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Ref: FSAR 4.6.6

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August 2, 2002

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

SUBJECT: COMANCHE PEAK STEAM ELECTRIC STATION (CPSES)
DOCKET NOS. 50-445 AND 50-446
UNIT 2 CYCLE 7 STARTUP REPORT

Gentlemen:

TXU Generation Company LP (TXU Energy) loaded eight (8) Westinghouse fuel assemblies into the reactor core as lead use assemblies in Unit 2 Cycle 7. These assemblies were loaded as the beginning of a transition to Westinghouse fuel from Framatome.

In accordance with the FSAR Section 4.6.6, attached is a summary report of the unit startup and power escalation testing following installation of fuel that has a different design or has been manufactured by a different fuel supplier.

This communication contains no new licensing basis commitments regarding CPSES Units 1 and 2.

IE26

Please contact Mr. J. D. Seawright at (254) 897-0140 (email jseawright@txu.com) should you have any questions.

Sincerely,

TXU Generation Company LP
By: TXU Generation Management Company LLC,
Its General Partner

C. L. Terry
Senior Vice President and Principal Nuclear Officer

By: Fred W Madden
Fred W. Madden
Nuclear Licensing Manager

JDS/js
Attachment

c - E. W. Merschoff, Region IV
W. D. Johnson, Region IV
D. H. Jaffe, NRR
Resident Inspectors, CPSES

TXU Generation Company LP
COMANCHE PEAK STEAM ELECTRIC STATION

ENGINEERING REPORT

Unit 2 Cycle 7
STARTUP REPORT

ERX-2002-004
Revision 0

7/15/02

Prepared By: ______________________ Date: 7-25-02
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Core Performance Engineering

Reviewed By: Norman L. Terrel Date: 7-25-2002
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Approved By: Mickey R. Killgore Date: 8/1/2002
Mickey R. Killgore
Reactor Engineering Manager

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1.0 INTRODUCTION

This report presents a summary of the startup of Comanche Peak Steam Electric Station (CPSES), Unit 2, Cycle 7. Cycle 7 contains 84 fresh assemblies supplied by Framatome ANP (FRA-ANP) (formerly Siemens Power Corporation), as well as 8 fresh "Lead Use" assemblies of Westinghouse supplied fuel.

This report satisfies the requirements of CPSES FSAR section 4.6.6, which states 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.

CPSES, 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 9. 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. Unit 1 is rated for a thermal reactor power level of 3411 MWth, and Unit 2 is rated at 3458 MWth. The plant is operated by TXU Generation Company LP.

Cycle 7 initial criticality occurred on May 3, 2002, and Low Power Physics Testing was completed later that day. The plant was synchronized to the grid on May 4. Power ascension testing continued, and 100% RTP was reached on May 9, but power was then reduced to 54% due to equipment problems. Full power was again reached on May 14, and power ascension testing was completed with the performance of a full power flux map on May 17.

2.0 DISCUSSION OF THE WESTINGHOUSE FUEL DESIGN

The CPSES Unit 2 Cycle 7 reactor core is comprised of 193 fuel assemblies arranged in a similar core configuration as found in recent CPSES cycles. The cycle 7 core contains 101 partially spent FRA-ANP fuel assemblies (Regions 7A, 7B, 7C, and 8), 84 fresh Region 9 fuel assemblies supplied by FRA-ANP, and 8 Region 9W "Lead Use" assemblies supplied by Westinghouse. The Region 9W assemblies are of the Optimized Fuel Assembly (OFA) design, similar to the design used for Unit 2 Cycles 1 and 2. A summary of the Cycle 7 fuel inventory is provided in Table 1.

The energy content of the Cycle 7 core has been designed to accommodate a refueling interval of approximately 18 months.

The CPSES Unit 2 Cycle 6 core configuration was comprised of 191 FRA-ANP (formerly Siemens Power Corporation) fuel assemblies (Regions 5, 6, 7A, 7B, and 7C), as well as 2 partially spent Westinghouse fuel assemblies (Regions 2 and 6W). The Cycle 7 configuration includes 185 FRA-ANP fuel assemblies and 8 W OFA fuel assemblies. Both the FRA-ANP and W fuel designs have a nominal outside rod diameter of 0.360 inches, and utilize a 17 x 17 lattice configuration.

