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SOUTHERN COMPANY
Energy to Serve Your World℠

May 18, 2006

Docket No.: 50-425

NL-06-0995

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

Vogtle Electric Generating Plant - Unit 2
Licensee Event Report 2-2006-002
Reactor Coolant Pressure Boundary Leakage Leads 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 March 20, 2006.

Sincerely,

Don E. Grissette

DEG/RJF/daj

Enclosure: LER 2-2006-002

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	APPROVED BY OMB: NO. 3150-0104 EXPIRES: 06/30/2007
LICENSEE EVENT REPORT (LER) (See reverse for required number of digits/characters for each block)		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
Vogtle Electric Generating Plant – Unit 2	05000-425	1 OF 5

4. TITLE
Reactor coolant pressure boundary leakage leads to shutdown required by the 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
03	20	2006	2006	002	00	5	18	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 Specify in Abstract below or in NRC Form 366A
	20.2203(a)(2)(vi)		50.73(a)(2)(i)(B)		50.73(a)(2)(v)(D)		

12. LICENSEE CONTACT FOR THIS LER

NAME	TELEPHONE NUMBER (Include Area Code)
Tim Schmidt, Performance Analysis	(706) 826-3471

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
	YES *(If yes, complete 15. EXPECTED SUBMISSION DATE)*	X	NO				

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

At 0356 EST and 0603 EST on March 20, 2006, Unit 2 control room operators received alarms indicative of increased radioactivity in containment atmosphere. Sampling revealed the presence of the isotopes Na-24 and Co-58, which indicated the possibility of an active Reactor Coolant System (RCS) leak. On March 20, 2006, at 1300 EST a robotic camera observed leakage inside the bioshield wall in the area of RCS loop 1. The leakage was determined to be RCS pressure boundary leakage located at the ¾" bypass line around the Residual Heat Removal (RHR) loop suction valve, 2HV-8701B, and shutdown to Mode 5 (Cold Shutdown) was initiated and achieved on March 21, 2006 at 1800 EST. Unit 2 was previously in Mode 5 to repair pressure boundary leakage at the 2HV-8701B bypass line on December 10, 2005 and February 5, 2006.

The source of the leakage was a crack in a ¾" socket weld between a flow-restrictor and half-coupling. Several factors caused vibration-induced fatigue, which led to the weld failure: RCS vortex shedding frequency, acoustic frequency of the RHR Loop 1 piping, structural frequency of the 2HV-8701B bypass line, no axial support for the bypass line and performance of an activity to reduce accumulator in-leakage by reseating a Safety Injection (SI) primary hot leg check valve. Prior to returning the unit to power operations, the 2HV-8701B bypass and leak-off lines were removed and the connections plugged. Manual isolation valves adjacent to and upstream of SI test header check valves on Unit 2 Loops 1 and 4 were installed to reduce leakage to the accumulators. Previously installed temporary instrumentation was optimized on Unit 2.

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 March 21, 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

Prior to the start of this event, Unit 2 was in Mode 1 (Power Operations) at 100% rated thermal power. Other than that described herein, there was no inoperable equipment that contributed to the occurrence of this event.

C) DESCRIPTION OF EVENT

Following the Unit 2 refueling outage completed on October 15, 2005, a leaking Safety Injection (SI) primary check valve and SI test header line AOVs led to SI accumulator in-leakage. Reseating the primary check valves was performed to stop accumulator level increase. Prior to Unit 2 being shutdown on December 10, 2005 to repair RCS pressure boundary leakage, this activity was performed multiple times. The leakage occurred at a ¾" socket weld between a ¼" flow restrictor and a ¾" half-coupling on a bypass line around a 12" Residual Heat Removal (RHR) loop suction valve 2HV-8701B. The weld was replaced, with the cause for the failure being attributed to a weld fabrication flaw. Unit 2 returned to Mode 1 (Power Operations) on December 19, 2005.

After December 19, 2005, accumulator in-leakage issues continued, and RCS primary check valve reseating again was performed to stop accumulator level increases. Unit 2 was placed in Mode 5 on February 5, 2006 to repair RCS pressure boundary leakage at the same 2HV-8701B bypass line where a previous leak had occurred. Two leaking connections were discovered on the bypass line, with the first being at a butt welded elbow downstream of a ¾" globe valve and the second being at a ¾" socket welded coupling upstream of a ¾" check valve. The cause for the failures was indeterminate, but believed to be due to acoustic vibration of the RHR line, 2HV-8701B bypass line vibration, piping support configuration for the bypass line, and weld quality.

Temporary instrumentation was installed on the bypass line to monitor vibration and displacement. The entire bypass line was replaced with the exception of the flow-restrictor, full-coupling and two-half-couplings. Weld taper profiles at the flow-restrictor/half-coupling and flow-restrictor/full-coupling were strengthened with a 2:1 profile, following satisfactory dye penetrant testing. Unit 2 returned to Mode 1 on February 15, 2006.

