H. L. Sumner, Jr.
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SOUTHERN COMPANY
Energy to Serve Your World℠

May 26, 2006

Docket No.: 50-366

NL-06-1089

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

Edwin I. Hatch Nuclear Plant – Unit 2
Licensee Event Report
Inadequate Procedure Results in Automatic Reactor Scram

Ladies and Gentlemen:

In accordance with the requirements of 10 CFR 50.73(a)(2)(iv)(A), Southern Nuclear is submitting the enclosed Licensee Event Report concerning an inadequate procedure which resulted in an automatic reactor scram on turbine control fast closure.

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

Sincerely,

H. L. Sumner, Jr.

HLS/OCV/daj

Enclosure: LER 2-2006-002

cc: Southern Nuclear Operating Company
Mr. J. T. Gasser, Executive Vice President
Mr. D. R. Madison, General Manager – Plant Hatch
RTYPE: CHA02.004

U. S. Nuclear Regulatory Commission
Dr. W. D. Travers, Regional Administrator
Mr. C. Gratton, NRR Project Manager – Hatch
Mr. D. S. Simpkins, Senior Resident Inspector – Hatch

IE22

NRC FORM 366 (6-2004) | U.S. NUCLEAR REGULATORY COMMISSION | APPROVED BY OMB: NO. 3150-0104 | EXPIRES: 06/30/2007

LICENSEE EVENT REPORT (LER)

(See reverse for required number of digits/characters for each block)

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.

1. FACILITY NAME	2. DOCKET NUMBER	3. PAGE
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4. TITLE
Unit Scram on Turbine Control Valve Fast Closure

5. EVENT DATE			6. LER NUMBER			7. REPORT DATE			8. OTHER FACILITIES INVOLVED	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	MONTH	DAY	YEAR	FACILITY NAME	DOCKET NUMBER(S)
										05000
04	05	2006	2006	002	0	5	26	2006	FACILITY NAME	DOCKET NUMBER(S)
										05000

9. OPERATING MODE
Mode 1

10. POWER LEVEL
100 %

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)	X 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 A. Underwood, Performance Analysis Supervisor	(912) 537-5931

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

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO EPIX		CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO EPIX

14. SUPPLEMENTAL REPORT EXPECTED

YES (If yes, complete 15. EXPECTED SUBMISSION DATE)	X	NO

15. EXPECTED SUBMISSION DATE

MONTH	DAY	YEAR

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

On 4/5/2006 at 00:16 EDT, Unit 2 was in the Run mode at a power level of 2804 CMWT (100 percent rated thermal power). At that time 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. Reactor pressure spiked to approximately 1125 psig, which resulted in an additional trip signal due to high reactor pressure. All control rods fully inserted. Eight of the eleven Safety Relief Valves opened as designed to mitigate the pressure transient. Pressure did not reach the nominal actuation set points for the remaining three Safety Relief Valves. Reactor water level initially decreased due to void collapse and then increased to a maximum value of approximately 60 inches due to swell after both Recirculation Pumps tripped automatically on End of Cycle-Recirculation Pump Trip per design and eight SRVs opened. Reactor water level subsequently decreased when the reactor feed pumps automatically tripped on high reactor water level per design. The minimum reactor water level decreased to approximately seven inches and was recovered with manual initiation of the Reactor Core Isolation Cooling and High Pressure Coolant Injection systems.

This event was caused by a power-load unbalance created during the calibration of the KVR Generator and Main Transformer Protection Recorder. The calibration procedure was found to be inadequate in that it allowed this calibration to be performed with the main turbine on-line and at greater than 40 percent rated thermal power. The calibration procedure was revised.

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

At 0016 EDT, on 4/5/06, a reactor scram occurred on Turbine Control Valve (TCV) (EIIS Code TA) fast closure. Reactor water level initially decreased due to void collapse and then increased to 60 inches. Reactor pressure spiked to approximately 1125 psig, which resulted in an additional trip signal due to high reactor pressure. All control rods fully inserted (EIIS Code JD). Eight of the eleven Safety Relief Valves (SRVs) (EIIS Code SB) opened as designed to mitigate the pressure transient. The remaining three SRVs did not open and should not have opened since the reactor pressure did not reach their lift set points. Reactor water level (RWL) increased to a maximum value of approximately 60 inches above instrument zero. This swell in RWL was due to a combination of the SRVs lifting and both recirculation pumps (EIIS Code AD) tripping automatically on End of Cycle-Recirculation Pump Trip (EOC-RPT) per design. Reactor water level subsequently decreased when the reactor feed pumps (RFP) (EIIS Code SJ) automatically tripped on high reactor water level per design. The 'A' RFP was restarted and subsequently tripped per design due to low vacuum in the main condenser (EIIS Code SG). It was once again restarted but subsequently tripped again on low condenser vacuum when the reactor feed pump turbine speed exceeded 3500 rpm. (The feed pump turbine low vacuum trip may be reset, and the feed pumps re-started with the low vacuum condition in effect, provided the turbine speed does not exceed 3500 rpm). As reactor water level (RWL) continued to decrease, Operations manually initiated the Reactor Core Isolation Cooling (RCIC) system (EIIS Code BN) which slowed the rate of RWL decrease. At this point, Operations also initiated a pressure reduction by reducing the turbine pressure set. The 'A' RFP was again re-started but could not inject due to the main condenser vacuum limiting injection pressure of the RFP to a value below the actual reactor pressure. Operations manually initiated the High Pressure Coolant Injection (HPCI) system (EIIS Code BJ) at a RWL of approximately seven inches above instrument zero. RWL increased and HPCI was secured at approximately 28 inches above instrument zero. RWL continued to increase due to the RCIC injection and the swell effect of decay heat heating the cooler water injected by HPCI and RCIC. As a result, the 'A' RFP subsequently tripped on high RWL, and was subsequently re-started. Operations continued to decrease reactor pressure until pressure reached approximately 475 psig. At that point, the 'A' RFP was secured and RWL was controlled by the condensate booster pump (EIIS Code SD).

