Dennis R. Madison
Vice President - Hatch

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SOUTHERN COMPANY

May 10, 2012

Docket Nos.: 50-321

NL-12-0944

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

Edwin I. Hatch Nuclear Plant
Licensee Event Report 2012-003-0
Leak in Reactor Pressure Boundary
at Small Bore Piping Fillet Weld

Ladies and Gentlemen:

In accordance with the requirements of 10CFR50.73(a)(2)(i)(B), Southern Nuclear Operating Company hereby submits the enclosed Licensee Event Report concerning an event of non-compliance with Technical Specification 3.4.4 for Reactor Coolant System operational leakage from a through-wall crack in a small bore piping fillet weld.

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

Respectfully submitted,

Dennis Madison

D. R. Madison
Vice President – Hatch

DRM/WEB/

Enclosure: LER 1-2012-003, Revision 0

cc: Southern Nuclear Operating Company
Mr. S. E. Kuczynski, Chairman, President & CEO
Mr. D. G. Bost, Executive Vice President & Chief Nuclear Officer
Mr. B. L. Ivey, Vice President – Regulatory Affairs
Mr. B. J. Adams, Vice President – Fleet Operations
Mr. M. J. Ajluni, Nuclear Licensing Director
RTYPE: CHA02.004

cc: (continued)

<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

Enclosure

NL-12-0944

Edwin I. Hatch Nuclear Plant – Unit 1

Licensee Event Report 2012-003-0

Leak in Reactor Pressure Boundary at Small Bore Piping Fillet Weld

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

APPROVED BY OMB: NO. 3150-0104 EXPIRES: 10/31/2013

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.

LICENSEE EVENT REPORT (LER)

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4. TITLE
Leak in Reactor Pressure Boundary at Small Bore Piping Fillet Weld

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
03	13	2012	2012	- 003 -	00	05	10	2012	FACILITY NAME	DOCKET NUMBER 05000

9. OPERATING MODE: 4

10. POWER LEVEL: 0

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 – Lead 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	PSF	N/A	Y					

14. SUPPLEMENTAL REPORT EXPECTED		15. EXPECTED SUBMISSION DATE	MONTH	DAY	YEAR
☒ YES *(If yes, complete 15. EXPECTED SUBMISSION DATE)*	☐ NO		12	01	2012

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

On 3/13/2012, during the Reactor Pressure Vessel (RPV) Pressure Test walk-down with the unit in Mode 4 for leakage testing, a through-wall leak was identified in a small bore line located upstream of a High Pressure Coolant Injection (HPCI) valve inboard of the piping's connection to the Main Steam piping. The leak identified during this event was in the thicker portion of the fillet weld adjacent to the socket elbow. Initial evaluation of the crack in the subject HPCI piping and elbow by an experienced site Quality Control inspector and a senior Southern Nuclear metallurgist concluded that the most apparent cause of the weld defect that led to the leak was inadequate root penetration in the weld. The actual cause has not yet been fully determined. The section of piping in which the leak was located will be sent to a vendor for inspection and expert determination of the most probable cause of the through-wall leak. Following removal and repair of the subject piping, a leak test was performed at 920 psig with no leaks identified. The Technical Specification (TS) definition of pressure boundary leakage is leakage through a non-isolable fault in the reactor coolant system. By its location, the leak met this definition. Inspection of the weld and adjacent areas determined the leak had existed when the Unit was in Mode 1, and no such leakage is allowed by TS.

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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 3/13/2012, with the unit in Mode 4 for conducting a Reactor Pressure Vessel (RPV) Pressure Test inspection following refueling, a through-wall leak was identified in a small bore line (specifically in a ¾ inch elbow) located inboard of the HPCI 1E41-F002 valve prior to the point at which the line connects to the Main Steam Line "B" piping (EIIS Code SB). The physical discoloration of the pipe and surrounding insulation surrounding the leak appear to support the judgment that the leak had existed for some period of time during the previous cycle. Initial evaluation of the crack in the subject HPCI piping and elbow concluded that the most apparent cause of the weld defect that led to the leak was inadequate root penetration in the weld that over time likely propagated through the wall of the pipe. The actual cause has not yet been fully determined.