In the CPSES Unit 2 Cycle 6 core, solid burnable absorbers (B_4C - Al_2O_3) encased in a Zircaloy-4 clad and manufactured by FRA-ANP were used to shape the power distribution and to achieve a desirable moderator temperature coefficient. Cycle 7 uses both the FRA-ANP burnable absorbers as well as W Wet Annular Burnable Absorbers (WABA). The WABAs consist of B_4C - Al_2O_3 pellets encased between inner and outer Zircaloy-4 clad. WABAs were previously used in CPSES Unit 2 Cycles 1 and 2.

TABLE 1

Fuel Assembly Design Parameters

CPSES Unit 2 Cycle 7

Region	7A	7B	7C	8	9	9W
Enrichment (w/o U_{235}) Central Zone*	4.20	4.55	4.80	4.74	4.65	4.55
Geometric Density (% theoretical)	95.0	95.0	95.0	95.0	95.0	95.5
Number of Assemblies	1	4	8	88	84	8
Pellet Diameter (inches)	0.3035	0.3035	0.3035	0.3035	0.3035	0.3088

*All Cycle 7 fuel regions, except region 7A, employ 2.0 w/o enriched axial blankets in the top and bottom six inch zones of each fuel rod. Region 7A axial blankets are at a natural uranium enrichment.

All enrichments and densities are design values.

2.1 MECHANICAL DESIGN

The W 17 x 17 fuel assembly design, used for the Region 7W fuel assemblies, contains 264 fuel rods which are supported by eight grid spacers in the fuel assembly structure. Mid-span grids are composed of ZIRLO™ while the top and bottom grids are composed of Inconel-718. The fuel assembly structure consists of an upper nozzle, a lower nozzle, twenty-four guide tubes, one instrument tube and eight spacer grids. Similar to the FRA-ANP fuel assemblies, the W OFA fuel assemblies contain 2.0 w/o enriched axial blankets.

The major differences between the W fuel assembly (Region 9W) design and the FRA-ANP fuel assembly (Region 9) design are:

- The W cladding, Guide Tube, Instrumentation Thimble, and mid-span grid assembly material is ZIRLO™, while the FRA-ANP fuel uses bimettalic (Zircaloy-4/Inconel-718) grid assemblies, with Zircaloy-4 Instrumentation Thimbles and Guide Tubes.

- The W fuel has a clad thickness of 0.0062 inches, while the FRA-ANP clad has a thickness of 0.0065

- The W fuel has a nominal density of 95.5 (percent of theoretical), while the FRA-ANP fuel has a nominal density of 95.0.

- The W fuel pellets measure 0.370 inches in length with a 0.3088 inch diameter. FRA-ANP fuel pellets measure 0.350 inches in length with a 0.3035 inch diameter.

- The FRA-ANP fuel assemblies are equipped with the FUELGUARD™ enhanced debris filtering bottom nozzles for improved debris filtering performance. The W assemblies are equipped with the W "Small Hole" debris filtering bottom nozzle, similar to those used in earlier W fuel at CPSES.

- The top nozzle design of the W fuel is incompatible with standard thimble plugs, and must use dually compatible thimble plugs. FRA-ANP fuel can use either the standard or the dually compatible thimble plugs.

In other respects, the FRA-ANP and W fuel designs are similar. Both are provided with unique serial numbers engraved on the top nozzle. Both use removable top nozzles. All locator holes in the top and bottom nozzles are compatible with the upper and lower core support plates.

Along with the fuel assemblies, W provided 192 WABA rodlets distributed among 8 clusters. These WABAs are similar to those used in W fuel in previous CPSES cycles.

The physical (including geometrical) properties of the W OFA fuel are compatible with the FRA-ANP fuel assembly designs and with the CPSES reactor vessel internals, spent fuel racks, and fuel handling equipment. CPSES has previously operated with mixed cores of FRA-ANP / W OFA fuel designs, and successfully demonstrated compatibility with existing rod control clusters and fuel handling equipment.

The mechanical design criteria to which the W fuel rods, fuel assemblies, and burnable absorber and thimble plug clusters have been designed are consistent with the design criteria used for the FRA-ANP fuel assemblies. Compliance with these mechanical design criteria has been demonstrated through mechanical analyses of the W fuel rod and fuel assembly designs, using W methodologies which have been approved by the NRC.

