ALABAMA POWER COMPANY

JOSEPH M. FARLEY NUCLEAR PLANT

UNIT NUMBER 2, CYCLE 4

STARTUP TEST REPORT

PREPARED BY PLANT REACTOR ENGINEERING GROUP

APPROVED:

C. D. Nesbitt ______ Technical Superintendent

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DISK: CYCLE 2

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1.0 INTRODUCTION

The Joseph M. Farley Unit 2 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 2 of the Joseph M. Farley Nuclear Plant is a Westinghouse three loop pressurized water reactor rated at 2652 MWth. The Cycle 4 core loading consists of 157 17 x 17 fuel assemblies.

The Unit began commercial operations on July 30, 1981, completed Cycle 1 on October 22, 1982 with an average core burnup of 15350.5 MWD/MTU, completed Cycle 2 on September 17, 1983 with an average core burnup of 10371.2 MWD/MTU, and completed cycle 3 on January 4, 1985 with an average core burnup of 14,639.0 MWD/MTU.

2.0 UNIT 2 CYCLE 4 CORE REFUELING

REFERENCES

1. Westinghouse Refueling Procedure FP-APR-R3
2. Westinghouse WCAP 10674 (The Nuclear Design and Core Management of the Joseph M. Farley Unit 2 Power Plant Cycle 4)

The refueling commenced on 1/12/85 and was completed in 10 days on 1/22/85. The as-loaded Cycle 4 core is shown in Figures 2.1 through 2.4, which give the location of each fuel assembly and insert, including the burnable poison insert locations and configurations. The Cycle 4 core has a nominal design lifetime of 15240 MWD/MTU and consists of 21 region 4 assemblies, 64 region 5 assemblies, and 72 region 6 assemblies. Fuel assembly inserts include 48 full length control rod clusters, 40 burnable poison inserts, two secondary sources, and 67 thimble plug inserts.

Figure 2.1

APR Unit 2, Cycle 4 Reference Loading Pattern

180°

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

90° (left) — 270° (right)

0°

COLOR CODE	ID XX	ID XX	ID XX
REGION	4	5	6
FROM CYCLE	3	3	FEED
[illegible] U-235	3.100	3.402	3.45

LEGEND:

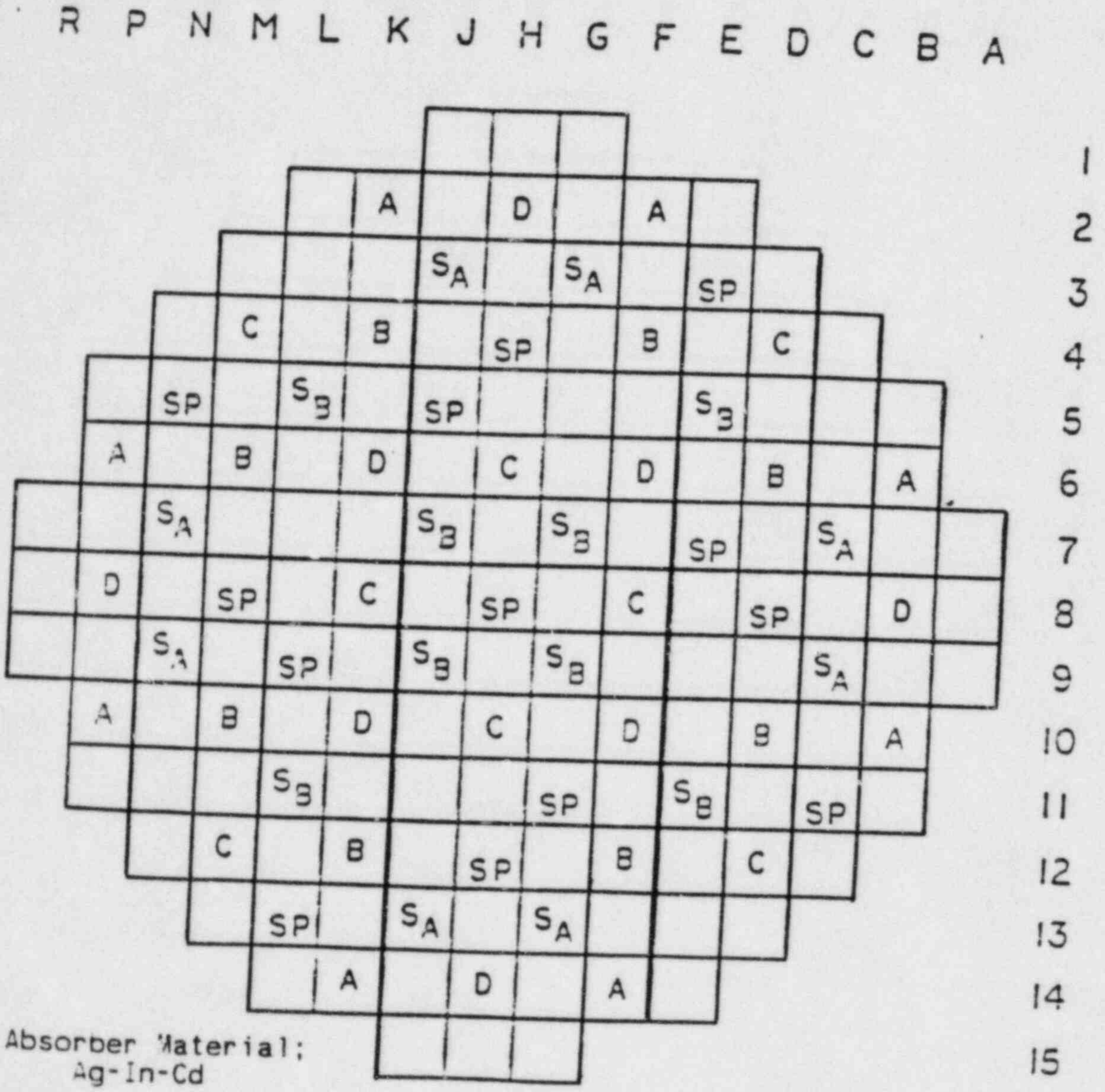
ID - ASSEMBLY IDENTIFICATION

SS - SECONDARY SOURCE LOCATION

XX - PREVIOUS CYCLE 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 SB	8
Shutdown Bank SA	8
SP (Spare Rod Locations)	13

FIGURE 2-3

BURNABLE ABSORBER AND SOURCE ASSEMBLY LOCATIONS

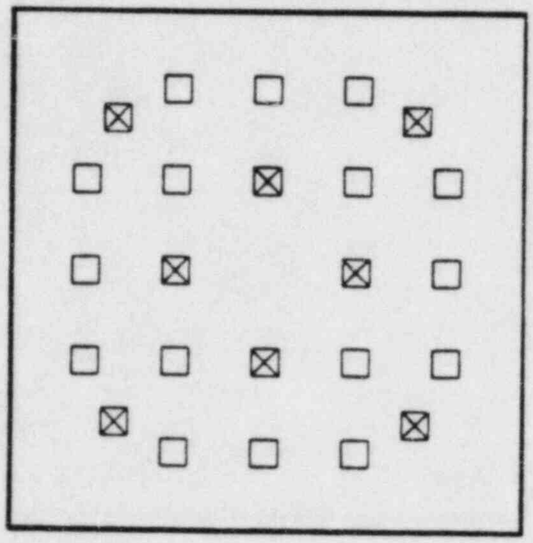
	R	P	N	M	L	K	J	H	G	F	E	D	C	B	A
1															
2															
3					8			SS		8					
4					12		12		12		12				
5				12		12		12		12		12			
6			8		12						12		8		
7				12				8				12			
8					12		8		8		12				
9				12				8				12			
10			8		12						12		8		
11				12		12		12		12		12			
12					12		12		12		12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