Following removal and repair of the subject piping, a leak test was performed at 920 psig with no leaks identified.

CAUSE OF EVENT

The leak identified during this event was in the thicker portion of the fillet weld adjacent to the socket elbow. The subject weld was an original construction Class 1 Tungsten Inert Gas (TIG) weld that was inspected both visually and by Penetration Test (PT) at the time the weld was performed. Initial evaluation of the crack in the subject HPCI piping and elbow by a Southern Nuclear (SNC) Metallurgist Principal Engineer (PE) and a person from Quality Control (QC), each with over thirty years of experience with weld processes and inspections, concluded that the most apparent cause of the weld defect that led to the leak was inadequate root penetration in the weld. This conclusion was reached based on the characteristics of the failed weld after consideration of High Cycle Fatigue (HCF), Intergranular Stress Corrosion Cracking (ICSCC), possible imposed stress from work conducted in the course of the subject outage, improper weld fusion, original weld defect caused by slag or porosity at the root, or inadequate root penetration.

At the time of the performance of the subject weld some years ago, it was permissible to use a welding rod with a diameter of 1/8 inch in such welds, and this practice may have contributed to the suspected inadequate root penetration. Applicable welding procedures were revised in 2008 to specify that the root pass of all socket welds be performed using filler material with a maximum diameter of 3/32 of an inch to provide proper root penetration. A subsurface or root defect resulting from inadequate root penetration most likely propagated over time through the wall, and eventually caused the leak. The actual cause of the weld failure that led to the observed leak has not yet been fully determined. The section of piping in which the leak was located will be sent to an appropriate vendor (Altran) for inspection and expert determination of the most probable cause of the through-wall leak. When that evaluation of the apparent cause has been completed, this report will be revised to disclose the final results of the analysis.

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REPORTABILITY ANALYSIS AND SAFETY ASSESSMENT
Assessment Information:
This report is required per 10 CFR 50.73(a)(2)(i)(B) because a condition existed which was prohibited by the plant's Technical Specification (TS). The TS definition of pressure boundary leakage is "LEAKAGE through a non-isolable fault in a Reactor Coolant System (RCS) component body, pipe wall, or vessel wall." Due to its location, this leak met this definition. Based on inspection of the weld and adjacent areas, it was determined the leak had existed when the Unit was in Mode 1, and no such leakage is allowed by TS. Therefore, this event comprises an operation or condition which was prohibited by LCO 3.4.4.a and is thus reportable under the requirements of 10CFR50.73(a)(2)(i)(B).

The reactor coolant system (RCS) includes systems and components that contain or transport the coolant to and from the reactor core. The pressure retaining components of the RCS and the portions of connecting systems out to and including the isolation valves define the reactor coolant pressure boundary. During plant life, the joint and valve interfaces can produce varying amounts of reactor coolant leakage through either normal operational wear or mechanical deterioration. Limits on RCS operational leakage are required to ensure appropriate action is taken before the integrity of the reactor coolant pressure boundary is compromised. The TS delineate the limits on the specific types of leakage. The unidentified leakage flow limit allows time for corrective action to be taken before the reactor coolant pressure boundary can be compromised significantly. The five gallons per minute (gpm) limit is a small fraction of the calculated flow from a critical crack in the primary system piping. A critical crack is one large enough to propagate rapidly, ultimately leading to failure of the affected component. As discussed in the FSAR, crack behavior from experimental programs shows that leakage rates of over a hundred gallons per minute will precede crack instability.

