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Joseph M. Farley Nuclear Plant - Unit 2
Cycle 13 Startup Test Report

Ladies and Gentlemen:

In accordance with the reporting requirements of Technical Specification 6.9.1.1, Southern Nuclear Operating Company is submitting a Startup Test Report for Farley Nuclear Plant Unit 2 Cycle 13.

Should you have any questions, please advise.

Respectfully submitted,

Dave Morey

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Enclosure

cc: Mr. L. A. Reyes, Region II Administrator
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SOUTHERN NUCLEAR OPERATING COMPANY
JOSEPH M. FARLEY NUCLEAR PLANT

Startup Test Report
Unit 2 Cycle 13

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1.0 INTRODUCTION

The Joseph M. Farley Unit 2 Cycle 13 Startup Test Report addresses the tests performed as required by plant procedures following core refueling. The report provides a brief synopsis of each test and gives a comparison of measured parameters with design predictions, Technical Specifications, Core Operating Limits Report, or values in the FSAR safety analysis.

The Unit 2 Cycle 13 core has been uprated to increase the NSSS power to 2785 MWth (core full power of 2775 MWth plus 10 MWth pump heat). The uprate design change was accomplished under 10 CFR 50.59 and associated Technical Specifications Amendment Number 129. The Cycle 13 fresh fuel has also been designed to provide: 1) improved fuel skeleton stability under irradiation; 2) improved corrosion performance; and 3) additional measures to control fuel rod internal pressures at high burnups. The additional Vantage + fuel assembly design features adopted for Cycle 13 to obtain improved fuel performance include ZIRLO Mid and IFM grids, ZIRLO guide thimble and instrument tubes, annular fuel pellets in the top six inches of IFBA rods, and 1.25X IFBA at 100 psig Helium backfill pressure. Also to reduce corrosion and the propensity for axial offset anomaly (AOA), insertion of thimble plugs was re-introduced for Cycle 13. The reload design for this cycle utilizes 68 fresh feed ZIRLO clad VANTAGE + assemblies with the above design features, 65 once burned ZIRLO clad VANTAGE + fuel assemblies and 24 twice burned Zircaloy-4 clad VANTAGE 5 fuel assemblies. ZIRLO cladding is designed to provide better corrosion resistance than the Zircaloy-4 cladding and has, based upon oxide measurements as discussed in Section 2.0, performed as expected. The secondary sources are located at D-08 & M-08 within once burned assemblies, as was the case with the previous cycle. The loading pattern places RCCAs into fuel assemblies which will not exceed 40,000 MWD/MTU burnup at EOL. The design depletion of reactivity of the Cycle 13 core is 18,300 MWD/MTU with an allowed power coast down of up to 19,500 MWD/MTU.

2.0 UNIT 2 CYCLE 12 - 13 CORE REFUELING

Unloading of the Cycle 12 core into the spent fuel pool commenced on 4/4/98. During the offload, each fuel assembly was inspected with binoculars for indications of damage or other problems. No indications of physical damage were found. White or grayish deposits were observed but to a much lesser degree than seen in the previous cycle core offload.

As follow-up to previous oxide measurements related to zinc addition, oxide measurements were again performed. The inspections utilized a moveable eddy current probe in a special underwater fixture mounted on the fuel racks in the spent fuel pool. In order to provide baseline data and to validate the accuracy of the eddy current equipment, a discharged fuel assembly which had previously been subjected to oxide thickness measurements was re-tested. Following core unload, oxide thickness measurements were performed on 12 additional fuel assemblies. These measurements showed that the oxide deposits on the fuel rods were within the expected, normal tolerances. Crud scrapings were not performed.

Since the fuel inspections revealed no fuel damage or defects and oxide measurements were within acceptance criteria, no revisions to the original Cycle 13 core reload pattern were required. Cycle 13 Core reload commenced on 4/26/98 and was completed on 4/28/98.

REFERENCES

1. Procedure FP-APR-R12, J. M. Farley Unit 2 Cycle 12-13 Refueling.

2. Westinghouse WCAP 15035, The Nuclear Design and Core Management of the Joseph M. Farley Unit 2 Power Plant, Cycle 13.

3.0 CONTROL ROD DROP TIME MEASUREMENT (FNP-2-STP-112)

PURPOSE

The purpose of this procedure was to measure the drop time of all control rods under hot full-flow conditions in the reactor coolant system in order to ensure compliance with Technical Specification requirements.

