2C38	2B05	2C40	2C02 R38	2D32	2A55
8	2B04	2D06 R24	2D13	2C08 SS09	2D10	2C36 R16	2C53	J24	2C48	2C56 R34	2D11	2C22 SS10	2D24	2D07 R48	2B23
9	2A62	2D39	2C30 R39	2C54	2B27	2C63	2D27 R06	2C43	2D28 R22	2C37	2B38	2C41	2C25 R4[illegible]	2D40	2A61
10		2C29 R18	2D37	2C17 R[illegible]7	2D15	2C06 R25	2C57	2C49 R30	2C62	2C14 R04	2D20	2C20 R10	2D38	2C07 R17	
11		2B59	2C52	2D51	2B6[illegible] R15	2D21	2B09	2D12	2B35	2D18	2B39 R28	2D45	2C64	2B53	
12			2B51	2D03 R27	2D52	2C16 R08	2C42	2C03 SS06	2C35	2C09 R20	2D47	2D04 R32	2B61		
13				2B49	2C44	2D33	2C32 R31	2D19	2C26 R33	2D35	2C33	2A14			
14					2B47	2C23 R23	2D29	2D08 R40	2D41	2[illegible]4 R11	2B60				
15							2A67	2B20	2A42						

↓ North

← Fuel Assembly Serial Number
← Insert Serial Number

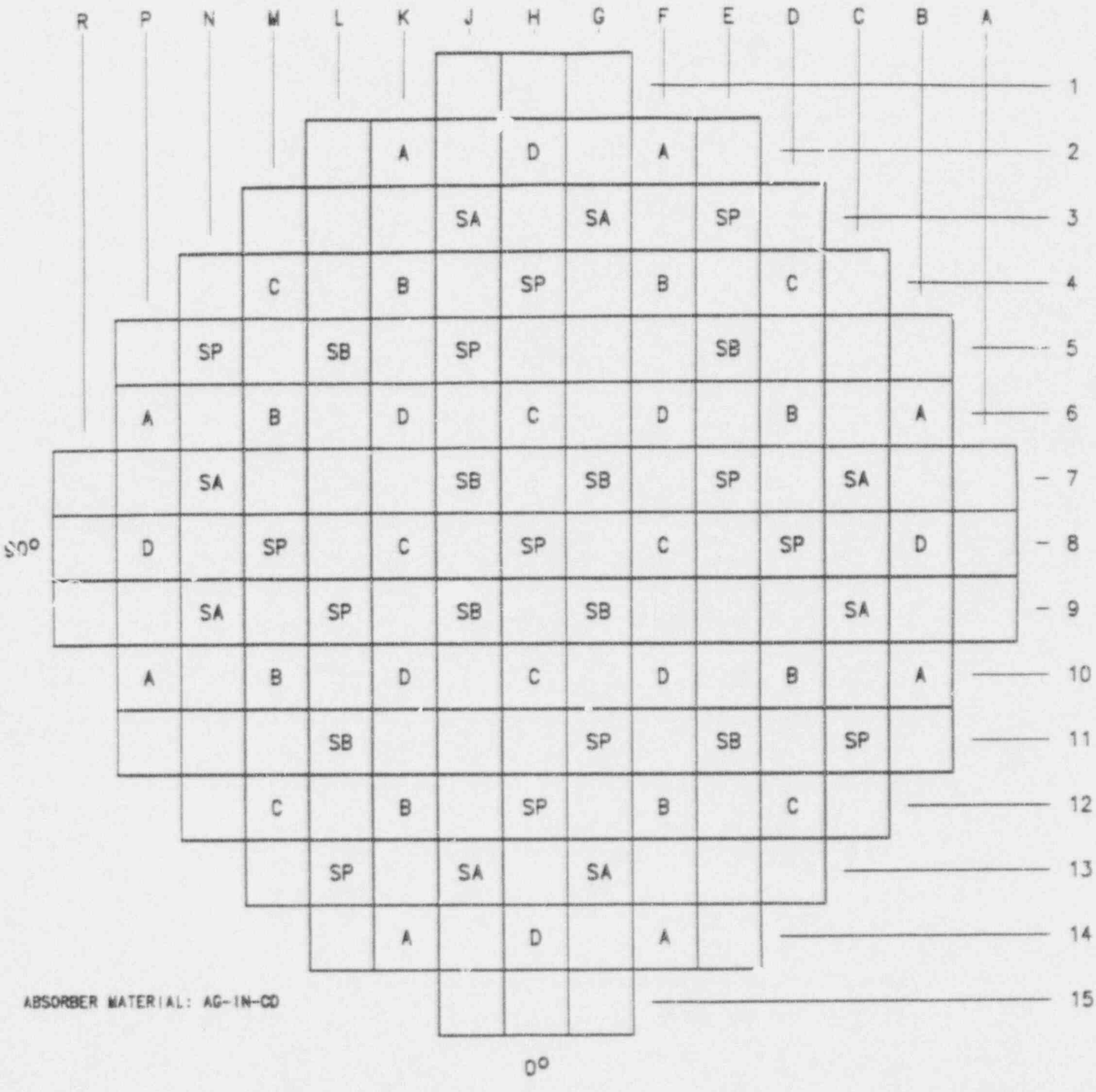
The Original w/o U-235 enrichments were:

Region 9A (J) assemblies......3.597%
Region 11A (2A) assemblies....3.805%
Region 11B (2A) assemblies....4.207%
Region 12A (2B) assemblies....3.803%
Region 12B (2B) assemblies....4.193%
Region 13A (2C) assemblies....3.801%
Region 13B (2C) assemblies....4.195%
Region 14A (2D) assemblies....3.800%
Region 14B (2D) assemblies....4.200%

No. of Fuel Assemblies:

Region 9A......1
Region 11A.....1
Region 11B.....8
Region 12A....16
Region 12B....15
Region 13A....32
Region 13B....32
Region 14A....28
Region 14B....24
Total 157

FIGURE 2.2: Control and Shutdown Rod Locations



BANK IDENTIFIER	NUMBER OF LOCATIONS
A	8
B	8
C	8
D	8

BANK IDENTIFIER	NUMBER OF LOCATIONS
SA	8
SB	8
SP (SPARE)	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							32I	32I	32I						
3						32I		80I		32I					
4					64I			4SSA			64I				
5				64I		80I		64I		80I		64I			
6			32I		80I						80I		32I		
7		32I					104I		104I					32I	
8		32I	80I	4SSA	64I						64I	4SSA	80I	32I	
9		32I					104I		104I					32I	
10			32I		80I						80I		32I		
11				64I		80I		64I		80I		64I			
12					64I			4SSA			64I				
13						32I		80I		32I					
14							32I	32I	32I						
15															