These evaluations are valid for peak fuel rod exposures of 62,000 MWD/MTU (for W fuel with ZIRLO™ clad). This exposure bounds the expected EOC burnup for the 8 lead use assemblies. The assumed power histories used in the mechanical design are consistent with those histories expected for Cycle 7 operation. An appropriate number of transients (load changes, trips, etc.) has been considered in the fatigue evaluations.

2.2 NUCLEAR DESIGN

The nuclear design of the CPSES Unit 2 Cycle 7 core was performed by TXU in accordance with methodologies approved by the NRC.

The differences between the W OFA fuel assembly design and the FRA-ANP fuel assembly designs are appropriately modeled in the core design and safety analysis codes.

The Cycle 7 core configuration is designed to meet at F_Q x P / K(z) limit of $\leq$ 2.42 for an axial flux difference (ΔI) within Technical Specification limits, where P is the reactor power normalized to rated thermal power.

The Cycle 7 core configuration is presented in Figures 1 and 2. The core contains a total of 1440 solid B_4C burnable absorber rodlets located in the Region 9 fuel assemblies, and 192 WABA rodlets located in the Region 9W fuel assemblies.

FIGURE 1
CORE LOADING PATTERN
CPSES Unit 2 Cycle 7

Quad IV N41 Loop 4 — Quad I N43 Loop 1 — Quad II N42 Loop 2 — Quad III N44 Loop 3

90° (left) — 0° (bottom)

	R	P	N	M	L	K	J	H	G	F	E	D	C	B	A
1					GG24 B-12	JJ70 9	HH85 K-15	JJ49 9	HH32 F-15	JJ78 9	GG33 P-12				
2			HH66 J-10	HH51 D-13	JJ16 9	HH17 P-11	JJ91 9	HH76 H-01	JJ29 9	HH22 B-11	JJ32 9	HH46 M-13	GG84 N-14		
3		GG64 B-13	JJ40 9	JW08 9W	HH58 L-10	JJ59 9	HH41 P-07	JJ76 9	HH33 B-07	JJ88 9	HH71 E-10	JW03 9W	JJ51 9	HH31 K-07	
4		HH65 C-12	JW05 9W	HH57 F-07	JJ46 9	HH11 J-12	JJ85 9	HH86 H-11	JJ84 9	HH35 G-12	JJ87 9	HH21 G-10	JW04 9W	HH62 N-12	
5	GG32 D-14	JJ61 9	HH77 F-05	JJ24 9	HH06 N-10	JJ45 9	HH13 D-11	JJ23 9	HH16 M-11	JJ80 9	HH45 K-03	JJ60 9	HH50 K-05	JJ09 9	GG07 M-14
6	JJ54 9	HH87 E-02	JJ82 9	HH04 D-07	JJ42 9	HH83 K-13	JJ64 9	HH12 H-13	JJ53 9	HH82 N-06	JJ43 9	HH55 M-07	JJ19 9	HH42 L-02	JJ39 9
7	HH39 A-06	JJ41 9	HH07 J-02	JJ90 9	HH63 E-12	JJ50 9	HH74 J-08	JJ36 9	HH03 H-07	JJ77 9	HH70 L-12	JJ73 9	HH84 G-02	JJ33 9	HH54 R-06
8	JJ15 9	HH15 R-08	JJ34 9	HH01 E-08	JJ81 9	HH48 C-08	JJ26 9	GG01 C-03	JJ67 9	HH34 N-08	JJ37 9	HH20 L-08	JJ25 9	HH72 A-08	JJ72 9
9	HH69 A-10	JJ28 9	HH49 J-14	JJ55 9	HH43 E-04	JJ31 9	HH38 H-09	JJ38 9	HH14 G-08	JJ57 9	HH81 L-04	JJ27 9	HH78 G-14	JJ18 9	HH61 R-10
10	JJ89 9	HH10 E-14	JJ62 9	HH27 D-09	JJ68 9	HH02 C-10	JJ74 9	HH23 H-03	JJ14 9	HH80 F-03	JJ71 9	HH75 M-09	JJ48 9	HH08 L-14	JJ69 9
11	GG37 D-02	JJ17 9	HH30 F-11	JJ75 9	HH56 F-13	JJ79 9	HH73 D-05	JJ86 9	HH24 M-05	JJ52 9	HH25 C-06	JJ13 9	HH44 K-11	JJ22 9	GG26 M-02
12		HH19 C-04	JW06 9W	HH26 J-06	JJ20 9	HH64 J-04	JJ66 9	HH36 H-05	JJ21 9	HH53 G-04	JJ63 9	HH79 K-09	JW02 9W	HH40 N-04	
13		HH52 F-09	JJ47 9	JW07 9W	HH68 L-06	JJ44 9	HH28 P-09	JJ92 9	HH18 B-09	JJ30 9	HH37 E-06	JW01 9W	JJ65 9	GG74 P-03	
14			GG79 C-02	HH60 D-03	JJ58 9	HH05 P-05	JJ11 9	HH29 H-15	JJ35 9	HH09 B-05	JJ10 9	HH47 M-03	HH67 G-06		
15					GG17 B-04	JJ83 9	HH59 K-01	JJ12 9	HH88 F-01	JJ56 9	GG19 P-04	ASSEMBLY ID CYCLE 6 LOCATION			

