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

Edwin I. Hatch Nuclear Plant - Unit 1
Startup Test Report

Ladies and Gentlemen:

In accordance with regulatory commitments, Southern Nuclear Operating Company (SNC) hereby submits the Unit 1 Startup Test Report for Cycle 21. This report summarizes the startup testing performed on Unit 1 following the seventeenth refueling outage. The report is required due to the first use, other than as lead use assemblies, of GE14 fuel assemblies loaded for Cycle 21.

The tests demonstrate the successful operation of the Plant E. I. Hatch Unit 1 reactor with the introduction of the GE14 fuel.

Should you have any questions in this regard, please contact this office.

Respectfully submitted,

Lewis Sumner

H. L. Sumner, Jr.

OCV/eb

Enclosure: Edwin I. Hatch Nuclear Plant - Unit 1
Startup Test Report for Cycle 21

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Enclosure

Edwin I. Hatch Nuclear Plant - Unit 1
Startup Test Report for Cycle 21

1.0 INTRODUCTION

1.1 Purpose and Summary

The Plant Edwin I. Hatch Unit 1 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 1 following the twentieth refueling outage. This report is being submitted due to a reload batch of 224 GE14 fuel assemblies that were loaded for Cycle 21. The GE14 fuel design has not previously been utilized in Unit 1. Four GE14 Lead Use Assemblies (LUAs) are currently in their third cycle on Unit Two.

This report consists of a summary of selected static and dynamic reactor core performance tests conducted prior to and during the beginning-of-cycle startup of Plant Hatch Unit 1 Cycle 21. These tests demonstrate the successful operation of the Unit 1 reactor with the introduction of the GE14 fuel design into production use.

1.2 Plant Description

The Edwin I. Hatch Nuclear Power Plant Unit 1 is a General Electric design single-cycle boiling water reactor (BWR/4). Plant Hatch Unit 1 is rated at 2763 MW(th) with a generator rating at this power of 900 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 1 resumed commercial operation on April 23, 2002, after completing a 32 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
- Full Core Shutdown Margin Demonstration
- 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 CYCLE DESIGN SUMMARY

2.1 Core Design Summary

The Unit 1 Cycle 21 core is designed to operate approximately 613 effective full power days (EFPDs) at rated thermal power conditions, which includes extension from increased core flow. The fuel is arranged in a conventional core loading designed to achieve 15780 MWD/sT incremental energy exposure. There are 152 fresh GE14 assemblies at 3.98 w/% enrichment and 72 fresh GE14 assemblies at 3.99 w/% enrichment, with the remaining reload bundles having an enrichment of 3.78 w/%. The loading pattern in this cycle is octant symmetric.

2.2 Reactivity/Thermal Limit Margins

The two parameters that describe the global behavior of the core throughout the cycle are hot excess reactivity (HER) and cold shutdown margin (CSDM).

The beginning of cycle (BOC) + 200.0 MWD/sT HER is 1.54%, and the peak HER is 2.58% at 9000.0 MWD/sT.

The expected minimum CSDM of 1.85% ΔK occurs at 0.0 MWD/sT for the as-burned, as-loaded core. In-sequence critical calculations do not identify any high notch or control rod worths around the expected critical rod pattern at BOC. The Hatch-1 Cycle 21 calculated core parameters are delineated in Table 2.1.

Target rod pattern recommendations are calculated in 0.5 GWD/sT exposure increments. Thermal margin design goals of 10%, 10%, and 7% for MFLPD, MAPRAT, and MFLCPR, respectively, are met throughout the cycle for these rod patterns. However, some MAPRAT problems are expected late in the cycle when notching rods between 12 and 20. This issue can be addressed through additional load reductions for pattern adjustments or by purchasing relaxed LHGR limits for the residual 9x9 fuel from the vendor.

2.3 Fuel Summary

All fuel assemblies loaded in Cycle 21 have barrier cladding. A set of "soft-startup" preconditioning guidelines have been established and are applied to selected fuel assemblies during the first sequence of cycle operation. These fuel assemblies are chosen for preconditioning because they have been moved from lower power regions of the core in the previous cycle to higher power regions this cycle.

Table 2.2 provides a list of all batches loaded in Plant Hatch Unit 1 Cycle 21. Note that the fresh fuel contains axially varying fuel types.

