GEORGIA POWER COMPANY
VOGTLE NUCLEAR PLANT
UNIT 2, CYCLE 3
STARTUP TEST REPORT

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1.0 INTRODUCTION

The Vogtle Nuclear Plant Unit 2 Cycle 3 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 2 of the Vogtle Nuclear Plant is a four loop Westinghouse pressurized water reactor rated at 3411 MWth. The Cycle 3 core loading consists of 193 17 x 17 fuel assemblies.

Unit 2 began commercial operations on May 19, 1989 and has completed the first two cycles with the following average burnups:

Cycle 1	Complete 09/14/90	17,161 MWD/MTU
Cycle 2	Complete 03/09/92	17,008 MWD/MTU

Seventy-six of the 193 assemblies comprising Cycle 3 are based upon the VANTAGE 5 design.

2.0 UNIT 2 CYCLE 3 CORE REFUELING

REFERENCES

Westinghouse WCAP 13119 (The Nuclear Design Report for the Vogtle Electric Generating Plant, Unit 2, Cycle 3)

SUMMARY

Unloading of the Cycle 2 core into the spent fuel pool commenced on 03/20/92 and was completed on 03/23/92.

Core reload commenced on 04/04/92 and was completed on 04/07/92. The as-loaded Cycle 3 core is shown in Figures 2.1 through 2.5, which give the location of each fuel assembly and insert. The Cycle 3 core has a nominal design lifetime of 18,500 MWD/MTU and consists of 17 Region 2 assemblies from Unit 1, 16 Region 3 assemblies, 68 Region 4A assemblies, 16 Region 4B assemblies, 48 Region 5A assemblies and 28 Region 5B assemblies. Fuel assembly inserts consist of 53 full length control rod clusters, two secondary sources and 138 thimble plug inserts. The assemblies in Regions 5A and 5B contain 4672 fuel rods with Integral Fuel Burnable Absorbers (IFBA).