GG	REGION 7A (FRA-ANP, 4.20 w/o, Central Zone)	HH 8	REGION 8 (FRA-ANP, 4.74 w/o, Central Zone)
GG	REGION 7B (FRA-ANP, 4.55 w/o, Central Zone)	JJ	REGION 9 (FRA-ANP, 4.65 w/o, Central Zone)
GG	REGION 7C (FRA-ANP, 4.80 w/o, Central Zone)	JW	REGION 9W (West., 4.55 w/o, Central Zone)

FIGURE 2
BURNABLE ABSORBER AND SOURCE ROD LOCATIONS
CPSES Unit 2 Cycle 7

	R	P	N	M	L	K	J	H	G	F	E	D	C	B	A
1					6S										
2					12B		24B		24B		12B				
3				24W		24B		24B		24B		24W			
4			24W		24B		24B		24B		24B		24W		
5		12B		24B		24B		24B		24B		24B		12B	
6			24B		24B		24B		24B		24B		24B		
7		24B		24B		24B		24B		24B		24B		24B	
8			24B		24B		24B		24B		24B		24B		
9		24B		24B		24B		24B		24B		24B		24B	
10			24B		24B		24B		24B		24B		24B		
11		12B		24B		24B		24B		24B		24B		12B	
12			24W		24B		24B		24B		24B		24W		
13				24W		24B		24B		24B		24W			
14					12B		24B		24B		12B				
15					6S										

Quad IV
N41
Loop 4

Quad I
N43
Loop 1

Quad III
N44
Loop 3

Quad II
N42
Loop 2

90°

0°

12B	12 B_4C RODLETS (16)
6S	SECONDARY SOURCES (2)
24B	24 B_4C RODLETS (52)
24W	24 WABA RODLETS (8)

3.0 DISCUSSION OF THE CYCLE 7 STARTUP TESTS

The objectives, methods, and results of each startup test is described in the following sections. The purpose of the overall test program is to ensure the new cycle reactor core behaves in a manner consistent with the design and safety analyses.

3.1 CORE LOADING

OBJECTIVES

Control the loading sequences to ensure that the nuclear fuel assemblies are loaded in a safe and cautious manner, and that the final core configuration is in agreement with the specified design.

TEST METHODOLOGY

Refueling was performed by completely offloading the Cycle 6 core to the Spent Fuel Pool, changing out fuel inserts, and then loading the Cycle 7 core. Cycle 6 had no indications of leaking fuel, and therefore no inmast sipping inspections or UT inspections of fuel were performed.

The first assembly (one of two source assemblies) to be reloaded was latched on March 17, 2002 and the last assembly to be loaded was unlatched on March 20. Inverse Count Rate Ratio (ICRR) plots were maintained during fuel loading.

The Cycle 7 core configuration is presented in Figure 1.

SUMMARY OF RESULTS

Prior to reload, fuel assembly insert number/type were verified in the spent fuel pool by Core Performance Engineering and Quality Control. There were no discrepancies identified. Fuel assemblies identifications were again verified via underwater camera for each assembly as it was loaded into the core.

Core loading was completed on March 20, 2002. All 193 assemblies were loaded into the core without incident.

Following reload, the core loading pattern verification process was completed for the Cycle 7 loading pattern by Core Performance Engineering and Quality Control.

3.2 CONTROL ROD DROP TIME MEASUREMENTS

OBJECTIVE

To determine the drop time of each Rod Control Cluster Assembly (RCCA) under hot, full flow conditions in accordance with Technical Specification SR 3.1.4.3.

