COMANCHE PEAK STEAM ELECTRIC STATION

UNIT 2 CYCLE 3

STARTUP REPORT

MAY 1996

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1.0 INTRODUCTION

This report presents a summary of the startup of Comanche Peak Steam Electric Station, Unit 2, Cycle 3. Cycle 3 contains the first batch of Siemens Power Corporation (SPC) fuel for Unit 2.

This report satisfies the requirements of the Comanche Peak Technical Specifications, Section 6.9.1.1, that a summary report of unit startup and power escalation testing shall be submitted following installation of fuel that has been manufactured by a different supplier. This report shall be submitted within 90 days following completion of the startup testing.

Comanche Peak Steam Electric Station, located in North Central Texas, is a two unit nuclear power plant. Unit 1 completed initial startup in 1990 and was declared to be in commercial operation on August 13, 1990. Unit 1 is in Cycle 5. Unit 2 completed initial startup in 1993 and was declared to be in commercial operation on August 3, 1993. Each unit utilizes a four loop Westinghouse (W) Pressurized Water Reactor as the Nuclear Steam Supply System. The Nuclear Steam Supply System is designed for a thermal power output of 3425 MWth (3411 MWth reactor power). The plant is operated by TU Electric Company.

Cycle 3 initial criticality occurred on May 5, 1996 and Low Power Physics Testing was completed on May 6. The plant was synchronized to the grid on May 7. Power ascension testing was completed with the performance of a full power flux map at 96.4% RTP on May 17. Full power testing was conducted at 96.4% RTP due to secondary problems that prevented achieving 100% RTP.

2.0 DISCUSSION OF THE SIEMENS POWER CORPORATION FUEL DESIGN

The CPSES Unit 2 Cycle 3 reactor core is comprised of 193 fuel assemblies arranged in a low leakage core configuration. During the refueling prior to operation of Cycle 3, 96 fresh Region 5 SPC fuel assemblies replaced 41 Region 2 W Optimized Fuel Assembly (OFA) assemblies, and 55 Region 3 OFA assemblies. A summary of the Cycle 3 fuel inventory is provided in Table 2.0-1.

The nominal core design parameters used for Cycle 3 are:

Core Power (Mwt)	3411
System Pressure (psia)	2250
Core Inlet Temperature (°F)	560.8
RCS Thermal Design Flow (gpm)	400,800
Average Linear Power Density (kw/ft)	5.434

The energy content of the Cycle 3 core has been designed to accommodate a refueling interval of approximately 18 months.

The CPSES Unit 2 Cycle 2 core configuration was comprised of 193 W OFA 17 x 17 fuel assemblies. The nominal outer diameter of the OFA fuel rods is 0.360". The Cycle 3 configuration includes 97 OFA fuel assemblies and 96 SPC 17 x 17 fuel assemblies. The nominal diameter of the SPC fuel rods is 0.360".

In the CPSES Unit 2 Cycle 2 core configuration, W Wet Annular Burnable Absorber rodlets (WABAs) were used to shape the power distribution and to achieve a desirable moderator temperature coefficient. Solid burnable absorber rodlets manufactured by SPC and consisting of B_4C - Al_2O_3 pellets are used in the CPSES Unit 2 Cycle 3 core design.

TABLE 2.0-1

Fuel Assembly Design Parameters

CPSES Unit 2 Cycle 3

Region	3	4A	4B	5
Enrichment (w/o U_{235})	3.088	3.6076	4.0073	4.9435
Geometric Density (% theoretical)	96.08	95.46	95.40	94.774
Number of Assemblies	9	48	40	96
Expected EOC Assembly Average Burnups (MWD/MTU)	30500	40700	42100	26900

Note: All enrichments and densities are as-built. Region 4 and 5 enrichments represent the value used for the central 132 inches. Regions 4 and 5 have natural uranium top and bottom axial blankets (each 6 inches in length).

2.1 MECHANICAL DESIGN

The SPC 17 x 17 fuel assembly design, used for the Region 5 fuel assemblies and designated CPB-1 by SPC, contains 264 fuel rods and one instrument tube which are supported by eight bi-metallic grid spacers in the fuel assembly structure. The fuel assembly structure consists of an upper nozzle, a lower nozzle, twenty-four guide tubes, and eight spacer grids. Similar to the Region 4 W OFA fuel assemblies, the Region 5 SPC fuel assembly design includes a debris filter bottom nozzle, a removable top nozzle, and axial blankets of natural uranium.

The major differences between the SPC fuel assembly design and the W OFA fuel assembly design are:

- the SPC fuel rod cladding thickness is 0.025", relative to the 0.0225" thick cladding used in the W fuel assembly design.

- The SPC grids are bi-metallic; i.e., Inconel springs imbedded in Zircaloy-4 grids are used to hold the fuel rods in place. The W OFA fuel assembly mid-grids are composed of Zircaloy-4; the top and bottom grids are Inconel.

In other respects, the SPC and W fuel designs are similar. Both designs include dished and chamfered pellets which are sintered to 95% of theoretical density. The SPC fuel assemblies are provided with unique serial numbers engraved on the top nozzle. All locator holes in the top and bottom nozzles are compatible with the upper and lower core support plates.

Along with fuel assemblies, SPC provided 1584 B_4C burnable absorber rodlets distributed among 76 clusters. Thimble plugs are placed in the positions in the burnable absorber assembly not taken by a burnable absorber rodlet.

The physical (including geometrical) properties of the SPC fuel have been designed to be compatible with the W OFA fuel assembly designs and with the CPSES reactor vessel internals, spent fuel racks, and fuel handling equipment. The following considerations apply:

- the fuel assembly design provides adequate clearance with the core support plates at EOL;

- the fuel assembly design is dimensionally compatible with the existing rod control cluster assemblies;

- the top nozzle was verified by testing to be dimensionally compatible with the fuel handling tools;

- dimensional comparisons between the SPC and W fuel assemblies verified the physical compatibility of the SPC fuel assembly with the new and spent fuel storage racks; and,

- co-resident fuel dimensional compatibility was verified by evaluation of the fuel assembly envelope and the centerline locations of the eight spacer grids. The assembly envelope and spacer grid locations are within design limits.

The mechanical design criteria to which the SPC fuel rods, fuel assemblies, and burnable absorber and thimble plug clusters have been designed are consistent with the design criteria used for the W OFA fuel assemblies in the CPSES Unit 2, Cycle 2 core configuration. Compliance with these mechanical design criteria has been demonstrated through mechanical analyses of the SPC fuel rod and fuel assembly designs, using SPC methodologies which have been approved by the NRC.

These evaluations are valid for peak fuel assembly exposures of 54,200 MWD/MTU. This exposure bounds the expected EOC burnup for the CPSES Unit 2, Cycle 3 fuel assemblies. The assumed power histories used in the mechanical design are consistent with those histories expected for Cycle 3 operation. An appropriate number of transients (load changes, scrams, etc.) has been considered in the fatigue evaluations.