FIGURE 2.1 UNIT 2 CYCLE 3 REFERENCE LOADING PATTERN

	A	B	C	D	E	F	G	H	J	K	L	M	N	P	R
15					5B36 99DS	5P43 173DS	5N02 77DS	5B63 162DS	5N09 134DS	5P41 166DS	5B31 50DS				
14			5?15 139DS	5P48 R585	5B07 319DS	5P61 R566	5R64 08640	5R76 R578	5B72 316DS	5R65 R605	5B06 326DS	5P09 R576	5N20 156DS		
13		5N40 146DS	5B09 301DS	5N67 296DS	5R46 R556	5P33 113DS	5P70 R591	5P13 129DS	5P83 R586	5P03 81DS	5R16 R582	5N70 384DS	5B08 320DS	5N73 145DS	
12		5P17 R559	5B53 307DS	5P58 R598	5P21 133DS	5N47 300DS	5P76 146DS	5P06 R587	5P74 170DS	5N43 278DS	5P51 100DS	5P24 R608	5B54 290DS	5P39 R596	
11	5B09 98DS	5B03 292DS	5N41 R562	5P52 112DS	5B27 280DS	5P65 175DS	5B38 294DS	5B37 160DS	5N24 298DS	5P25 11DS	5R17 317DS	5P50 59DS	5N44 R589	5R12 322DS	5B16 157DS
10	5P01 152DS	5R66 R570	5P31 22DS	5R19 325DS	5P16 37DS	5P35 R588	5P30 02DS	5R39 R597	5P59 163DS	5P67 R603	5P11 126DS	5B30 314DS	5P55 19DS	5R69 R595	5P57 159DS
9	5N25 104DS	5B57 287DS	5P73 R594	5P79 174DS	5B33 295DS	5P60 169DS	5N48 293DS	5P29 178DS	5N34 307DS	5P37 150DS	5N40 299DS	5P80 172DS	5P81 R602	5B55 304DS	5P56 90DS
8	5N45 168DS	5R75 R565	5P24 153DS	5P56 R561	5B59 155DS	5N42 R558	5P04 149DS	5N62 R569	5P28 63DS	5N20 R574	5B23 76DS	5P08 R564	5P63 78DS	5R74 R577	5B25 84DS
7	5N53 120DS	5R60 308DS	5P71 R568	5P64 180DS	5B21 313DS	5P68 158DS	5B35 323DS	5P18 58DS	5R13 305DS	5P66 10DS	5B25 291DS	5P72 179DS	5P82 R592	5B59 310DS	5N65 89DS
6	5P10 166DS	5R63 R606	5P27 107DS	5R22 312DS	5P53 167DS	5P07 R560	5P34 151DS	5R29 R583	5P22 161DS	5P12 R593	5P14 165DS	5B28 283DS	5P20 14DS	5R62 R590	5P61 160DS
5	5B24 21DS	5R01 286DS	5R32 R572	5P40 41DS	5B31 282DS	5P38 154DS	5R18 306DS	5N44 138DS	5N45 318DS	5P45 96DS	5R14 324DS	5P44 65DS	5B37 R599	5R11 281DS	5B01 40DS
4		5P62 R563	5R52 289DS	5P19 R601	5P54 38DS	5N36 285DS	5P69 142DS	5P05 R557	5P75 28DS	5R26 321DS	5P36 97DS	5P66 R579	5R71 311DS	5P67 R581	
3		5N59 144DS	5R10 304DS	5R50 302DS	5R23 R567	5P23 57DS	5P78 R584	5P64 176DS	5P77 R575	5P32 111DS	5R15 R600	5N68 277DS	5B05 279DS	5N05 143DS	
2			5N50 147DS	5P?5 R573	5R04 297DS	5N67 R604	5R58 08630	5R73 R571	5R56 315DS	5R51 R607	5R02 288DS	5P49 R580	5N64 161DS		
1					5B42 87DS	5P02 171DS	5N01 61DS	5B33 47DS	5N12 31DS	5P62 54DS	5B51 132DS				

5B Region 2 (2.601 w/o)

5P Region 4B (4.190 w/o)

