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June 12, 2015

Docket Nos.: 50-366 NL-15-0905

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D. C. 20555-0001

Edwin I. Hatch Nuclear Plant - Unit 2
<u>Startup Test Report</u>

Ladies and Gentlemen:

In accordance with FSAR requirements, Southern Nuclear Operating Company (SNC) hereby submits the Unit 2 Startup Test Report for Cycle 24. This report summarizes the startup testing performed on Unit 2 following the twenty-third refueling outage. The report is required due to the first use, other than as lead use assemblies, of GNF2 fuel assemblies loaded for Cycle 24.

The tests demonstrate the successful operation of the Plant E. I. Hatch Unit 2 reactor with the introduction of the GNF2 fuel.

This letter contains no NRC commitments. If you have any questions, please contact Ken McElroy at (205) 992-7369.

Respectfully submitted,

C. R. Pierce

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Regulatory Affairs Director

CRP/RMJ

Enclosure: Edwin I. Hatch Nuclear Plant - Unit 2
GNF2 New Fuel Introduction Startup Test Report
for Cycle 24

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Edwin I. Hatch Nuclear Plant - Unit 2
Startup Test Report

Enclosure

GNF2 New Fuel Introduction Startup Test Report
for Cycle 24

1.0 INTRODUCTION

1.1 Purpose and Summary

The Plant Edwin I. Hatch Unit 2 Startup Test Report is submitted to the Nuclear Regulatory Commission (NRC) in accordance with regulatory commitments contained in the Plant Edwin I. Hatch Unit 2 Final Safety Analysis Report (FSAR) Section 13.6.4. This report summarizes the startup testing performed on Unit 2 following the twenty-third refueling outage. This report is being submitted due to a reload batch of 224 GNF2 fuel assemblies that were loaded for Cycle 24. The GNF2 fuel design has not previously been utilized in Unit 2 except as Lead Test Assemblies (LTAs).

This report consists of a brief summary of the core design followed by summaries of selected static and dynamic reactor core performance tests conducted prior to and during the beginning-of-cycle startup of Plant Hatch Unit 2 Cycle 24. These tests demonstrate the successful operation of the Unit 2 reactor with the introduction of the GNF2 fuel design into production use.

1.2 Plant Description

The Edwin I. Hatch Nuclear Power Plant Unit 2 is a General Electric design boiling water reactor (BWR/4). Plant Hatch Unit 2 is rated at 2804 MW(th) with a generator rating at this power of 920 MW(e). The plant is located on the south side of the Altamaha River, Southeast of the intersection of the river with U. S. Highway #1 in the Northwestern sector of Appling County, Georgia.

1.3 Post-Refueling Outage Startup Test Description

The Edwin I. Hatch Nuclear Power Plant Unit 2 resumed commercial operation on March 14, 2015, after completing a 33-day refueling/maintenance outage. The following core performance tests were performed as part of the post-refueling outage startup test program:

- Core Verification
- Control Rod Drive Timing
- In-Sequence Critical Shutdown Margin Demonstration
- Cold Critical Eigenvalue Comparison
- LPRM Calibration
- APRM Calibration
- Control Rod Scram Time Testing
- Core Performance
- Reactivity Anomaly Calculation

The purpose for, a brief description of, and the acceptance criteria for each of the tests listed above is enumerated in Section 3 of this report.

1.4 Post-Refueling Outage Startup Test Acceptance Criteria

Where applicable, a definition of the relevant acceptance criteria for the test is given and is designated either "Level 1" or "Level 2."

Acceptance Criteria:

Level 1 criteria: Data trend, singular value, or information which relates to Technical Specifications margin and/or plant design in such a manner that requires strict observance.

Level 2 criteria: Data trend, singular value, or information relative to system or equipment performance which does not fall under the definition of Level 1 criteria.

