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

Southern Nuclear Operating Company, Inc.
Plant Edwin I. Hatch
11028 Hatch Parkway, North
Baxley, Georgia 31513

Tel 912.537.5859
Fax 912.366.2077

SOUTHERN COMPANY
Energy to Serve Your World℠

August 30, 2011

Docket No.: 50-366

NL-11-1720

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

Edwin I. Hatch Nuclear Plant
Licensee Event Report 2011-002-0
Corrosion-Induced Binding Results in
Multiple Safety Relief Valves Setpoint Drift

Ladies and Gentlemen:

In accordance with the requirements of 10CFR 50.73(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 due to corrosion-induced binding 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,

Dennis Madison

D. R. Madison
Vice President - Hatch

DRM/WEB/msc

Enclosure: LER 2011-002-0

cc: <u>Southern Nuclear Operating Company</u>
Mr. S. E. Kuczynski, Chairman, President & CEO
Mr. J. T. Gasser, Executive Vice President
Ms. P. M. Marino, Vice President – Engineering
RTYPE: CHA02.004

<u>U. S. Nuclear Regulatory Commission</u>
Mr. V. M. McCree, Regional Administrator
Mr. R. E. Martin, NRR Project Manager -Farley, Hatch and Vogtle
Mr. E.D. Morris, Senior Resident Inspector – Hatch

Edwin I. Hatch Nuclear Plant

Corrosion-Induced Binding Results in
Multiple Safety Relief Valves Setpoint Drift

Enclosure 1 to NL-11-1720

Licensee Event Report 2011-002-0

LICENSEE EVENT REPORT (LER)

1. FACILITY NAME	2. DOCKET NUMBER	3. PAGE
Edwin I. Hatch Nuclear Plant Unit 2	05000 366	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
07	05	2011	2011	- 002 -	0	08	30	2011	FACILITY NAME	DOCKET NUMBER
										05000

9. OPERATING 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)	☐ 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 July 5, 2011, at approximately 1000 EDT, Unit 2 was at 100 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 eight of eleven SRVs had experienced setpoint drift which resulted in their allowable Tech Spec limits being exceeded.

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 were removed from Unit 2 in April 2011, and preemptively replaced with 3-stage SRVs as the long term corrective action for the historically observed setpoint drift. The use of 3-stage SRVs is regarded as an industry-wide solution for the corrosion-induced bonding phenomenon which has been a historic industry issue since the early 1980s.

1. FACILITY NAME	2. DOCKET	6. LER NUMBER			3. PAGE
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
Edwin I. Hatch Nuclear Plant Unit 2	05000-366	2011	- 002 -	0	2 OF 5

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 July 5, 2011, at approximately 1000 EDT, Unit 2 was at 100 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 eight of eleven SRVs (EIIS Code SB) had experienced setpoint drift which resulted in their allowable Tech Spec 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
2B21-F013A	**1003**	**1194**	**103.83**
2B21-F013B	1011	1183	102.87
2B21-F013C	**1009**	**1195**	**103.91**
2B21-F013D	**312**	**1207**	**104.96**
2B21-F013E	**1227**	**1276**	**110.96**
2B21-F013F	**311**	**1271**	**110.52**
2B21-F013G	1188	1179	102.52
2B21-F013H	**1190**	**1243**	**108.09**
2B21-F013K	305	1177	102.35
2B21-F013L	**1008**	**1309**	**113.83**
2B21-F013M	**301**	**1226**	**106.61**

These eleven valves were removed from service during the Spring 2011 refueling outage and preemptively replaced with 3-stage SRVs that had been 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 seating surface. This conclusion is based on previous root cause analyses and the repetitive nature of this condition at 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 1000 psig and 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 (iaw) Title 10 of the Code of Federal Regulations (CFR), Part 50.73(a)(2)(i)(B) because an event occurred which is prohibited by the Technical Specifications (TS). 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 iaw TS surveillance requirement 3.4.3.1 in which the valves are tested as directed by the In-Service Testing Program to verify lift setpoints 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 H2C21 "as-found" condition of the SRVs. The results from this analysis showed a small increase in peak pressures relative to the Hatch-2 Cycle 21 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, the larger actual valve bore size was used in the calculations for nine of the valves rather than the smaller bore size which was conservatively assumed in the RLA. Therefore, higher steam flow capacities than those assumed in the RLA were used in this analysis for those nine valves. 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 27.7 psig and the minimum margin to the 1325 psig Tech Spec Safety Limit for the reactor steam dome pressure would have been 2.9 psig during an MSIVF event during Cycle 21 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 the Cycle 21 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

All eleven 2-stage SRV pilot valves were preemptively replaced with new 3-stage SRV pilot valves as the long term corrective action. The use of the 3-stage SRVs is regarded as an industry-wide solution for the corrosion-induced bonding phenomenon which has been a historic industry issue since the early 1980s.

ADDITIONAL INFORMATION

Other Systems Affected: None

Failed Components Information:

Master Parts List Number:2B21-F013A, B, C, D, E, F, G, H, K, L, M **EIIS System Code**: SB
Manufacturer: Target Rock **Reportable to EPIX**: Yes
Model Number: 7567F **Root Cause Code**: B
Type: Relief Valve **EIIS Component Code**: RV
Manufacturer Code: T020

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

Previous Similar Events:
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.

The replacement of the 2-stage SRVs with 3-stage SRVs should resolve this BWR industry issue, and this assertion will be confirmed during the performance of future "as found" testing during the next scheduled refueling outage.