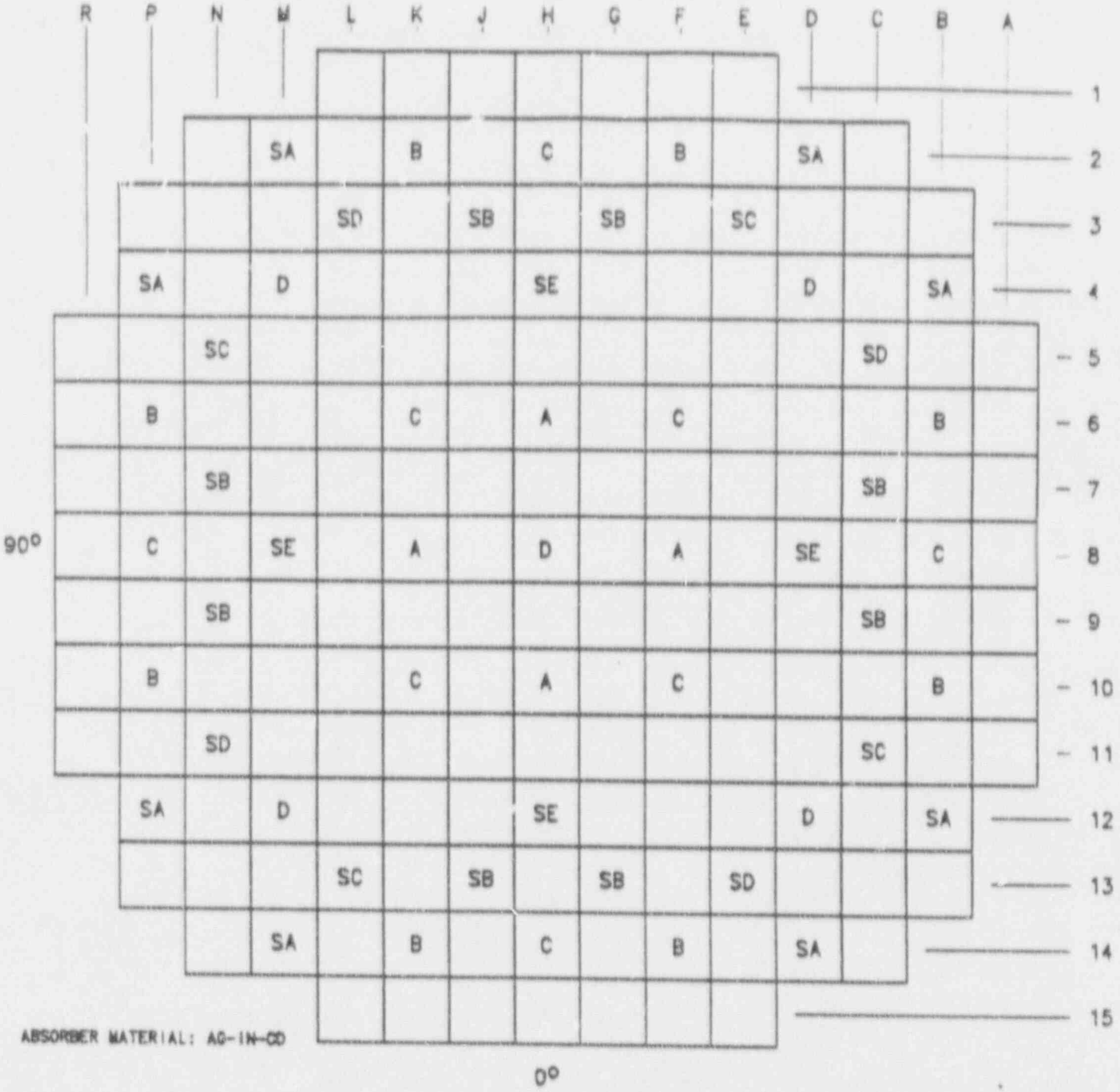
5N Region 3 (3.118 w/o)

5R Region 5A (3.805 w/o)

5P Region 4A (3.807 w/o)

5R Region 5B (4.198 w/o)