TEST METHODOLOGY

The Plant Process Computer (PPC) method was used to determine the rod drop times for Unit 2 Cycle 7. This involves withdrawing each rod bank and opening the reactor trip breakers. The difference between the time the reactor trip breakers open and the time a RCCA has entered the dashpot (according to PPC DRPI indications) is used to determine the rod drop time. This process is repeated for the remaining banks.

SUMMARY OF RESULTS

Technical Specification SR 3.1.4.3 requires that the drop time for each RCCA from the fully withdrawn position to 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 500°F and all reactor coolant pumps running. Under these conditions, the longest drop time was 2.09 seconds for RCCAs at locations C07, G13, N09, and J03.

All rod banks satisfied review and acceptance criteria.

3.3 INITIAL CRITICALITY

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 2054 ppm, the Shutdown and Control Banks were withdrawn to the full out position (FOP) in proper overlap sequence. Inverse Count Rate Ratio (ICRR) plots were maintained during bank withdrawal.

Reactor Coolant System (RCS) dilution was initiated. During dilution, ICRR was plotted. Criticality was declared on May 3rd, 2002, and dilution was terminated. Control Bank D (CBD) motion was used to stabilize flux level.

SUMMARY OF RESULTS

Cycle 7 initial criticality was achieved in a controlled manner on May 3, 2002 at 0532 hours.

3.4 LOW POWER PHYSICS TESTING

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 the Digital Reactivity Computer (DRC) resident on the Plant Process Computer (PPC) to indicate reactivity changes below the point of adding heat.

The individual tests completed during the initial criticality and the low power test sequences are discussed in the following sections of this report. All required tests were satisfactorily completed.

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

3.4.1 DETERMINATION OF THE RANGE FOR PHYSICS TESING

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 primary IR channel), approximately +40 pcm of positive reactivity was added by withdrawal of Control Bank D. Flux was allowed to increase until fuel temperature feedback effects were observed by a decrease in the indicated core reactivity, as indicated on strip chart recorders.

The physics testing range upper limit was set at 30% of the flux level at which the point of adding heat was observed. The LPPT lower limit is 3% of this point, giving a one decade range in which to perform LPPT.

SUMMARY OF RESULTS

Fuel temperature reactivity feedback was observed at flux levels similar to past CPSES cycles. The LPPT range was set appropriately. There are no review or acceptance criteria for this test.

3.4.2 ARO BORON ENDPOINT MEASUREMENT

OBJECTIVES

To measure the critical boron concentration at the All Rods Out configuration.

TEST METHODOLOGY

Conditions were established with Control Bank D within 30-50 pcm of its full out position configuration with the reactor critical in the low power physics testing range. The control bank was withdrawn to the full out position 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

The ARO boron endpoint measurement satisfied the review and acceptance criteria.

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 Generator blowdown flow and increasing Auxiliary Feedwater Flow to compensate. The resulting reactivity changes were measured and used to calculate the ITC. 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

The measurement of ITC met the review criteria of being within $\pm$ 2 pcm/°F of the design value. The difference between the measured value and design value was similar to past CPSES cycles. MTC met the acceptance criteria of < +5.0 pcm/°F.

3.4.4 REFERENCE BANK WORTH MEASUREMENT

3.4.5 BANK REACTIVITY WORTH MEASUREMENTS (ROD SWAP)

OBJECTIVE

To infer the integral reactivity worth of each Control and Shutdown Bank based on the known IRW of the Reference Bank measurement.

TEST METHODOLOGY

Integral bank worths were measured using the rod swap method. The subject bank was inserted then compensated for 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 measured worth.

SUMMARY OF RESULTS

The following review and acceptance criteria were satisfied.

Review Criteria:

> Individual Banks within 15% or within 100 pcm of design worths, whichever is greater.
>
> Total Worth is $\leq$ 110% of design.

Acceptance Criteria:

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

The differences between the measured values and design values were similar to past CPSES cycles.

3.5 FLUX MAPPING

OBJECTIVE

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

TEST METHODOLOGY

Flux maps were taken at the 28%, 80%, and 100% RTP plateaus. All acceptance criteria were met for the flux maps.

SUMMARY OF TEST RESULTS

Based on the results of the AFD Monitor Check at 28% RTP, an Intercept Current Alignment of the excore Power Range channels was required prior to exceeding 50% RTP. The maximum allowable power level extrapolated above 80% (the next target plateau) based on peaking factors. A check of the core loading pattern was performed by comparing the Relative Power Densities (RPD) from the flux map to design predicted values. All RPD values satisfied review criteria limits.