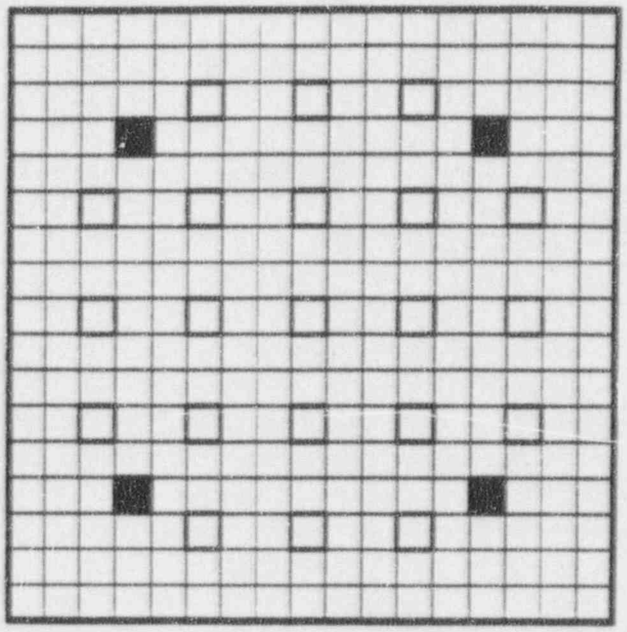
90°

0°

TYPE	TOTAL
###I..(NUMBER OF IFBA RODS)..................	2784
#SSA..(NUMBER OF SECONDARY SOURCE RODLETS)...	16

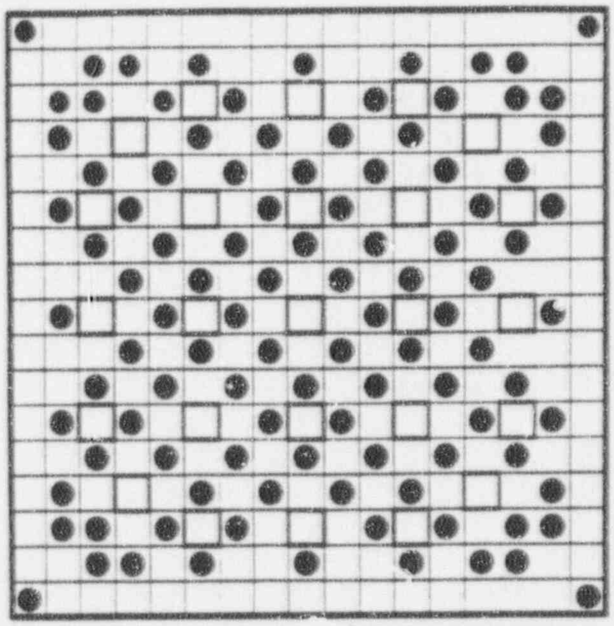
Note - Locations M-8 and D-8 contain new dually-compatible Secondary Sources for first time irradiation in Cycle 12