FIGURE 2.2 CONTROL ROD LOCATIONS



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

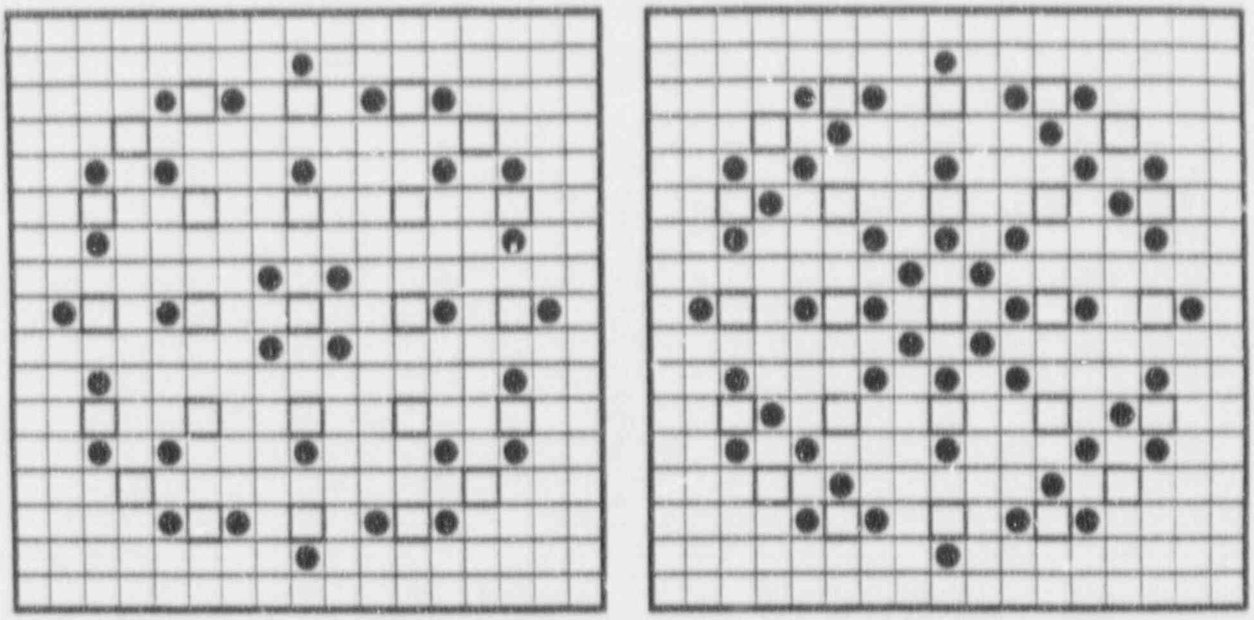
ABSORBER MATERIAL: AG-IN-CD

BANK IDENTIFIER	NUMBER OF LOCATIONS
A	4
B	8
C	8
D	5

BANK IDENTIFIER	NUMBER OF LOCATIONS
SA	8
SB	8
SC	4
SD	4
SE	4

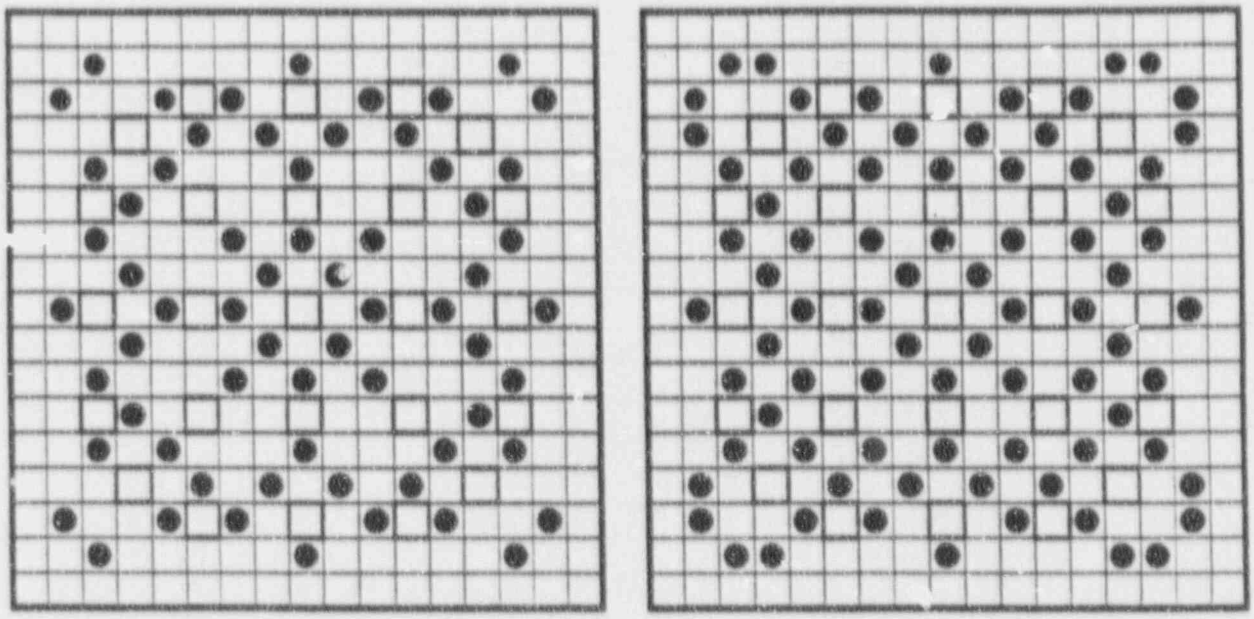
FIGURE 2.3

Burnable Absorber Configurations



32 IFBA ASSEMBLY

48 IFBA ASSEMBLY



64 IFBA ASSEMBLY

80 IFBA ASSEMBLY

LEGEND :

