D. R. Madison (Dennis)
Vice President - Hatch

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SOUTHERN COMPANY
Energy to Serve Your World℠

August 25, 2008

Docket No.: 50-321

NL-08-1279

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

Edwin I. Hatch Nuclear Plant – Unit 1
Licensee Event Report
Sensed Low EHC Pressure Causes Turbine Trip Resulting in a Reactor Scram

Ladies and Gentlemen:

In accordance with the requirements of 10 CFR 50.73(a)(2)(iv)(A), Southern Nuclear Operating Company is submitting the enclosed Licensee Event Report (LER) concerning an event that resulted in an automatic reactor scram

This letter contains no NRC commitments. If you have any questions, please advise.

Sincerely,

D. R. Madison
Vice President – Hatch

DRM/MJK/daj

Enclosure: LER 1-2008-003

cc: Southern Nuclear Operating Company
Mr. J. T. Gasser, Executive Vice President
Mr. D. R. Madison, Vice President – Hatch
Mr. D. H. Jones, Vice President – Engineering
RTYPE: CHA02.004

U. S. Nuclear Regulatory Commission
Mr. L. A. Reyes, Regional Administrator
Mr. R. E. Martin, NRR Project Manager – Hatch
Mr. J. A. Hickey, Senior Resident Inspector – Hatch

NRC FORM 366 (9-2007) U.S. NUCLEAR REGULATORY COMMISSION

APPROVED BY OMB: NO. 3150-0104 EXPIRES: 08/31/2010

Estimated burden per response to comply with this mandatory collection request: 50 hours. Reported lessons learned are incorporated into the licensing process and fed back to industry. Send comments regarding burden estimate to the Records and FOIA/Privacy Service Branch (T-5 F52), U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, or by internet e-mail to infocollects@nrc.gov, and to the Desk Officer, Office of Information and Regulatory Affairs, NEOB-10202, (3150-0104), Office of Management and Budget, Washington, DC 20503. If a means used to impose an information collection does not display a currently valid OMB control number, the NRC may not conduct or sponsor, and a person is not required to respond to, the information collection.

LICENSEE EVENT REPORT (LER)

1. FACILITY NAME	2. DOCKET NUMBER	3. PAGE
Edwin I. Hatch Nuclear Plant Unit 1	05000 321	1 OF 3

4. TITLE

Sensed Low EHC Pressure Causes Turbine Trip Resulting in a Reactor Scram

5. EVENT DATE			6. LER NUMBER			7. REPORT DATE			8. OTHER FACILITIES INVOLVED	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REV NO.	MONTH	DAY	YEAR	FACILITY NAME	DOCKET NUMBER
										05000
07	04	2008	2008	003	0	08	25	2008	FACILITY NAME	DOCKET NUMBER 05000

9. OPERATING MODE

1

10. POWER LEVEL

99.7

11. THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR§: *(Check all that apply)*

☐ 20.2201(b)	☐ 20.2203(a)(3)(i)	☐ 50.73(a)(2)(i)(C)	☐ 50.73(a)(2)(vii)
☐ 20.2201(d)	☐ 20.2203(a)(3)(ii)	☐ 50.73(a)(2)(ii)(A)	☐ 50.73(a)(2)(viii)(A)
☐ 20.2203(a)(1)	☐ 20.2203(a)(4)	☐ 50.73(a)(2)(ii)(B)	☐ 50.73(a)(2)(viii)(B)
☐ 20.2203(a)(2)(i)	☐ 50.36(c)(1)(i)(A)	☐ 50.73(a)(2)(iii)	☐ 50.73(a)(2)(ix)(A)
☐ 20.2203(a)(2)(ii)	☐ 50.36(c)(1)(ii)(A)	☒ 50.73(a)(2)(iv)(A)	☐ 50.73(a)(2)(x)
☐ 20.2203(a)(2)(iii)	☐ 50.36(c)(2)	☐ 50.73(a)(2)(v)(A)	☐ 73.71(a)(4)
☐ 20.2203(a)(2)(iv)	☐ 50.46(a)(3)(ii)	☐ 50.73(a)(2)(v)(B)	☐ 73.71(a)(5)
☐ 20.2203(a)(2)(v)	☐ 50.73(a)(2)(i)(A)	☐ 50.73(a)(2)(v)(C)	☐ OTHER
☐ 20.2203(a)(2)(vi)	☐ 50.73(a)(2)(i)(B)	☐ 50.73(a)(2)(v)(D)	Specify in Abstract below or in NRC Form 366A

12. LICENSEE CONTACT FOR THIS LER

FACILITY NAME	TELEPHONE NUMBER (Include Area Code)
Edwin I. Hatch / Kathy Underwood, Performance Analysis Supervisor	912-537-5931

13. COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT

CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX	CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX

14. SUPPLEMENTAL REPORT EXPECTED

☐ YES *(If yes, complete 15. EXPECTED SUBMISSION DATE)* ☒ NO

15. EXPECTED SUBMISSION DATE	MONTH	DAY	YEAR

ABSTRACT (Limit to 1400 spaces, i.e., approximately 15 single-spaced typewritten lines)

On July 4, 2008 at 08:39 EDT, Unit 1 was in the Run mode at a power level of approximately 2797 CMWT, 99.7 percent rated thermal power. At that time testing of the 'Electro-hydraulic Control (EHC) pump auto start was in progress. During the testing, a Low EHC Pressure was sensed which resulted in a turbine trip. The reactor tripped on turbine control valve fast closure. Following the reactor scram, reactor pressure peaked at approximately 1120 psig, resulting in four of the eleven safety relief valves (SRV's) opening as designed to reduce pressure. Water level decreased due to void collapse resulting in the closure of the Group 2 primary containment isolation valves, and a Reactor Feed Pump Turbine speed increase, both per design. The feedwater level control system controlled reactor water level with a minimum water level of approximately 2.5 inches above instrument zero (about 160 inches above the top of active fuel). All control rods fully inserted. Pressure did not reach the nominal actuation set points for the remaining seven SRV's.