SUMMARY OF RESULTS

For the hot full-flow condition (Tavg ≥ 541°F and all reactor coolant pumps operating), Technical Specification 3.1.3.4 requires verification that each control rod will insert in ≤ 2.7 seconds when tripped from the fully-withdrawn position. For this test, an automatic data acquisition system provided by the Analysis and Measurement Services Corporation (AMS) was used to obtain drop time data from an entire rod bank (8 rods) at a time. Individual control rod drop times are shown in Figure 3.1. All control rod drop times were measured to be less than 2.7 seconds. The longest drop time recorded was 2.14 seconds for rod B6. Mean drop times are summarized below.

RCS Conditions	Mean Time to Dashpot Entry	Mean Time to Dashpot Bottom
Hot full-flow	1.43 sec.	1.95 sec.

To confirm normal CRDM operation prior to conducting the rod drop test, the Verification of Rod Control System Operability (FNP-0-ETP-3643) was performed also using the AMS System to acquire stepping data for an entire rod bank at a time. In this test, the stepping waveforms of the stationary, lift and movable gripper coils were examined for anomalies; rod speed was measured; and the functioning of the Digital Rod Position Indicator (DRPI) and bank overlap unit were checked. In addition, the bank overlap unit settings for the fully withdrawn rod position to 226 steps were verified to be correct. Timing measurements were also performed on the stepping waveforms for CRDM Logic Cabinet performance testing. All results were satisfactory.

In addition to control rod drop time measurement, RCCA Eddy Current Testing was performed on each RCCA once the fuel had been offloaded to the Spent Fuel Pool. The purpose of the eddy current testing was to identify and, if necessary, characterize excessive wear on the cladding of the RCCA rodlets. The testing was performed to continue trending previously identified wear scars. All RCCAs met inspection criteria for continued use. Based upon these inspection results, the fully withdrawn position was retained at 226 steps for Cycle 13.

Figure 3.1

FNP Unit 2 Cycle 13 Control Rod Drop Times

	R	P	N	M	L	K	J	H	G	F	E	D	C	B	A
1															
2						1.46 2.04		1.44 1.95		1.42 1.95					
3							1.40 1.94		1.45 2.03						
4				1.44 1.97		1.40 1.97				1.40 1.93		1.41 1.92			
5					1.41 1.93						1.42 2.00				
6		1.45 2.00		1.38 1.90		1.41 1.92		1.40 1.89		1.42 1.90		1.39 1.88		1.57 2.14	
7			1.41 1.90				1.36 1.83		1.38 1.94				1.46 1.98		
8		1.42 1.96				1.38 1.95				1.38 1.91				1.52 2.01	
9			1.42 1.95				1.39 1.94		1.36 1.86				1.43 1.92		
10		1.42 1.95		1.40 1.89		1.39 1.96		1.41 1.96		1.40 1.92		1.39 1.88		1.44 2.00	
11					1.42 1.91						1.43 1.95				
12				1.41 1.91		1.39 1.91				1.43 1.40		1.42 1.92			
13							1.42 1.95		1.42 1.89						
14						1.52 2.06		1.47 1.97		1.43 2.01					
15															

↓ North

X.XX	←Breaker "opening" to Dashpot entry
X.XX	←Breaker "opening" to Dashpot bottom

RCS Temperature:	548	°F
RCS Pressure:	2235	psig
% Flow:	100	%

4.0 INITIAL CRITICALITY (FNP-2-STP-101)

PURPOSE

The purpose of this evolution is to achieve initial criticality under carefully controlled conditions, establish the upper flux limit for conducting zero power physics tests, and operationally verify the calibration of the reactivity computer.

SUMMARY OF RESULTS

Initial reactor criticality for Cycle 13 was achieved during dilution mixing on 5/15/98. The reactor was allowed to stabilize at the following conditions.

RCS Pressure	2235 psig
RCS Temperature	547.4 °F
Intermediate Range Power	1.1×10^{-8} amps
RCS Boron Concentration	1499 ppm
Bank D position	197 Steps

Once criticality was achieved, the point of adding nuclear heat was determined in order to define the flux range for physics testing. The point of adding nuclear heat was determined to be 2.263×10^{-7} amps on Power Range Nuclear Instrumentation (PRNI) channel N-44 that was connected to the reactivity computer. Low power physics testing reactivity measurements were performed with flux level at least 30% below the point of adding nuclear heat. The reactivity computer calibration was verified by making reactivity changes and comparing the reactivity indicated by the reactivity computer with values calculated from the Inhour Equation.

