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SOUTHERN COMPANY
Energy to Serve Your World℠

April 6, 2006

Docket No.: 50-425

NL-06-0686

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

Vogtle Electric Generating Plant – Unit 2
Licensee Event Report 2-2006-001
Two Reactor Coolant Pressure Boundary Leaks Lead to Shutdown
Required by Technical Specifications

Ladies and Gentlemen:

In accordance with the requirements of 10 CFR 50.73, Southern Nuclear Operating Company hereby submits a Vogtle Electric Generating Plant licensee event report for a condition that was determined to be reportable on February 3, 2006.

Sincerely,

Don E. Grissette

DEG/RJF/sdl

Enclosure: LER 2-2006-001

cc: Southern Nuclear Operating Company
Mr. J. T. Gasser, Executive Vice President
Mr. T. E. Tynan, General Manager – Plant Vogtle
RType: CVC7000

U. S. Nuclear Regulatory Commission
Dr. W. D. Travers, Regional Administrator
Mr. C. Gratton, NRR Project Manager – Vogtle
Mr. G. J. McCoy, Senior Resident Inspector – Vogtle

NRC FORM 366 (6-2004) **U.S. NUCLEAR REGULATORY COMMISSION**

LICENSEE EVENT REPORT (LER)

(See reverse for required number of digits/characters for each block)

1. FACILITY NAME	2. DOCKET NUMBER	3. PAGE
Vogtle Electric Generating Plant – Unit 2	05000-425	1 OF 5

4. TITLE
Two Reactor Coolant Pressure Boundary Leaks Lead to Shutdown Required by Technical Specifications

5. EVENT DATE			6. LER NUMBER			7. REPORT DATE			8. OTHER FACILITIES INVOLVED	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	MONTH	DAY	YEAR	FACILITY NAME	DOCKET NUMBER(S) 05000
02	03	2006	2006	001	00	04	06	2006	FACILITY NAME	DOCKET NUMBER(S) 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)	X	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)
Tom Webb, Performance Analysis	(706) 826-3105

13. COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO EPIX		CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO EPIX

14. SUPPLEMENTAL REPORT EXPECTED				15. EXPECTED SUBMISSION DATE	MONTH	DAY	YEAR
X	YES (*If yes, complete 15. EXPECTED SUBMISSION DATE*)		NO		07	11	2006

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

On February 1, 2006, control room operators received indication of an increase in radioactivity in the containment atmosphere. On February 3, 2006, a robotic camera observed leakage inside the bioshield wall in the area of reactor coolant system (RCS) loop 1. Unit 2 was placed in Mode 3 (Hot Standby) at 1806 EST, on February 3, 2006, to allow further investigation of specific leakage locations. At 2124 EST, this investigation found RCS pressure boundary leakage at two welded connections on a ¾" bypass line around the Residual Heat Removal (RHR) loop suction valve, 2HV 8701B, and shutdown to Mode 5 (Cold Shutdown) was initiated. On February 5, 2006, at 0035 EST, Unit 2 entered Mode 5 to comply with Technical Specification 3.4.13.a. due to the RCS pressure boundary leakage.

Although the cause of the weld failures is undetermined at this time, the root cause investigation is still underway. Issues being addressed include high vibration of the bypass line, support design, excessive weld stress, and weld quality. The theory presented by the root cause team is the cause of the cracks was from high cycle fatigue. Corrective actions included replacement of the Loop 1 bypass line, inspection of supports and snubbers for both Unit 2 bypass lines, inspection of welds on the Loop 4 bypass line, and installation of monitoring instrumentation. Following additional monitoring, the cause determination will be completed, follow-up corrective actions implemented and a revised LER will be submitted by July 11, 2006.

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

A) REQUIREMENT FOR REPORT

This report is required per 10 CFR 50.73 (a)(2)(i)(A), because a unit shutdown to Mode 5 (Cold Shutdown) was completed on February 5, 2006, to comply with Technical Specification 3.4.13.a., following the discovery of Reactor Coolant System (RCS) pressure boundary leakage.

B) UNIT STATUS AT TIME OF EVENT

On February 1, 2006, Unit 2 was de-rated from full power to approximately 20% rated thermal power for repair of an Electrohydraulic Control (EHC) leak on the main turbine front standard. The EHC leak was repaired and power ascension commenced to restore the unit to full power. During the power ascension, containment radiation monitor 2RE2562A went into intermediate alarm on three different occasions. Prior to the discovery of RCS leakage on February 3, 2006, Unit 2 had returned to Mode 1 (Power Operations) at 100% rated thermal power.

C) DESCRIPTION OF EVENT

On February 1, 2006, control room operators received indication of an increase in radioactivity in the containment atmosphere. Containment entries were performed on February 2nd & 3rd, 2006, to determine the source of increased activity. On February 3, 2006, a robotic camera observed leakage inside the bioshield wall in the area of reactor coolant system (RCS) loop 1. On February 3, 2006, at 1806 EST, Unit 2 was taken to Mode 3 (Hot Standby) to allow further investigation of specific leakage locations. On February 3, 2006, at 2124 EST, RCS pressure boundary leakage was detected at two welded connections on a ¾" bypass line around the Residual Heat Removal (RHR) loop suction valve, 2HV 8701B. The unit was transitioned to Mode 5 (Cold Shutdown) on February 5, 2006, at 0035 EST, to comply with Technical Specification 3.4.13.a. due to RCS pressure boundary leakage. The NRC Operations Center was notified of this condition on February 3, 2006, at 2305 EST.