2.2 NUCLEAR DESIGN

The nuclear design of the CPSES Unit 2 Cycle 3 core was performed by TU Electric in accordance with methodologies approved by the NRC.

The differences between the SPC fuel assembly design and the W OFA fuel assembly designs are explicitly modeled in the neutronics codes.

The Cycle 3 core configuration is designed to meet a F^T_Q x P ECCS limit of $\leq$ 2.42 x K(z) for a flux difference (ΔI) bandwidth during normal operation conditions of +3, -12% ΔI, where P is the reactor power normalized to rated thermal power.

The core loading pattern for Cycle 3 is shown in Figure 2.2-1. The Cycle 3 core configuration contains a total of 1584 burnable absorber rodlets located in the Region 5 fuel assemblies. The locations of the burnable absorbers and source rods are shown in Figure 2.2-2. The locations of the control and shutdown rods are shown in Figure 2.2-3.

FIGURE 2.2-1
CORE LOADING PATTERN
CPSES Unit 2 Cycle 3

R	P	N	M	L	K	J	H	G	F	E	D	C	B	A	
				DD07 M-9	EE08 5	EE91 5	DD25 H-5	EE92 5	EE78 5	DD40 D-9					1
Quad IV N41 Loop 4		CC44 D-2	DD55 K-11	EE42 5	DD02 F-3	EE85 5	DD52 M-13	EE12 5	DD34 K-3	EE56 5	DD64 F-11	CC29 M-2		Quad I N43 Loop 1	2
	CC45 P-12	EE40 5	EE24 5	DD74 R-9	EE59 5	DD05 P-11	EE76 5	DD06 B-11	EE95 5	DD70 A-9	EE71 5	EE27 5	CC06 B-12		3
	DD75 E-6	EE82 5	DD38 G-8	EE81 5	DD76 G-2	EE07 5	DD35 H-13	EE29 5	DD66 J-2	EE32 5	DD37 H-9	EE13 5	DD49 L-6		4
DD29 G-4	EE05 5	DD63 G-1	EE90 5	DD44 C-13	EE45 5	DD04 G-10	EE23 5	DD47 J-10	EE88 5	DD21 N-13	EE63 5	DD51 J-1	EE65 5	DD08 J-4	5
EE54 5	DD42 N-10	EE96 5	DD88 P-9	EE68 5	DD86 D-13	EE64 5	DD81 L-12	EE84 5	DD67 N-12	EE93 5	DD73 B-9	EE51 5	DD46 C-10	EE22 5	6
EE15 5	EE41 5	DD17 E-2	EE55 5	DD23 F-9	EE50 5	DD65 D-5	EE36 5	DD83 E-12	EE80 5	DD18 K-9	EE10 5	DD03 L-2	EE09 5	EE39 5	7
90° DD31 L-8	DD87 C-12	EE34 5	DD43 C-8	EE20 5	DD78 D-11	EE26 5	CC56 A-8	EE53 5	DD68 M-5	EE87 5	DD48 N-8	EE19 5	DD85 N-4	DD09 E-8	8
EE89 5	EE28 5	DD12 E-14	EE75 5	DD39 F-7	EE31 5	DD56 L-4	EE03 5	DD84 M-11	EE77 5	DD10 K-7	EE60 5	DD24 L-14	EE33 5	EE72 5	9
EE61 5	DD16 N-6	EE06 5	DD58 P-7	EE58 5	DD72 C-4	EE74 5	DD71 E-4	EE67 5	DD77 M-3	EE43 5	DD82 B-7	EE25 5	DD33 C-6	EE69 5	10
DD22 G-12	EE17 5	DD60 G-15	EE16 5	DD45 C-3	EE79 5	DD01 G-6	EE38 5	DD19 J-6	EE57 5	DD20 N-3	EE73 5	DD54 J-15	EE18 5	DD14 J-12	11
	DD57 E-10	EE62 5	DD41 H-7	EE70 5	DD59 G-14	EE66 5	DD36 H-3	EE14 5	DD53 J-14	EE37 5	DD28 J-8	EE48 5	DD79 L-10		12
	CC49 P-4	EE47 5	EE46 5	DD61 R-7	EE44 5	DD30 P-5	EE86 5	DD27 B-5	EE52 5	DD80 A-7	EE04 5	EE02 5	CC53 B-4		13
Quad III N44 Loop 3		CC27 D-14	DD69 K-5	EE94 5	DD26 F-13	EE11 5	DD50 D-3	EE21 5	DD11 K-13	EE01 5	DD62 F-5	CC26 M-14		Quad II N42 Loop 2	14
				DD15 M-7	EE83 5	EE49 5	DD13 H-11	EE35 5	EE30 5	DD32 D-7	ASSEMBLY ID CYCLE 2 LOCATION				15

0°

CC	REGION 3 (3.1 w/o)	DD	REGION 4B (4.0 w/o, Central Zone)
DD	REGION 4A (3.6 w/o, Central Zone)	EE 5	REGION 5 (4.95 w/o, Central Zone)