SS Secondary Source

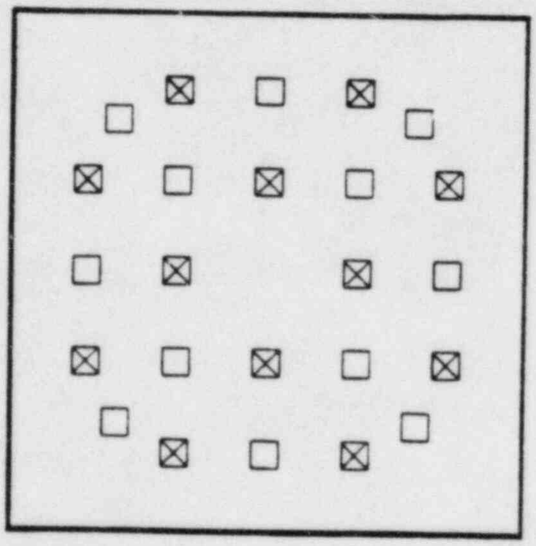
432 Fresh Standard BA's

FIGURE 2-4

BURNABLE ABSORBER CONFIGURATIONS



8 Fresh BA
Configuration



12 Fresh BA
Configuration

3.0 CONTROL ROD DROP TIME MEASUREMENT (FNP-2-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 ensure 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 < 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 1.47 seconds for rod B-6 and K-14. 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	1.38 sec	1.89 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 moveable gripper coils were examined, and the functioning of the Digital Rod position indicator and the bank overlap unit was checked. Rod stepping speed measurements were conducted. All results were satisfactory.

NORTH

UNIT 2 CYCLE 4

	15	14	13	12	11	10	9	8	7	6	5	4	3	2	1
R								90°							
P						1.40 1.82		1.37 1.89		1.40 1.87					
N							1.38 1.89		1.35 1.90						
M				1.38 1.85		1.36 1.88				1.37 1.87		1.38 1.87			
L					1.38 1.91						1.37 1.91				
K		1.47 1.96		1.38 1.87		1.35 1.87		1.36 1.87		1.35 1.88		1.38 1.89		1.43 1.91	
J			1.32 1.85				1.33 1.86		1.36 1.91				1.38 1.91		
H	0°	1.41 1.94				1.39 1.91				1.36 1.88				1.37 1.89	180°
G			1.38 1.86				1.34 1.83		1.37 1.88				1.42 1.93		
F		1.43 1.89		1.39 1.91		1.38 1.92		1.34 1.85		1.39 1.91		1.36 1.90		1.41 1.90	
E					1.35 1.85						1.38 1.88				
D				1.38 1.82		1.36 1.87				1.33 1.86		1.37 1.85			
C							1.40 1.90		1.38 1.89						
B						1.41 1.88		1.42 1.92		1.47 2.04					
A								270°							

DRIVE LINE "DROP TIME" TABULATION

TEMPERATURE - 547°F PRESSURE - 2235 psig % FLOW - 100

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

DATE - 3-7-85

FIGURE 3.1

4.0 INITIAL CRITICALITY (FNP-2-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 4 was achieved during dilution mixing at 0735 hours on March 8, 1985 The reactor was allowed to stabilize at the following critical conditions: RCS pressure- 2235 psig, RCS temperature 548.0°F, intermediate range power 2 x 10^{-8} amp, RCS boron concentration 1900 ppm, and Control Bank D position- 181.5 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 ALL-RODS-OUT ISOTHERMAL TEMPERATURE COEFFICIENT, BORON ENDPOINT AND FLUX DISTRIBUTION (FNP-2-ETP-3601)

PURPOSE

The objectives of these measurements were 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. (optional)

SUMMARY OF RESULTS

The measured ARO, hot zero power temperature coefficients and the ARO boron endpoint concentration are shown in Table 5.1. The isothermal temperature coefficient was measured to be +1.22 pcm/°F which meets the design acceptance criteria. This gives a calculated moderator temperature coefficient of +3.92 pcm/°F which is within the Technical Specification limit of +5.0 pcm/°F. Thus, no rod withdrawal limits are needed to ensure the +5.0 pcm/°F limit is met. The design acceptance criterion for the ARO critical boron concentration was also satisfactorily met.

Flux distribution was determined by the performance of a flux map at 34% power (see section 7.0 for results).