Table 2.1

Cycle 21 Calculated Parameters

BOC Core Average Exposure	12566.0 MWD/sT
Cycle 21 Core Weight	107.918 sT
Cycle Energy (Rated Power)	15780.0 MWD/sT
	(613.0 EFPDs)
Uncertainty in Energy	+320.0 MWD/sT
	-335.0 MWD/sT
Cold Shutdown Margin	
BOC	1.847 %
R	0.000 %
Hot Excess Reactivity	
200 MWD/sT	1.54%
9000 MWD/sT	2.58%

Table 2.2

Fuel Batches Loaded in Hatch-1 Cycle 21

Batch	IAT Type	Bund Qty	ID Range	Bundle Nomenclature
H1R18	11	26	YJS163-YJS182 YJS187-YJS192	GE13-P9HTB378-6G5.0/6G4.0/1G2.0-100T-146-T
H1R18	12	78	YJS029-YJS032 YJS037-YJS040 YJS057-YJS060 YJS073-YJS076 YJS081-YJS096 YJS105-YJS116 YJS125-YJS136 YJS141-YJS144 YJS149-YJS158	GE13-P9HTB378-6G5.0/6G4.0-100T-146-T
H1R19	13	136	YJW489-YJW624	GE13-P9DTB378-6G5.0/6G4.0-100T-146-T-2411
H1R19	14	48	YJW441-YJW488	GE13-P9DTB378-6G5.0/6G4.0/1G2.0-100T-146-T
H2R20	15	152	JLC353-JLC504	GE14-P10DNAB398-15GZ-100T-150-T-2518
H1R20	16	72	JLC281-JLC352	GE14-P10DNAB399-16GZ-100T-150-T-2517
H1R18	17	6	YJS161-YJS162 YJS183-YJS186	GE13-P9HTB378-6G5.0/6G4.0/1G2.0-100T-146-T
H1R18	18	42	YJS001-YJS016 YJS049-YJS052 YJS097-YJS104 YJS117-YJS124 YJS137-YJS140 YJS159-YJS160	GE13-P9HTB378-6G5.0/6G4.0-100T-146-T

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 as specified by the General Electric 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 1 Cycle 21 core verification was performed by use of an underwater TV camera 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

A full core verification was completed on April 11, 2002, in accordance with engineering procedures for fuel movement. The verification showed all bundles were in their correct locations and in the correct orientation. Only one peripherally-located bundle required reseating during the core-verification.

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: N/A

Level 2 Criteria: The insert and withdrawal drive time for each CRD must be between 38.4 and 57.6 seconds. In the event that a CRD fails to meet this criteria, the applicable drive must be adjusted and a new criteria of 43.2 to 52.8 seconds is applied to the adjusted drive.

3.2.3 Test Description

Control rod drive timing is generally 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 April 17, 2002, 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, with the following exceptions:

(1) Thirty-four control rods could not initially be moved from the full-in position using normal drive water pressure. By procedure, drive water pressure was increased in discrete steps until control rod movement was successful. Once rod movement was attained, these rods were successfully timed at normal drive water pressure.

(2) One control rod, 34-27, was found to have an excessively fast withdraw speed of 38.6 seconds when timed from notch position 00 to position 48 after all allowable speed adjustments were completed. This condition, although deficient, was analyzed to be within the acceptable limits of the Rod Withdrawal Error transient analysis (>28.8 seconds).

Note: These CRD mechanisms have been documented via the Corrective Action Program and are currently being trended and evaluated for repair and/or replacement.

3.3 Full Core Shutdown Margin Demonstration

3.3.1 Purpose

To demonstrate the reactor can be made subcritical for any reactivity condition during Cycle 21 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 full core shutdown margin demonstration was performed analytically during the Plant Hatch Unit 1 Cycle 21 BOC in-sequence critical 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 are used to adjust the startup condition to cold conditions with the highest worth control rod fully withdrawn.

3.3.4 Test Results

The full core shutdown margin demonstration was performed on April 18, 2002, in accordance with core calculation procedures for shutdown margin demonstration. Results of this calculation yielded a cold shutdown margin of 1.249% ΔK. The minimum cold shutdown margin was also 1.249% ΔK, because cold shutdown margin for Cycle 21 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: N/A

Level 2 Criteria: N/A

3.4.3 Test Description

The cold critical eigenvalue is the assumed value of the PANACEA 3-D simulator model K_{eff} at which criticality is achieved with the reactor in a xenon-free state and the coolant is 68 degrees Farenheight. 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 is 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 initial beginning of cycle startup for Plant Hatch Unit 1 Cycle 21 was performed on April 18, 2002. The observed reactor core conditions when a critical state was achieved are listed in Attachment 1.

A cold critical eigenvalue of 1.00128 was calculated from the actual critical data given above. This compares to the initial estimate for the cold critical eigenvalue of 1.0040. The actual cold critical indicates that the design value is non-conservative by as much as 0.27% Δk. Given that this was the first reload of GE14 (10x10) fuel and the reload size was 224 bundles (up from a 184) the eigenvalue is considered to be acceptable. In addition, because of the changes in this cycle, extra shutdown margin (SDM) was built in to the design. Collectively, the added SDM and the non-conservatism of the design eigenvalue result in an actual SDM of 1.28% as compared to the licensing requirement (Tech Spec) of 0.38% $\Delta k/k$.

3.5 Local Power Range Monitor (LPRM) Calibration

3.5.1 Purpose

To calibrate the local power range monitors (LPRMs).

3.5.2 Acceptance Criteria

Level 1 Criteria: Per plant procedures.