5.0 ALL-RODS-OUT ISOTHERMAL TEMPERATURE COEFFICIENT AND BORON ENDPOINT (FNP-2-STP-101)

PURPOSE

The objectives of these measurements were to determine the hot zero power (HZP) isothermal and moderator temperature coefficients for the all-rods-out (ARO) configuration and to measure the ARO, HZP critical boron (boron endpoint) concentration.

SUMMARY OF RESULTS

The ARO, HZP temperature coefficients and the ARO boron endpoint concentration are tabulated below.

ARO, HZP ISOTHERMAL AND MODERATOR TEMPERATURE COEFFICIENT

Rod Configuration	Boron Conc. (ppm)	Measured ITC (pcm/°F)	ITC Design Acceptance Criterion (pcm/°F)	Calculated MTC (pcm/°F)
All Rods Out	1508	−0.285	−0.199 ± 2	+3.38

Where:

ITC = Isothermal Temperature Coefficient: (includes Doppler Coefficient of −1.595 pcm/°F), and

MTC = Cycle maximum Moderator Temperature Coefficient.

The MTC calculated from testing (+1.31 pcm/°F) was normalized to the ARO design-predicted critical boron concentration (1524 ppm) and was corrected for the predicted MTC increase with burnup (+2.2 pcm/°F) to obtain the +3.38 pcm/°F maximum MTC for Unit 2, Cycle 13.

ARO, HZP BORON ENDPOINT CONCENTRATION

Rod Configuration	Measured C_B (ppm)	Design-Predicted C_B (ppm)
All Rods Out	1508	1524 ± 50

Since the maximum Cycle 13 MTC (+3.38 pcm/°F) was less positive than the Technical Specification limit of +7.0 pcm/°F, no rod withdrawal limits were required. The design review criterion for the ARO critical boron concentration was also satisfied.

6.0 CONTROL AND SHUTDOWN BANK WORTH MEASUREMENTS (FNP-2-STP-101)

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 Dynamic Rod Worth Measurement (DRWM) method. During this measurement each bank was driven continuously from the fully withdrawn position to the fully inserted position at the maximum attainable stepping speed, without changing the boron concentration. The integral of the reactivity change for each bank was measured using the reactivity computer. The measured bank worths satisfied the review criteria both for the banks measured individually and for the total worth of all banks combined.

Summary Of Control And Shutdown Bank Worth Measurements

Control or Shutdown Bank	Predicted Bank Worth (pcm)	Standard Review Criteria (pcm)	Measured Bank Worth (pcm)	Percent Difference From Predicted
A	450.0	± 100	475.4	5.6
B (Ref)	1287.8	± 193	1409.7	9.5
C	790.4	± 119	812.8	2.8
D	1110.8	± 167	1164.2	4.8
SD-A	1166.2	± 175	1260.9	8.1
SD-B	939.3	± 141	952.3	1.4
All banks	5744.5	± 574.4	6075.3	5.76

7.0 POWER ASCENSION ACTIVITIES

Upon completion of HZP physics testing, power ascension testing was conducted. Unit ramping and testing proceeded very smoothly with plant control systems exhibiting very stable performance. The sequence of these activities was controlled under plant procedures FNP-0-SOP-103, Return To Service Checklist and Return To Service Systems Lineup, and FNP-2-ETP-4441, Power Ascension Following Unit Uprate, and completed on June 15,1998. Key activities performed during power ascension or at full power included the following.

1. Correlation of auxiliary reactor power indications to reactor power level determined from calorimetric measurements at various reactor power plateaus.
2. Verification of proper steam generator level control dynamic response at low power.
3. Power Range Nuclear Instrumentation (PRNIS) Axial Offset Calibration and initial core hot channel factors surveillance.
4. Secondary side walk-downs to confirm expected system response at plateaus of approximately 48%, 70%, 80%, 90%, 95% and 100% reactor power.
5. Verification of instrument scaling projections for specified control and protection loops at various reactor power plateaus.
6. Confirmation of proper main control board (MCB) indications, MCB annunciation and plant computer response at various reactor power plateaus.
7. Verification of proper steam generator level control dynamic response at high power.
8. Determination of optimum Median T_{avg} and corresponding main turbine governor valve position.
9. Reactor coolant system flow measurement, core hot channel factors surveillance and Excore Detector calibration surveillance at 100% power.
10. Evaluation of the OPΔT and OTΔT protection loops scaling based upon the 100% loop ΔTs measured during the RCS flow test.
11. Turbine performance testing and final confirmation of proper MCB indications, MCB annunciation and plant computer response at 100% reactor power.

