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SOUTHERN COMPANY
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April 18, 2008

Docket No.: 50-366

NL-08-0622

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

Edwin I. Hatch Nuclear Plant – Unit 2
Licensee Event Report 2-2008-001
Leak in Vent Pipe to Process Pipe
Sockolet in RHRSW System

Ladies and Gentlemen:

In accordance with the requirements of 10 CFR 50.73 (a)(2)(v), Southern Nuclear Operating Company (SNC) is submitting the enclosed Licensee Event Report concerning a leak in a vent pipe sockolet attached to the Residual Heat Removal Service Water (RHRSW) system. A Supplemental Report will be submitted by May 16, 2008 regarding this event.

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

Sincerely,

D. R. Madison
Vice President – Hatch

DRM/RDB/daj

Enclosure: LER 2-2008-001

cc: Southern Nuclear Operating Company
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. S. Nuclear Regulatory Commission
Mr. V. M. McCree, Acting 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
Leak in Vent Pipe to Process Pipe Sockolet in RHRSW System

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
02	18	2008	2008	- 001 -	00	04	18	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)
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

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	BI	PSP		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 February 18, 2008 at 1800 EST, a crack was identified at a weld on a ¾ inch vent line of the Residual Heat Removal Service Water (RHRSW) system. An immediate operability determination was conducted based on available information with the conclusion reached at that time that there was reasonable assurance the RHRSW system could continue to perform its safety function. Subsequent engineering review resulted in a determination on February 20, 2008 at 1505 EST that this condition could have prevented the RHRSW system from meeting its 30-day mission time. On February 20, 2008 at 1505 EST, the RHRSW system was declared inoperable, and both loops of RHRSW were removed from service in order to make the necessary repairs. It was postulated that a complete shear of the vent line at the crack location could have prevented continued operation of the system for 30 days, as required by the Unit 2 Final Safety Analysis Report for design basis accident (DBA) conditions. The resulting leakage from the postulated shearing of the line could have exceeded the capacity of the Reactor Building sumps, and over a prolonged period of operation, could have resulted in excessive water levels in the Reactor Building Torus Room area. The cause of this event is attributed to high cycle fatigue due to vibration resulting from cavitation downstream of the 2E11F068B flow control valve for the "B" loop of the RHRSW system. The vent line was repaired and the system returned to service on February 20, 2008.

Plant and System Identification

General Electric Boiling Water Reactor
Energy Industry Identification System Codes appear in the text as (EIIS Code BI)

Description of the Event

On February 18, 2008 at 1800 EST, a small leak was identified near a 3/4 inch vent valve (2E11-FV001) downstream of the 'B' RHR (RHR, EIIS Code BO) heat exchanger (2E11B001B) flow control valve (2E11-F068B). At that time, Hatch Nuclear Plant (HNP) Unit 2 was in Mode 1 at a power level of 100%. The vent line containing 2E11-FV001 is 0.75 inches in diameter and has an external OD of 1.05 inches, and the valve is welded by a socket weld to a process pipe sockolet. The crack was approximately 120 degrees of the weld circumference at the vent pipe to process pipe sockolet. The valve is at a high point in the “B” subsystem of the Residual Heat Removal Service Water (RHRSW, EIIS Code BI) system within the reactor building, and is at the approximate height of the discharge of the RHRSW system to the atmosphere at the cooling tower flume.

