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the southern electric system

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

Joseph M. Farley Nuclear Plant - Unit 2
Unit 2 Cycle - Startup Report

Gentlemen:

Enclosed is the Startup Report for Unit 2 Cycle 10. If you have any questions, please advise.

Respectfully submitted,

Dave Morey

REM/clt:U2CYCL10.DOC

Enclosure

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SOUTHERN NUCLEAR OPERATING COMPANY
JOSEPH M. FARLEY NUCLEAR PLANT

STARTUP TEST REPORT
UNIT 2 CYCLE 10

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APPROVED:

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1.0 INTRODUCTION

The Joseph M. Farley Unit 2 Cycle 10 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 2 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 July 30, 1981. The Cycle 10 core loading consists of 157 17 x 17 fuel assemblies, of which 42 are Westinghouse Low Parasitic (LOPAR) assemblies and the remaining 115 are Westinghouse Vantage 5 fuel assemblies.

Each of the 60 new Vantage 5 fuel assemblies loaded in the Cycle 10 core contains fresh Westinghouse Integral Fuel Burnable Absorbers (IFBAs). In addition, the core contains two double encapsulated secondary source inserts and 12 previously burned wet annular burnable poison (WABA) inserts. No thimble plug inserts were used. The design depletion of reactivity of the Cycle 10 core is 16500 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	05-08-81	05-27-81	10-22-82	15350	416.50	1.141
2	11-30-82	12-03-82	09-16-83	10371	281.68	1.913
3	10-22-83	10-24-83	01-05-85	14639	397.73	3.002
4	03-08-85	03-20-85	04-04-86	13183	359.48	3.987
5	05-11-86	05-13-86	10-03-87	16674	457.67	5.241
6	12-02-87	12-05-87	03-24-89	16138	444.09	6.458
7	05-18-89	05-21-89	10-13-90	17051	468.76	7.742
8	01-03-91	01-06-91	03-06-92	14757	405.69	8.853
9	05-08-92	05-12-92	09-24-93	17352	462.00	10.120

2.0 UNIT 2 CYCLE 10 CORE REFUELING

REFERENCES

1. Westinghouse Refueling Procedure FP-APR-R9.
2. Westinghouse WCAP 13842, Rev. 1 (The Nuclear Design and Core Management of the Joseph M. Farley Unit 2 Power Plant Cycle 10)

Unloading of the Cycle 9 core into the spent fuel pool commenced on 10/1/93 and was completed on 10/3/93. During the offload, each fuel assembly was inspected with binoculars for indications of damage or other problems. Two fuel assemblies scheduled for reload (Y39 and 2L35) were noted to have grid damage.

Following the cycle 9 core unload, an EPRI funded fuel inspection to gather baseline data for the Zinc Addition program was conducted by Westinghouse on 20 fuel assemblies. The fuel inspections consisted of oxide thickness measurements, high-magnification TV examinations and crud sampling. During the inspection program, the fuel inspectors performed

TV visual examinations of the two grid damaged assemblies identified during the core offload. In addition, the assemblies which were adjacent to the damaged assemblies during the previous core cycles were inspected. During these examinations, it was confirmed that fuel assemblies Y39 and 2L35 had severe grid damage, and a third fuel assembly, 2L47, also scheduled for reload, was found to have a gouge-like defect on a corner rod. The cycle 10 core was redesigned to exclude these fuel assemblies from reload. 2L35 and 2L47 are Vantage 5 assemblies.

The Cycle 10 Core reload commenced on 10/29/93 and was completed on 11/2/93. Due to an error in the core reload procedure, fresh fuel assembly 2M25 was mistakenly loaded into core location D5 (vice assembly 2M26). Since these fresh assemblies have the same enrichment, Management opted to leave 2M25 in location D5 in order to avoid the risk of damaging the assembly by moving it, and assembly 2M26 was loaded into location M11 (vice 2M25).

The as-loaded, redesigned Cycle-10 core, control rod locations, locations of burnable absorbers and source inserts, and the burnable absorber configurations are shown in Figures 2.1 through 2.5.