At 80% RTP, a base case flux map (U2C07M04) and six quarter-core flux maps (U2C07M05 thru U2C07M10) were taken for the Confirmation of the Calibration Standard. The Confirmation of the Calibration Standard met all review and acceptance criteria. Based on the results of the AFD Monitor Check of the base case flux map, an Intercept Current Alignment was required. Peaking factor extrapolation resulted in a most limiting allowable power level in excess of 100% RTP.

Reactor power reached 100% RTP on May 9, 2002 at about 1700 hours. Problems with a Main Feedwater Pump did not allow equilibrium xenon conditions to be reached before commencing a power reduction on May 11, 2002. Power ascension recommenced from 54% RTP on May 13, 2002 and 100% RTP was reached on May 14, 2002 at about 1100 hours. Power was held at 100% RTP long enough to establish xenon equilibrium. A full core flux map was performed. Based on the results of the AFD Monitor Check, an Intercept Current Alignment was required.

The differences between the measured values and design values were similar to past CPSES cycles. All flux maps taken during power ascension displayed adequate flux symmetry and power distributions.

3.6 INCORE/EXCORE DETECTOR CALIBRATION

OBJECTIVES

The objective of this surveillance is to check the validity of the current incore/excore detector calibration equations. The incore axial flux difference (AFD) is measured with a full core flux map and compared to the AFD indicated by the control board indicators, the plant process computer, and the NIS power range excore detector currents. This procedure satisfies Technical Specifications Surveillance Requirements 3.3.1.3.6 and 3.3.1.6.6 for Overtemperature N-16 function.

TEST METHODOLOGY AND RESULTS

Pre-critical adjustment ratios from the Unit 2 Cycle 7 Startup and Operations Report were used to adjust the latest calibration currents from the previous cycle.

A full core flux map was taken at 28% power. AFD Monitor Check calculations passed acceptance criteria, but did not pass review criteria. Therefore, excore detector calibrations were required. Power ascension was allowed to continue as excore detectors were calibrated.

At the next calibration plateau, power was held near 80% for a sufficient amount of time to reach xenon stability. A full core flux map was performed on May 8, 2002. It was determined that AFD indications were within acceptance criteria, but did not pass review criteria. Therefore, excore calibrations were performed.

Quarter Core flux maps U2C07M05 through U2C07M10 were performed on May 8, 2002 through May 9, 2002 to be used in the Confirmation of the Calibration Standard. The flux maps were measured over a total change of 19.5% in incore axial offset. The measurements confirmed that the Calibration Standard could be used in place of multipoint measurements for the calibration of the power range NIS throughout Unit 2 Cycle 7 operation.

Neutron Streaming Gains were determined and transmitted to I&C for calibration of the N16 system.

A full core flux map was performed on May 17, 2002 with the reactor at 100% RTP. The AFD Monitor check satisfied acceptance criteria, but did not satisfy review criteria. Therefore, both the Intercept Current and Delta Q alignments for each excore NIS channel were performed.

3.7 CORE REACTIVITY BALANCE

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 100% 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 within the acceptance criteria of 1000 pcm, as required by Technical Specification SR 3.1.2.1. The difference between the measured value and design value was similar to past CPSES cycles.

4.0 SUMMARY

This report is submitted as required following installation of fuel that has been manufactured by a different supplier. Cycle 7 contains 84 fresh assemblies supplied by FRA-ANP, as well as eight "Lead Use" assemblies of Westinghouse supplied fuel. Comanche Peak has not previously loaded fresh fuel of this specific Westinghouse design.

Comanche Peak has previously used fuel of the Westinghouse OFA design. Since 1993, however, Siemens Power Corporation (now FRA-ANP) has been the primary fuel supplier. The design of this Westinghouse fuel, including the WABA burnable absorbers, is similar to the previous fuel used at CPSES; however, it uses ZIRLO™ materials to replace zircalloy.

Unit 2 Cycle 7 reload, startup, and physics tests were performed without incident. All required testing was performed, and all acceptance criteria were satisfied. The differences between the measured values and design values were similar to past CPSES cycles. Based on the results, the lead use Westinghouse assemblies were properly modeled in the design of the core, and there was no need to perform further testing.