During the HNP Unit 1 Spring 2008 refueling outage, damaged supports had been found in the Unit 1 RHRSW system. Consequently, a RHRSW system walkdown was in progress for Unit 2 to determine the extent of condition regarding similar supports on that unit. At the time the leak was discovered, there were no Unit 2 structures, systems, or components that were known to be inoperable. Additionally, the Unit 2 RHRSW system walkdown identified one U-bolt for a pipe support to have loose nuts and one support (2E11-RSW-R15) was found to have a broken paddle. An immediate operability determination was conducted based on available information with the conclusion reached at that time that there was reasonable assurance the RHRSW system could continue to perform its safety function. Subsequent engineering review resulted in a determination on February 20, 2008 at 1505 EST that this condition could have prevented the RHRSW system from meeting its 30-day mission time. Pending additional metallurgical examination and analysis, it was postulated that a complete shear of the vent line at the crack location could have prevented continued operation of the system for 30 days, as required by the Unit 2 Final Safety Analysis Report. Credit for manual actions to plug the leak or improve water removal efforts was not considered. The resulting leakage from the postulated shearing of the line could have exceeded the capacity of the building sumps, and over a prolonged period of operation could have resulted in excessive water levels in the Reactor Building (Reactor Building, EIIS Code NG) Torus Room area. Both loops of RHRSW come together in a common discharge line downstream of 2E11FV001. Since the damaged line was in an unisolable section of the discharge piping for the Residual Heat Removal Service Water (RHRSW) system, shutdown of the RHRSW system would have been required to mitigate the resultant leakage and repair the failure. The RHRSW system is required to operate continuously for 30 days post-LOCA. The RHRSW system was declared inoperable, and both loops of the RHRSW system were removed from service in order to make the necessary repairs. The line was repaired and the system returned to service on February 20, 2008 at 2120 EST.

Causes of the Event

A laboratory assessment of the cracked vent line was performed. The failure of the weld was a result of vibration that culminated in high cycle fatigue failure of the valve to sockolet weld for vent valve 2E11-FV001. The direct causes of vibration are related to cavitation of the RHR heat exchanger flow control valve at flow rates near 7,000 gpm. This cavitation was caused by the RHR heat exchanger flow control valve flow characteristics and the dynamics of flow through the system, which created significant piping vibration downstream.

The RHR heat exchanger flow control valve is a Control Components Inc. (CCI) 10 x 10 300 lb globe/ flow control valve and was installed in 1994. The valve is Model No. DRAG-EXG3-X7-10IF-10IF-XXAKB2. The HNP Unit 2

RHRSW system piping was constructed to ASME, Boiler and Pressure Vessel Code, Section III, Class 3, 1971 Edition.

Reportability Analysis and Safety Assessment

The containment heat removal system consists of the shutdown cooling and the containment cooling modes of the residual heat removal (RHR) system for planned and abnormal operation. Four RHR pumps can take water from the suppression pool in two loops, each containing two RHR pumps and one heat exchanger. Heat absorbed in these cooling modes is exchanged to the RHRSW system via the RHR heat exchangers. The RHRSW system is comprised of four pumps, with two pumps in each of two loops. The RHRSW loops pump river water through the secondary side of the two RHR heat exchangers to remove suppression pool heat. A LOCA or an LOSP provides an automatic stop signal to the RHRSW pumps. The RHRSW system is manually operated. If offsite power is available, the RHRSW pumps can be started immediately. If offsite power is not available, the RHRSW pumps are manually started from the main control room (MCR) as required and as other loads on the essential buses are secured.

In the event of a postulated LOCA, the short-term energy release is discharged to the suppression pool, causing a pool temperature increase of ~45°F. Subsequent to the accident, fission product decay heat results in a continuing energy release to the suppression pool which, unless removed, could result in unacceptable suppression pool temperatures and containment pressures. The containment cooling mode of the RHR system is used to remove heat from the suppression pool. At the design basis river temperature of 95°F and an RHRSW system flow rate of 7000 gpm, RHR heat exchanger performance is such that peak suppression pool temperature is reached in less than nine hours into the event and one RHRSW pump is capable of removing the post-LOCA heat load within approximately 12 hours following the postulated LOCA. The design basis credits only one RHR heat exchanger and the two RHRSW pumps that communicate with that heat exchanger in order to allow for the postulation of a single failure, such as the failure of a RHR heat exchanger flow control valve to open. Because failure of the vent line with no subsequent repair actions could have exceeded the capacity of the Reactor Building sumps, and over a prolonged period of operation, could have resulted in excessive water levels in the Reactor Building Torus Room Area, this event is being reported pursuant to 10 CFR 50.73(a)(2)(v) – Event or Condition That Could Have Prevented Fulfillment of a Safety Function.