The two (2) leaking welded connections on the ¾" RHR loop suction valve bypass line included a butt welded elbow downstream of a ¾" globe valve and a ¾" socket welded coupling upstream of a check valve.

D. CAUSE OF EVENT

Although the cause of the weld failures is undetermined at this time, the root cause investigation is still underway. Issues being addressed include high vibration of the bypass line during certain conditions, support design that allowed axial vibration to be amplified in the bypass line, excessive weld stress experienced during heat up and cooldown, and weld quality. The theory presented by the root cause team is the cause of the cracks was from cyclic fatigue. The fatigue appears to be high cycle and is driven by a set of conditions that can and likely do exist on the RHR and Safety Injection (SI) piping. Following additional monitoring, the cause determination will be refined and a revised LER will be submitted by July 11, 2006.

After replacement of the bypass line between the half coupling connection on the RHR pump end and the coupling connection to a flow restrictor on the RCS end of the piping, the failed socket weld at the RHR pump end was not available for inspection. However, forensic examination of the failed butt weld has concluded it failed from a fatigue-type mechanism. This failure mechanism is supported through the presence of striations and the fact that the crack progressed from the inside surface to the outside in nearly a straight line.

The bypass line is supported by 2 horizontal struts and 1 vertical strut attached to the building steel independent of the RHR line, and there is also a vertical Safety Injection System (SI) line in close proximity to the bypass line that may have caused some interference with the insulation. There are no axial supports on the bypass line. The direction of spray from weld leaks is consistent in location and direction with the bypass line not moving to the same extent as the RHR line. The RHR piping expands away from the reactor coolant system as the piping heats up. Thermal growth of the RHR line at the bypass line location is over 1.5". Stress analysis has indicated that if the bypass line is partially restrained from moving with the RHR line, the highest stressed locations are the welds that have cracked.

Contributors to the weld failures are believed to be a combination of stresses due to:

- acoustic vibration (organ pipe) in the RHR line due to RCS flow past the RHR line at a frequency near a structural frequency of the bypass line
- bypass line vibration caused by actions taken in connected systems to mitigate the effects of small leakage past a Safety Injection check valve that excite the normal vibration of the bypass line. The excitation is believed to be caused by steam bubble collapse, or column separation/rejoining in connecting piping, that results in a ¾" Y piston spring check valve beginning to chatter in sync with the acoustic vibration of the RHR line, and a
- piping support configuration with no axial supports for the bypass line that allowed the bypass line to amplify axial vibration of the RHR line, (contributor for high cycle fatigue crack growth)
- lateral supports on the bypass line that could have restrained it from moving axially as far as the RHR line moves, (contributor for low cycle fatigue crack growth)
- weld quality that caused stress concentration points in the root of the original welds and a location for the start of the crack.

Position transducers and accelerometers have been installed that measure vibration at a frequency consistent with the acoustic frequency from RCS flow past the RHR nozzle that is near a natural structural frequency of the bypass line. An evaluation of the measured vibration from instruments installed after the cracks in early February indicates that the vibration alone is not believed to be at sufficient levels to have caused the bypass line welds to crack.

E. ANALYSIS OF EVENT

Operators properly responded to determine the source of the leakage by initiating a plant shutdown and then initiating a unit cooldown to cold shutdown when it was determined to involve the RCS pressure boundary.

Had a complete rupture of either of the ¾" welded piping connections occurred, reactor coolant would have been lost through the ruptured opening after passing through a ¼" flow restrictor. However, the Chemical Volume and Control System (CVCS) is fully capable of making-up this quantity of coolant loss, plus any additional leakage through the 2HV-8701B valve as allowed by Technical Specifications, and the RCS would have retained its full volume and pressure. Based on these considerations, there was no adverse effect on plant safety or on the health and safety of the public as a result of this event.

This event does not represent a safety system functional failure.

F. CORRECTIVE ACTIONS

1) The bypass line piping was replaced with new materials and new welds between the half-coupling connection to the RHR on one end and the coupling connection to the ¼" flow restrictor at the RHR pipe on the other end. This eliminated any fatigue accumulation that may have occurred at these welds.

2) Snubbers and pipe supports in the vicinity of the Loop 1 RHR line and 2HV-8701B bypass line in Unit 2 were inspected and one degraded snubber was replaced. In addition, one of the two lateral pipe supports was removed from the bypass line. This was done to reduce the calculated stresses in the bypass line at key locations.

3) A walkdown of the other RHR bypass line (Loop 4) in Unit 2 resulted in a snubber replacement on the bypass line for 2HV8702B.

4) Temporary instrumentation was installed on both Unit 2 bypass lines to monitor vibration and displacement. The data collection and evaluation is ongoing and may result in recommendations for additional corrective measures.

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

5) To verify the possible extent of condition of the leakage, the other RHR bypass line (Loop 4) in Unit 2 was inspected using dye penetrant and tested by radiography. No leakage or leakage precursors were found.

6) Both RHR bypass lines in Unit 1 were inspected with a robotic camera and no leakage was found.

7) Upon completion of the cause determination, permanent corrective measures will be implemented.

G. ADDITIONAL INFORMATION

1) Failed Components:
None

2) Previous Similar Events:
There has been one previous similar event in the last three years, LER 50-425-2005-003-00, dated January 30, 2006.

3) Energy Industry Identification System Codes:
Reactor Coolant System – AB
Chemical Volume and Control System – CB
Plant Effluent Radiation Monitoring System – IL
Residual Heat Removal System - BP