FIGURE 2.4: Secondary Source Rod Configurations

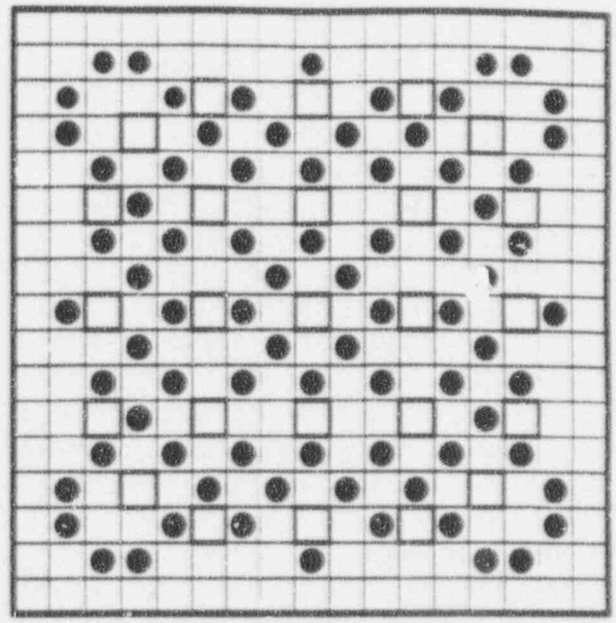


SECONDARY SOURCE ASSEMBLY

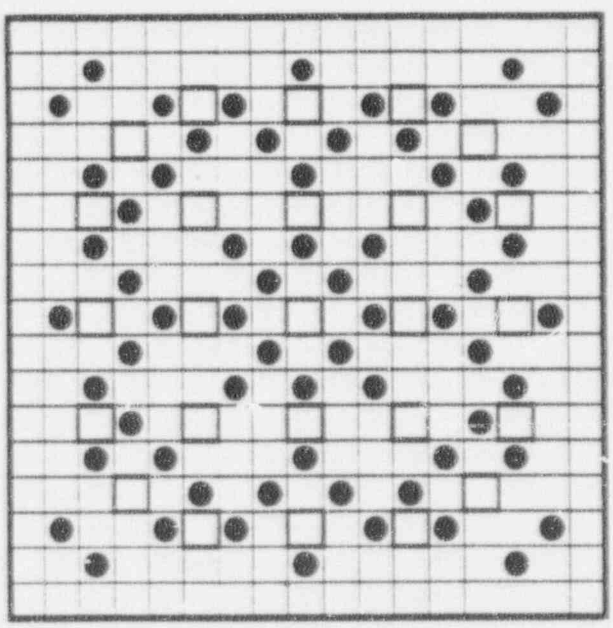
FIGURE 2.5: Burnable Absorber Configurations



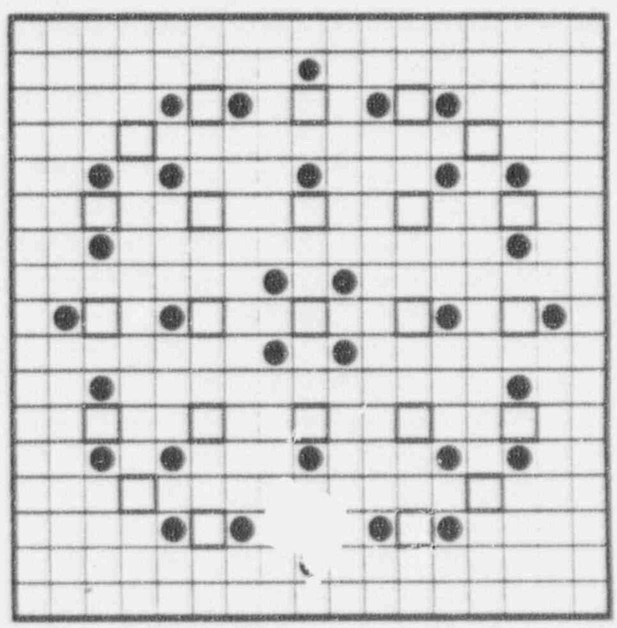
104 IFBA ASSEMBLY



80 IFBA ASSEMBLY



64 IFBA ASSEMBLY



32 IFBA ASSEMBLY

3.0 CONTROL ROD DROP TIME MEASUREMENT (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 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 drop time from the fully withdrawn position shall be $\leq$ 2.7 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 2.7 seconds. The longest drop time recorded was 1.85 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	1.616 sec.	2.200 sec.

To confirm normal rod mechanism operation prior to conducting the rod drop test, the Verification of Rod Control System Operability (FNP-0-ETP-3643) was performed. In this test, the stepping waveforms of the stationary, lift and movable gripper coils were examined for anomalies, rod speed was measured, the functioning of the Digital Rod Position Indicator (DRPI) and bank overlap unit were checked, and the bank overlap unit switch settings and functions were verified to be correct.

FIGURE 3.1: UNIT 1 CYCLE 12 DRIVE LINE "DROP TIME" TABULATION

R	P	N	M	L	K	J	H	G	F	E	D	C	B	A	
															1
					1.633 2.183		1.583 2.150		1.617 2.283						2
						1.583 2.183		1.650 2.183							3
			1.633 2.217		1.617 2.182				1.620 2.196		1.617 2.200				4
				1.600 2.183						1.633 2.250					5
	1.617 2.167		1.643 2.200		1.617 2.200		1.567 2.150		1.567 2.150		1.570 2.167		1.850 2.517		6
		1.617 2.200				1.550 2.183		1.583 2.167				1.650 2.200			7
	1.617 2.200				1.550 2.083				1.567 2.117				1.700 2.317		8
		1.617 2.217				1.533 2.150		1.550 2.133				1.617 2.200			9
	1.617 2.183		1.582 2.183		1.617 2.200		1.600 2.133		1.600 2.183		1.612 2.233		1.683 2.250		10
				1.600 2.167						1.617 2.150					11
			1.567 2.167		1.548 2.133				1.617 2.185		1.633 2.233				12
						1.600 2.167		1.667 2.283							13
					1.733 2.367		1.633 2.217		1.667 2.250						14
															15