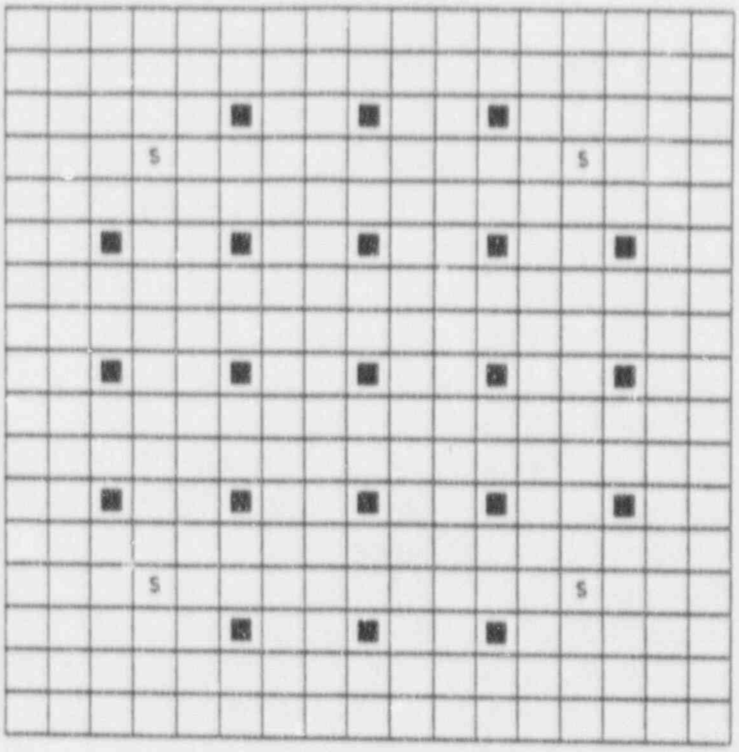
☐ FUEL ROD

☐ GUIDE TUBE OR INSTRUMENTATION TUBE

● IFBA ROD

FIGURE 2.4

Secondary Source Rod Configuration

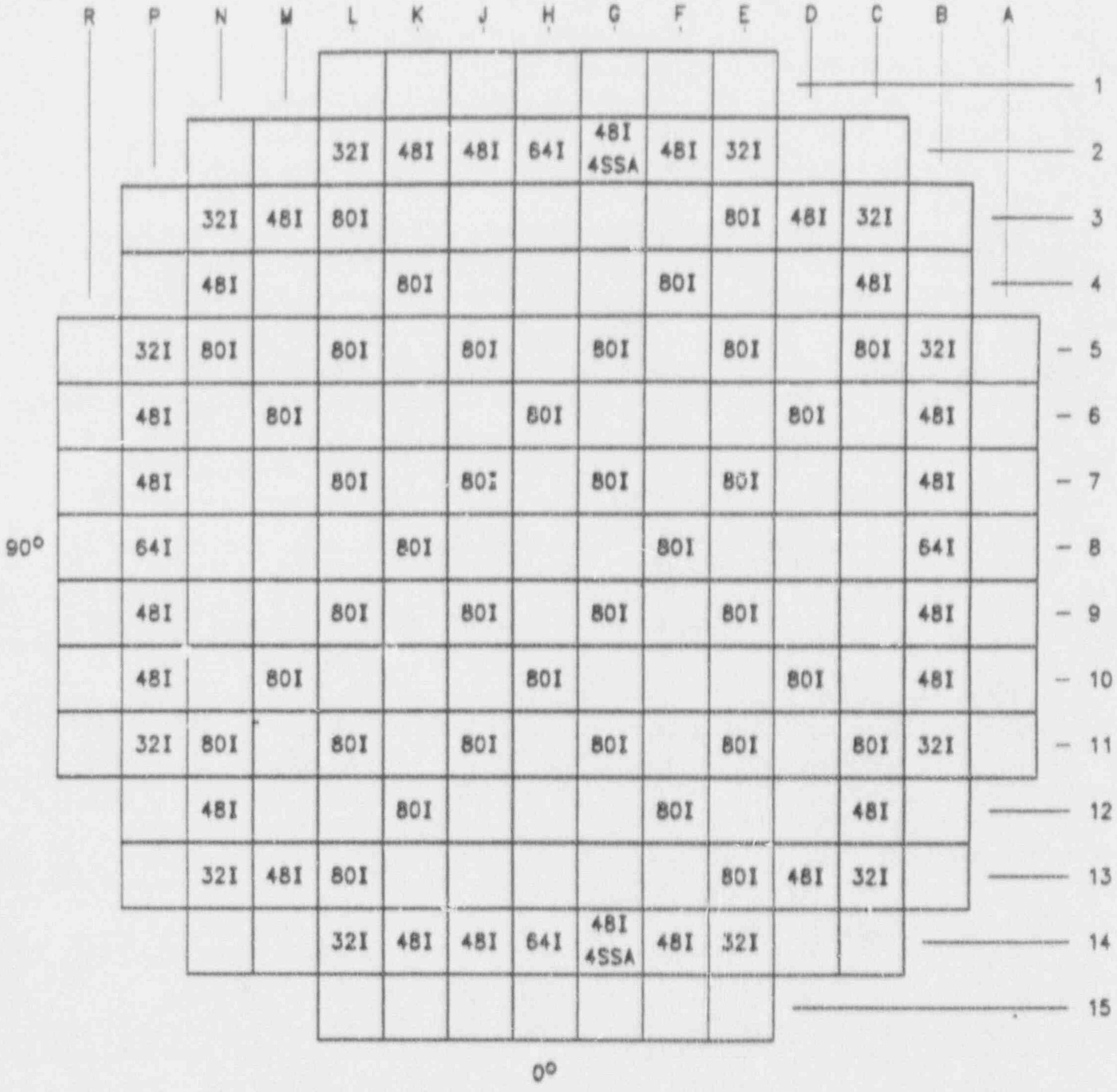


Secondary Sources

S Secondary Source Rod

FIGURE 2.5

Burnable Absorber and Source Rod Locations



	R	P	N	M	L	K	J	H	G	F	E	D	C	B	A
1															
2					32I	48I	48I	64I	48I 4SSA	48I	32I				
3			32I	48I	80I						80I	48I	32I		
4			48I			80I				80I			48I		
5		32I	80I		80I		80I		80I		80I		80I	32I	
6		48I		80I				80I				80I		48I	
7		48I			80I		80I		80I		80I			48I	
8		64I				80I				80I				64I	
9		48I			80I		80I		80I		80I			48I	
10		48I		80I				80I				80I		48I	
11		32I	80I		80I		80I		80I		80I		80I	32I	
12			48I			80I				80I			48I		
13			32I	48I	80I						80I	48I	32I		
14					32I	48I	48I	64I	48I 4SSA	48I	32I				
15															

TYPE	TOTAL
###I..(NUMBER OF IFBA RODS)..................	4672
#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 1.562 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	1588
B12	1556	H06	1602
M14	1560	H10	1542
P04	1582	F08	1592
B04	1600	K08	1584
D14	1616	F02	1578
P12	1576	B10	1624
M02	1564	K14	1528
G03	1542	P06	1570
C09	1540	B06	1544
J13	1544	F14	1600
N07	1532	P10	1554
C07	1558	K02	1528
G13	1558	H02	1570
N09	1556	B08	1554
J03	1592	H14	1518
E03	1524	P08	1570
C11	1538	F06	1580
L13	1606	F10	1538
N05	1540	K10	1542
C05	1570	K06	1548
E13	1566	D04	1554
N11	1556	M12	1500
L03	1578	D12	1526
H04	1540	M04	1572
D08	1594	H08	1562
H12	1552		

SAMPLE SIZE = 53 MEAN = 1.562 SIGMA = 0.026 2SIGMA = 0.052
MEAN-2SIGMA = 1.510 MEAN + 2SIGMA = 1.614

Control rods in locations D14, B10 and M12 fell outside of the 2SIGMA limit and were therefore dropped an additional six times. The drop times for the additional drops were all measured within the 2.7 second Tech. Spec. limit. A summary of the additional rod drop data is provided below.

