GEORGIA POWER COMPANY

VOGTLE NUCLEAR PLANT

UNIT NUMBER 1, CYCLE 4

STARTUP TEST REPORT

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1.0 INTRODUCTION

The Vogtle Nuclear Plant Unit 1 Cycle 4 Startup Test Report summarizes results for tests performed as required by plant procedures following a core refueling. The report provides a brief synopsis of each test and gives a comparison of measured values with design parameters, Technical Specifications, or values assumed in the FSAR safety analysis.

Unit 1 of the Vogtle Nuclear Plant is a four loop Westinghouse pressurized water reactor rated at 3411 MWth. The Cycle 4 core loading consists of 193 17 x 17 fuel assemblies.

Unit 1 began commercial operations on May 31, 1987 and has completed the first three cycles with the following average burnups:

Cycle 1	Complete 10/8/88	15,852 MWD/MTU
Cycle 2	Complete 2/23/90	15,789 MWD/MTU
Cycle 3	Complete 9/15/91	18,504 MWD/MTU

Seventy-two of the 193 assemblies comprising Cycle 4 are based upon the VANTAGE 5 design. Two of the VANTAGE 5 assemblies contain a total of 24 demonstration rods clad in ZIRLO (Reference Tech Spec 5.3.1) and are located in core locations K2 and F14.

2.0 UNIT 1 CYCLE 4 CORE REFUELING

REFERENCES

Westinghouse WCAP 13023 (The Nuclear Design Report for the Vogtle Electric Generating Plant, Unit 1, Cycle 4)

SUMMARY

Unloading of the Cycle 3 core into the spent fuel pool commenced on 09/26/91 and was completed on 09/28/91.

Core reload commenced on 10/18/91 and was completed on 10/23/91. 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. The Cycle 4 core has a nominal design lifetime of 17,500 MWD/MTU and consists of 37 Region 4 assemblies, 48 Region 5A assemblies, 36 Region 5B assemblies, 56 Region 6A assemblies and 16 Region 6B assemblies. Fuel assembly inserts consist of 53 full length control rod clusters, two secondary sources and 138 thimble plug inserts. The assemblies in Regions 6A and 6B contain 3200 fuel rods with Integral Fuel Burnable Absorbers (IFBA).

During the reload, assembly 5F48 (core location L5) was discovered to have a bolt and nut attached to the second grid strap from the top of the assembly. This discovery occurred in the spent fuel pool as the assembly was raised from storage for transport to the core. The bolt and nut were removed and the assembly was inspected and cleaned of discoloration in the spent fuel pool area by a representative from Westinghouse's Fuel Plant in Columbia, South Carolina. After cleaning and inspection, the assembly was approved for core loading.