Additional Details:

Main condenser vacuum decreased because the turbine steam sealing pressure was in manual control. The regulator for the Steam Seal Feed Valve Controller was replaced during the 2005 spring refueling outage. Later in 2005, additional problems were identified indicating problems with the steam seal feed valve. These problems were documented in the corrective action program. The Steam Seal System was placed in alternate seal steam pressure control per plant procedures. During normal operation manual control maintained main condenser vacuum. During the forced shut-down the valve was repaired.

In addition, the reactor coolant system (RCS) temperature in the bottom head region decreased by more than the 100 degrees Fahrenheit in one hour limit provided in the Technical Specifications. An evaluation of this condition was performed and the RCS and the reactor vessel were found acceptable for continued operation.

CAUSE OF EVENT

The direct cause of this event was a main turbine trip from a sensed power load unbalance (PLU) due to the performance of procedure 57CP-CAL-010-2, "Esterline Angus Megavar & KV Recorder," in Mode 1 (Run). The power-load unbalance caused the TCV fast closure. I&C was calibrating recorder 2S32-R017, KVR Generator & Main Transformer Protection Recorder, at the time of the trip. Review of plant and vendor drawings determined that shorting the CTs to ground during the performance of 57CP-CAL-010-2 was the cause of the turbine trip. Shorting the CTs to ground in 2H11-P658 caused the current to go to zero. The PLU module in the Mark I turbine control system looks at the current in each generator output phase and determines the electrical power based on the percent current. The PLU then compares the electrical power to the mechanical power and trips the turbine if a greater than 40 percent unbalance is detected. With the CTs grounded, the PLU percent current was zero and mechanical power was 100 percent. Therefore, the turbine tripped per design even though no actual power unbalance condition existed. This procedure was found to be inadequate in that it allowed this calibration to be performed with the main turbine on-line and at greater than 40 percent rated thermal power.

REPORTABILITY ANALYSIS AND SAFETY ASSESSMENT

This event is reportable per 10 CFR 50.73 (a)(2)(iv)(A) because of unplanned actuations of Engineered Safety Feature systems. First, the reactor protection system actuated on turbine control valve fast closure when the main turbine tripped following a sensed power load unbalance. Second, HPCI and RCIC were manually initiated since the RFP could not inject due to the main condenser vacuum limiting RFP operation to less than or equal to 3500 rpm.

Fast closure of the turbine control valves is initiated whenever a power load unbalance is sensed. The turbine control valves close as rapidly as possible to prevent overspeed 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 safety/relief valves will briefly discharge steam to the suppression pool (EIIS Code BL).

Reactor Pressure did not increase to the level of the Safety Relief Valve's mechanical lift setpoint of 1150 psig. However, SRVs B, D, F, G, A, C, K, and M lifted due to reactor pressure reaching their electrical lift setpoints. Reactor pressure did not reach the electrical setpoint for the other three SRVs, E, H, and L. The solenoid operated electrical lift feature for the SRVs is intended as a non-safety related, back-up system to the safety related mechanical lift system.

In this event, the reactor recirculation pumps tripped on End-of-Cycle Recirculation Pump Trip, initiated by the turbine control valve fast closure. Reactor scram and recirculation pump trip initiation by turbine control valve fast closure prevents the core from exceeding thermal hydraulic safety limits following a main generator or main turbine trip. A reactor scram is initiated on turbine control valve fast closure. The scram, along with the reactor recirculation pump trip system, ensures that the minimum critical power ratio safety limit is not exceeded.

The main turbine tripped as designed in response to the sensed power load unbalance. The turbine trip actuated the reactor protection system and scrammed the reactor. Vessel water level was maintained well above the top of the active fuel throughout the transient and never decreased to the reactor scram actuation setpoint. The water level decrease was terminated prior to reaching the automatic initiation set point for HPCI and RCIC when Operations personnel manually initiated these systems. Therefore, there were no automatic safety system actuations on low water level.

Typically, the bottom head region of the pressure vessel experiences rapid cooling following a scram coincident with a trip of the reactor recirculation pumps. This cooling is the result of the loss of effective water mixing due to the trip of the recirculation pumps and increased cold water flow from the control rod drive (EIIS Code AA) system following a scram. In this event, the temperature in the vessel bottom head region decreased by 129 degrees Fahrenheit in one hour. However, a bounding analysis indicated cool down up to 165°F in one hour will not place unacceptable stress on components of the reactor coolant system.

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

Maintenance procedure 57CP-CAL-010-2, "Esterline Angus Megavar and KV Recorder," has been revised to prevent performance while the reactor is operating.

2N33-F001, steam seal feed valve has been repaired.

ADDITIONAL INFORMATION

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

A review for impact on Unit 1 was performed. Procedure 57CP-CAL-010-1 was reviewed by system engineering and it was determined that due to the Mark VI turbine controls modification the same vulnerability to cause a turbine control valve fast closure and resulting reactor scram does not exist.

LICENSEE EVENT REPORT (LER)

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

This event was due to an inadequate procedure. There were no equipment failures that caused this event.

Failed Components Information: None

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

Previous Similar Events in the last two years in which the reactor scrammed automatically while critical were reported in the following Licensee Event Reports:

1-2005-002, dated 12-21-2005

Previous Similar Events in the last two years in which an ECCS system has actuated either automatically of manually:

1-2004-004, dated 5-5-2004