CONTROL ROD LOCATION	MAXIMUM DASHPOT ENTRY TIME (MSEC)	MINIMUM DASHPOT ENTRY TIME (MSEC)	AVERAGE DASHPOT ENTRY TIME (MSEC)
D14	1.600	1.552	1.585
B10	1.592	1.564	1.577
M12	1.564	1.494	1.527

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 performance of zero power physics tests, and operationally verify the calibration of the reactivity computer.

SUMMARY OF RESULTS

Initial reactor criticality for Cycle 3 was achieved by dilution at 0315 on May 7, 1992. 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 2089 ppm, and Control Bank D position at 187.5 (188/187) 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 +3.02 pcm/°F which meets the design acceptance criteria. This gives a calculated moderator temperature coefficient of +4.81 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.

Because of the high burnup core design of the Unit 2 Cycle 3 core, it was expected that the design acceptance criteria of ± 50 ppm for the ARO critical boron concentration would not be met. As shown in Table 5.1, the measured ARO critical boron concentration was 85 ppm below the predicted ARO critical boron concentration. The boron concentration corresponding to the 1000 pcm Technical Specification acceptance criterion is approximately 134 ppm. Westinghouse conducted a review of the Reload Safety Evaluation (RSE) and determined that the RSE conclusions would remain valid using 134 ppm for the acceptance criterion as long as the remaining HZP physics test results met the relevant design review criteria. Since all of the remaining HZP physics test results met the relevant design review criteria, we conclude that the RSE remains valid.

TABLE 5.1

ARO HZP ISOTHERMAL AND MODERATOR TEMPERATURE COEFFICIENT

Rod Configuration	Boron Concentration	Measured ITC	ITC Design Acceptance	Calculated MTC
All Rods Out	2078 ppm	+3.02 pcm/°F	+3.47 pcm/°F	+4.81 pcm/°F

ITC - Isothermal temperature coefficient, includes -1.7[illegible] 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	2091	2176

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	268 ± 100	209.2	-21.9
Control B	794 ± 119	820.4	+3.3
Control C	778 ± 117	748.1	- 3.8
Control D	502 ± 100	466.8	- 7.0
Shutdown A	244 ± 100	234.9	- 3.7
Shutdown B (Reference)	957 ± 96	996.0	+4.1
Shutdown C	456 ± 100	447.1	- 2.0
Shutdown D	451 ± 100	438.8	- 2.7
Shutdown E	452 ± 100	427.3	- 5.5
All Banks Combined	4902 + 490	4788.6	- 2.3

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 account for the effect of a low leakage core.

2. Performance 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 plateaus of approximately 31%, 49%, 76% and 80% 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. An incore-excore recalibration test was performed at 76% RTP. Reactor coolant flow was determined from calorimetric measurements at 94.8% RTP. Delta T calibration constants were determined at 94.8% 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 67	Map 68	Map 69	Map 71
Avg. % Power	31.2	48.9	75.9	80.3
LOPAR FDHN Limit	1.902	1.831	1.681	1.659
VANTAGE 5 FDHN Limit	1.991	1.903	1.769	1.748
LOPAR FDHN Measured	1.4716	1.4418	1.4256	1.4182
VANTAGE 5 FDHN Measured	1.6209	1.5829	1.5537	1.5454
Core Avg. AFD	4.0	6.0	4.0	3.3
Avg. Core % A.O.	12.796	12.287	5.309	4.065
Most Limiting FQ(Z) +2%	2.1594	2.2008	1.7851	1.7830
Transient FQ Limit	4.0702	4.2654	2.3825	2.2032

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 415,033 gpm.