Figure 2.1 Unit 2 Cycle 10 Reference Loading Pattern

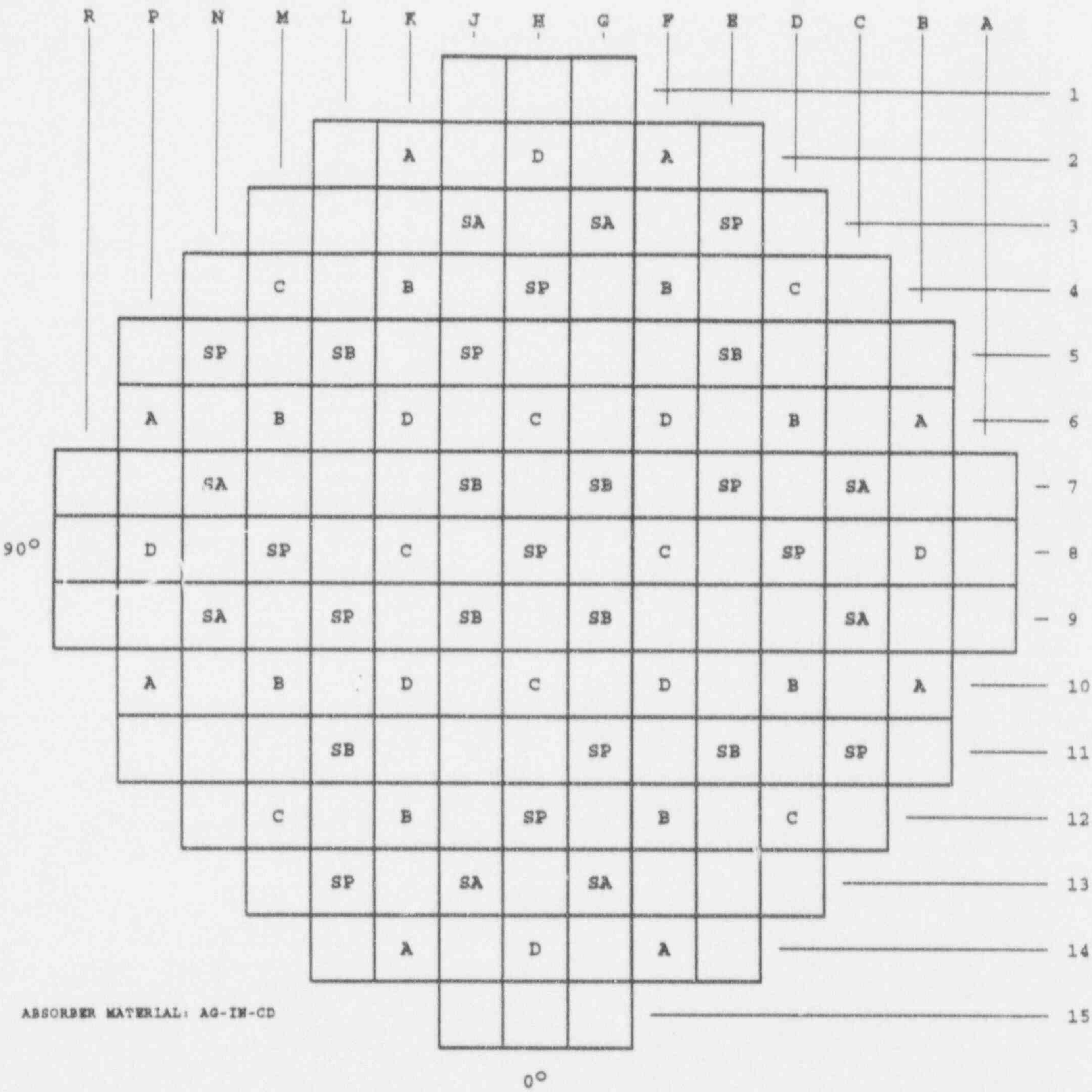
	R	P	N	M	L	K	J	H	G	F	E	D	C	B	A
1							W62	2L32	W57						
2					Y34	R102 2L16	2M43	R142 2M48	2M50	R140 2L44	Y13				
3				Y43	2M56	2M11	R129 2L06	Y11	R107 2L05	2M18	2M57	Y51			
4			Y37	R145 2L33	4W24D 2M21	R119 2L15	2M31	2L18	2M32	R114 2L13	4W20D 2M10	R103 2L39	Y55		
5		Y23	2M42	4W26D 2M39	R128 2L48	2L52	2L24	12W62D 2M13	2L25	2L57	R118 2L43	4W23D 2M25	2M60	Y07	
6		R110 2L30	2M27	R124 2L11	2L50	R135 Y46	2M02	R108 Y12	2M04	R115 Y56	2L56	R130 2L14	2M23	R137 2L38	
7	W58	2M44	R117 2L08	2M24	2L21	2M05	R144 Y52	2M17	R116 Y47	2M06	2L20	2M20	R106 2L03	2M53	W60
8	2L41	R127 2M54	Y25	SS07 2L26	12W67D 2M09	R113 Y30	2M30	Y48	2M16	R148 Y08	12W63D 2M15	SS08 2L22	Y29	R131 2M51	2L42
9	W53	2M52	R112 2L02	2M37	2L28	2M07	R146 Y50	2M35	R134 Y42	2M01	2L23	2M29	R123 2L04	2M47	W51
10		R139 2L49	2M40	R121 2L10	2L55	R133 Y45	2M08	R104 Y06	2M03	R138 Y41	2L54	R125 2L12	2M28	R141 2L31	
11		Y16	2M49	4W25D 2M26	R122 2L40	2L53	2L27	12W69D 2M34	2L19	2L51	R136 2L37	4W28D 2M14	2M41	Y22	
12			Y44	R109 2L34	4W21D 2M38	R120 2L17	2M12	2L29	2M19	R126 2L09	4W29D 2M33	R147 2L36	Y38		
13				Y53	2M55	2M22	R105 2L01	Y04	R143 2L07	2M36	2M46	Y40			
14					Y21	R111 2L46	2M45	R132 2M59	2M58	R101 2L45	Y03				
15							W56	Y49	W52						