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. Average 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 will be obtained from the process computer programs OD3 (Core Thermal Power/Flow Log) or CTP (Core Thermal Power), or from the Official Monitor case, or from a manual heat balance in accordance with plant operating procedures.

3.6.4 Test Results

APRM calibration was performed in accordance with plant operating procedures at approximately 7%, 20%, 45%, 60%, 69% and 98%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 greater than or equal to 800 psig shall not exceed the following:

Notch Position from Fully Withdrawn	Average 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.

3. No more than 20% of the control rods tested are determined to be SLOW.

(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. Scram times for all control rods were acceptable. A summary of the results is given in Attachment 2 of this report.

3.8 Core Performance

3.8.1 Purpose

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

3.8.2 Acceptance Criteria

Level 1 criteria: The following thermal limits are ≤ 1.000:

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 was increased, core thermal limits were evaluated at various levels up to 100% rated thermal power. 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 regularly monitored during power ascension. The surveillance procedure was performed satisfactorily at various levels as indicated below:

Power Level	Thermal Limit		
	MFLCPR	MFLPD	MAPRAT
22.4%	0.586	0.281	0.468
40.1%	0.618	0.463	0.669
48.6%	0.622	0.505	0.701
72.0%	0.887	0.671	0.796
84.9%	0.867	0.773	0.833
92.7%	0.869	0.817	0.842
96.5%	0.855	0.849	0.869
99.7%	0.920	0.877	0.843
99.9%	0.907	0.915	0.871

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 corrected control rod density shall not differ from the predicted control rod density equivalent by more than $\pm$ 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. Since anticipated operation or unanticipated events may place the reactor in a condition other than that for which the baseline anomaly curve was developed, the actual control rod density is corrected for off-rated conditions. The corrected control rod density is then compared to the reactivity anomaly curve provided in the Cycle Management Report to ensure that the corrected control rod density is within a $\pm$ 1% Δ K acceptance band about the curve.

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 April 26, 2002. The corrected control rod density 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 the results of the startup testing indicate, operation of the Plant E. I. Hatch Unit 1 reactor is successful with the introduction of the GE14 fuel.

ATTACHMENT 1

FULL CORE SHUTDOWN
MARGIN DEMONSTRATION

Sequence	A2
RWM Group 1	Fully Withdrawn
RWM Group 2	11 control rods withdrawn full out (notch 48), 5 control rods withdrawn to notch 24, and the 12th control rod in the group (38-19) withdrawn to notch 32
K_{SRO}	0.98153
K_{CRIT}	1.00035
Control Rod Density	0.7725
Reactor Coolant Temperature	144° F
Reactivity Correction for Temperature	-0.0029 ΔK
Reactor Period	155.1 sec.
Reactivity Correction for Period	0.00043 ΔK
Cold Shutdown Margin	1.249% ΔK
Value of R	0.0% ΔK
Value of B (conservative bias)	0.003 ΔK
Minimum Cold Shutdown Margin	1.249% ΔK
Tech Spec Required Shutdown Margin	0.38% ΔK

ATTACHMENT 2

SCRAM TIME TESTING

LOCATIONS	TIME IN SECONDS TO NOTCH POSITION			
	46	**36**	**26**	**06**
Slowest Rods with slowest notch identified in bold				
22-11	**0.372**	0.902	1.427	2.559
42-27	0.348	**0.957**	**1.594**	**2.855**
Fastest Rods with fastest notch identified in bold				
22-51	**0.258**	0.781	1.322	2.421
10-43	0.271	**0.751**	**1.254**	**2.312**
No control rods met the criteria to be declared "SLOW"				
Average (All Rods)	**0.286**	**0.827**	**1.385**	**2.544**

ATTACHMENT 3

REACTIVITY ANOMALY CALCULATION

UNIT 1 CYCLE 21
SEQUENCE: A2

DATE PERFORMED		04/26/02
THERMAL POWER (MW_{th})	CMWT	2758.2
RATED THERMAL POWER (MW_{th})		2763.0
CORE FLOW (Mlb/hr)	WT	74.45
RATED CORE FLOW (Mlb/hr)		78.50
DOME PRESSURE (psia)	PR	1049.3
RATED PRESSURE (psia)		1050.0
SUBCOOLING (BTU/lb)	DHS	26.44
DESIGN INLET SUBCOOLING (BTU/lb)		24.53
CYCLE EXPOSURE (MWD/sT)		76.15
CONTROL ROD DENSITY	CRD	0.0736

CORRECTED CRD = CRD + CORRECTION

CORRECTION = -2.2316E-1 x (1.0-(CMWT/RATED CORE THERMAL POWER))
+1.4036E-1 x (1.0-(WT/RATED CORE FLOW))
+1.68026E-3 x (DESIGN INLET SUBCOOLING-DHS)
+3.4708E-5 x (RATED PRESSURE - PR)

CORRECTION = 0.0036

CORRECTED CRD = 0.0736 + 0.0036 = 0.0772

PREDICTED CRD = 0.0749

+1% VALUE = 0.1180 -1% VALUE = 0.0319