↓ North

TEMPERATURE - 547.667 PRESSURE - 2220 psig %FLOW - 100

← BREAKER "OPENING" TO DASHPOT ENTY - IN SECONDS
← BREAKER "OPENING" TO DASHPOT BOTTOM - IN SECONDS

DATE: 11-28-92

4.0 INITIAL CRITICALITY (FNP-0-ETP-3601)

PURPOSE

The purpose of this procedure was to achieve initial crit..ality 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 12 was achieved during dilution mixing at 1325 hours on November 29, 1992. The reactor was allowed to stabilize at the following conditions:

RCS Pressure	2230.0 psig
RCS Temperature	546.5 °F
Intermediate Range Power	1.49×10^{-8} Amp
RCS Boron Concentration	1916.5 ppm
Bank D Position	184.5 steps

Once criticality was achieved, the pc'nt of adding nuclear heat was determined in order to define the flux r nge for physics testing, and the reactivity computer calibration was verified by making positive and negative reactivity changes and comparing the reactivity indicated by the reactivity computer with values determined using the Inhour Equation.

5.0 ALL-RODS-OUT ISOTHERMAL TEMPERATURE COEFFICIENT AND BORON ENDPOINT (FNP-0-ETP-3601)

PURPOSE

The objectives of these measurements were to determine the hot, zero power isothermal and moderator temperature coefficients for the all-rods-out (ARO) configuration and to measure the ARO boron endpoint concentration.

SUMMARY OF RESULTS

The ARO, hot zero power temperature coefficients and the ARO boron endpoint concentration are tabulated below:

ARO, HZP ISOTHERMAL AND MODERATOR TEMPERATURE COEFFICIENT

Rod Configuration	Boron Conc. ppm	Measured ITC pcm/°F	ITC Design Acc. Criterion pcm/°F	Calculated MTC pcm/°F
All Rods Out	1949.0	+1.15	+0.92 ± 2	+3.40*

where:

ITC = Isothermal Temperature Coefficient, includes -1.96 pcm/°F Doppler coefficient

MTC = Moderator Temperature Coefficient, corrected to the ARO condition

* MTC result was normalized to all rods out (ARO) and to the ARO critical boron concentration (1919 ppm).

ARO, HZP BORON ENDPOINT CONCENTRATION

Rod Configuration	Measured C_B (ppm)	Design-predicted C_B (ppm)
All Rods Out	1952.0	1919 ±50

Since the measured MTC (+3.40 pcm/°F) was less positive than the Technical Specification limit of +7.0 pcm/°F, no rod withdrawal limits were required. The design review criterion for the ARO boron concentration was also satisfied.

6.0 CONTROL AND SHUTDOWN BANK WORTH MEASUREMENTS (FNP-0-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; then (2) the worths of the remaining control and shutdown banks are derived from the change in the reference bank reactivity needed to offset full insertion of the bank being measured. For Cycle 12, control bank D was the reference bank. The measured bank worths satisfied the review criteria both for the banks measured individually and for the total worth of all banks combined.

SUMMARY OF CONTROL AND SHUTDOWN BANK WORTH MEASUREMENTS

Control or Shutdown Bank	Predicted Bank Worth & Review Criteria (pcm)	Measured Bank Worth (pcm)	Percent Difference
A	296 ± 100	314.7	+6.32
B	1122 ± 168	1106.9	-1.35
C	1017 ± 152	973.2	-4.31
D (Ref.)*	1174 ± 117	1138.9	-2.99
SD - A	902 ± 135	930.7	+3.18
SD - B	1052 ± 158	972.8	-7.53
All Banks	5563 ± 556.3	5437.2	-2.26

*The reference bank worth was measured by the dilution method.