FIGURE 2.2-2
BURNABLE ABSORBER AND SOURCE ROD LOCATIONS
CPSES Unit 2 Cycle 3

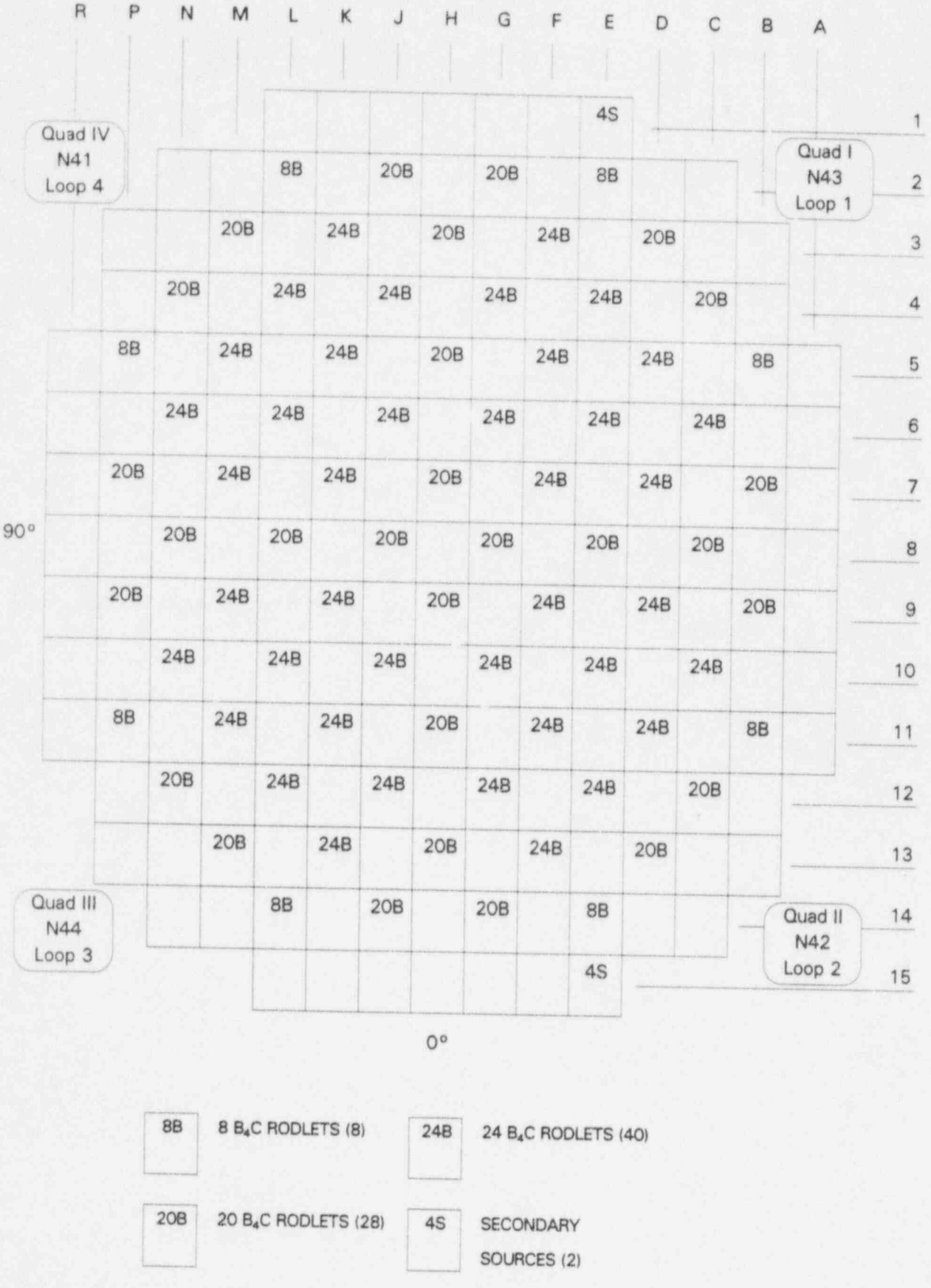
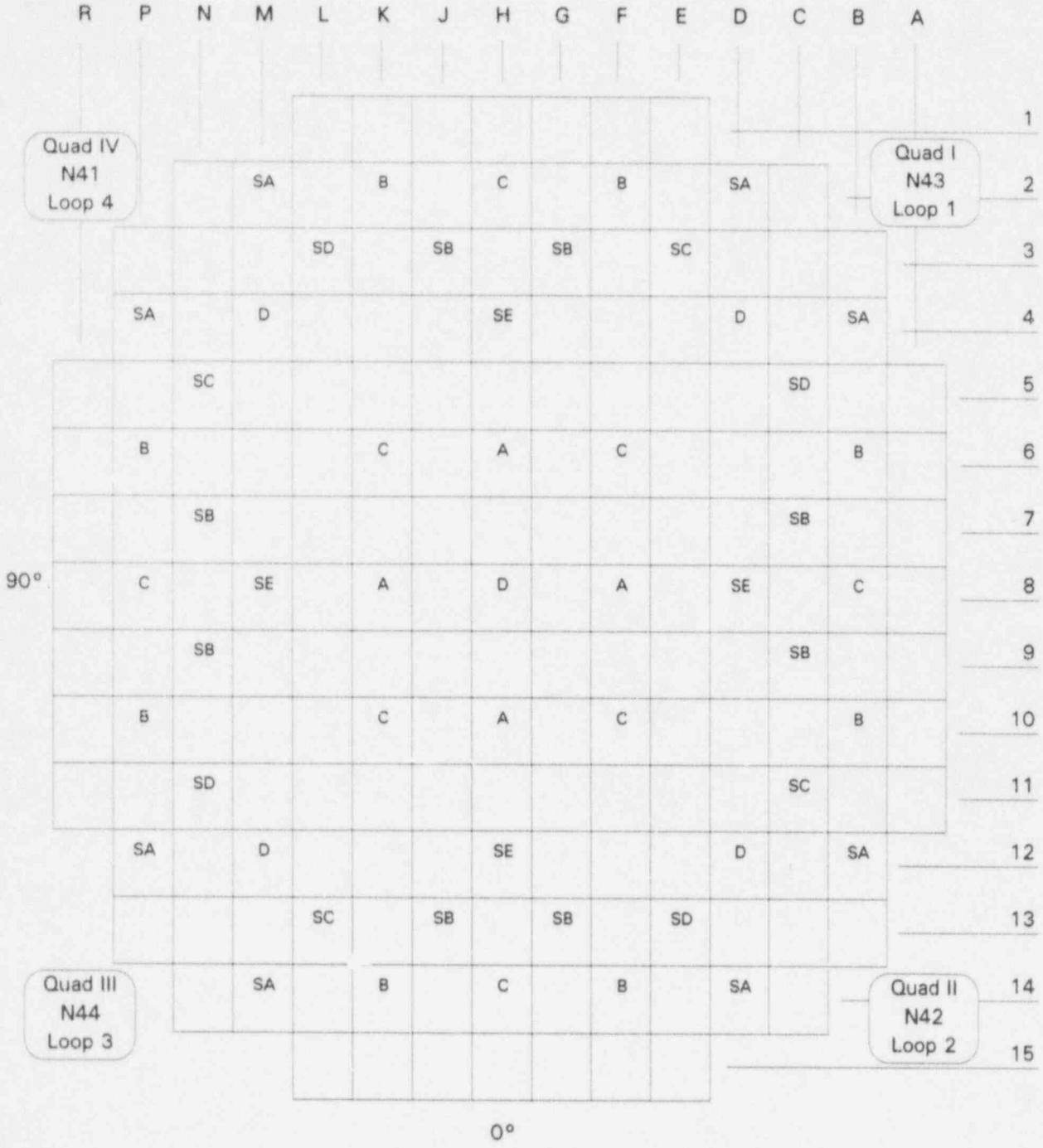


FIGURE 2.2-3
CONTROL AND SHUTDOWN ROD LOCATIONS
CPSES Unit 2 Cycle 3



CONTROL BANK	NUMBER OF RODS
A	4
B	8
C	8
D	5

SHUTDOWN BANK	NUMBER OF RODS
SA	8
SB	8
SC	4
SD	4
SE	4

3.1 CORE LOADING
(RFO-106)

OBJECTIVES

To specify the final core configuration and to control the loading sequences to ensure that the nuclear fuel assemblies are loaded in a safe and cautious manner.

TEST METHODOLOGY

Refueling was performed by completely offloading the Cycle 2 core to the Spent Fuel Pool, changing out fuel inserts, and then loading the Cycle 3 core. Inmast sipping inspection was performed in the containment refueling machine mast during offload. Visual inspection of irradiated assemblies and UT inspection of suspect assemblies were performed in the Spent Fuel Pool. No known leaking fuel assemblies were reloaded for Cycle 3 operation.