↓ North

XXX	← Insert Serial Number
XXX	← Fuel Assembly Serial Number

REGION	ORIGINAL w/o U-235 ENRICHMENT	No. of FUEL ASSEMBLIES
Region 9B (W) assemblies	4.202%	8
Region 10A (Y) assemblies	3.806%	16
Region 10B (Y) assemblies	4.185%	18
Region 11A (2L) assemblies	3.606%	29
Region 11B (2L) assemblies	4.005%	26
Region 12A (2M) assemblies	4.201%	40
Region 12B (2M) assemblies	4.415%	20
	Total	157

FIGURE 2.2: Control and Shutdown Rod Locations



BANK IDENTIFIER	NUMBER OF LOCATIONS	BANK IDENTIFIER	NUMBER OF LOCATIONS
A	8	SA	8
B	8	SB	8
C	8	SP	13
D	8		

FIGURE 2.3: Burnable Absorber and Source Assembly Locations

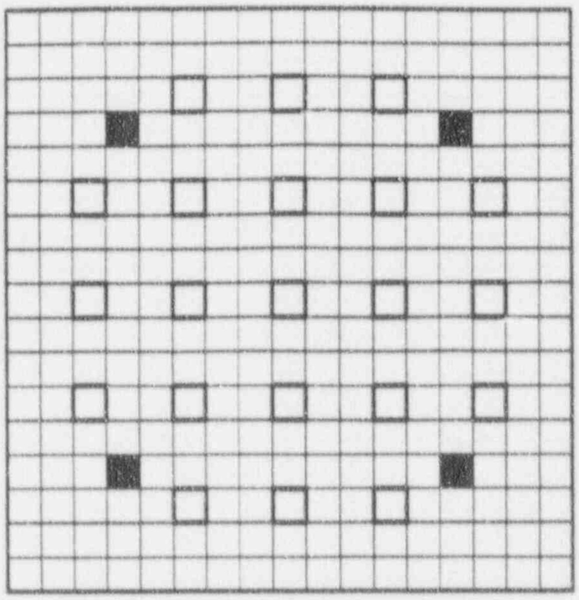
	R	P	N	M	L	K	J	H	G	F	E	D	C	B	A
1															
2							64I	64I	64I						
3					64I	104I				104I	64I				
4					4W 104I		104I		104I		4W 104I				
5			64I	4W 104I				12W 104I				4W 104I	64I		
6			104I				80I		80I				104I		
7		64I		104I		80I		104I		80I		104I		64I	
8		64I		4SSA	12W 104I		104I		104I		12W 104I	4SSA		64I	
9		64I		104I		80I		104I		80I		104I		64I	
10			104I				80I		80I				104I		
11			64I	4W 104I				12W 104I				4W 104I	64I		
12					4W 104I		104I		104I		4W 104I				
13					64I	104I				104I	64I				
14							64I	64I	64I						
15															