FIGURE 2.1

A. W. VOGTLE UNIT 1, CYCLE 4

REFERENCE LOADING PATTERN

R	P	N	M	L	K	J	H	G	F	E	D	C	B	A	
				5D54 F-15	5D04 N-14	5E44 E-12	5E14 H-11	5E11 L-12	5D21 C-14	5D14 K-15					1
		5D45 J-15	5D36 L-15	5F02 FEED	5F59 FEED	5F14 FEED	5F16 FEED	5F17 FEED	5F62 FEED	5F04 FEED	5D31 E-15	5D78 G-15			2
	5D19 A-7	5F05 FEED	5F64 FEED	5F27 FEED	5E63 F-13	5E79 M-14	5E15 M-7	5E82 D-14	5E60 K-13	5F34 FEED	5F70 FEED	5F10 FEED	5D22 R-7		3
	5D56 A-5	5F71 FEED	5E81 H-3	5E19 G-6	5E62 E-2	5E05 E-10	5F37 FEED	5E47 L-10	5E59 L-2	5E16 J-6	5E78 C-8	5F58 FEED	5D64 R-5		4
5D57 A-10	5F11 FEED	5F47 FEED	5E18 K-9	5F48 FEED	5E03 J-2	5F26 FEED	5D28 H-1	5F55 FEED	5E17 G-2	5F29 FEED	5E23 F-9	5F42 FEED	5F01 FEED	5D30 R-10	5
5D11 B-3	5F72 FEED	5E70 C-10	5E84 P-11	5E33 P-7	5E07 M-9	5E73 M-3	5F43 FEED	5E67 D-3	5E34 J-4	5E48 B-7	5E69 B-11	5E76 N-10	5F60 FEED	5D18 P-3	6
5E21 D-11	5F22 FEED	5E58 B-4	5E22 F-11	5F39 FEED	5E64 N-4	5F40 FEED	5E41 H-7	5F33 FEED	5E65 C-4	5F36 FEED	5E06 K-11	5E68 P-4	5F23 FEED	5E13 M-11	7
5E09 E-8	5F13 FEED	5E35 J-12	5F25 FEED	5D40 R-8	5F46 FEED	5E28 J-8	5D46 F-14	5E38 G-8	5F32 FEED	5D24 A-8	5F31 FEED	5E36 G-4	5F24 FEED	5E31 L-8	8
5E25 D-5	5F15 FEED	5E80 B-12	5E30 F-5	5F53 FEED	5E75 N-12	5F51 FEED	5E37 H-9	5F45 FEED	5E55 C-12	5F56 FEED	5E08 K-5	5E66 P-12	5F20 FEED	5E45 M-5	9
5D09 B-13	5F67 FEED	5E57 C-6	5E61 P-5	5E43 P-9	5E26 G-12	5E53 M-13	5F44 FEED	5E77 D-13	5E32 D-7	5E42 B-9	5E49 B-5	5E71 N-6	5F68 FEED	5D07 P-13	10
5D73 A-6	5F12 FEED	5F41 FEED	5E10 K-7	5F50 FEED	5E24 J-14	5F30 FEED	5D52 H-15	5F54 FEED	5E12 G-14	5F28 FEED	5E39 F-7	5F35 FEED	5F03 FEED	5D69 R-6	11
	5D76 A-11	5F65 FEED	5E56 N-8	5E46 G-10	5E52 E-14	5E01 E-6	5F49 FEED	5E27 L-6	5E83 L-14	5E04 J-10	5E74 H-13	5F66 FEED	5D53 R-11		12
	5D33 A-9	5F08 FEED	5F61 FEED	5F38 FEED	5E54 F-3	5E51 M-2	5E40 D-9	5E72 D-2	5E50 K-3	5F52 FEED	5F63 FEED	5F09 FEED	5D16 R-9		13
		5D66 J-1	5D05 L-1	5F06 FEED	5F57 FEED	5F21 FEED	5F18 FEED	5F19 FEED	5F69 FEED	5F07 FEED	5D38 E-1	5D77 G-1			14
				5D72 F-1	5D63 N-2	5E02 E-4	5E29 H-5	5E20 L-4	5D35 C-2	5D62 K-1					15