The first assembly (one of two source assemblies) to be reloaded was latched on March 22, 1996 at 0734 and the last assembly to be loaded was unlatched on March 24, 1996 at 1049. Inverse Count Rate Ratio (ICRR) plots were maintained during fuel loading (see Figure 3.1-1).

The Cycle 3 core configuration is presented in Figures 2.2-1, 2.2-2, and 2.2-3.

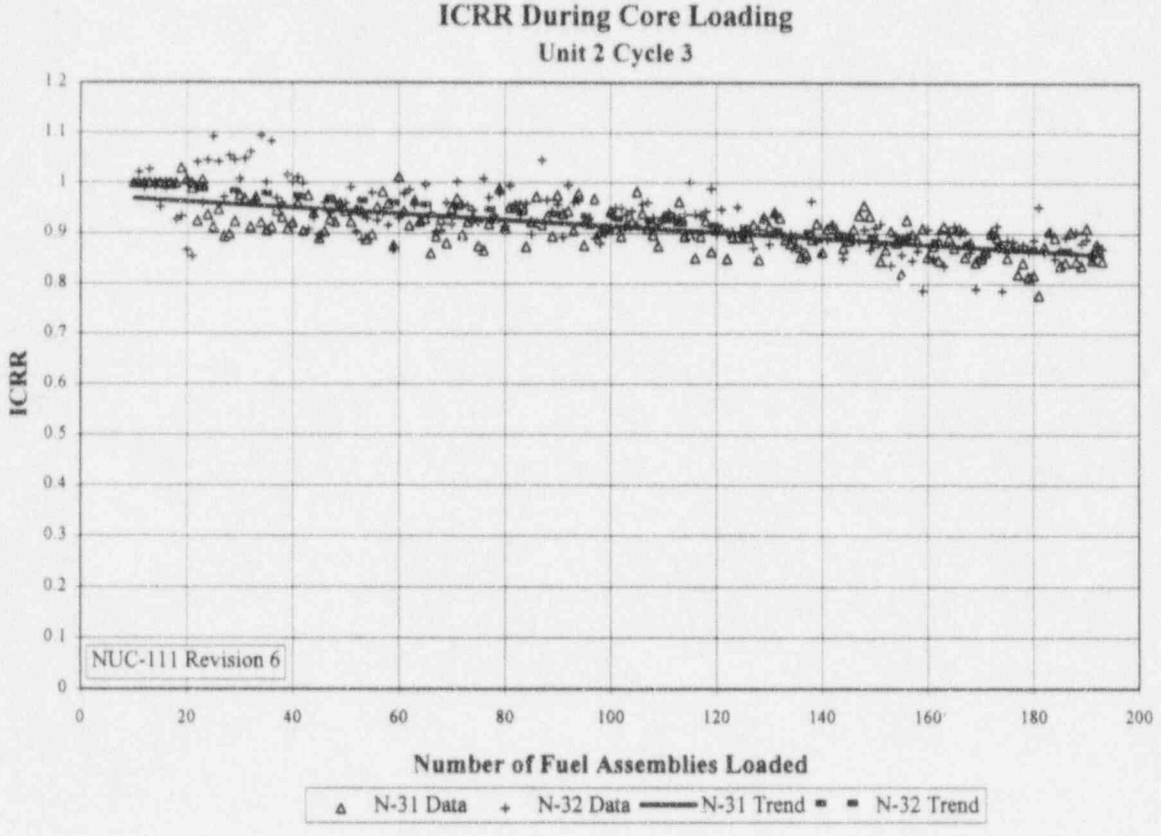
SUMMARY OF RESULTS

On March 15, 1996 (prior to core loading), fuel assembly insert number/type were verified in the spent fuel pool per RFO-204-2 by Reactor Engineering and Quality Control. There were no discrepancies identified.

Core loading was completed on March 24, 1996 at 1049. All 193 assemblies were loaded in the core without incident.

On March 24, 1996, at approximately 1500, the core loading pattern verification process was completed for the Cycle 3 core loading pattern per RFO-204. RFO-204-1 was completed by Reactor Engineering and Quality Control. Westinghouse refueling personnel, Reactor Engineering, and the Operations Fuel Handling Supervisor completed the fuel assembly alignment verifications per RFO-204.

FIGURE 3.1-1


ICRR During Core Loading
Unit 2 Cycle 3
ICRR
1.2
1.1
1
0.9
0.8
0.7
0.6
0.5
0.4
0.3
0.2
0.1
0
NUC-111 Revision 6
0
20
40
60
80
100
120
140
160
180
200
Number of Fuel Assemblies Loaded
N-31 Data
N-32 Data
N-31 Trend
N-32 Trend

3.2 CONTROL ROD DROP TIME MEASUREMENTS (NUC-206)

OBJECTIVE

To measure the drop time of each Rod Control Cluster Assembly (RCCA) under hot, full flow conditions in accordance with Technical Specification 3/4.1.3.4.

TEST METHODOLOGY

The test set consists of two remote Gould strip chart recorders connected to each Digital Rod Position Indication (DRPI) cabinet inside containment via interconnecting cables at the rod control cabinets in the Safeguards Building. A rod bank is withdrawn and the reactor trip breakers are opened, causing the selected group of rods to drop. Analog signals created by the rods passing through the DRPI coils are fed to the Gould strip chart recorders, where the resultant printout is analyzed for drop times. Following completion of data transmission, the process is repeated for the remaining banks.

SUMMARY OF RESULTS

Technical Specification 3/4.1.3.4 requires that the drop time for each RCCA from the fully withdrawn position be less than or equal to 2.4 seconds from the beginning of decay of stationary gripper coil voltage to dashpot entry with Tavg greater than or equal to 531°F and all reactor coolant pumps operating. Under these conditions, the longest drop time was 1.49 seconds for RCCAs at locations H-8, M-2, and C-11.

A summary of all rod drop times is presented as Table 3.2-1. The mean drop time to dashpot entry was 1.45 seconds.

The drop times for RCCAs at locations H8, M2, C11, and L13 were greater than two standard deviations from the mean (1.42 to 1.49 seconds) but were within acceptance criteria. All four fuel assemblies are W OFA assemblies. H8 is a Region 3 assembly, and M2, C11, and L13 are Region 4B assemblies.