Following unit restart, accumulator in-leakage continued and the primary check valves were reseated once on February 22, 2006. Temporary instrumentation was monitored during this activity and measured vibration levels of the bypass line were found to have increased, but determined to be within

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endurance limits and plant operation continued. At 0356 EST and 0603 EST on March 20, 2006, Unit 2 control room operators received alarms indicative of increased radioactivity in the containment atmosphere. Chemistry sampling revealed the presence of the isotopes Na-24 and Co-58, which indicated the possibility of an active RCS leak. A containment entry was made to determine the source of the increase in radioactivity. On March 20, 2006 at 1300EST a robotic camera observed leakage inside the bioshield wall in the area of RCS loop 1.

The leakage was determined by an inspection team to be RCS pressure boundary leakage located at the same 2HV-8701B bypass line where previous leaks had occurred, and shutdown to Mode 5 was initiated. Unit 2 was placed in Mode 3 (Hot Standby) at 1655 EST on March 20, 2006 and in Mode 5 on March 21, 2006 at 1800 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 March 20, 2006 at 1551 EST.

A closer inspection of the bypass line determined that the leak occurred in a ¾" socket weld between a ¼" flow restrictor and a ¾" ANSI 6000 lb. half-coupling. The ¾" half-coupling is drilled for a 3/8" flow opening. The ¾" piping is part of a bypass line used to equalize pressure on either side of 2HV-8701B.

Dye penetrant testing revealed a circumferential crack which ran approximately from the 3 o'clock to the 4 o'clock position. The crack was approximately 5/8" in length and approximately 3/16" from the toe of the 2:1 weld reinforcement. Forensic analysis revealed that the RHR bypass line socket to flow restrictor weld most likely failed due to a fatigue mechanism and was already cracked when the 2:1 reinforcement weld was added.

D. CAUSE OF EVENT

A detailed analytical model was constructed to analyze the conditions and stresses in the effected piping. Several factors identified must be simultaneously present to create vibration severe enough to result in the observed cracking of the 2HV-8701B bypass line welded connection:

- Data has shown that the RCS vortex shedding frequency across the RHR nozzle at the RCS, acoustic frequency of the Loop 1 RHR piping, and structural frequency of the 2HV-8701B bypass line had approximately the same resonance frequency. Alone, this resonance condition was within calculated endurance limits of the 2HV-8701B bypass line. In addition, with there not being any axial support for the 2HV-8701B bypass line, the resonance condition was not dampened. Even with this condition, calculated stresses were still within endurance limits.

- Reseating of the hot leg SI primary check valve on Loop 1 to mitigate accumulator level increases is believed to have resulted in an impulse at the 2HV-8701B bypass line and initiating valve chatter with the 2HV-8701B bypass line check valve.

With the stated frequencies having approximately the same resonance frequency and lack of axial support for the bypass line, the added stimuli of the SI primary check valve reseating caused piping to vibrate with enough amplitude to exceed endurance limits, resulting in the cracked socket weld. Data collected on the Loop 4 RHR and bypass lines for Unit 2 determined that a resonance condition does not exist, and that the lack of axial support for this bypass line is not a concern.

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 the ¾" connection occurred, coolant would have been lost through the 3/8" opening of the half-coupling. 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. 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 entire 2HV-8701B bypass line and bonnet leak-off lines were removed and respective ¾" and ½" connections were plugged. With the bypass line removed, no additional corrective action involving the lack of axial support is necessary for Loop 1 on Unit 2.
2) Previously installed temporary instrumentation on Loops 1 and 4 RHR and bypass lines on Unit 2 was refined, along with a monitoring plan that included established limits and prescribed actions. Data collection and evaluation is on-going and may result in recommendations for additional corrective actions. Applicable procedures were revised to restrict primary hot leg check valve reseating on Unit 2.
3) Dye penetrant and ultra-sonic testing was performed on a 12" x 12" x 6" reducing tee, which connects the SI hot leg injection line to the loop 1 RHR line on Unit 2 with acceptable results. This testing was done to address broadness aspects and extent of condition.
4) Dye penetrant testing was performed on select small-bore welded connections on the loop 1 SI hot leg injection line on Unit 2 with acceptable results. This testing was done to address broadness aspects and extent of condition.
5) Supports and snubbers in the vicinity of the loop 1 RHR line on Unit 2 and adjacent SI hot leg injection line were inspected with satisfactory results.

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

6) Manual isolation valves adjacent to and upstream of SI test header check valves on Unit 2 Loops 1 and 4 were installed to reduce leakage to the accumulators.

G. ADDITIONAL INFORMATION

1) Instrumentation was also subsequently installed on Loops 1 and 4 RHR and bypass lines on Unit 1. Data collection and evaluation is on-going and may result in recommendations for additional corrective actions. Applicable procedures were revised to restrict primary hot leg check valve reseating on Unit 1.

2) Options are being considered for both Unit 1 and 2 to determine a permanent resolution to the repeated RHR bypass line failures on Unit 2.

3) Failed Components:
None

4) Previous Similar Events:
There have been two previous similar events in the last three years: LER 50-425-2005-003-00, dated January 30, 2006 and LER 50-425-2006-001-00, dated April 6, 2006.

5) 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