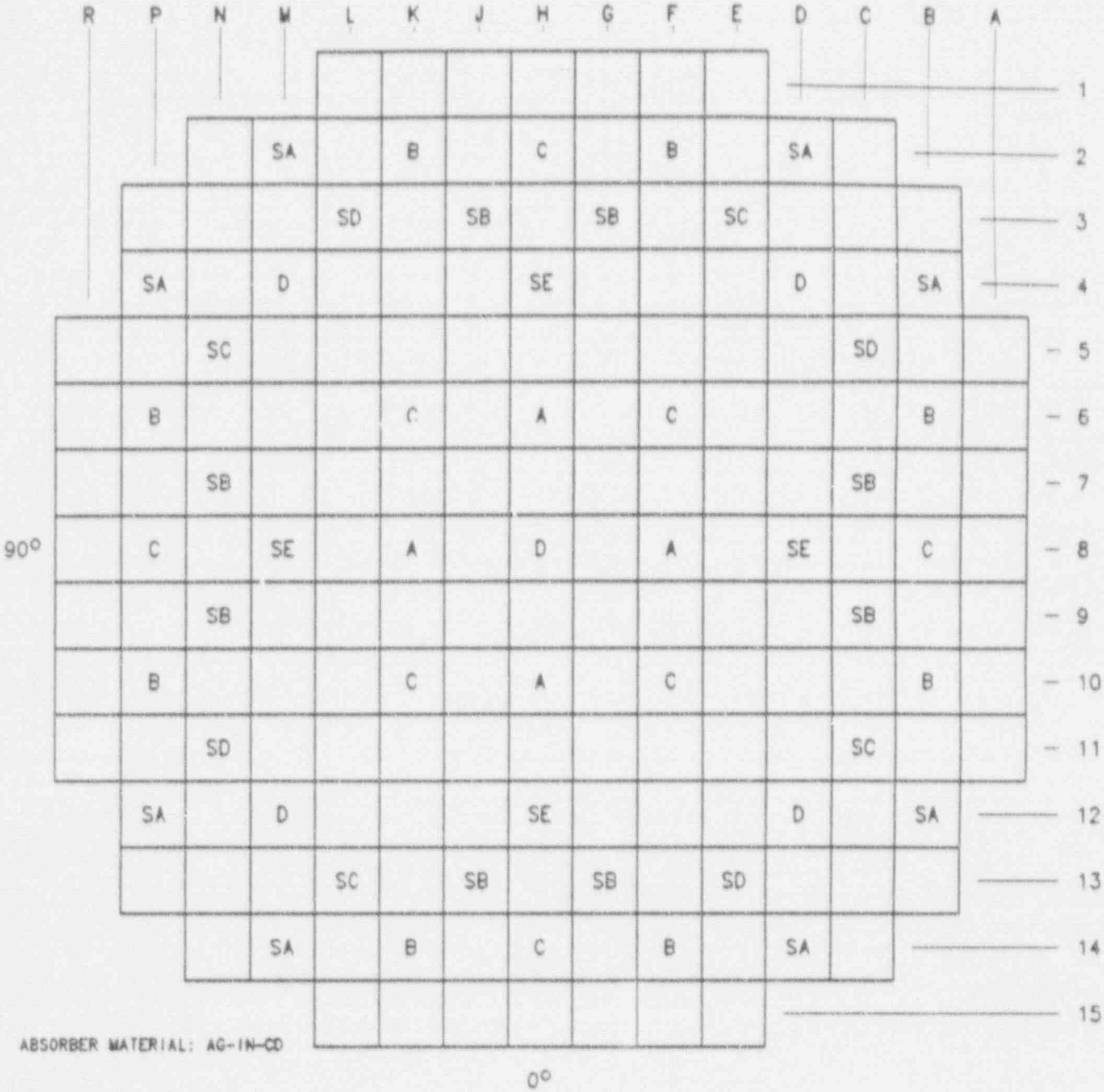
90°

0°

5D	REGION 4 (3.405 w/o)	5F	REGION 6A (3.800 w/o)
5E	REGION 5A (3.997 w/o)	5F	REGION 6B (4.100 w/o)
5E	REGION 5B (4.305 w/o)		