TABLE 3.2-1

ROD DROP TIME MEASUREMENT

CPSES UNIT 2 CYCLE 3

Date: 5-5-96

No. of RCS Pumps Running: 4

RCS Temperature: 557°F

RCS Pressure: 2240 psig

RCCA Bank/Group	Core Location	Assy ID	Fuel Region	Guide Tube Material	Assy Burnup (MWD/MTU)	Drag Force (lbs)*		Drop Time (secs)
						In Dashpot	Out Dashpot	
CBA/1	H-6	DD81	4B	W Improved Zirc-4	23184	15	15	1.45
CBA/1	H-10	DD71	4B	W Improved Zirc-4	23238	15	10	1.44
CBA/2	F-8	DD68	4B	W Improved Zirc-4	22974	35	15	1.47
CBA/2	K-8	DD78	4B	W Improved Zirc-4	23209	15	15	1.44
CBB/1	F-2	DD34	4A	W Improved Zirc-4	22363	15	15	1.45
CBB/1	B-10	DD33	4A	W Improved Zirc-4	22264	20	15	1.45
CBB/1	K-14	DD26	4A	W Improved Zirc-4	22209	15	15	1.45
CBB/1	P-6	DD42	4A	W Improved Zirc-4	21828	30	20	1.46
CBB/2	B-6	DD46	4A	W Improved Zirc-4	22305	20	10	1.44
CBB/2	F-14	DD11	4A	W Improved Zirc-4	22068	10	15	1.45
CBB/2	P-10	DD16	4A	W Improved Zirc-4	21897	30	20	1.44
CBB/2	K-2	DD02	4A	W Improved Zirc-4	21931	15	10	1.45
CBC/1	H-2	DD52	4B	W Improved Zirc-4	20895	10	10	1.47
CBC/1	B-8	DD85	4B	W Improved Zirc-4	20375	15	10	1.44
CBC/1	H-14	DD50	4B	W Improved Zirc-4	20456	15	15	1.47
CBC/1	P-8	DD87	4B	W Improved Zirc-4	21281	20	15	1.44
CBC/2	F-6	DD67	4B	W Improved Zirc-4	20667	20	20	1.46
CBC/2	F-10	DD77	4B	W Improved Zirc-4	21020	10	10	1.44
CBC/2	K-10	DD72	4B	W Improved Zirc-4	20990	15	15	1.45
CBC/2	K-6	DD86	4B	W Improved Zirc-4	21034	15	15	1.45
CBD/1	D-4	DD37	4A	W Improved Zirc-4	22690	10	15	1.45
CBD/1	M-12	DD41	4A	W Improved Zirc-4	22816	15	15	1.46
CBD/2	D-12	DD28	4A	W Improved Zirc-4	22578	15	15	1.44
CBD/2	M-4	DD38	4A	W Improved Zirc-4	22814	15	10	1.45
CBD/2	H-8	CC56	3	W Standard Zirc-4	21219	10	10	1.49

TABLE 3.2-1

ROD DROP TIME MEASUREMENT

RCCA Bank/Group	Core Location	Assy ID	Fuel Region	Guide Tube Material	Assy Burnup (MWD/MTU)	Drag Force (lbs)*		Drop Time (secs)
						In Dashpot	Out Dashpot	
SBA/1	D-2	DD64	4B	W Improved Zirc-4	22484	10	10	1.48
SBA/1	B-12	DD79	4B	W Improved Zirc-4	22635	15	20	1.45
SBA/1	M-14	DD69	4B	W Improved Zirc-4	22900	25	25	1.48
SBA/1	P-4	DD75	4B	W Improved Zirc-4	22686	10	10	1.45
SBA/2	B-4	DD49	4B	W Improved Zirc-4	22694	10	10	1.46
SBA/2	D-14	DD62	4B	W Improved Zirc-4	22951	10	15	1.45
SBA/2	P-12	DD57	4B	W Improved Zirc-4	22713	20	15	1.43
SBA/2	M-2	DD55	4B	W Improved Zirc-4	22633	10	10	1.49
SBB/1	G-3	DD06	4A	W Improved Zirc-4	18645	10	15	1.44
SBB/1	C-9	DD24	4A	W Improved Zirc-4	18065	10	10	1.45
SBB/1	J-13	DD30	4A	W Improved Zirc-4	17855	15	15	1.46
SBB/1	N-7	DD17	4A	W Improved Zirc-4	17748	15	15	1.46
SBB/2	C-7	DD03	4A	W Improved Zirc-4	17765	10	15	1.47
SBB/2	G-13	DD27	4A	W Improved Zirc-4	17710	10	10	1.46
SBB/2	N-9	DD12	4A	W Improved Zirc-4	18076	15	10	1.47
SBB/2	J-3	DD05	4A	W Improved Zirc-4	18235	15	15	1.44
SBC/1	E-3	DD70	4B	W Improved Zirc-4	13933	15	15	1.45
SBC/1	C-11	DD54	4B	W Improved Zirc-4	13740	10	10	1.49
SBC/1	L-13	DD61	4B	W Improved Zirc-4	14355	15	15	1.42
SBC/1	N-5	DD63	4B	W Improved Zirc-4	14308	15	10	1.45
SBD/1	C-5	DD51	4B	W Improved Zirc-4	14229	15	20	1.46
SBD/1	E-13	DD80	4B	W Improved Zirc-4	14134	10	15	1.45
SBD/1	N-11	DD60	4B	W Improved Zirc-4	14312	15	15	1.44
SBD/1	L-3	DD74	4B	W Improved Zirc-4	14072	15	15	1.45
SBE/1	H-4	DD35	4A	W Improved Zirc-4	21928	15	10	1.45
SBE/1	D-8	DD48	4A	W Improved Zirc-4	21891	10	10	1.43
SBE/1	H-12	DD36	4A	W Improved Zirc-4	21940	15	15	1.45
SBE/1	M-8	DD43	4A	W Improved Zirc-4	21751	15	15	1.46

* RCCA drag force measurements taken during latching of control rod drive shafts per RFO-403. Drag force limits are as follows:

- Rods completely out of "dashpot" . . . . . ± 40 lbs
- Tips of Rods in "dashpot" . . . . . ± 100 lbs

3.3 INITIAL CRITICALITY
(NUC-101 / IPO-002B)

OBJECTIVE

To achieve initial criticality following refueling in a deliberate and controlled manner.