90°

0°

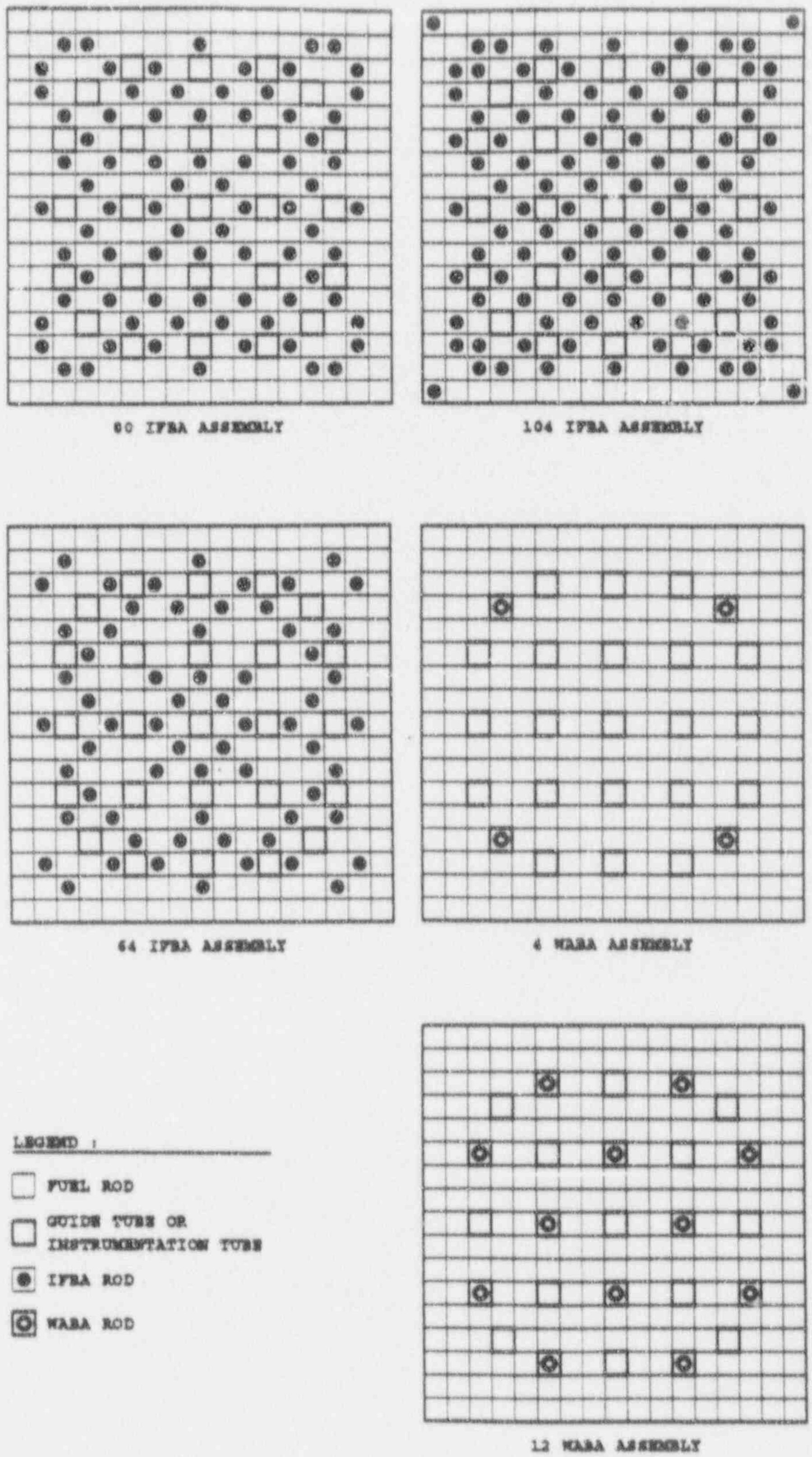
TYPE	TOTAL
##W...(NUMBER OF WABA RODLETS)...............	80
###I..(NUMBER OF IFBA RODS)..................	5248
#SSA..(NUMBER OF SECONDARY SOURCE RODLETS)...	8

FIGURE 2.4: Secondary Source Rod Configurations



SECONDARY SOURCE ASSEMBLY

FIGURE 2.5: Burnable Absorber Configurations



3.0 CONTROL ROD DROP TIME MEASUREMENT (FNP-2-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 ensure 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 1.55 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	1.363 sec.	1.829 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, and the functioning of the Digital Rod Position Indicator (DRPI) and bank overlap unit were checked. In addition, the bank overlap unit switch settings and functions were verified to be correct. No abnormal indications were found during this test.

