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Vice President - Hatch

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SOUTHERN COMPANY
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Docket Nos.: 50-366

NL-08-1042

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

Edwin I. Hatch Nuclear Plant – Unit 2
Licensee Event Report
Crack in Reactor Pressure Boundary Piping Due to High Cycle Fatigue Results in Leakage

Ladies and Gentlemen:

In accordance with the requirements of 10 CFR 50.73(a)(2)(i)(B), Southern Nuclear Operating Company is submitting the enclosed Licensee Event Report (LER) concerning an unisolable leak in the reactor pressure boundary.

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

Sincerely,

Dennis Madison

D. R. Madison
Vice President – Hatch

DRM/mjk

Enclosure: LER 2-2008-003

cc: <u>Southern Nuclear Operating Company</u>
Mr. J. T. Gasser, Executive Vice President
Mr. D. R. Madison, Vice President – Hatch
Mr. D. H. Jones, Vice President – Engineering
RTYPE: CHA02.004

<u>U. S. Nuclear Regulatory Commission</u>
Mr. L. A. Reyes, Regional Administrator
Mr. R. E. Martin, NRR Project Manager – Hatch
Mr. J. A. Hickey, Senior Resident Inspector – Hatch

NRC FORM 366 (9-2007)	U.S. NUCLEAR REGULATORY COMMISSION	APPROVED BY OMB: NO. 3150-0104 EXPIRES: 08/31/2010
LICENSEE EVENT REPORT (LER)		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
Crack in Reactor Pressure Boundary Piping Due to High Cycle Fatigue Results in Leakage

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
5	19	2008	2008	003 -	0	07	07	2008	FACILITY NAME	DOCKET NUMBER 05000

9. OPERATING MODE	11. THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR§: *(Check all that apply)*			
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)
99.9	☐ 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 Underwood, Performance Analysis Supervisor	912-537-5931

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	BN	N/A	N/A	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 19, 2008, it was determined that a condition discovered on March 8, 2005, consisting of a leak on a non-isolable portion of the reactor coolant boundary identified during the reactor vessel leakage test was reportable under 10 CFR 50.73. The leak was located in a section of one inch stainless steel instrumentation piping associated with the four inch steam supply to the reactor core isolation cooling (RCIC) system. The Technical Specification (TS) definition of pressure boundary leakage is "leakage through a non-isolable fault in the reactor coolant system." Due to its location, the leak met this definition. Pressure boundary leakage in modes 1, 2, or 3 is not allowed by the plant's TS. Although the leak was discovered when the unit was in Mode 4, review of the failure analysis report provided indication that the leak most likely started at some point during the run cycle (i.e. during Modes 1, 2 and/or 3) leading up to the time of discovery.

The piping was sent to an independent testing facility which determined the cause of the leak was failure of the pipe due to High Cycle Fatigue.

Corrective actions for this event included replacing the failed weld, evaluating similar small bore class 1 piping for additional action, and revising The Corrective Action Program to elevate the severity level even if the unit is in an operating mode at the time of condition discovery that does not require operability, and regardless of how the condition was discovered.

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

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 8, 2005, a leak on a non-isolable portion of the reactor coolant boundary was discovered during the reactor vessel leakage test (performed in Mode 4). The leak was located in a section of one inch stainless steel instrumentation piping associated with the four inch steam supply to the Reactor Core Isolation Cooling (RCIC) system (EIIS Code BN). The Technical Specification (TS) definition of pressure boundary leakage is "leakage through a non-isolable fault in the reactor coolant system". Due to its location, the leak met this definition. The leak was discovered when the unit was in Mode 4. A condition report was created, assigned a low severity level, and corrective action was taken. The damaged pipe was replaced and similar Class 1 small bore piping was inspected at that time for evidence of leakage or damage with no problems identified. The pipe was sent to an independent testing laboratory for failure analysis which determined the cause of the failure to be high cycle fatigue. Documentation of a reportability evaluation based on the findings from the failure analysis report could not be found. Subsequent review of the failure analysis report provided indication that the leak most likely started at some point during the run cycle (i.e. during Modes 1, 2 and/or 3) leading up to the time of discovery and as a result the determination was made on March 19, 2008 that this condition should be reported under 10 CFR 50.73.