TEST METHODOLOGY

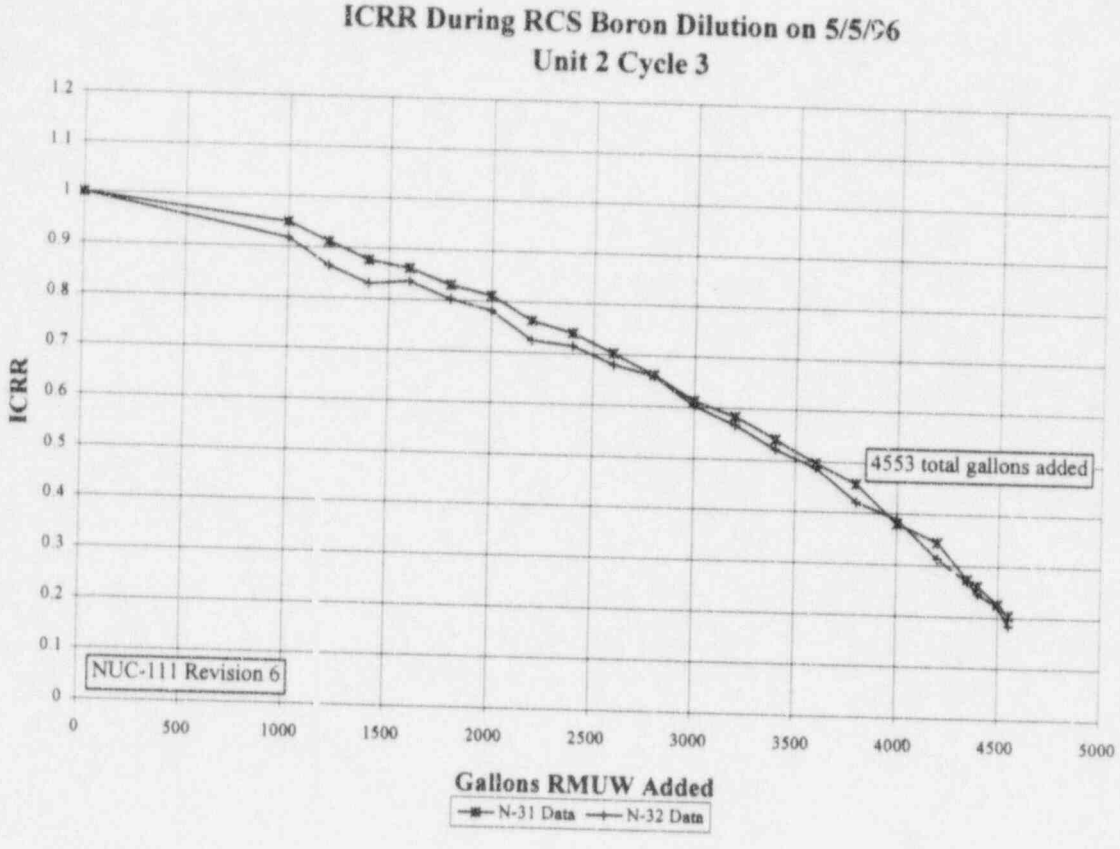
From an initial condition of all rods in and a boron concentration of 2028 ppm, the Shutdown and Control Banks were withdrawn to a desired critical position of Control Bank D at 187 steps in proper overlap sequence. Inverse Count Rate Ratio (ICRR) plots were maintained during bank withdrawal.

Reactor Coolant System dilution was initiated at 60 gpm. During dilution, the ICRR was plotted. When the ICRR decreased to 0.184, dilution was stopped; subsequent mixing with Control Bank D rod pulls to 202 steps brought the reactor critical.

SUMMARY OF RESULTS

Cycle 3 initial criticality was achieved on May 5, 1996 at 2146. Criticality was achieved with Control Bank D at 202 steps and a boron concentration of 1911 ppm. The design estimated critical boron concentration with Control Bank D at 187 steps was 1921 ppm. Figure 3.3-1 is a plot of ICRR for initial criticality plotted versus Reactor Makeup Water (RMUW) added.

FIGURE 3.3-1


ICRR During RCS Boron Dilution on 5/5/96
Unit 2 Cycle 3
ICRR
1.2
1.1
1
0.9
0.8
0.7
0.6
0.5
0.4
0.3
0.2
0.1
0
0
500
1000
1500
2000
2500
3000
3500
4000
4500
5000
4553 total gallons added
NUC-111 Revision 6
Gallons RMUW Added
N-31 Data
N-32 Data

3.4 LOW POWER PHYSICS TESTING
(NUC-301)

Low Power Physics Testing (LPPT) verifies the design of the reactor by performing a series of selected measurements including control/shutdown bank worths, moderator temperature coefficient and boron worth. These measurements are performed by using an ABB/CE Reactivity Monitoring System to indicate reactivity changes below the point of adding heat. LPPT also ensures the surveillance requirements for Technical Specification Special Test Exception 3.10.3 are met.

Results of individual tests completed during the initial criticality and low power test sequence are discussed in the following sections of this report. A tabulation of key physics measurement results is also included in Table 3.4-1. All required tests were performed. Initial criticality was achieved without incident on May 5, 1996. A low power physics testing power range was determined and the reactivity monitoring system was verified to be operating properly. Boron endpoint concentration measurements and associated core reactivity balance calculations were performed. The Isothermal Temperature Coefficient was then measured in order to calculate the Moderator Temperature Coefficient. The Moderator Temperature Coefficient, while slightly positive, was less positive than the Technical Specification limit and rod withdrawal limits were not required. Control rod worths were inferred using the bank exchange method (rod swap method). All required testing was satisfactorily completed.

Upon completion of LPPT, the plant was aligned as directed by the Shift Manager for power operations and additional power ascension testing.

TABLE 3.4-1

HZP PHYSICS TESTING RESULTS

REVIEW CRITERIA

TEST PARAMETER	Design	Actual	Review Criteria
All Rods Out Critical Boron (ppm)	1928	1894	1878 to 1978
Isothermal Temperature Coefficient (pcm/°F)	-0.44	-1.02	-2.44 to +1.56
Differential Boron Worth (pcm/ppm)	-7.25	-7.44	-8.34 to -6.16
Reactivity Computer Error	NA	*1.52%	4%
Reference Rod Bank Worth Error	NA	-0.24%	-10% to +10%
All Other Banks Worth Error (max.)	NA	7.94%	-15% to +15%
Total Rod Bank Worth Error	NA	1.18%	≤ +10%

*The average of the absolute values of the difference between the reactivity indicated by the reactivity computer and the corresponding reactivity calculated from the reactor period or doubling time for three separate checkouts of the ABB/CE Reactivity Monitoring System.

ACCEPTANCE CRITERIA

TEST PARAMETER	Design	Actual	Acceptance Criteria
All Rods Out Critical Boron Difference (pcm)	NA	246	< 1000
Moderator Temperature Coefficient (pcm/°F)	+1.10	+0.52	< +5
Reference Rod Bank Worth (pcm)	819	821	696 to 942
Total Rod Bank Worth (pcm)	4652	4597	≥ 4187

NOTE: pcm means percent millirho, equivalent to a reactivity value of 10^{-5} $\Delta K/K$

3.4.1 DETERMINATION OF THE RANGE FOR PHYSICS TESTING

OBJECTIVE

To determine the neutron flux level at which detectable reactivity feedback from fuel heating occurs and to establish the flux range for low power physics testing.

TEST METHODOLOGY

With the reactor critical at a power level of approximately 1.0 E-8 amps (as indicated by the Intermediate Range channels), approximately 40 pcm of positive reactivity was added by withdrawal of Control Bank D. Flux was allowed to increase until reactivity feedback effects were observed by a decrease in the indicated core reactivity, as indicated on strip chart recorders.

The lower physics testing range limit was set at the low end of the decade which was below the decade in which the point of adding heat was observed.

SUMMARY OF RESULTS

Detectable reactivity feedback was observed at approximately 9.5 E-7 amps as indicated by the Intermediate Range channels. The low power physics testing range was set at 2.0 E-8 to 2.0 E-7 amps.

3.4.2 ARO BORON ENDPOINT MEASUREMENTS

OBJECTIVES

To measure the critical boron concentration at various Control Bank configurations.

