Mark J. Ajluni, P.E.
Nuclear Licensing Director

Southern Nuclear Operating Company, Inc.
40 Inverness Center Parkway
Post Office Box 1295
Birmingham, Alabama 35201

Tel 205.992.7673
Fax 205.992.7885

SOUTHERN COMPANY

May 25, 2012

Docket Nos.: 50-321

NL-12-1092

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

Edwin I. Hatch Nuclear Plant Unit 1
Licensee Event Report 2012-004-0
Corrosion-Induced Bonding Results in
Multiple Safety Relief Valves Setpoint Drift

Ladies and Gentlemen:

In accordance with the requirements of 10CFR50.73(a)(2)(i)(B), Southern Nuclear Operating Company (SNC) hereby submits the enclosed Licensee Event Report which addresses setpoint drift in excess of that allowed by Technical Specification SR 3.4.3.1, occurring in eight safety relief valves(SRVs) due to corrosion-induced bonding between the pilot disc and associated seating surfaces.

This letter contains no NRC commitments. If you have any questions, please contact Doug McKinney at (205) 992-5982.

Respectfully submitted,

Mark J. Ajluni

M. J. Ajluni
Nuclear Licensing Director

MJA/SBT/lac

Enclosure: LER 1-2012-004, Revision 0

cc: <u>Southern Nuclear Operating Company</u>
Mr. S. E. Kuczynski, Chairman, President & CEO
Mr. D. G. Bost, Executive Vice President & Chief Nuclear Officer
Mr. D. R. Madison, Vice President – Hatch
Mr. B. L. Ivey, Vice President – Regulatory Affairs
Mr. B. J. Adams, Vice President – Fleet Operations
RTYPE: CHA02.004

<u>U. S. Nuclear Regulatory Commission</u>
Mr. V. M. McCree, Regional Administrator
Mr. P. G. Boyle, NRR Senior Project Manager - Hatch
Mr. E. D. Morris, Senior Resident Inspector – Hatch

Edwin I. Hatch Nuclear Plant Unit 1
Licensee Event Report 2012-004-0
Corrosion-Induced Bonding Results in
Multiple Safety Relief Valves Setpoint Drift

Enclosure

Licensee Event Report 2012-004-0

NRC FORM 366 (9-2007) U.S. NUCLEAR REGULATORY COMMISSION	APPROVED BY OMB: NO. 3150-0104 EXPIRES: 10/31/2013
LICENSEE EVENT REPORT (LER)	Estimated burden per response to comply with this mandatory collection request: 80 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 F53), U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, or by internet e-mail to infocollects.resources@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
Edwin I. Hatch Nuclear Plant Unit 1	05000 321	1 OF 5

4. TITLE
Corrosion-Induced Bonding Results in Setpoint Drift for Multiple Safety Relief Valves

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
3	26	2012	2012	004	0	5	25	2012	FACILITY NAME	DOCKET NUMBER 05000

9. OPERATING MODE: 1

10. POWER LEVEL: 98.6%

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 / Steven Tipps – Principal Engineer – Licensing	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	Yes					

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 March 26, 2012, at approximately 1000 EDT, Unit 1 was at 98.6 percent rated thermal power (RTP) when the “as-found” testing results of the two-stage main steam safety relief valves (SRVs) were received which indicated that eight 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% 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 stellite-21 seats were removed from Unit 1 during the 2012 refueling outage and replaced with 2-stage SRVs with the application of platinum to the pilot discs as the long term corrective action for the historically observed corrosion-induced bonding with consequential setpoint drift. The use of 2-stage SRVs with platinum coated discs is regarded as a proven industry-wide solution for the corrosion-induced bonding phenomenon with excellent results noted for the first operating cycle after installation.