Y-YY	WESTINGHOUSE ASSEMBLY ID
Z-ZZ	PREVIOUS CYCLE LOCATION

FIGURE 2.2 CONTROL ROD LOCATIONS



ABSORBER MATERIAL: AG-IN-CD

BANK IDENTIFIER	NUMBER OF LOCATIONS	BANK IDENTIFIER	NUMBER OF LOCATIONS
A	4	SA	8
B	8	SB	8
C	8	SC	4
D	5	SD	4
		SE	4

FIGURE 2.3

Burnable Absorber and Secondary Source Rod Configurations

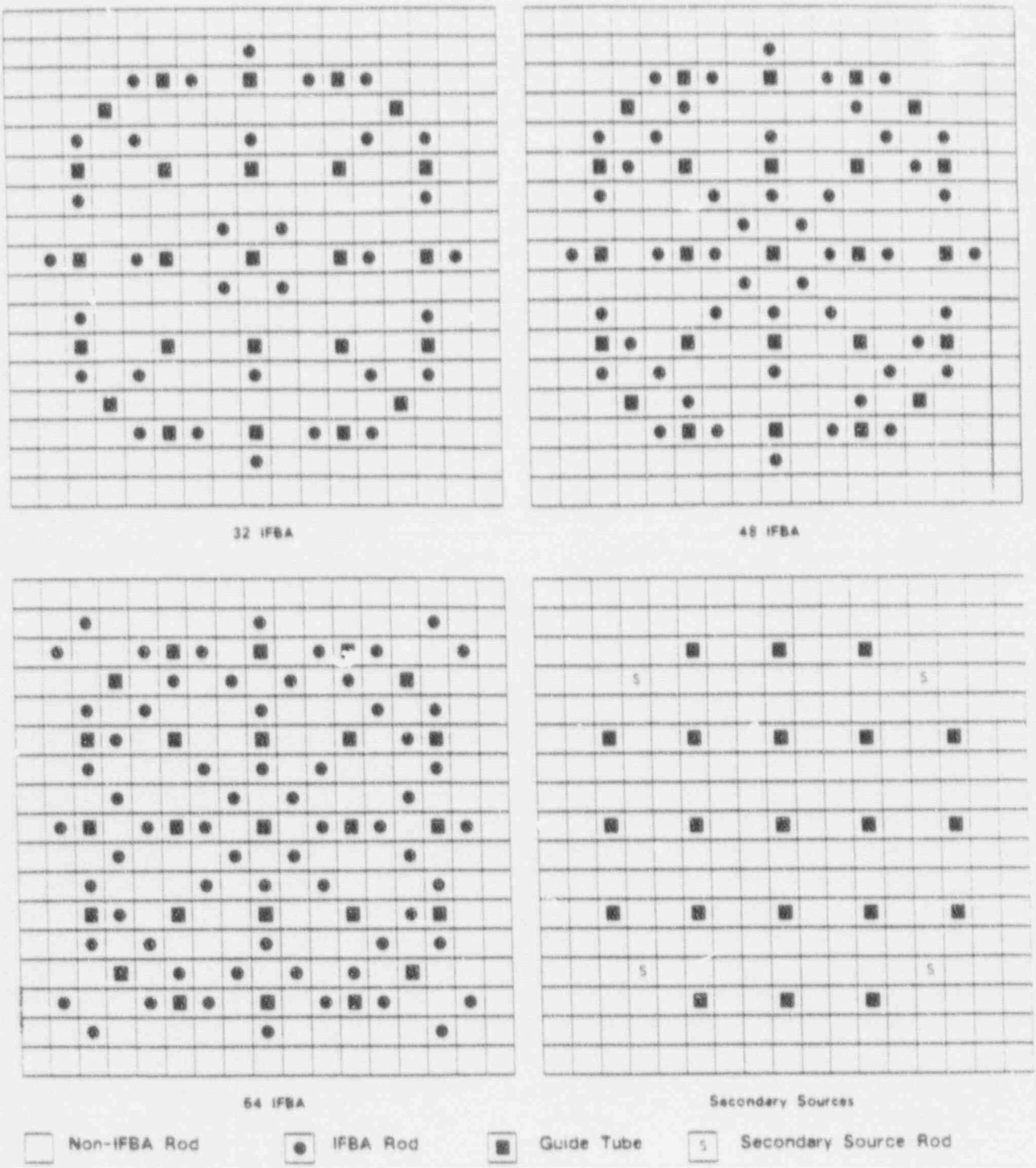


FIGURE 2.4

Burnable Absorber and Source Rod Locations

	R	P	N	M	L	K	J	H	G	F	E	D	C	B	A	
																1
						48I	32I	32I	32I 4SSA	48I						2
				48I	64I						64I	48I				3
			48I					64I					48I			4
			64I		64I		64I		64I		64I		64I			5
		48I						64I						48I		6
		32I			64I		64I		64I		64I			32I		7
90°		32I		64I		64I				64I		64I		32I		8
		32I			64I		64I		64I		64I			32I		9
		48I						64I						48I		10
			64I		64I		64I		64I		64I		64I			11
			48I					64I					48I			12
				48I	64I						64I	48I				13
						48I	32I	32I	32I 4SSA	48I						14
																15