TEST METHODOLOGY

Conditions were established with Control Bank D greater than or equal to 190 steps (within 30-50 pcm of its endpoint configuration) with the reactor critical in the low power physics testing range. The control bank was inserted/withdrawn as applicable to its endpoint configuration while monitoring reactivity. The changes in reactivity due to bank movement and Tavg deviation from Tref were converted to equivalent boron concentration units and used to correct the initial boron concentration, yielding the endpoint boron concentration.

SUMMARY OF RESULTS

All boron endpoint measurements met the review criteria, as summarized below.

Bank Position	Design (ppm)	Measured (ppm)	Difference (ppm/pcm)	Review Criteria	Acceptance Criteria (pcm)
All Rods Out	1928	1894	34 / 253	± 50	< 1000
Reference Bank In	1815	1784	31 / NA	±272	NA

3.4.3 MODERATOR TEMPERATURE COEFFICIENT MEASUREMENTS

OBJECTIVE

To measure the Isothermal Temperature Coefficient (ITC) and calculate the Moderator Temperature Coefficient (MTC).

TEST METHODOLOGY

The ITC measurement was performed by first decreasing, then increasing Tavg using steam dump control and measuring the resulting reactivity changes. The ITC is the change in reactivity divided by the associated change in temperature.

The MTC was determined by subtracting the design Doppler Temperature Coefficient from the ITC.

SUMMARY OF RESULTS

As required by Technical Specification 3/4.1.1.3.a, the BOL/ARO/HZP-MTC must be less positive than +5 pcm/°F. The subject MTC was determined to be +0.52 pcm/°F.

All measurements met the review criteria, as summarized below.

	Design (pcm/°F)	Measured (pcm/°F)	Difference (pcm/°F)	Review Criteria (pcm/°F)	Acceptance Criteria (pcm/°F)
ITC	-0.44	-1.02	0.58	± 2.0	NA
MTC	+1.10	+0.52	0.58	NA	< +5.0

3.4.4. BANK REACTIVITY WORTH MEASUREMENTS

OBJECTIVE

To measure integral reactivity worth of each Control and Shutdown Bank.

TEST METHODOLOGY

Integral bank worths were measured using the rod swap method. The subject bank was inserted then countered by pulling the reference bank in response to the change in reactivity caused by the insertion of the measured bank. Each bank's worth was determined by comparison to the Reference Bank's worth of 821.0 pcm.

SUMMARY OF RESULTS

The review criteria was met, as summarized below.

Review Criteria:

Individual Banks

≤ 15% of design or ≤ 100 pcm, whichever is greater,

Total Worth

≤ 110% of design.

Acceptance Criteria:

Sum of measured bank worths shall be no less than 90% of the design sum of bank worths.

BANK	DESIGN (pcm)	MEASURED (pcm)	DIFFERENCE	
			[d-m] (pcm)	[(d-m)/d] (%)
SBB Ref	819.00	821.00	-2.00	-0.24
CBA	315.74	290.67	25.07	7.94
SBA	314.53	304.63	9.91	3.15
SBE	313.02	312.02	0.99	0.32
SBD	439.73	439.50	0.23	0.05
SBC	440.79	438.00	2.79	0.63
CBD	531.29	512.50	18.79	3.54
CBB	679.49	687.58	-8.10	-1.19
CBC	798.56	791.50	7.06	0.88
TOTAL	4,652.14	4,597.40	54.74	1.18

3.5 FLUX MAPPING
(NUC-101)

OBJECTIVE

To verify adequate flux symmetry and power distribution during initial startup following refueling.

TEST METHODOLOGY

Flux maps were taken at 27.87%, 78.42% and 96.4% RTP. All acceptance criteria were met for the flux maps.

SUMMARY OF TEST RESULTS

Low power flux map results are presented as Table 3.5-1. Incore tilts for lower and upper quadrant 2 exceeded 1.02. NUC-208 states "If the largest quadrant incore tilt is greater than 1.02 but less than or equal to 1.04 the incore tilt is acceptable and reactor power can be increased with normal plant procedures. However, the core designers shall be notified of the tilt. This notification is not a prerequisite for power escalation." The excore tilts matched the quadrants showing incore tilts. Discussions with Reactor Physics (core designers) and Nuclear Engineering Management led to performing intercept current alignments on all four power range Nuclear Instrumentation channels per NUC-203 to zero the excore indicated tilts. New 120% currents were transmitted to I&C for the intercept current alignments. Management expected that further contingency monitoring of tilts may be required up to the 80% plateau. FQ(Z) and FDHN were less than applicable limits. The extrapolated maximum allowable power level based on FQ(Z) was 97.43% RTP, and based on FDHN it was 97.02% RTP.

Intermediate power flux map results are presented as Table 3.5-2. Upper quadrant 2 incore tilt still slightly exceeded 1.02. Further discussions with Reactor Physics and Nuclear Engineering Management resulted in an administrative power level limit of 95% RTP until the core load verification tape could be reviewed for the possibility of a misloaded assembly. Subsequent review of the tape confirmed a properly loaded core. No technical concerns remained once the tape had been reviewed and power ascension to 100% RTP was authorized. NUC-203 base case map and quarter core maps were taken for the Incore/Excore Detector Calibration. See Section 3.6 for Incore/Excore Detector Calibration. FQ(Z) and FDHN were less than applicable limits. The extrapolated maximum allowable power level based on FQ(Z) was 102.78% RTP, and based on FDHN it was 101.38% RTP.

Full power flux map results are presented as Table 3.5-3. Power level was stabilized at 96.4% RTP due to valve problems in the secondary plant. This power level satisfied the NUC-101 full power flux map prerequisite of 91-100% RTP. Upper quadrant 2 incore tilt still slightly exceeded 1.02. New 120% currents were transmitted to I&C for Nuclear Instrumentation alignment. See Section 3.6 for Incore/Excore Detector Calibration. FQ(Z) and FDHN were less than applicable limits. The extrapolated maximum allowable power level based on FQ(Z) was 105.06% RTP, and based on FDHN it was 103.33% RTP.

TABLE 3.5-1

LOW POWER FLUX MAP RESULTS

(NUC-201, NUC-208)

Map ID:	U2C03M06
Date Performed:	05/09/1996
Power Level:	27.99%
Cycle Burnup:	18.6 MWD/MTU or 0.4 EFPD
Boron Concentration:	1633 ppm @ 0010 hours on 05/09/1996
Rod Position:	CBD at 180
Core Axial Offset:	4.04%
Measured Peak FQ(Z):	2.2098
FQ(Z) Limit:	4.7432
Maximum Measured FDHN:	1.5638
FDHN Limit:	1.8848

Incore Tilts: Review Criteria ≤ 1.02. If ≥ 1.02 but ≤ 1.04 the tilt is acceptable, however, the core designer is notified.