Failure to meet Level 1 criteria constitutes failure of the specific test. The Test Lead is required to resolve the problem, and if necessary, the test is repeated. Level 2 criteria do not constitute a test failure or acceptance; they serve as information only.

2.0 CORE DESIGN SUMMARY

2.1 Cycle/Core Summary

The Cycle 24 design achieves a full power energy of 15.616 GWd/ST or 625.3 effective full power days (EFPDs) at 2804 MWth. This energy includes cycle extension from increased core flow. Two hundred and twenty-four (224) fresh GNF2 bundles, divided into four streams having enrichment that varies from 3.98 w% to 4.11 w% U-235 enrichment, were loaded in a conventional core configuration for a 24-month fuel cycle.

2.2 Calculated Reactivity/Thermal Limit Margins

The two parameters which describe the global behavior of the core throughout the cycle are hot excess reactivity (HER) and cold shutdown margin (CSDM). The 200.0 MWd/ST hot excess reactivity is 1.494%, the early cycle minimum HER is 1.478% at 500.0 MWd/ST, and the mid-cycle peak HER is 1.647% at 9,200.0 MWd/ST. The minimum cold shutdown margin of 1.64% occurs at BOC for the as-to-be-loaded core loading based upon an EOC 23 shutdown at 17.056 GWd/ST cycle exposure. Calculated core parameters are delineated in Table 2.1.

Target rod patterns were developed at reasonable exposure increments and 2,300 MWd/ST sequence exchange intervals. Design margins to thermal limits were met for all exposures.

2.3 Fuel Summary

Table 2.2 provides a list of all fuel batches loaded in Cycle 24. Note that all fuel contains axially varying fuel lattice types.

All returning once-burned fuel and twice-burned assemblies are equipped with the Defender™ Debris Filter LTP, leaving four (4) returning thrice-burned assemblies with the standard GE14 debris filter LTP.

Four thrice-burned GE14 bundles, equipped with standard debris filters, identified below were originally loaded in Hatch-2 Cycle 18 and discharged after Cycle 20. They are being incorporated into Hatch-2 Cycle 24 for a fourth cycle of operation on the core periphery.

One once-burned GE14 bundle identified below was originally loaded in Hatch-1 Cycle 25. It was discharged for post outage inspection after one cycle of operation and reloaded into Hatch-2 Cycle 23. It will complete a third cycle of operation during Cycle 24.

Four GNF2 Lead Test Assemblies (LTAs) were originally loaded in Hatch-2 Cycle 22. The GNF2 LTAs are designed to mimic a sibling GE14 bundle design. The GNF2 LTAs contain ZIRON clad fuel rods and standard NSF channels. They will complete their third cycle of operation during Cycle 24.

Table 2.1
Cycle Calculated Parameters

BOC Core Average Exposure[1]			15,016.56	MWd/ST
Cycle Core Weight			112.2764	ST
Daily Full Power Exposure Capability			24.9741	MWd/ST
Cycle Energy		EFPDs	Exposure	
EUP (rated)[2,3]		629.9	15,730	MWd/ST
Achieved (rated)[2]		625.3	15,616	MWd/ST
Total Energy (with coastdown)[3]		663.0	16,558	MWd/ST
Uncertainty in Energy			± 428	MWd/ST
Cold Shutdown Margin				
BOC			1.94	% Δk
R			0.00	% Δk
B			0.30	% Δk
Hot Excess Reactivity				
BOC	200.00 MWd/ST		1.49	% Δk
Early Cycle Min	500.00 MWd/ST		1.48	% Δk
Mid-Cycle Peak	9,200.00 MWd/ST		1.65	% Δk

1- BOC CAVEX based on projection to an EOC 23 cycle exposure of 17,056 MWd/ST.
2- Rated power at 105% core flow.
3- Energy based on July 16, 2014 Nuclear Fuel Plan of Record