TABLE 5.1

ARO, 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	1909.5	+1.22	+3.92	+0.6 ± 3

α_T - Isothermal temperature coefficient, includes -2.7 pcm/°F doppler 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	1900	1911 ± 191.1

6.0 CONTROL AND SHUTDOWN BANK WORTH MEASUREMENTS (FNP-2-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; 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	622 ± 100	598.8	-3.7
Control B (Ref.)	1308 ± 131	1223.8*	-6.4
Control C	9[illegible]3 ± 140	885.4	-5.1
Control D	983 ± 147	955.6	-2.8
Shutdown A	1113 ± 167	1051.6	-5.5
Shutdown B	900 ± 135	833.3	-7.4
All Banks Combined	5859 ± 585	5548.5	-5.3

*Measured by dilution method

7.0 POWER ASCENSION PROCEDURE (FNP-2-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

Westinghouse fuel warranty provisions recommends that the power ramp rate be limited to 3% of full power per hour between 20% and 100% power until full power is achieved for 72 cumulative hours out of any seven-day operation period. This ramp rate was followed throughout the ascension to 100% power except for 3 occasions when the indicated NIS power went up between 3 and 4% in one hour.

Alignment of the incore movable detector system normal, calibrate and emergency paths was accomplished during power ascension (at power levels above 5%) prior to performing the flux map at 34% power.

Full flux maps were taken at 34%, and 78% power. As summarized in Table 7.1, all results were within Technical Specification Limits.

An incore/excore calibration check was performed at 34% power. Channel N44 currents were found to be out of calibration and channels N42 and N44 showed QPTR's >1.02, therefore a correction to the incore-excore calibration data was performed and revised currents were issued for calculating AFD (STP-7.1) and QPTR (STP-7.0). At approximately 75% power, a complete incore-excore recalibration was performed to comply with Technical Specification requirements. The incore-excore recalibration is described in Section 8.0.

TABLE 7.1
SUMMARY OF POWER ASCENSION FLUX MAP DATA

Parameters	Map 100	Map 101
Date	3/22/85	3/25/85
Time	10:30	11:00
Avg. % Power	34%	78%
Max FΔH	1.4888	1.4114
Max. Power Tilt*	1.0093	1.0055
Avg. Core % A. O.	+9.579	+7.0C8

*Calculated power tilts based on assembly FΔHN from all 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 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

A preliminary verification of excore AFD channel calibration was performed at 34% power to insure that an AFD target band could be defined for ascension to 78% power. Flux maps for incore-excore recalibration were run at approximately 78% power at average percent core axial offsets of +7.008, -12.182, -17.478 and +14.324 as determined from the INCORE code 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 to the overtemperature ΔT setpoint. (See Table 8.1)

TABLE 8.1

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

CHANNEL N41:

I-Top = 0.8465*AO + 198.7294 μa
I-Bottom = -1.3662*AO + 200.7462 μa

CHANNEL 42:

I-Top = 0.8655*AO + 194.4327 μa
I-Bottom = -1.4081*AO + 199.3029 μa

CHANNEL N43:

I-Top = 0.8296*AO + 203.0991 μa
I-Bottom = -1.5133*AO + 201.7735 μa

CHANNEL N44:

I-Top = 0.9614*AO + 219.2990 μa
I-Bottom = -1.5816*AO + 225.0805 μa

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 Unit 2 Technical Specifications.

SUMMARY OF RESULTS

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 265,500 gpm for three loop operation. From the average of six calorimetric heat balance measurements, the total core flow was determined to be 281,443.2 gpm, which meets the above criterion.

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R. P. McDonald
Senior Vice President
Flintridge Building

Alabama Power
the southern electric system

June 10, 1985

Docket No. 50-364

Director, Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Attention: Mr. S. A. Varga

Joseph M. Farley Nuclear Plant - Unit 2
Cycle 4 Startup Report

Gentlemen:

Enclosed is the Startup Report for Unit 2 Cycle 4 as required by the December 18, 1984 letter from Mr. R. P. McDonald to Mr. S. A. Varga.

If you have any questions, please advise.

Yours very truly,

R. P. McDonald

RPM/MDR:cl, D-5
Enclosure

cc: Mr. L. B. Long
Dr. J. N. Grace
Mr. E. A. Reeves
Mr. W. H. Bradford

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