SUMMARY OF RESULTS

During power escalation, stabilization plateaus of approximately 18%, 33%, 48%, 70%, 80%, 90%, 95% and 100% were selected to confirm agreement of diverse reactor power level indication and for conducting of selected system performance testing or required surveillance testing. Reactor power level indications from PRNIS, RCS loop ΔTs, turbine first stage impulse pressure (P_{imp}), and calorimetric power were compared under stable conditions at each of the above plateaus. Agreement of these diverse indications of power level was within the review criteria for continued power escalation without any instrument channel re-calibrations being required except for the Excore PRNIS axial offset calibration normally performed at approximately 33% to 48% power.

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The low power portion of FNP-2-ETP-4440, Steam Generator Water Level Control Testing, was performed during the chemistry hold at approximately 31% power. The purpose of this procedure was to verify proper dynamic response of the steam generator level control system. The sequence of this procedure placed the level control system in "Manual." The last NCB card in the loop was removed and placed on an extender board and the level control system was then returned to "Auto." With the NCB card on an extender board, a 5% setpoint change was introduced at the "setpoint" input. The level control system performance was monitored and performance data was collected and reviewed by SNC and Westinghouse personnel who were present during testing. The acceptance criteria were that the steam generator water level control system would return steam generator median level to the desired setpoint ±2% with dampening oscillations occurring within approximately 3 time constants. Testing at this power level showed that steam generator water level control system was stable and did not require tuning. Overall, the S/G median level overshot the setpoint by 1 to 2% and dampened out to the setpoint ±2% in approximately 6 minutes or 1.5 time constants (based on a 250 second time constant).

Also at this plateau, a full core flux map was obtained as the "base case" map for the Incore-Excore cross-calibration test. Five additional (quarter-core) flux maps were performed at various positive and negative axial offsets in order to develop equations relating detector current to incore axial offset.

While awaiting a chemistry hold at this power level, the PRNIS channels were calibrated. Also a full core flux map at equilibrium Xenon conditions was obtained for evaluation of hot channel factors and confirmation of PRNIS detector axial offset calibration. These results were satisfactory and are summarized in Tables 7.1 and 7.2.

Table 7.1
Detector Current Versus Axial Offset Equations Obtained From Incore-Excore Calibration Test

CHANNEL N41:

I-Top	=	0.9691	* AO	+	173.2089	uA
I-Bottom	=	−1.0480	* AO	+	169.2889	uA

CHANNEL N42:

I-Top	=	0.9539	* AO	+	174.2237	uA
I-Bottom	=	−1.0970	* AO	+	167.8269	uA

CHANNEL N43:

I-Top	=	0.9843	* AO	+	177.9756	uA
I-Bottom	=	−1.0744	* AO	+	171.6638	uA

CHANNEL N44:

I-Top	=	1.0736	* AO	+	191.5263	uA
I-Bottom	=	−1.2757	* AO	+	190.7116	uA

Table 7.2
Summary Of Power Ascension Full Core Flux Map Data

PARAMETER	MAP 328	MAP 329
Avg. % Power	31%	100%
Max. Power Tilt*	1.0090	1.0077
Avg. Core AO	1.056	−3.463
Max. FΔH	1.5274	1.475
FΔH Limit	2.049	1.70
$F_Q(Z)$ Steady State	2.1179	1.9052
$F_Q(Z)$ SS Limit	5.0000	2.4975
$F_Q(Z)$ Transient	2.1568	1.9410
$F_Z(Z)$ Tran. Limit	4.2598	2.2342

*Calculated Power Tilts based on assembly FΔH from all assemblies.

At approximately 48%, 70%, 80%, 90%, 95% and 100% reactor power, secondary side walk-downs, instrument scaling data comparisons, MCB indicator response comparisons, and control systems response and stability evaluations were performed. Assessments of RCS loop T_{hot} and ΔT scaling and margin to OPΔT and OTΔT reactor trip and high steam flow ESF setpoints were also performed during power ascension as designated by FNP-2-ETP-4441. Systems evaluated included: main steam; feedwater and feedwater heater vents and drains; steam generator level control; steam generator feed pump speed control; rod control; pressurizer level control; and pressurizer pressure control. Evaluations performed at the designated reactor power plateaus confirmed that plant instrument loop scaling, MCB indicator response and control systems performance criteria were met. Ample margins to OPΔT, OTΔT and high steam flow setpoints were also confirmed at designated plateaus. The initial calibration of the steam flow instrument channels incorporated conservative scaling with respect to the high steam flow setpoint. Based on operating full power performance data, the steam flow instrument channels were rescaled to the specified setpoint and were renormalized to provide closer agreement with feedwater flow indication. Secondary side walkdowns identified various dump controllers and drain valves that were contributing to thermal losses and affecting secondary side thermal efficiency. These deficiencies were investigated and determined to be equipment maintenance issues and not related to process parameter changes associated with the Unit uprate. Maintenance repairs were initiated and implemented to correct the secondary side thermal efficiency losses identified.

