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SOUTHERN COMPANY
Energy to Serve Your World℠

April 5, 2005

Docket No.: 50-366

NL-05-0611

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

Edwin I. Hatch Nuclear Plant – Unit 2
Licensee Event Report
Inboard and Outboard Main Steam Isolation Valves Fail Local Leak Rate Test

Ladies and Gentlemen:

In accordance with the requirements of 10 CFR 50.73(a)(2)(ii), Southern Nuclear Operating Company (SNC)is submitting the enclosed licensee event report (LER) concerning a failed Local Leak Rate Test on both main steam isolation valves on the 'A' main steam line.

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

Sincerely,

Lewis Sumner

H. L. Sumner, Jr.

HLS/OCV/sdl

Enclosure: LER 50-366/2005-001

cc: Southern Nuclear Operating Company
Mr. J. T. Gasser, Executive Vice President
Mr. G. R. Frederick, General Manager – Plant Hatch
RTYPE: CHA02.004

U. S. Nuclear Regulatory Commission
Dr. W. D. Travers, Regional Administrator
Mr. C. Gratton, NRR Project Manager – Hatch
Mr. D. S. Simpkins, Senior Resident Inspector – Hatch

IE22

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)

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4. TITLE
Both Inboard and Outboard Main Steam Isolation Valves for the 'A' Penetration Failed Local Leak Rate Test

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
2	20	2005	2005	001	0	4	8	2005	FACILITY NAME	DOCKET NUMBER(S) 05000

9. OPERATING MODE: Mode 5

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)	X	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 / Kathy A. Underwood, Performance Analysis Supervisor	(912) 537-5931

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
X	SB	SHV	R344	Yes						

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)

On 2/20/2005 at 1435 ET, Unit 2 was in the Refuel mode with fuel in the vessel and the reactor cavity flooded for refueling operations. At that time, plant engineers and technicians were performing Local Leak Rate Testing (LLRT) on Primary Containment Isolation Valves (PCIVs) when it was discovered that two Main Steam Isolation Valves (MSIVs) had failed their associated LLRT. Both valves are located in the same penetration involving the 'A' Main Steam Line. The leakage acceptance criteria for the MSIVs are contained in the plant's Technical Specifications.

The primary cause of the failure for the inboard (2B21-F022A) MSIV appears to be degradation over a period of time. The failure of the outboard (2B21-F028A) MSIV appears to be the methodology used for testing the valve. Corrective actions for this event included repairing the valves and performing another LLRT with successful results. Finally, additional procedural guidance is being provided for the performance of Local Leak Rate Testing when using blocking pressure against the Main Steam Line plugs as a course of action to prevent recurrence.

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17. 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 2/20/2005 at 1435 ET, Unit 2 was in the Refuel mode with fuel in the vessel and the reactor cavity flooded for refueling operations. At that time, engineers and technicians were performing Local Leak Rate Testing (LLRT) on valves 2B21-F022A and 2B21-F028A per Unit 2 Technical Specifications surveillance requirement (SR) 3.6.1.3.11. This SR addresses the leakage restrictions through the Main Steam Isolation Valves (MSIVs, EIIS Code SB). The MSIVs have specific leakage rates established in the plant's Technical Specifications to ensure that the assumptions of the safety analysis are met. The maximum leakage rate allowed for all of the Main Steam Lines is 250 standard cubic feet per hour (SCFH). The as found measured LLRT leakage for 2B21-F022A and 2B21-F028A was greater than 30,000 actual cubic centimeters (ACCM) (off scale of the measuring device being used).

The test that failed was performed by pressurizing the space between the inboard (2B21-F022A) and outboard (2B21-F028A) valves with the piping upstream of the inboard valve vented. All four Main Steam Line plugs were installed and the piping upstream of the inboard isolation valves was drained and vented. The Main Steam Lines at Plant Hatch are all interconnected with a common vent line. This vent ties all of the inboard Main Steam Lines together and is not isolable. During this failed test, the vent line was open and an inflatable device was placed over the vent line opening to determine whether the inboard (2B21-F022A) valve was leaking by during the test. The inflatable device did not exhibit any evidence of leakage (i.e., did not inflate) during this test and therefore the test leakage was conservatively attributed to the outboard (2B21-F028A) valve. Because this test did not absolutely eliminate the potential that there was leakage past valve 2B21-F022A (e.g., leakage through the other main steam lines could cause the inflatable device to not inflate), it was proposed that blocking pressure be placed upstream of the inboard valve and the test be performed again. The Local Leak Rate Testing procedure (42SV-TET-001-0) provides guidance to ensure that 2B21-F022A was not leaking. The procedural guidance provided states:

> IF any one of the four leakage values is unacceptable, the reactor water level is raised up to approximately 692" above vessel zero. By doing this, a water head in the steam lines equivalent to the test pressure can be produced, thereby negating leakage through the inboard MSIVs. By performing another LLRT on each set of MSIVs whose initial test was unacceptable, the outboard MSIV test leakage can be determined. The value can then be compared to the previous test results to determine the inboard MSIV leakage rates. This test is done for diagnostic purposes only unless the volume is drained after the test AND less than 1 gallon of water has accumulated. The volume of water drained will be recorded in the comments section of the data sheet.

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A decision was made to not retest this valve using this portion of the procedure (this would have required the removal of the Main Steam Line plugs) and instead all of the test leakage was attributed to the outboard valve. After this decision was made, the outboard valve was subsequently disassembled and repaired. It was noted during the repair of the outboard valve (2B21-F028A) that there were no obvious defects found with the valve internals that explained the LLRT failure. After the repair of the outboard valve, it was retested with blocking pressure (using air) applied between the inboard MSIV and the MSL plugs. The LLRT procedure did not explicitly contain steps for applying blocking pressure with the MSL plugs in place, consequently, the instructions were written into the procedure data package. The results of this test indicated that there was 0 SCFH leakage through valve 2B21-F028A.

After the outboard valve was repaired and determined to have 0 SCFH leakage, the inboard MSIV was tested by applying pressure between the inboard and the outboard valve without blocking pressure applied upstream of the inboard valve. The required test pressure could not be maintained and the flow meter indicated leakage of approximately 28,000 ACCM. There were no activities performed that adequately explained why the 2B21-F022A valve was leaking during this test if it was not leaking during the previous test. The inboard valve was stroked and the test was again performed by applying pressure between the inboard and outboard valves without blocking pressure applied upstream of the inboard valve, with similar results. The valve was subsequently disassembled and repaired. The amount of leakage measured during the 2003 refueling outage LLRT for valve 2B21-F022A, while within acceptance criteria limits, was higher than the other MSIVs. It was noted during the repair of the inboard valve (2B21-F022A) that the seats had indications 180 degrees apart (i.e., low areas in the seating area) and it was concluded that the valve had degraded over a period of time. After the repair of the inboard valve, it was retested and found to have 0 SCFH leakage through the valve.

Additional testing for the other Main Steam Lines found that the 'B' line had a leakage rate of 13.9 SCFH (after the outboard valve's packing had been tightened), 'C' line had a leakage rate of 13.9 SCFH (after the outboard valve's packing had been tightened) and the minimum pathway leakage through penetration 10 (for valves 2E51-F007 and 2E51-F008 (EIIS Code BN)) was 662 ACCM (approximately 6 SCFH). All of these leakage paths were connected to the vent line that was initially used to determine whether or not valve 2B21-F022A was leaking. These additional leakage paths could explain why the inflatable device did not inflate during the original leakage test on the 'A' MSL.

CAUSE OF EVENT

The primary cause of the failure of the inboard 2B21-F022A valve appears to be degradation of the valve over a period of time. The last successful LLRT for this valve indicated that leakage (while within limits) had increased through this valve and that the valve was degrading. The primary cause of the failure of the outboard valve was concluded to be the method used to test the valve. Subsequent to the valve repair, it was determined that the inboard valve was leaking and the method used to determine that the inboard valve was not leaking did not account for the vent paths through the 'B' MSL, 'C', MSL, and the 2E51-F007 and F008 valves. Furthermore, the procedure used for LLRT had provisions for applying blocking pressure by raising water level but did not establish criteria for using blocking pressure against the MSL plugs.