Table 2.2
Fuel Batches Loaded in Cycle 24

QTY	Bundle Type Label
	Fresh Fuel
32	GNF2-P10DG2B398-15GZ-100T2-150-T6-4314
32	GNF2-P10DG2B402-14GZ-100T2-150-T6-4315
72	GNF2-P10DG2B400-13GZ-100T2-150-T6-4316
80	GNF2-P10DG2B411-14GZ-100T2-150-T6-4317
8	GNF2-P10DG2B411-14GZ-100T2-150-T6-4317
	Once Burned Fuel
64	GE14-P10DNAB393-14GZ-100T-150-T6-4182
64	GE14-P10DNAB406-18GZ-100T-150-T6-4183
24	GE14-P10DNAB418-16GZ-100T-150-T6-4184
72	GE14-P10DNAB423-15GZ-100T-150-T6-4185
	Twice Burned Fuel
48	GE14-P10DNAB419-16GZ-100T-150-T6-3392
28	GE14-P10DNAB395-14GZ-100T-150-T6-3391
23	GE14-P10DNAB402-15GZ-100T-150-T6-3389
4	GE14-P10DNAB423-15GZ-100T-150-T6-2876
4	GNF2-P10DG2B401-14GZ-100T2-150-T6-3394
1	GE14-P10DNAB423-15GZ-100T-150-T6-2876
	Thrice Burned Fuel
4	GE14-P10DNAB398-4G7.0/11G6.0/1G2.0-100T-150-T6-2620

3.0 SUMMARY OF POST-REFUELING OUTAGE STARTUP TEST RESULTS

3.1 Core Verification

3.1.1 Purpose

To verify all fuel assemblies have been properly loaded into the reactor core as per the licensed final loading pattern, including fuel bundle location, orientation, and seating.

3.1.2 Acceptance Criteria

Level 1 criteria: Each fuel assembly must be verified to be in its proper location and orientation as specified by the final loading pattern (Licensed Core) and be correctly seated in its respective cell.

Level 2 criteria: N/A

3.1.3 Test Description

The Hatch Unit 2 Cycle 24 core verification was performed by use of underwater TV cameras to visually inspect the location (by bundle serial number identification), orientation, and seating of each of the 560 fuel assemblies that comprise the as-loaded core.

3.1.4 Test Results

Core verification was performed on February 28, 2015, in accordance with engineering procedures for fuel movement. The visual inspection confirmed all bundles were in their correct location and orientation, and no bundles required reseating.

3.2 Control Rod Drive (CRD) Timing

3.2.1 Purpose

To demonstrate the CRD system operates properly following the completion of a core alteration. In particular, this functional test verifies that the insert and withdrawal capability of the CRD system is within acceptable limits.

3.2.2 Acceptance Criteria

Level 1 Criteria: The insert and withdrawal drive time for each CRD must be between 43.2 and 52.8 seconds. In the event that a CRD fails to meet these criteria, the applicable drive must be adjusted and new criteria of 45.4 to 50.2 seconds are applied to the adjusted drive.

Level 2 Criteria: N/A

3.2.3 Test Description

Control rod drive timing is performed once per operating cycle on all CRDs. Normal withdrawal and insertion times are recorded for each of the drives under normal drive water pressure. If acceptable withdrawal and/or insertion cannot be obtained with normal drive water pressure, then the respective needle valve for the applicable withdrawal and/or insertion stroke must be adjusted until an acceptable drive time is achieved in accordance with the above criteria.

3.2.4 Test Results

Control rod drive timing was completed on March 8, 2015 for all 137 CRDs in accordance with plant operating procedures for CRD timing. Each CRD was determined to have, or was adjusted (where necessary) to have, a normal insertion and withdrawal speed as required.

3.3 In-Sequence Critical Shutdown Margin Demonstration

3.3.1 Purpose

To demonstrate the reactor can be made subcritical for any reactivity condition during Cycle 24 operation with the analytically determined highest worth control rod capable of withdrawal, fully withdrawn and all other rods fully inserted.