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 March 26, 2012, at approximately 1000 EDT, Unit 1 was at 98.6 percent rated thermal power (RTP) when the report of the "as-found" testing results of the two-stage main steam SRVs was received which indicated that eight 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	Percent Drift
1B21-F013A	**1005**	**1254**	**109.04**
1B21-F013B	**1001**	**1264**	**109.91**
1B21-F013C	309	1159	100.78
1B21-F013D	302	1156	100.52
1B21-F013E	**314**	**1239**	**107.74**
1B21-F013F	**1230**	**1206**	**104.87**
1B21-F013G	1010	1184	102.96
1B21-F013H	**370**	**1222**	**106.26**
1B21-F013K	**1002**	**1218**	**105.91**
1B21-F013L	**315**	**1226**	**106.61**
1B21-F013M	**308**	**1237**	**107.57**

These eleven valves were removed from service during the Spring 2012 refueling outage and replaced with two-stage SRVs whose pilot discs had undergone a platinum surface treatment. These SRVs were properly setup and tested at Wyle Laboratories prior to installation.

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 2, "Event Reporting Guidelines 10CFR50.72 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 notes that "discrepancies found in technical specifications 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 10CFR50.73(a)(2)(i)(B).

There are eleven (11) SRVs located on the four main steam lines within the drywell (EIIS Code NH) 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 by the Nuclear Fuels Department for the Hatch-1 Cycle 25 "as-found" condition of the SRVs. The results from this analysis showed a small increase in peak pressures relative to the Hatch-1 Cycle 25 reload licensing analysis (RLA) results. The higher peak pressures were due to the fact that eight of the eleven SRVs opened at pressures higher than that which was assumed in the RLA. It should be noted that in this analysis, eight of the SRVs tested had the larger actual valve bore size, and therefore higher steam flow capacity than what was conservatively assumed in the RLA. Therefore, higher steam flow capacities than those assumed in the RLA were used in this analysis for those eight valves with the larger valve bores. Based on the analysis, the calculated minimum margin to the 1375 psig ASME Boiler and Pressure Vessel Code overpressure limit for peak vessel pressure would have been 33.7 psig and the minimum margin to the 1325 psig Tech Spec Safety Limit for the reactor steam dome pressure would have been 15.5 psig during an MSIVF event during Cycle 25 operation. Therefore, the analysis of the "as found" test results showed that the peak pressure at the bottom of the vessel remained below the ASME 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 25 this redundant electrical logic system was fully functional.

Based on the analysis by the Nuclear Fuels Department, 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.

CORRECTIVE ACTIONS

The 2-stage SRVs with stellite-21 seats were removed from Unit 1 during the 2012 refueling outage and replaced with 2-stage SRVs containing pilot discs that had undergone a platinum surface treatment that results in the application of platinum material to the seating surface of the valve disc, thereby creating an area with enhanced resistance to surface oxidation. This minimal amount of implanted platinum creates the desired surface oxidation resistance without affecting the mechanical properties of the bulk disc material and is currently considered the interim corrective action for the historically observed setpoint drift. The platinum surface treatment has been proven to be durable enough to withstand the harsh operating conditions of an operating power plant with minimal degradation, wear, or corrosion.

ADDITIONAL INFORMATION

Other Systems Affected: None

Failed Components Information:

Master Parts List Number:1B21-F013A, B, E, F, H, 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 new permanent licensing commitments.

Previous Similar Events:
LER 2-2011-002, identified multiple SRV setpoint drift for 8 of the 11 SRVs. Corrective actions included replacement of the two stage SRVs with three 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 three 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 three-stage SRVs were replaced with two stage SRVs containing pilot discs that had undergone the platinum surface treatment previously discussed.

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.

Multiple examples of SRV setpoint drift occurred and were also reported in LERs 2-2008-004, 1-2008-002, 2-2007-006 and 1-2006-003. These instances of SRV setpoint drift occurred due to like causes which have been noted to be similar to those of the ongoing industry issues with these type SRVs. In each of these cases SNC concluded that the overpressure protection system would have performed its required safety function had it been challenged during its respective operating cycle.