Figure 3.1: Cycle 10 Drive Line "Drop Time" Tabulation

	R	P	N	M	L	K	J	H	G	F	E	D	C	B	A
1															
2						1.417 1.867		1.367 1.833		1.400 1.850					
3							1.350 1.833		1.367 1.850						
4				1.367 1.850		1.333 1.833				1.367 1.833		1.350 1.817			
5					1.333 1.800						1.367 1.850				
6		1.417 1.900		1.333 1.783		1.433 1.883		1.317 1.833		1.317 1.800		1.333 1.783		1.550 2.017	
7			1.350 1.817				1.283 1.750		1.317 1.783				1.367 1.850		
8		1.350 1.833				1.300 1.767				1.350 1.800				1.417 1.833	
9			1.350 1.800				1.317 1.817		1.383 1.783				1.350 1.800		
10		1.383 1.867		1.350 1.817		1.317 1.767		1.350 1.817		1.317 1.800		1.333 1.800		1.417 1.850	
11					1.350 1.817						1.350 1.817				
12				1.350 1.800		1.350 1.833				1.350 1.833		1.350 1.800			
13							1.350 1.817		1.350 1.800						
14						1.500 1.950		1.383 1.867		1.417 1.883					
15															

↓ North

X.XX	← Breaker "opening" to dashpot entry (seconds)
X.XX	← Breaker "opening" to dashpot bottom (seconds)

TEMPERATURE 546.98 PRESSURE 2272.54 % FLOW 100

DATE 11-27-93

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

PURPOSE

The purpose of this procedure was to achieve initial 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 10 was achieved during dilution mixing at 2251 hours on November 29, 1993. The reactor was allowed to stabilize at the following conditions:

RCS Pressure	2236.8 psig
RCS Temperature	547.0 °F
Intermediate Range Power	1.2×10^{-8} Amp
RCS Boron Concentration	1628.5 ppm
Bank D Position	206 steps

Once criticality was achieved, the point of adding nuclear heat was determined in order to define the flux range 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 from 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	1643.5	+0.345	+0.38 ± 2	+2.30*

* MTC result was normalized to all rods out (ARO) and to the ARO critical boron concentration (1643 ppm).

where:

ITC = Isothermal Temperature Coefficient, includes -1.92 pcm/°F Doppler coefficient.

MTC = Moderator Temperature Coefficient, corrected to the ARO condition.

NOTE: The objective of the MTC determination is to verify that the most positive MTC that occurs during the cycle at power levels below 70% does not exceed the Technical Specification limit of +7 pcm/°F. In the Cycle 10 design, it was determined that the MTC would reach its maximum value following BOL, at which time it would exceed the measured BOL value by 0.4 pcm/°F. Thus, at powers less than 70%:

Cycle 10 maximum MTC = (2.3 pcm/°F + 0.4 pcm/°F) = +2.7 pcm/°F

ARO, HZP BORON ENDPOINT CONCENTRATION

Rod Configuration	Measured C_B (ppm)	Design-predicted C_B (ppm)
All Rods Out	1646.8	1643 ± 50

Since the maximum Cycle 10 MTC (+2.7 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 10, control bank B 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	384 ± 100	383.99	-0.003
B (Ref.)*	1191 ± 119	1162.50	-2.39
C	855 ± 128	827.78	-3.18
D	999 ± 150	995.21	-0.38
SD - A	974 ± 146	949.09	-2.56
SD - B	1008 ± 151	976.38	-3.14
All Banks	5411 ± 541.1	5294.95	-2.14

* 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 full power:

1. Incore movable detector system alignment.

2. Measurement of NIS intermediate range (IR) channel currents in order to determine IR high flux trip and rod stop setpoints.

3. Incore-excore AFD channel recalibration.

4. Core hot channel factor surveillance.

5. Reactor coolant system flow measurement.

6. Generation of rescaling data for the OPΔT and OTΔT protection loops based on the 100% loop ΔTs measured during the RCS flow test.

At approximately 10% - 30% power, the determinat... of the incore system core limit settings (FNP-2-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. Since NIS intermediate range detector N36 was replaced, the preliminary N36 channel trip setpoint and rod stop currents were set to 80% of the previous, Cycle 9 currents for this channel.

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.) A power ascension limitation of 30.31% (derived from the projected Cycle 9 - 10 change in core neutron leakage) was recommended prior to the first thermal power measurement to prevent inadvertently exceeding 35% power prior to channel calibration. Intermediate Range detector currents measured at 29.5% power were used to generate rod stop and high flux trip setpoints for Intermediate Range NIS Channels N35 and N36.