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REPORTABILITY ANALYSIS AND SAFETY ASSESSMENT

This event is reportable per 10 CFR 50.73 (a)(2)(ii) because an event occurred which resulted in the degradation of one of the plant's principal safety barriers. Specifically, the 'A' Main Steam line minimum pathway leakage exceeded the allowable leakage established by the plant's Technical Specifications.

The function of the Primary Containment is to isolate and contain fission products released from the Reactor Primary System following a design basis accident (DBA) and to confine the postulated release of radioactive material. The Primary Containment consists of a steel vessel which surrounds the Reactor Primary System and provides a barrier against the uncontrolled release of radioactive material to the environment. Some leakage from the Primary Containment is assumed to occur, although the majority of the leakage is assumed to be released into the Secondary Containment. The total allowable leakage rate for the Primary Containment is designated L_a and is equal to 1.2 percent by weight of the contained air volume per day. For Plant Hatch Unit 2, this equates to a total allowable leakage of 61,000 ACCM, most of which is assumed to occur within the Secondary Containment where it will be treated by the Standby Gas Treatment System (EIIS Code BH) before being released at an elevated point through the Main Stack (EIIS Code VL).

The Main Steam Lines lead outside of Secondary Containment and have their own specific limits for leakage established in the plants Technical Specifications of 250 SCFH maximum pathway leakage for all four steam lines. The leakage rates measured in this event were greater than this amount. The allowable leakage for the Main Steam Lines has been factored into the plant's safety analysis.

Primary Containment leakage criteria was established using conservative licensing basis evaluation methods in accordance with NRC Regulatory Guide 1.3. These methods conservatively assume that the postulated accident results in fuel damage with 100 percent of the core noble gas activity and 50 percent of the iodine activity released. Consequently, the leakage rate predicted for the valves based on the results of the flawed LLRT would likely have resulted in exceeding the values set forth in 10 CFR 100 during a postulated design basis accident that assumes fuel damage per NRC Regulatory Guide 1.3.

The Final Safety Analysis Report (FSAR) for Plant Hatch Unit 2 designates the Design Basis Accident (DBA) as the break of a Reactor Recirculation System (EIIS Code AD) pipe which results in the rapid depressurization of the reactor vessel to the Primary Containment. However, the FSAR analysis shows that, for such an accident, resulting peak fuel cladding temperatures would be less than those required to produce damage to the fuel. The plant-specific SAFER/GESTR analysis for this accident scenario shows that no damage to the fuel cladding would occur even if additional failures are postulated, such as failures of certain power supplies and certain low pressure emergency core cooling systems. Therefore, by this analysis, the only radioactive materials present in the released coolant would be those already present due to normal operation and the small additional amount of contaminated or activated crud released from vessel internals and primary system piping during the initial stages of the transient. Realistically therefore, the 10 CFR 100 off-site dose limits would likely not have been exceeded during an actual event.

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Based on this analysis contained in the FSAR, it is concluded that this event did not result in any adverse impact on nuclear safety. This analysis applies to all operating conditions.

CORRECTIVE ACTIONS

Valves 2B21-F022A and 2B21-F028A were disassembled and repaired. The failure of both MSIVs in one penetration to pass LLRT has not occurred within the last ten years of plant operation. Therefore, the repair of the valves and subsequent successful LLRT is considered to be a satisfactory solution.

The Local Leak Rate Testing procedure for both Unit 1 and Unit 2 will be revised to include the requirements for using air to provide blocking pressure between the Main Steam Line plugs and the inboard MSIVs. These procedure revisions will be completed before they are needed to be used during the next respective Unit's refueling outage.

ADDITIONAL INFORMATION

No systems other than those already mentioned in this report were affected by this event.

Failed Component Information:

Master Parts List Number: 2B21-F022A&F028A
Manufacturer: Rockwell International
Model Number: 1612 JM MNTY
Type: Valve, Shutoff
Manufacturer Code: R344

EIIS System Code: SB
Reportable to EPIX: Yes
Root Cause Code: X
EIIS Component Code: SHV

Previous Similar Events: No events have been reported in the past two years in which both the inboard and outboard Main Steam Isolation Valves for a single penetration failed LLRT.

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