In this event, a small leak was identified and investigated as a result of a RPV Pressure Test walk-down. This leak was determined to meet the TS definition of pressure boundary leakage due to its location in a portion of the RCS piping which could not be isolated from the reactor coolant pressure boundary. At the time it was discovered and corrective action taken, the leak was not unstable and would not have resulted in catastrophic failure of the line. However, a worst-case instantaneous and complete severing of the line due to the presence of such a leak would not result in a significant loss of reactor coolant or present any challenge to core cooling. In addition, even if the inventory loss were completely water as compared to steam or a steam-water mix, the break would still be bounded by both the Loss of Coolant Accident analysis and the Feedwater Line break analysis. This hypothetical leak would be significantly less than the rated capacity of the High Pressure Coolant Injection, HPCI (EIIS Code BJ) system, which is sized to provide adequate coolant make-up for pipe breaks up to four inches, and approximates the rated capacity of the Reactor Core Isolation Cooling, RCIC (EIIS Code BN) system. Consequently, either system would have been capable of indefinitely maintaining normal reactor water level. Additionally, a leak of several hundred gallons per minute (gpm) would be adequately accommodated by the feedwater system (EIIS Code SJ), which has a flow rate capacity margin at rated conditions of at least 10 percent. Therefore, any one of three diverse and independent high pressure injection systems could have provided sufficient make-up flow to maintain water level well above the top of the active fuel. Based upon the preceding considerations, it is concluded that this event had no adverse impact on nuclear safety. This analysis is applicable to all operating conditions under which the subject leak might have propagated to line failure.

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CORRECTIVE ACTIONS
Short-term corrective actions included removal and replacement of that portion of the piping which contained the leak, and the inspection of the repaired piping to assure that the leak had been properly corrected. Inspection of other potential leak sites had been performed as part of the Reactor Pressure Vessel Pressure Test walk-down during which the subject leak was identified.

Longer-term corrective actions include the determination of the most probable cause of the cracking and once the metallurgical evaluation of the piping/elbow has been performed by Altran, any necessary actions prompted by that evaluation will be considered. The apparent cause of the defective weld, performed many years ago, is likely due to the weld not having proper root penetration. Appropriate corrective actions were taken in the past (albeit after the welding of the subject piping) to ensure the Weld Process Control Procedure was revised to mitigate the probability of future small bore piping weld failures.

The extent of condition potentially includes the Class 1 small bore piping in the Unit 1 and Unit 2 drywells. The Class 1 drywell small bore piping is tested in accordance with the RPV Leak test performed during refueling outages. Therefore, any small bore piping leaks will be identified during these walkdowns. It should also be noted that drywell leakage is monitored daily, and any significant leak would be identified by an increase in drywell leakage.

ADDITIONAL INFORMATION

Other Systems Affected: None

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

Previous Similar Events:
Non-isolable ASME Class-1 pressure boundary leaks were discovered during three previous refueling outages; one each in Unit 2 piping in years 2005 and 2007, and one in Unit 1 piping in 2008. All three leaks were identified during the RPV System Leakage Tests at the end of refueling outages. The three leaks occurred on non-isolable 1" stainless steel instrumentation piping associated with the main steam system. In each of these instances, the plant had to be returned to cold shutdown in order to perform the repairs. In each case, it was apparent that the leak, although small in magnitude, had existed for some period of time during the preceding run cycle. Upon completion of the failure analysis for the affected pipe elbow this section will be updated to determine if there were actually any identified previous similarities.

The piping leak discovered on Unit 2 in the Spring of 2005, occurred in the 1" stainless steel instrumentation piping associated with the 4" steam supply to the RCIC system. The failed component was shipped to a vendor (Altran) for failure analysis. High Cycle Fatigue (HCF) was determined to be the predominant failure mechanism. The failed piping section was replaced with the identical material and original weld geometry.

The second piping leak discovered on Unit 2 in the Spring of 2007, occurred in 1" stainless steel instrumentation piping associated with the main steam flow-measurement manifold, on

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the downstream side of a condensing chamber. A root cause analysis was performed and the failed component was shipped to Altran for failure analysis. HCF was again determined to be the predominant failure mechanism. The failed piping section was replaced with the identical material.

During the Spring 2008 Unit 1 refueling outage, a piping leak was discovered on a 1" stainless steel instrumentation line associated with the main steam flow-measurement manifold on the upstream side of a condensing chamber. A root cause investigation was performed and the failed component was shipped to Altran for analysis. In this case, IGSCC was determined to be the predominant failure mechanism. Additionally, a piping strap (restraint) shown on the isometric drawing was found missing during inspections following this failure. The failed piping section was replaced with the same material using the original weld geometry.