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SOUTHERN COMPANY

July 7, 2014

Docket Nos.: 50-321 NL-14-0996

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

Edwin I. Hatch Nuclear Plant
Licensee Event Report 2014-003-00
Safety Relief Valves As Found Settings Resulted in Not Meeting Tech Spec Surveillance Criteria

Ladies and Gentlemen:

In accordance with the requirements of 10 CFR 50.73(a)(2)(i)(B) Southern Nuclear Operating Company hereby submits the enclosed Licensee Event Report.

This letter contains no NRC commitments. If you have any questions, please contact Greg Johnson at (912) 537-5874.

Respectfully submitted,

D. R. Vineyard
Vice President – Hatch

DRV/mr

Enclosures: LER 2014-003-00

cc: Southern Nuclear Operating Company
Mr. S. E. Kuczynski, Chairman, President & CEO
Mr. D. G. Bost, Executive Vice President & Chief Nuclear Officer
Mr. D. R. Vineyard, Vice President – Hatch
Mr. B. L. Ivey, Vice President – Regulatory Affairs
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Mr. G. L. Johnson, Regulatory Affairs Manager - Hatch
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NRC FORM 366 (02-2014)

U.S. NUCLEAR REGULATORY COMMISSION

APPROVED BY OMB: NO. 3150-0104 EXPIRES: 01/31/2017

LICENSEE EVENT REPORT (LER)

(See Page 2 for required number of digits/characters for each block)

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4. TITLE

Safety Relief Valves As Found Settings Resulted in Not Meeting Tech Spec Surveillance Criteria

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
05	07	2014	2014	- 003 -	00	7	7	2014	FACILITY NAME	DOCKET NUMBER

9. OPERATING MODE	11. THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: *(Check all that apply)*			
Mode 1	☐ 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)
10. POWER LEVEL	☐ 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)
100	☐ 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

LICENSEE CONTACT	TELEPHONE NUMBER *(Include Area Code)*
Edwin I. Hatch / Steven Tipps – Licensing Supervisor	912-537-5880

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
B	SB	RV	T020	Y					

14. SUPPLEMENTAL REPORT EXPECTED	15. EXPECTED SUBMISSION DATE	MONTH	DAY	YEAR
☐ YES *(If yes, complete 15. EXPECTED SUBMISSION DATE)* ☒ NO				

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

On May 7, 2014, at approximately 0837, Unit 1 was at 99.9 percent rated thermal power (RTP) when the "as-found" testing results of the 2-stage main steam safety relief valves (SRVs) were received which indicated that five of eleven SRVs had experienced setpoint drift during the previous operating cycle which resulted in their failing to meet the Technical Specification (TS) opening setpoints of 1150 psig +/- 3 percent as required by TS surveillance requirement 3.4.3.1.

The root cause of the SRV setpoint drift is attributed to corrosion-induced bonding between the pilot disc and seating surfaces. This conclusion is based on previous root cause analyses and the repetitive nature of this condition at Hatch and within the BWR industry. The 2-stage SRVs with platinum coated pilot seats were removed from Unit 1 during the 2014 refueling outage and replaced with 3-stage SRVs with a modified pilot. 3-stage SRVs typically do not exhibit set point drift, additionally the modified pilot reduces instances of vibration induced spurious openings and leak-by.

A 3-stage SRV with a similar modified pilot was installed on Unit 2 during the 2013 outage. Current plans are to replace the remaining ten valves at Unit 2 with the same modified pilot valves during the next outage in 2015.

LICENSEE EVENT REPORT (LER) CONTINUATION SHEET

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NARRATIVE

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 May 7, 2014, at approximately 0837, Unit 1 was at 99.9 percent rated thermal power (RTP) when the report of the "as-found" testing results of the 2-stage main steam safety relief valves (SRVs) were received which indicated that five of eleven SRVs (EIIS Code SB) had experienced setpoint drift during the previous operating cycle which resulted in their allowable TS surveillance requirement (SR) 3.4.3.1 limits of 1150 +/- 34.5 psig (+/- 3 percent) being exceeded. The following is a tabulation of the test results of the eleven SRVs:

MPL Number	Pilot Serial Number	As-Found Lift Pressure (psig)	Percent Drift
1B21-F013A	1007	1175	2.17
1B21-F013B	1010	1172	1.91
1B21-F013C	1003	1223	6.35
1B21-F013D	311	1197	4.09
1B21-F013E	1226	1181	2.70
1B21-F013F	1011	1168	1.57
1B21-F013G	314	1181	2.70
1B21-F013H	312	1166	1.39
1B21-F013J	304	1204	4.70
1B21-F013K	301	1198	4.17
1B21-F013L	306	1201	4.43

All eleven valves were removed from service during the spring 2014 refueling outage and replaced with 3-stage SRVs. The 3-stage SRVs have a modified pilot that helps reduce the possibility of inadvertent lift and leak by.

The 2-stage SRVs installed on Unit 2 prior to the 2013 refueling outage utilized platinum coated pilot discs. Even though these valves are identical to the valves that were used in Unit 1 Cycle 26, there is confidence that the 2-stage SRVs currently installed on Unit 2 will function reliably for the remainder of the cycle. This is based on the recent Unit 2 operating experience from the previous outage (I0 of 11 SRVs met Tech Spec acceptance criteria), the fact that the SRVs were successfully tested prior to installation and that new platinum coated pilots were installed. Plans are to replace the existing 2-stage SRVs on Unit 2 with 3-stage SRVs during the next scheduled refueling outage as a long term corrective action.

CAUSE OF EVENT

The root cause of the SRV setpoint drift is attributed to corrosion-induced bonding between the pilot disc and its seating surface. This conclusion is based on previous root cause analyses and the repetitive nature of this condition at Plant Hatch and in the industry. In General Electric (GE) service information letter (SIL) 196, Supplement 16, GE determined that condensation of steam in the pilot chamber of Target Rock 2-stage SRVs can cause oxygen and hydrogen dissolved in the steam to accumulate. As steam condenses in the relatively stagnant pilot chamber, the dissolved gases are released. In a volume such as the pilot chamber which is normally at approximately a 1000 psig pressure and a temperature of 545 degrees F, the total pressure consists primarily of water vapor partial pressure because 544.6 degrees F is the saturation temperature at 1000 psig. This wet, hot, high-oxygen atmosphere can be very corrosive and can increase the likelihood of corrosion-induced bonding of the pilot disk to its seat. It was also noted that proper insulation minimizes the accumulation rate of non-condensable gases and the steady-state oxygen partial pressure. Despite improvements made in maintaining the integrity of insulation for the previously installed 2-stage SRVs the corrosion-induced bonding continued to occur as evidenced by the test results from this most recent outage.

REPORTABILITY ANALYSIS AND SAFETY ASSESSMENT

This event is reportable in accordance with Title 10 of the Code of Federal Regulations (CFR), Part 50.73(a)(2)(i)(B) because an event occurred which is prohibited by TS surveillance requirement (SR) 3.4.3.1. Specifically, an example of multiple test failures is given in NUREG-1022, Revision 3, "Event Reporting Guidelines 10 CFR 50.72 and 50.73" which describes the sequential testing of safety valves. This example notes that "Sometimes multiple valves are found to lift with set points outside of technical specification limits."

NUREG-1022 further states in the example that "discrepancies found in TS surveillance tests should be assumed to occur at the time of the test unless there is firm evidence, based on a review of relevant information (e.g., the equipment history and the cause of failure), to indicate that the discrepancy occurred earlier. However, the existence of similar discrepancies in multiple valves is an indication that the discrepancies may well have arisen over a period of time and the failure mode should be evaluated to make this determination." Based on this guidance and the fact that the development of the corrosion occurred over a period of time of plant operation, the determination was made that this "as found" condition is reportable under the reporting requirements of 10 CFR 50.73(a)(2)(i)(B).

There are eleven SRVs located on the four main steam lines within the drywell (EIIS Code NH) in between the reactor pressure vessel (EIIS Code AD) and the inboard main steam isolation valves (MSIV, EIIS Code SB). These SRVs are required to be operable during Modes 1, 2 and 3 to limit the peak pressure in the nuclear system such that it will not exceed the applicable ASME Boiler and Pressure Vessel Code Limits for the reactor coolant pressure boundary. The SRVs are tested in accordance with TS surveillance requirement 3.4.3.1 in which the valves are tested as directed by the In-Service Testing Program to verify lift set points are within their specified limits to confirm they would perform their required safety function of overpressure protection. The SRVs must accommodate the most severe pressurization transient which, for the purposes of demonstrating compliance with the ASME Code Limit of 1375 psig peak vessel pressure, has been defined by an event involving the closure of all MSIVs with a failure of the direct reactor protection system trip from the MSIV position switches with the reactor ultimately shutting down as the result of a high neutron flux trip (a scenario designated as MSIVF).