This event was caused by a sensed low pressure in the EHC system tripping the main turbine which results in an automatic reactor scram.

Performance of the weekly EHC pump auto start procedure has been suspended pending additional corrective actions. Installation of a time delay circuit, relocation of the pressure sensor tap to its previous location, along with additional corrective actions will be considered and tracked in the corrective action program.

NARRATIVE (If more space is required, use additional copies of NRC Form 366A)

PLANT AND SYSTEM IDENTIFICATION

General Electric - Boiling Water Reactor
Energy Industry Identification System codes appear in the text as (EIIS Code XX).

DESCRIPTION OF EVENT

On July 4, 2008 at 08:39 EDT, Unit 1 was in the Run mode at a power level of approximately 2797 CMWT, 99.7 percent rated thermal power. At that time testing of the 'Electro-hydraulic Control (EHC) pump auto start (EIIS Code TG) was in progress. During the testing, a Low EHC Pressure was sensed which resulted in a turbine trip. The reactor tripped on turbine control valve fast closure. Following the reactor scram, reactor pressure peaked at approximately 1120 psig, resulting in four of the eleven safety relief valves (SRV, EIIS Code SB) opening as designed to reduce pressure. Water level decreased due to void collapse from the rapid reactor pressure increase. The decrease in water level resulted in the closure of the Group 2 primary containment isolation valves, and a Reactor Feed Pump Turbine (RFPT) speed increase (EIIS Code SJ), both per design. As reactor pressure decreased water level increased to a maximum value of approximately 33 inches due to swell after both Recirculation Pumps (EIIS Code AD) tripped automatically on an End of Cycle-Recirculation Pump Trip, as per design. The feedwater level control system controlled reactor water level with a minimum water level of approximately 2.5 inches above instrument zero (about 160 inches above the top of active fuel). All control rods (EIIS Code AA) fully inserted. Pressure did not reach the nominal actuation set points for the remaining seven SRV's.

CAUSE OF EVENT

This event was caused by a sensed low pressure in the EHC system tripping the turbine which resulted in an automatic reactor scram. The turbine trip was the result of the combination of the following modifications. A new Mark VI turbine control system was installed in Spring 2006. During that modification the point at which the pressure sensors tap off of the EHC line was changed from the manifold where accumulators are attached to the smaller tubing which also feeds the auto start solenoid valve, this resulted in a larger sensed pressure drop during testing. The digital pressure transmitters which were installed with the original Mark VI modification were changed to analog transmitters during the Spring 2008 refueling outage. The analog transmitters have a significantly faster response time of 100 ms compared to the 225 ms response time of the digital transmitters.

REPORTABILITY ANALYSIS AND SAFETY ASSESSMENT

This report is required by 10 CFR 50.73 (a)(2)(iv)(A) because of the unplanned actuation of reportable system. Specifically, the reactor protection system actuated on turbine control valve fast closure when the main turbine tripped following the turbine control valve fast closure. Group 2 primary containment isolation valves closed and four of the SRV's opened on high vessel pressure. Fast closure of the turbine control valves initiates a reactor scram. The valves close as rapidly as possible to prevent over-speed of the turbine-generator rotor. Valve closing causes a sudden reduction in steam flow that, in turn, results in a reactor vessel pressure increase. If the pressure increases to the pressure relief setpoints, some or all of the SRV's will briefly discharge steam to the suppression pool (EIIS Code BL). All SRVs functioned properly

during the scram. Per design, SRVs A, C, G, H lifted since reactor pressure reached approximately 1120 psig. The Reactor scram initiation by turbine control valve fast closure prevents the core from exceeding thermal hydraulic safety limits following a main turbine trip. Closure of the valves results in the loss of the normal heat sink (main condenser, EIIS Code SG) thereby increasing reactor pressure, neutron flux, and heat flux transients that must be limited. A reactor scram is initiated on the valve fast closures in anticipation of these transients. The reactor trip ensures that the minimum critical power ratio safety limit is not exceeded. In this event, the main turbine tripped when low pressure in the EHC system was sensed. The turbine trip actuated the reactor protection system and resulted in a reactor scram. All required safety systems functioned as expected given the water level and pressure transients caused by the turbine, and reactor trips. Vessel water level was maintained well above the top of the active fue1 throughout the transient.

Based upon the preceding analysis, it is concluded this event had no adverse impact on nuclear safety. The analysis is applicable to all power levels.

CORRECTIVE ACTIONS

Performance of the weekly EHC pump auto start procedure has been suspended pending additional corrective actions.

Installation of a time delay circuit, relocation of the pressure sensor tap to its previous location, along with additional corrective actions will be considered and tracked in the corrective action program.

ADDITIONAL INFORMATION

Other Systems Affected: No systems other than those already mentioned in this report were affected by this event.

Failed Components Information: None

Commitment Information: This report does not create any permanent licensing commitments.

A previous similar event in the last two years in which the reactor scrammed automatically due to a main turbine trip was reported in the following Licensee Event Report:

2-2006-002, Unit Scram On Turbine Control Valve Fast Closure. A power-load unbalance was sensed resulting in a turbine control valve fast closure. The sensed power-load unbalance was a false indication introduced by performance of a calibration procedure. This event was the result of performing a surveillance procedure on-line that should not have been performed at that time. The corrective action for this event would not have prevented the current event which resulted from a plant modification.