0°

TYPE	TOTAL
###I..(NUMBER OF IFBA RODS)....................	3200
#SSA..(NUMBER OF SECONDARY SOURCE RODLETS)...	8

3.0 CONTROL ROD DROP TIME MEASUREMENT

PURPOSE

The purpose of this test was to measure the drop time of all 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 $\geq$ 551°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 $\leq$ 2.7 seconds from the beginning of stationary gripper coil voltage decay until dashpot entry. All rod drop times were measured to be less than 2.7 seconds. The rod drop time results for dashpot entry are presented in Table 3.1. The mean drop time was determined to be 1.554 seconds.

TABLE 3.1 CONTROL ROD DASHPOT ENTRY TIMES

CONTROL ROD LOCATION	DASHPOT ENTRY TIME (MSEC)	CONTROL ROD LOCATION	DASHPOT ENTRY TIME (MSEC)
D02	1588	M08	1536
B12	1540	H06	1548
M14	1582	H10	1556
P04	1538	F08	1576
B04	1564	K08	1592
D14	1618	F02	1550
P12	1580	B10	1546
M02	1648	K14	1582
G03	1516	P06	1518
C09	1550	B06	1554
J13	1558	F14	1546
N07	1530	P10	1534
C07	1528	K02	1558
G13	1544	H02	1574
N09	1570	B08	1512
J03	1554	H14	1570
E03	1528	P08	1528
C11	1556	F06	1544
L13	1556	F10	1552
N05	1534	K10	1546
C05	1544	K06	1522
E13	1546	D04	1570
N11	1536	M12	1544
L03	1604	D12	1558
H04	1540	M04	1538
D08	1546	H08	1586
H12	1552		

SAMPLE SIZE = 53 MEAN = 1.554 SIGMA = 0.025 2SIGMA = 0.050
MEAN-2SIGMA = 1.504 MEAN + 2SIGMA = 1.604

Since the control rods in locations D14 and M02 fell outside of the 2SIGMA limit, they were dropped an additional six times. The drop times for these extra six drops were all measured at less than 1.610 seconds which was within the 2.7 second Tech. Spec. limit.

4.0 INITIAL CRITICALITY

PURPOSE

The purpose of this test 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 by dilution at 1417 on November 19, 1991. The reactor was stabilized at the following critical conditions: RCS temperature 556.6°F, intermediate range power approximately 1 x 10-8 amps, RCS boron concentration 2094 ppm, and Control Bank D position at 166 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 initial criticality to demonstrate that adequate overlap existed.

5.0 ALL-RODS-OUT ISOTHERMAL TEMPERATURE COEFFICIENT AND BORON ENDPOINT MEASUREMENT

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 measured ARO, hot zero power temperature coefficients and the ARO boron endpoint concentration for HZP are shown in Table 5.1. The isothermal temperature coefficient was measured to be +2.77 pcm/°F which meets the design acceptance criteria. This gives a calculated moderator temperature coefficient of +4.55 pcm/°F which is within the Technical Specification limit of +7.0 pcm/°F at HZP. Thus, no rod withdrawal limits were needed to ensure the +7.0 pcm/°F limit was met. The design acceptance criterion for the ARO critical boron concentration was satisfactorily met using the revised criteria of +50/-70 ppm as recommeded by Westinghouse. This change reflects the high boron concentration at BOL required for extended life cores; however these limits remain within the Technical Specification requirements of 3.1.1.3.