7.0 POWER ASCENSION ACTIVITIES

Upon completion of HZP physics tests, the following activities were performed during power ascension, or at 100% power:

1. Incore movable detector system alignment.

2. Calorimetric thermal power measurement and adjustment of the power range NIS channel percent power indications.

3. Measurement of NIS intermediate range channel currents in order to determine high flux trip and rod stop setpoints.

4. Incore-excore AFD channel recalibration.

5. Core hot channel factor surveillance.

6. Reactor coolant system flow measurement.

7. Rescaling OPΔT and OTΔT protection loops to the 100% loop ΔTs measured during the RCS flow test.

At approximately 10% - 12% power, the determination of the incore system core limit settings (FNP-1-ETP-3606) was performed. The purpose of this procedure is to align the system so that the movable detectors stop at the correct core heights during flux mapping.

In order to invoke Technical Specification 3.10.3 test exceptions for HZP physics tests, preliminary intermediate and power range trip setpoints of less than or equal to 25% power were used for initial reactor startup and physics testing. In addition, since both NIS intermediate range detectors were replaced, the preliminary N35 and N36 channel trip setpoint and rod stop currents were set to 80% of the corresponding Cycle 11 setpoint currents.

Following the completion of physics tests, the NIS power range high range high flux trip setpoint was increased to 80% to allow power escalation above 25%. The 80% setpoint (vice 109%) was administratively imposed to address the possibility that the power range channels initially could be indicating nonconservatively. Due to the projected reduction in core neutron leakage between Cycle 11 and Cycle 12, it was recommended that power ascension be limited to 22.75% indicated power prior to performing the first thermal power measurement in order to prevent inadvertently exceeding 35% power. Therefore, a thermal power measurement was performed prior to increasing power above 22% and the NIS PR channels percent power indications were adjusted.

At 30% power, the thermal power measurement was repeated, the power range channels were recalibrated and currents were measured for determining the intermediate range high flux trip and rod stop setpoints. Then, the reactor was ramped to 33%, the Incore-Excore test (described in Par. 8.0) was performed and the power range N41 - N44 delta flux channels were recalibrated. Following calibration of the delta flux channels, a full-core flux map was performed at equilibrium xenon conditions at 33% power for core hot channel factor surveillance, and the power range NIS high flux trip setpoint was increased from 80% to 109%. Subsequent power escalation to 99% proceeded smoothly with no high quadrant power tilt ratio (QPTR) indications.

At approximately 99% power, a Loop 3 Overpower ΔT rod stop alarm was received. In addition, the percent ΔT channels had been indicating approximately 2% higher than NIS (calorimetric) power. Therefore, power escalation was stopped at 99%, the RCS flow test (described in Par. 9.0) was performed and the ΔT protection channels (OPΔT and OTΔT) were rescaled to the ΔT values (normalized to 100%) measured during the flow test.

As summarized in Table 7.1, core hot channel factor surveillance was initially performed under non-equilibrium conditions using the incore-excore base case full core flux map taken at 33% power, and then under equilibrium conditions using full-core flux maps performed at 33% and 100% power.

TABLE 7.1

SUMMARY OF POWER ASCENSION FULL CORE FLUX MAP DATA

Parameter	Fuel Type	Map 287	Map 294	Map 295
Avg. % power	N/A	33%	33%	100%
Max FDH	Lopar Vantage 5	1.4575 1.6273	1.4560 1.6279	1.4457 1.5570
Max power tilt*	N/A	1.0090	1.0070	1.0062
Avg. core % A.O.	N/A	+7.959	+8.365	+5.023
Limiting FQ(Z)**	Lopar Vantage 5	1.9844 2.2557	2.0312 2.2634	1.8184 1.9347
FQ Limit	Lopar Vantage 5	4.5364 4.8453	4.5571 4.8015	2.2779 2.4056
Flux map conditions	N/A	Non-equi-librium	Equi-librium	Equi-librium

*Calculated power tilts based on assembly FDHN from all assemblies.