At approximately 90% power, the high power portion of FNP-2-ETP-4440, Steam Generator Water Level Control Testing was conducted with results again demonstrating that the steam generator water level control system was stable and did not require tuning. Overall, the S/G median level overshot the setpoint by 1 to 2% and dampened out to the setpoint (± 2%) in approximately 6 minutes or 1.5 time constants (based on a 250 second time constant).

Following the high power level portion of the steam generator level control testing, the unit was ramped to 2652 MWth (approximately 95% core power level). Evaluation of indicated main turbine #4 governor valve position versus expected and projection of the #4 governor valve position for 100% power were performed. Design projections were 0% and 50% open at 95% and 100% power, respectively, versus measured values of 10% and 43%, respectively.

The unit was then ramped to 2775 MWth at approximately 1% reactor power per hour with the unit being stabilized during the ramp for confirmation of power level by calorimetric. Testing was then conducted to determine optimum median T_{avg} and turbine flow margin. Based on these measurements, the optimum median T_{avg} was determined to be 575 °F. Since this was also the design value specified for initial calibration of program T_{ref}, re-calibration of the Pressurizer Level, Rod Control, and Steam Dumps control loops was not required.

After reaching full power equilibrium Xenon conditions, procedure FNP-2-STP-115.1, RCS Flow Measurement, was performed. The purpose of this procedure was to measure the flow rate in each reactor coolant loop in order to confirm that the total core flow met the minimum flow requirement given in the Technical Specifications. In addition, the RCS loop 100% ΔTs measured during this test were used to evaluate the need to rescale the OPΔT and OTΔT protection channels. A full-core flux map was also concurrently performed to support surveillance for core hot channel factors, incore thermocouples, and PRNIS channel calibration (procedures FNP-2-STP-110, FNP-2-STP-108 and FNP-2-STP-121, respectively). RCS flow was determined by precision calorimetric method and for comparison purposes, by measurement of the loop elbow tap flow meter voltages. Results are given below in Table 7.3. Though lower than the previous cycle 12 measurement, flow results were very close to expected when re-insertion of thimble plugs and an increase in the equivalent steam generator plugging level are considered.

Table 7.3

Unit 2 Cycle 13

RCS Flow Measurement Comparison

		Cycle 12		Cycle 13 Calorimetric Method			Cycle 13 Elbow Tap Method		
	Baseline Flows (gpm)	Calorimetric Flows (gpm)	Elbow Tap Flows (gpm)	Measured Flow Data (gpm)	Difference From Baseline	Change From Previous Cycle	Measured Flow Data (gpm)	Difference From Baseline	Change From Previous Cycle
Loop-1 (A)	99169	94019	97858	92753	-6.5%	-1.3%	96486	-2.7%	-1.4%
Loop-2 (B)	93364	88568	92744	87864	-5.9%	-0.8%	90838	-2.7%	-2.1%
Loop-3 (C)	95997	93391	95298	93107	-3.0%	-0.3%	93400	-2.7%	-2.0%
Total RCS	288530	275978	285900	273725	-5.1%	-0.8%	280725	-2.7%	-1.8%

Notes: 1) Baseline flow was determined by precision calorimetric early in plant life. (Determined for Elbow Tap Methodology comparisons.)

2) Steam generator plugging equivalent increased from previous cycle by 0.6% to a total of 7.3% equivalent plugged (plugged and sleeved).

3) The Cycle 13 core design provided for thimble plugs to be reinstalled. Design predictions regarding thimble plug impact to RCS flow predicted a total flow decrease of 0.6%.

Following completion of median T_{avg} determination and RCS flow measurement, the unit was ramped down to approximately 95% core power (2652 MWth). Turbine baseline testing at pre-uprate conditions was performed as a part of the turbine uprate contract provisions. This confirmed that no unexplained degradations in unit performance had resulted from outage work.

The unit was then ramped back to 100% rated thermal power, and turbine uprate guarantee testing was performed. This testing determined that the unit uprate achieved an increase in electrical output of approximately 26 MWe. On June 15, 1998, following completion of the uprate guarantee testing evolutions, final assessment of unit main control board indications and plant computer alarms was conducted and confirmed no off-normal indications or alarms that were related to the unit uprate. This completed the testing activities specified by FNP-2-ETP-4441, Power Ascension Following Unit Uprate.