TABLE 5.1

ARO HZP ISOTHERMAL AND MODERATOR TEMPERATURE COEFFICIENT

Rod Configuration	Boron Concentration	Measured ITC	ITC Design Acceptance	Calculated MTC
All Rods Out	2107 ppm	+2.77 pcm/°F	+2.96 pcm/°F	+4.55 pcm/°F

ITC - Isothermal temperature coefficient, includes -1.78 pcm/°F doppler coefficient

MTC - Moderator only temperature coefficient, normalized to the ARO condition

ARO HZP BORON ENDPOINT CONCENTRATION

Rod Configuration	Measured C_B (ppm)	Design - predicted C_B (ppm)
All Rods Out	2112	2166

6.0 CONTROL AND SHUTDOWN BANK WORTH MEASUREMENTS

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 (the "Reference Bank") is carefully measured using the standard boron 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.

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 BANK WORTH MEASUREMENTS

Bank	Predicted Integral Bank Worth & Review Criteria (pcm)	Measured Bank Worth (pcm)	Percent Difference (pcm)
Control A	337 ± 100	267.5	-20.6
Control B	733 ± 110	717.8	-2.1
Control C	845 ± 127	768.8	-9.0
Control D	483 ± 100	459.0	-5.0
Shutdown A	210 ± 100	207.2	-1.3
Shutdown B (Reference)	940 ± 94	896.0	-4.7
Shutdown C	435 ± 100	423.9	-2.6
Shutdown D	436 ± 100	429.5	-1.5
Shutdown E	490 ± 100	434.7	-11.3
All Banks Combined	4909 + 491	4604.4	-6.2

7.0 STARTUP AND POWER ASCENSION

PURPOSE

The purpose of the power ascension program was to provide controlling instructions for:

1. NIS intermediate and power range calibration as required prior to startup and during power ascension to take into account the effect of a low leakage core.

2. Conduct of startup and power ascension testing, to include:

   a. HZP reactor physics tests
   b. Reactor coolant system flow measurement
   c. Core hot channel factor surveillance
   d. Incore-excore AFD channel calibration
   e. Reactor Coolant System Delta T Calibration

SUMMARY OF RESULTS

Full core flux maps were obtained at about 30%, 49%, 76% and 98% RTP. Hot Channel factors were evaluated at each power plateau and are shown in Table 7.1. The incore and excore delta-I were also evaluated at each plateau. Reactor coolant flow was determined from calorimetric measurements at 96% RTP. An incore-excore recalibration test was performed at 76% RTP.

Delta T calibration constants were determined at 98% power using the calorimetric measurements and measured values of T-HOT and T-COLD.

TABLE 7.1

SUMMARY OF POWER ASCENSION FLUX MAP DATA

Param	Map 130	Map 131	Map 135	Map 136
Avg. % Power	30	49	76	98
LOPAR FDHN Limit	1.899	1.807	1.683	1.580
VANTAGE 5 FDHN Limit	1.995	1.904	1.768	1.659
LOPAR FDHN Measured	1.5670	1.4560	1.4336	1.4093
VANTAGE 5 FDHN Measured	1.6084	1.5977	1.5447	1.5310
Core Avg. AFD	-0.1	8.4	5.1	4.9
Avg. Core % A.O.	-0.290	17.275	6.659	4.973
Most Limiting FQ(Z) +2%	1.8978	2.3219	1.8393	1.8089
Transient FQ Limit	3.4023	4.2511	2.4276	1.8311

8.0 REACTOR COOLANT SYSTEM FLOW MEASUREMENT

PURPOSE

The purpose of this test was to determine the flow rate in each reactor coolant loop in order to confirm that the total core flow met the minimum flow requirement given in Technical Specifications.

SUMMARY OF RESULTS

To comply with the Technical Specifications, the total reactor coolant system flow rate determined at normal operating temperature and pressure must equal or exceed 393,136 gpm. The total core flow was determined to be 400,231 gpm.