CAUSE OF EVENT

The piping was sent to an independent testing facility which determined the cause of the leak was failure of the pipe due to High Cycle fatigue.

REPORTABILITY ANALYSIS

This report is submitted, per 10 CFR 50.73(a)(2)(i)(B), as a condition prohibited by the plant's TS. The Unit 2 TS allows no pressure boundary leakage in modes 1, 2, or 3. Reportability of the condition turns on whether there is evidence that the condition existed prior to discovery, i.e. when the unit was in Mode 1, 2, and/or 3.

Guidance provided in NUREG 1022 Rev. 2 states in section 3.2.2:

"... For the purpose of evaluating the reportability of a discrepancy found during surveillance testing that is required by the technical specifications:

(1) For testing that is conducted within the required time (i.e., the surveillance interval plus any allowed extension), it should be assumed that the discrepancy occurred at the time of its discovery unless there is firm evidence, based on a review of relevant information such as the equipment history

and the cause of failure, to indicate that the discrepancy existed previously."

The condition report that identified this leak was classified at a low severity level and there is no documentation that a reportability evaluation took place at the time of discovery or once the failure analysis report was received. At the time of discovery it was recognized that the leak must be repaired prior to entering modes 1, 2 or 3 but consideration of impact during the prior operating cycle was not made. Recent review of the failure analysis report provided indication that the leak most likely started at some point during the run cycle (i.e. during Modes 1, 2 and/or 3) leading up to the time of discovery. As a result the determination was made on March 19, 2008 that this condition should be reported under 10 CFR 50.73.

SAFETY ASSESSMENT

The reactor coolant system (RCS) includes systems and components that contain or transport the coolant to or 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 TS unidentified leakage flow limit of five gallons per minute (gpm) 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. Crack behavior from experimental programs shows that leakage rates of hundreds of gallons per minute will precede crack instability (see Unit 2 Final Safety Analysis Report, section 5.2.7.5, "Nuclear System Leakage Detection and Leakage Rate Limits," and Unit 2 TS Bases B 3.4.4, "RCS Operational Leakage").

At the time the unit was shut down, the unidentified leakage rate was approximately one percent of the TS limit of five gpm. The size of the crack was much smaller than the "critical crack" (on which the TS limit is based) as evidenced by the low leakage rate. Therefore, at the time it was discovered and corrective action taken; the crack was not unstable and would not have resulted in catastrophic failure of the line. Additionally, in the unlikely event of the worst case instantaneous and complete severing of the one-inch line, due to the presence of a crack, the resulting loss of reactor coolant would not be of a significant amount.

A rupture of the one-inch RCIC line does not result in a significant decrease in water inventory within the vessel. In addition, the break is bounded by both the Loss of Coolant Accident analysis and the Feedwater Line break analysis. The proposed leak in the RCIC line is less than 10 percent of the rated capacity of the High Pressure Coolant Injection (EIIS Code BJ) system, which is sized to provide adequate coolant make-up for pipe breaks up to four inches. Additionally, a leak of several hundred gpm would have adequately been accommodated by the feedwater system (EIIS Code SJ), which has a flow rate capacity margin at rated conditions of at least 10 percent (over 2000 gpm). Therefore, either of these diverse and independent 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 analysis, it is concluded that this event had no adverse impact on nuclear safety.

CORRECTIVE ACTIONS

The failed weld was replaced.

Systems within the ASME Class 1 boundary were reviewed for lines which are small-bore, unisolable, and stainless steel. Sixteen Main Steam Flow connections on each unit, and the four flow measurement lines each for steam supply to RCIC and HPCI, were selected to be evaluated and corrective actions taken as determined appropriate. These actions will be tracked in the Corrective Action Program.

The Corrective Action Program has been revised to elevate the severity level even if the unit is in an operating mode at the time of condition discovery that does not require operability, and regardless of how the condition was discovered.

ADDITIONAL INFORMATION

Other Systems Affected: No systems were affected by this event other than those which have already been discussed in this report.

Failed Components Information:
Master Parts List Number: 2E51
Manufacturer: N/A
Model Number: N/A
Type: N/A
Manufacturer Code: N/A

EIIS System Code: BN
Reportable to EPIX: Yes
Root Cause Code: B
EIIS Component Code: N/A

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

Previous Similar Events: There were no similar events reported in the two years prior to the original event.