3.3.2 Acceptance Criteria

Level 1 Criteria: The loaded core must be subcritical by at least 0.38% Δ K with the analytically determined highest worth control rod capable of being withdrawn, fully withdrawn, and all other rods fully inserted at the most reactive condition during the cycle.

Level 2 Criteria: N/A

3.3.3 Test Description

The in-sequence critical shutdown margin demonstration was performed immediately following the Plant Hatch Unit 2 Cycle 24 BOC initial criticality with the reactor core in a xenon free state. To account for reactivity effects such as moderator temperature, reactor period, and the one rod out criterion, correction factors were used to adjust the startup condition to cold conditions with the highest worth control rod fully withdrawn.

3.3.4 Test Results

The in-sequence critical shutdown margin demonstration was performed on March 10, 2015 in accordance with core calculation procedures for shutdown margin demonstration. Results of this calculation yielded a cold shutdown margin of 1.861% Δ K. The minimum cold shutdown margin was also 1.861% Δ K because cold shutdown margin this operating cycle is a minimum at BOC. A summary of the shutdown margin demonstration is given in Attachment 1 of this report.

3.4 Cold Critical Eigenvalue Comparison

3.4.1 Purpose

To compare the critical eigenvalue calculated using the actual cold, xenon free critical control rod configuration (corrected for moderator temperature and reactor period reactivity effects) to the cold critical eigenvalue assumed in the cycle management analysis.

3.4.2 Acceptance Criteria

Level 1 Criteria: The cold critical eigenvalue calculated using actual critical data shall not differ from the design cold critical eigenvalue by more than ±1% ΔK

Level 2 Criteria: N/A

3.4.3 Test Description

The cold critical eigenvalue is the assumed value of the PANACEA 3-D core simulator model K_{eff} at which criticality is achieved with the reactor in a xenon free state and the coolant at 68 degrees F. This value is determined based on historical data and used for cycle management analysis by core analysis personnel. Once the actual critical state is achieved during the beginning of cycle startup, the applicable data are provided to core analysis personnel, and the actual (corrected for moderator temperature and reactor period reactivity effects) cold critical eigenvalue is calculated. This value is then compared to the assumed critical eigenvalue as a method of validating rod worths and shutdown margin calculations throughout the cycle. The actual critical eigenvalue is also entered into a database for predicting future cold critical eigenvalues.

3.4.4 Test Results

The beginning of cycle startup for Plant Hatch Unit 2 Cycle 24 was performed on March 10, 2015. The observed reactor core conditions when a critical state was achieved are listed in Attachment 1.

The results of the PANACEA case show the temperature and period-corrected cold eigenvalue to be 1.0022. This is 0.22% ΔK above the design value of 1.0000 and is well within the $\pm$ 1.0% ΔK acceptance criteria. This also compares favorably to the value of 1.0000 which was actually used in the core design as an extra margin of conservatism due to the introduction of the GNF-2 fuel type.

3.5 Local Power Range Monitor (LPRM) Calibration

3.5.1 Purpose

To calibrate the local power range monitors (LPRMs) by fine-tuning gain adjustment factors (GAFs) such that LPRM readings are equivalent to Traversing Incore Probe (TIP) detector readings. The TIP measurements, in turn, are proportional to the axial flux distribution at selected intervals over the regions of the core where the LPRMs are located. TIP readings are of high precision to allow reliable calibration of LPRM gains.

3.5.2 Acceptance Criteria

Level 1 Criteria: All detector GAFs ≤ 40.

Level 2 Criteria: N/A

3.5.3 Test Description

The LPRM channels were calibrated to make the LPRM readings proportional to the neutron flux in the narrow-narrow water gap at the chamber elevation. This calibration was performed in accordance with engineering procedures for LPRM calibration.

3.5.4 Test Results

Using site procedures, LPRMs were successfully calibrated at 100% power. LPRM Gain Adjustment Factor Values for all operable LPRM channels were within specified limits.