After ramping the reactor to 48% power, the Incore-excore test (described in par. 8.0) was performed and the power range N41-N44 delta flux channels were recalibrated. Following delta flux channel calibration, the power range NIS high flux trip setpoint was increased from 80% to 109%, and a full-core flux map was performed at equilibrium xenon conditions at 48% power for core hot channel factor surveillance (FNP-2-STP-110).

At approximately 99% power, the RCS flow test (described in par. 9.0) was performed and the 100% power loop ΔTs were determined. Since greater than a 1% difference existed between the new values and the ΔTs to which ΔT protection loops -1 and -3 were scaled, new OPΔT and OTΔT scaling data was given to I&C for recalibration of these channels.

As described in Table 7.1, core hot channel surveillance was initially performed under non-equilibrium conditions using the incore-excore base case full core flux map taken at 48% power, and then under equilibrium conditions using full-core flux maps performed at 48% and 99% power. As shown in Table 7.1, all results were satisfactory.

TABLE 7.1
SUMMARY OF POWER ASCENSION FULL CORE FLUX MAP DATA

Parameter	Fuel Type	Map 246	Map 252	Map 253
Avg. % power	N/A	47.4%	48.7%	99.6%
Max power tilt*	N/A	1.0143	1.0148	1.0141
Avg. core % A.O.	N/A	+4.177	+0.762	+0.552
Max FΔH	Lopar Vantage 5	1.125 1.5904	1.1344 1.5750	1.1453 1.5532
FΔH Limit	Lopar Vantage 5	1.791 1.907	1.789 1.904	1.551 1.651
Limiting FQ(Z)**	Lopar Vantage 5	1.5323 2.0970	1.5287 2.0859	1.4227 1.9474
FQ Limit	Lopar Vantage 5	4.5571 4.8125	4.5985 4.8453	2.3062 2.4574
Flux map conditions	N/A	Non-equilibrium	Equilibrium	Equilibrium

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

** Based on percent to FQ limit.

Fuel types referenced above are Lopar (low parasitic fuel, 42 assemblies) and Vantage 5 fuel (115 assemblies).

8.0 INCORE-EXCORE DETECTOR CALIBRATION (FNP-2-STP-121)

PURPOSE

The objective of this procedure was to determine the relationship between power range upper and lower excore detector currents and core axial offset for the purpose of calibrating the main control board and 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 48%, a full core base-case flux map was performed at the AO (+4.177%) obtained immediately following power ascension. Five additional (quarter-core) flux maps were performed at various positive and negative axial offsets ranging from -30.754% to +21.429% in order to develop equations relating detector current to incore axial offset. Prior to ascending above 48% power, the power range NIS channels were adjusted to incorporate the revised calibration data.

During the refueling outage preceding the cycle 10 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 99.6% power, the maximum QPTR was only 1.0029, well within the required limit of 1.02. The revised detector current vs AO equations resulting from the Incore-Excore recalibration are tabulated below:

TABLE 8.1

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

CHANNEL N41:

I-Top	=	0.8026 * AO	+	159.1434 uA
I-Bottom	=	-1.0551 * AO	+	158.8756 uA

CHANNEL N42:

I-Top	=	0.8079 * AO	+	163.2901 uA
I-Bottom	=	-1.0928 * AO	+	158.4032 uA

CHANNEL N43:

I-Top	=	0.8270 * AO	+	167.5974 uA
I-Bottom	=	-1.1082 * AO	+	165.0391 uA

CHANNEL N44:

I-Top	=	0.9013 * AO	+	176.7149 uA
I-Bottom	=	-1.2405 * AO	+	176.0114 uA

9.0 REACTOR COOLANT SYSTEM FLOW MEASUREMENT (FNP-2-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

In order to comply with the Unit 2 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 = 94,375.9 gpm
Loop B = 89,702.0 gpm
Loop C = 92,206.6 gpm

These combine to give a total measured core flow of 276,284.5 gpm, which satisfies the Technical Specification requirement.

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

Loop A: 63.762 °F
Loop B: 67.675 °F
Loop C: 64.054 °F

Since more than a 1% difference existed between these values and the ΔTs to which Loops -1 and -3 were scaled, scaling calculations were performed and provided to I&C for recalibration of the OPΔT and OTΔT protection channels.