**Based on percent to FQ limit.

Fuel types refererced above are low parasitic (Lopar) fuel (105 assemblies), and Vantage 5 fuel (52 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 axial offset for the purpose of calibrating the control board and the plant computer axial flux difference (AFD) channels, and for calibrating the delta flux penalty input to the overtemperature delta-T protection system.

SUMMARY OF RESULTS

At an indicated power of approximately 33%, a full core base-case flux map was performed at the AO (+7.959%) obtained immediately following power ascension. Five additional (quarter-core) flux maps were performed at various positive and negative axial offsets ranging from +24.6% to -35.4% in order to develop equations relating detector current to incore axial offset. (A sixth quarter-core map was performed, but the data was lost due to a plant computer malfunction.) In addition, data was taken from intermediate range channels N35 and N36 to determine the effects of Bank D insertion on the detector currents. Prior to ascending above 33% power, the power range NIS channels were adjusted to incorporate the revised calibration data.

During the refueling outage preceding the cycle 12 startup, the original analog power range channel detector current meters were replaced with permanently installed digital meters on all channels (N41 - N44). The digital meters enhanced the accuracy and precision of detector current readings and reduced the error in the incore-excore test. As a result, the excore quadrant power tilt ratio (QPTR) remained well within its limits during the ascension to full power and, at 100% power, the maximum QPTR was only 1.0037. In addition, an evaluation of the incore-excore calibration performed at 100% power using flux map data showed that the 33% power calibration was still within acceptable tolerance. Nevertheless, the detector equations were refined using full power flux map data and, once the core hot channel factors were verified to be satisfactory, the NIS delta-flux channels were adjusted to incorporate the revised calibration. The revised detector current vs AO equations (in which both the slopes and zero-offset currents were revised to account for core leakage changes between 33% and full power) are tabulated below:

TABLE 8.1

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

CHANNEL N41:

I-Top	=	0.7718 *AO	+	145.97 uA
I-Bottom	=	-0.8473 *AO	+	140.17 uA

CHANNEL N42:

I-Top	=	0.7911 *AO	+	143.50 uA
I-Bottom	=	-0.8752 *AO	+	136.69 uA

CHANNEL N43:

I-Top	=	0.7656 *AO	+	148.66 uA
I-Bottom	=	-0.9546 *AO	+	153.73 uA

CHANNEL N44:

I-Top	=	0.8105 *AO	+	142.63 uA
I-Bottom	=	-0.8786 *AO	+	139.44 uA

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 Technical Specifications. In addition, the RCS loop 100% delta-T values measured during this test are used to evaluate and, if necessary, to rescale the OPΔT and OTΔT protection channels.

SUMMARY OF RESULTS

The occurrance of a rod stop at 99% power and the disagreements noted between percent loop ΔT and NIS (calorimetric) percent power indicated the OPΔT and OTΔT protection loops required rescaling. Therefore, power escalation was stopped and the Unit 1 RCS flow measurement was performed at an average measured power of 99.75%. Since it was intended to use the data for rescaling the ΔT protection channels, 12 sets of flow test data (rather than the minimum of six) were taken to ensure accuracy.

In order 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 267,880 gpm for three loop operation. From the average of 12 sets of measurements, the measured RCS loop flows were:

Loop A = 94,476 gpm
Loop B = 92,997 gpm
Loop C = 94,431 gpm

This gave a total measured core flow of 281,904 gpm, which satisfied the Technical Specification requirement.

The measured 100% loop ΔTs (normalized to 100.0% power) obtained during the RCS flow test were:

Loop A: 62.837 °F
Loop B: 64.876 °F
Loop C: 64.126 °F

Scaling calculations were performed and the OPΔT and OTΔT protection channels were recalibrated to these revised 100% ΔT values.