3.6 APRM Calibration

3.6.1 Purpose

To calibrate the APRM system to actual core thermal power, as determined by a heat balance.

3.6.2 Acceptance Criteria

Level 1 criteria: The APRM readings must be within a tolerance of 2% of core thermal power as determined from a heat balance.

Level 2 criteria: N/A

3.6.3 Test Description

The APRM gains are adjusted after major power level changes, if required, to read the actual core thermal power as determined by a heat balance performed in accordance with plant operating procedures for APRM adjustment to core thermal power. The heat balance required for the calibration process was obtained from the process computer program OD3 (Core Thermal Power and APRM Calibration), or from the Official Monitor case in accordance with plant operating procedures.

3.6.4 Test Results

APRM calibration was performed in accordance with plant operating procedures at approximately 23%, 36%, 58%, 71%, 85%, 95%, and 100% of Rated Thermal Power. Each APRM was calibrated within a 2% tolerance to read core thermal power as calculated by the heat balance.

3.7 Control Rod Scram Time Testing

3.7.1 Purpose

To demonstrate that the CRD system functions as designed with respect to scram insertion times following the completion of core alterations.

3.7.2 Acceptance Criteria

Level 1 criteria:

(a) The individual scram insertion time for all operable control rods from the fully withdrawn position, based on de-energization of the scram pilot solenoids, with reactor steam dome pressure above 800 psig shall not exceed the following:

From Fully Withdrawn To Notch Position	Individual Rod Maximum Insertion Time (sec)
46	0.44
36	1.08
26	1.83
06	3.35

(b) The individual control rods with scram times in excess of those listed in (a) above are to be declared as SLOW with the following restrictions:

1. No more than 10 operable control rods are declared SLOW.

2. No more than 2 operable control rods that are declared SLOW occupy adjacent locations.

(c) The maximum scram insertion time of each control rod, from the fully withdrawn position to position 06, based on the de-energization of the scram pilot solenoid, shall not exceed 7.0 seconds.

Level 2 criteria: N/A

3.7.3 Test Description

The CRD scram time testing was performed in accordance with engineering procedures for control rod scram testing, with the steam dome pressure above 800 psig. The test consists of scramming each control rod, collecting the resulting scram time data, and analyzing the data in accordance with the acceptance criteria noted above.

3.7.4 Test Results

All CRDs were tested in accordance with engineering procedures for control rod scram testing, with the steam dome pressure greater than 800 psig. A summary of the results is given in Attachment 2 of this report.

3.8 Core Performance

3.8.1 Purpose

To evaluate core performance parameters to assure plant thermal limits are maintained during power ascension to rated conditions.

3.8.2 Acceptance Criteria

Level 1 criteria: The following thermal limits are ≤ 1.000 when ≥ 24% RTP:

1. MFLCPR (Maximum Fraction of Limiting Critical Power Ratio)
2. MFLPD (Maximum Fraction of Limiting Power Density)
3. MAPRAT (Maximum Average Planar Linear Heat Generation Ratio).

Level 2 criteria: N/A

3.8.3 Test Description

As power is increased, core thermal limits were evaluated at various levels up to 100%. In accordance with plant operating procedures for core parameter surveillance, demonstration of fuel thermal margin was performed. Fuel thermal margin was confirmed at each level before increasing reactor power further.

3.8.4 Test Results

Thermal limits were continuously monitored during power ascension. The surveillance procedure was performed satisfactorily at various levels as indicated below:

Thermal Limit	24%	36%	58%	70%	85%	96%	100%
MFLCPR	0.760	0.730	0.820	0.913	0.861	0.864	0.886
MFLPD	0.642	0.547	0.798	0.790	0.856	0.847	0.832
MAPRAT	0.358	0.359	0.615	0.615	0.630	0.666	0.699

3.9 Reactivity Anomaly Calculation

3.9.1 Purpose

To check for possible reactivity anomalies as the core excess reactivity changes with exposure.