Engineering review determined that one RHRSW pump was sufficient to provide the required heat removal whenever river temperature is below 74°F and given system capabilities. Most recently, river temperature decreased below 74°F on October 26, 2007, where it has remained since that time which encompasses the time frame in which the condition was identified and resolved. In addition, if the vent line had failed during normal plant operation, Reactor Building sump high level alarms would have alerted plant personnel to the leak, and allowed personnel to initiate repair actions before excessive water levels occurred. Vibration levels with one pump in service are very low and crack propagation rates would be considerably lower. Engineering evaluation of vibration levels indicated the levels were below the endurance limit for the cracked weld, and accordingly, there would be no driving force for further propagation during one-pump operation. There were no actual safety consequences to this event, since the leakage from the cracked weld was minor, and there were no events requiring the RHRSW system to perform in a design basis manner for the time period ending on repair of the line on February 20, 2008 at 2120 EST.

<u>Corrective Actions</u>

The pipe leak at vent valve 2E11-FV001 was repaired. Since the 2E11-FV001 failure showed that "leak before break" was evident, VT2 inspections (leak check) were performed for the vent valve and for drain piping downstream of the RHR heat exchanger including 2E11-FD001, 2E11-FV001, 2E11-FD003, and 2E11-FV003. Piping at 2E11-FV003 valve location was inspected even though the valve was previously removed and the piping capped. Inspections revealed no service related weld indications. The vent/drain lines listed above will be periodically observed during system operation (compensatory action) to determine if any leakage exists.

The broken strut on support 2E11-RSW-R15 was repaired. Evaluation of the system with failed support 2E11-RSW-R15 showed the system piping was still within stress limits even with the support broken.

As a proactive measure, rigid supports have been installed on Unit 2 on affected RHRSW vent and drain lines. These supports reduce susceptibility to vibration induced failures.

To identify any other broken supports visual inspections (VT-3) were performed on the piping supports for both loops of the Unit 2 RHRSW system downstream of the RHR Heat Exchangers within the Reactor Building. Deficient conditions identified impacting functionality were subsequently corrected.

After inspecting the system, and correcting damaged or degraded components, vibration analyses were conducted at various pump flow rates. Elevated levels of vibration were observed to be occurring in the process and attached piping during operation at the higher flowrates (two pump operation). Operational compensatory limits were put into place to assure unacceptable system vibration from cavitation does not occur by procedurally limiting system flow for this manually operated system.

For long term resolution, the design of this system will be modified to replace the RHR heat exchanger flow control valves on each unit to reduce cavitation induced vibration. These corrective actions will be completed by 6/30/2008.

Additionally, as a part of the long term resolution, the RHRSW process pipe support system will be modified with additional restraints to improve the vibration response of the system. These actions will be completed by 6/30/2008.

<u>Additional Information</u>

Other Systems Affected:

Unit 1 RHRSW hangers and supports were found damaged during the recent 2008 refueling outage. The damaged hangers and supports were restored to meet design requirements prior to startup from that outage. The operational compensatory limits that were put in place on Unit 2 are also currently in place on Unit 1 to assure unacceptable system vibration from cavitation does not occur, by procedurally limiting system flow for this manually operated system. Corrective measures are being taken on Unit 1 RHRSW with partial implementation of similar rigid supports on the affected RHRSW vent and drain lines. Actions to install the remaining rigid supports are in progress.

Failed Components Information: None

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

Previous Similar Events:

LER 2-1999-006 included the complete failure of a ¾ inch vent line near the weld where it joined the RHRSW system process piping. This vent line contained vent valve 2E11-FV003 which is the RHR ‘A’ heat exchanger outlet vent valve. The cause for the line failure was determined to be high cycle fatigue that was exacerbated by pipe geometry. Corrective actions included visual inspections of RHR, RHRSW and HPCI socket welds on both units with no “crack like” indications identified in December 2000. Vibration screening was performed for socket welds on RHR and RHRSW on both units with no problems identified with welds on the Unit 2 RHRSW system. Broadness actions taken for small bore piping on certain safety systems susceptible to similar high cycle fatigue were incomplete and too narrow. The root cause analyses were insufficient in determining the actual causes of the high cycle fatigue failures. Corrective actions for these issues are being addressed in the corrective action program.