QUADRANT	UPPER CORE TILT	LOWER CORE TILT
1	1.00440	1.00357
2	1.02498	1.02276
3	0.97508	0.98527
4	0.99555	0.98839

NOTE 1: Nuclear Peaking Factors include required uncertainties.

NOTE 2: FQ limit includes the K(Z) axial penalty factor.

TABLE 3.5-2

INTERMEDIATE POWER FLUX MAP RESULTS

(NUC-201, NUC-203)

Map ID:	UNC03M07
Date Performed:	05/12/1996
Power Level:	78.42%
Cycle Burnup:	95.2 MWD/MTU or 2.2 EFPD
Boron Concentration:	1408 ppm @ 0800 hours on 05/12/1996
Rod Position:	CBD at 183
Core Axial Offset:	-1.310%
Measured Peak FQ(Z):	2.1266
FQ(Z) Limit:	3.0859
Maximum Measured FDHN:	1.5436
FDHN Limit:	1.6503

Incore Tilts: Review Criteria ≤ 1.02. If ≥ 1.02 but ≤ 1.04 the tilt is acceptable, however, the core designer is notified.

QUADRANT	UPPER CORE TILT	LOWER CORE TILT
1	1.00545	1.00328
2	1.02081	1.01925
3	0.97876	0.98708
4	0.99498	0.99039

NOTE 1: Nuclear Peaking Factors include required uncertainties.

NOTE 2: FQ limit includes the K(Z) axial penalty factor.

TABLE 3.5-3

FULL POWER FLUX MAP RESULTS

(NUC-201, NUC-203)

Map ID:	U2C03M17
Date Performed:	05/15/1996
Power Level:	96.4%
Cycle Burnup:	206.4 MWD/MTU or 4.9 EFPD
Boron Concentration:	1340 ppm @ 1100 hours on 05/15/1996
Rod Position:	CBD at 200
Core Axial Offset:	-2.235%
Measured Peak FQ(Z):	2.0603
FQ(Z) Limit:	2.5104
Maximum Measured FDHN:	1.5345
FDHN Limit:	1.5667

Incore Tilts: Review Criteria ≤ 1.02. If ≥ 1.02 but ≤ 1.04 the tilt is acceptable, however, the core designer is notified.

QUADRANT	UPPER CORE TILT	LOWER CORE TILT
1	1.00433	1.00327
2	1.02052	1.01945
3	0.97806	0.98711
4	0.99709	0.99017

NOTE 1: Nuclear Peaking Factors include required uncertainties.

NOTE 2: FQ limit includes the K(Z) axial penalty factor.

3.6 INCORE/EXCORE DETECTOR CALIBRATION (NUC-203)

OBJECTIVES

The object of this test is to measure the Nuclear Instrumentation System (NIS) response to a varying core flux distribution for determination of the normalized detector currents and the excore NIS and plant computer constants for the Incore vs. Excore detector calibration of CPSES Unit 2, Cycle 3. This procedure satisfies Technical Specification surveillance requirements 4.3.1.1.2a and 4.3.1.1.7 for the NIS and N-16 systems.

TEST METHODOLOGY

An incore/excore calibration was performed at 96.4% RTP.

The base case flux map, U2C03M07, was taken with the AFD near the target AFD at approximately 78.42% RTP. The transient portion of the test was performed at 78.42% RTP by using bank D motion to vary the AFD between the lower and upper AFD limits. The RCS boron dilution was initiated at approximately 1115 on May 12, 1996. The AFD was reduced to and maintained near the lower target band limit for approximately two hours to allow the xenon concentration to redistribute. After this time, the rods were borated out and the first quarter core flux map (QCFM), U2C03M08, was taken with Control Bank D at 180 steps. A three-pass QCFM sequence was used. Additional three-pass QCFMs were taken at changes in the AFD of approximately 2-3%. A total of nine quarter core maps were obtained during the AFD transition. The axial flux difference approached the upper target band limit and rod motion and dilution were used to return the AFD to within the target band.

A total of ten flux maps were taken at the 78.42% RTP plateau and the data was processed in parallel with the testing. The data reduction was completed. Based on the AFD Monitor Check from map U2C03M07, no adjustments to the NI system were required to be performed at this power plateau. The confirmation of the calibration standard was performed per NUC-203 in accordance with methodology detailed in RXE-94-02. The confirmation resulted in a high degree of confidence that the standard is still valid.

Following confirmation of the calibration standard, reactor power was increased to about 96.4% RTP and stabilized with Xenon equilibrium. A higher power level was precluded due to valve problems in the secondary plant. U2C03M17 was taken at this power plateau and used to perform the final testing required per NUC-101. An AFD Monitor Check was performed per NUC-203 and revealed the need for an intercept current alignment and delta-Q alignment. New 120% currents were transmitted to I&C for the final power ascension Nuclear Instrumentation alignment.

SUMMARY OF RESULTS

The incore/excore detector calibration sheets were forwarded to I&C for calibration of the NIS channels on May 16, 1996 after they were reviewed and approved. Instrument and Control started and completed the calibrations of the NIS channels on May 17, 1996. This concluded the power ascension program outlined in NUC-101.

3.7 REACTOR COOLANT FLOW MEASUREMENTS (PPT-S2-7018)

OBJECTIVE

To determine the total Reactor Coolant System (RCS) flow rate by precision heat balance measurements, with the RCS at ≤ 592°F and ≥ 2219 psig, prior to exceeding 85% RTP.

TEST METHODOLOGY

The total RCS flow rate measurement was performed at 78.4% RTP. The instrumentation used for determination of steam pressure, feedwater temperature, feedwater pressure, and feedwater venturi delta-P was calibrated within ninety days of performing the calorimetric flow measurement.

The RCS N-16 Transit Time Flow Meter (TTFM) is also utilized to measure the RCS flow rate. The TTFM measures N-16 produced in the reactor by activation of O-16.

SUMMARY OF RESULTS

The measured total RCS flow rate was 419,550 gpm. This met the acceptance criteria that the total RCS flow rate be ≥ 408,000 gpm, per Technical Specification 3/4.2.5, which includes a flow measurement uncertainty of 1.8%.

For comparison, the measured total RCS flow rate for Unit 2, Cycle 2 was 419,080 gpm. This flow rate was also well above the minimum required Technical Specification limit stated above.

3.8 CORE REACTIVITY BALANCE
(NUC-205)

OBJECTIVE

To compare the overall core reactivity balance with predicted values at hot full power (HFP), all rods out (ARO), equilibrium Xenon/Samarium boron concentration.

TEST METHODOLOGY

Under equilibrium conditions at 96.4% RTP, the Reactor Coolant System measured boron concentration was corrected to yield the Hot Full Power, All Rods Out, Equilibrium Xenon/Samarium boron concentration for comparison with the predicted boron concentration.

SUMMARY OF RESULTS

The equivalent reactivity difference between measured and predicted boron concentration was 445.3 pcm (64.05 ppm) which met the acceptance criteria of 1000 pcm, as required by Technical Specification 4.1.1.1.2.