3.9.2 Acceptance Criteria

Level 1 Criteria: The monitored core k_{eff} shall not differ from the predicted core k_{eff} by more than ±1% Δ K.

Level 2 criteria: N/A

3.9.3 Test Description

After obtaining steady-state conditions following a BOC startup from a refueling outage and every month thereafter, a reactivity anomaly calculation is performed to monitor the core reactivity during the cycle. Verifying the reactivity difference between the monitored and predicted core k_{eff} is within limits provides assurance that plant operation is maintained within the assumptions of the DBA and transient analyses. The core monitoring system calculates the core k_{eff} for the reactor conditions obtained from plant instrumentation. A comparison of the monitored core k_{eff} to the predicted core k_{eff} at the same cycle exposure is used to ensure the difference is within a $\pm$ 1% Δ K acceptance band.

3.9.4 Test Results

The initial reactivity anomaly calculation for the cycle was performed in accordance with the engineering procedures for reactivity anomaly calculations on March 19, 2015. The monitored core k_{eff} was well within the acceptance criteria range as specified above. The results of this calculation are given in Attachment 3 of this report.

4.0 CONCLUSIONS

As indicated by the acceptable results of all the startup testing, operation of the Plant E. I. Hatch Unit 2 reactor is successful with the introduction of the GNF-2 fuel.

ATTACHMENT 1

IN-SEQUENCE CRITICAL COLD SHUTDOWN MARGIN DEMONSTRATION

Sequence	A2
RWM Group 1	Fully Withdrawn
RWM Group 2	15 control rods fully withdrawn
K_{SRO}	0.98063
K_{CRIT}	1.00455
Control Rod Density	0.7664
Reactor Coolant Temperature	125.0° F
Reactivity Correction for Temperature	-0.002 ΔK
Reactor Period	215 sec.
Reactivity Correction for Period	0.00031 ΔK
Corrected K_{CRIT}	0.99924 ΔK
Cold Shutdown Margin	1.861% ΔK
Value of R	0.0% ΔK
Value of B (conservative bias)	0.0030 ΔK
Minimum Cold Shutdown Margin	1.861% ΔK
Tech Spec Required Shutdown Margin	0.38% ΔK

ATTACHMENT 2

SCRAM TIME TESTING

LOCATIONS	TIME IN SECONDS TO NOTCH POSITION			
	<u>**46**</u>	<u>**36**</u>	<u>**26**</u>	<u>**06**</u>
Slowest Rods				
30-43	**0.358**	0.906	1.447	2.513
30-43	0.358	**0.906**	1.447	2.513
30-43	0.358	0.906	**1.447**	2.513
26-31	0.280	0.835	1.396	**2.591**
Fastest Rods				
06-19	**0.228**	0.700	1.198	2.257
42-27	0.249	**0.698**	1.159	2.083
42-27	0.249	0.698	**1.159**	2.083
42-27	0.249	0.698	1.159	**2.083**
Average (All Rods)	**0.256**	**0.760**	**1.276**	**2.328**

ATTACHMENT 3

REACTIVITY ANOMALY CALCULATION

UNIT 2 CYCLE 24
SEQUENCE: A2

DATE PERFORMED		03/19/2015
THERMAL POWER (MW_{th})	CMWT	2768.4
RATED THERMAL POWER (MW_{th})		2804.0
CORE FLOW (Mlb/hr)	WT	78.93
RATED CORE FLOW (Mlb/hr)		77.00
CORE XENON CONCENTRATION		-2.20
XE/RATED		0.987
CYCLE EXPOSURE (MWD/sT)		82.1
EIGENVALUE	k_{eff}	1.0056

PREDICTED k_{eff} = 1.0060

+1% VALUE = 1.0160 -1% VALUE = 0.9960