This MSIVF event analysis was performed for the Unit 1 Cycle 24 "as-found" condition of the SRVs showed that there was adequate margin to the vessel dome pressure and ASME vessel overpressure limits. Another analysis was performed which compared the Cycle 26 "as-found" SRV measured opening pressures to those of the Cycle 24 reload licensing analysis (RLA). The results from this analysis showed a decrease in peak pressures due to the fact that eight of the eleven SRVs opened at pressures lower than those which were measured in the Cycle 24 RLA.

Based on this comparison, recognizing that SRV opening pressures used in the Cycle 24 RLA were generally much larger than the Cycle 26 measured pressures, and since the Cycle 24 RLA showed sufficient margin to the Technical Specification 2.1.2 dome pressure safety limit and the ASME Pressure Vessel Code overpressure limit, the results of the Cycle 24 remain bounding. Therefore the peak pressure at the bottom of the vessel remained below the ASME

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Boiler and Pressure Vessel code limit, and the peak reactor pressure vessel dome pressure remained within the Tech Spec Safety Limits.

Additionally, a highly reliable, though non-credited, electrical actuation system serves as a redundant, independent method to actuate the SRVs. During Cycle 26 this redundant electrical logic system was fully functional.

Based on the analyses performed, the overpressure protection system would have continued to perform its required safety function if called upon in its "as found" condition. Therefore, this event had no adverse impact on nuclear safety and was of very low safety significance.

CORRECTIVE ACTIONS

The 2-stage SRVs with platinum-coated pilot discs were removed from Unit 1 during the 2014 refueling outage and replaced with 3-stage SRVs that have a modified pilot. 3-stage SRVs typically do not exhibit set point drift due to their design. The modified pilots will help reduce spurious openings and leak-by due to system vibration. Additionally, current plans are to replace the remaining 2-stage Unit 2 SRVs with 3-stage SRVs with the same modified pilots during the 2015 refueling outage.

ADDITIONAL INFORMATION

Other Systems Affected: None

Failed Components Information:

Master Parts List Number: 1B21-F013C, D, J, K, L
Manufacturer: Target Rock
Model Number: 7567F
Type: Relief Valve
Manufacturer Code: T020

EIIS System Code: SB
Reportable to EPIX: Yes
Root Cause Code: B
EIIS Component Code: RV

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

PREVIOUS SIMILAR EVENTS:

LER 1-2012-004, identified multiple SRV setpoint drift for 8 of the 11 SRVs. Corrective actions included replacement of the 2-stage SRVs with 2-stage SRVs whose pilot discs had undergone a platinum surface treatment which was considered at that time to be the long term fix for this corrosion bonding issue.

LER 2-2011-002, identified multiple SRV setpoint drift for 8 of the 11 SRVs. Corrective actions included replacement of the 2-stage SRVs with 3-stage SRVs during the Unit 2 Spring 2011 refueling outage which was considered at that time to be the long term fix for this corrosion bonding issue. Subsequent to that outage the 3-stage SRVs exhibited signs of unacceptable leakage which resulted in two separate outages that involved changing out four SRVs during the first outage and the remaining seven SRVs during the subsequent outage in May 2012. The 3-stage SRVs were replaced with 2-stage SRVs containing pilot discs that had undergone the platinum surface treatment.

LER 1-2010-001, identified multiple SRV setpoint drift for 5 of the 11 SRVs. Corrective actions included refurbishment of the pilot valves and included the replacement of the pilot discs with discs made from Stellite 21 material. Additionally, the insulation surrounding each SRV was upgraded to improve resistance to corrosion-induced bonding. These were the same actions that were taken following similar failures reported in LER 2-2009-001, since improved results had been seen to some degree in the industry for at least one operating cycle when these actions were implemented.