H. L. Sumner, Jr.
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SOUTHERN COMPANY
Energy to Serve Your World℠

April 15, 2005

Docket Nos.: 50-366

NL-05-0690

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

Edwin I. Hatch Nuclear Plant – Unit 2
Licensee Event Report
Secondary Containment Bypass Leakage Exceeded

Ladies and Gentlemen:

In accordance with the requirements of 10 CFR 10 CFR 50.73(a)(2)(ii), Southern Nuclear is submitting the enclosed Licensee Event Report (LER) concerning a failed Local Leak Rate Test which resulted in secondary containment bypass leakage being exceeded.

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

Sincerely,

H. L. Sumner, Jr.

HLS/OCV/daj

Enclosures: LER 2-2005-002

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)

1. FACILITY NAME	2. DOCKET NUMBER	3. PAGE
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4. TITLE
Secondary Containment Bypass Leakage Requirements Exceeded

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
3	2	2005	2005	002	0	4	15	2005	FACILITY NAME	DOCKET NUMBER(S) 05000

9. OPERATING MODE	11. THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR § : (Check all that apply)							
Mode 5		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)
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)
0%		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	BN	SHV	P305	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 3/02/2005 at 1500 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 Reactor Core Isolation Cooling (RCIC) valves had failed their associated LLRT. Both of these valves are located in the same penetration and are associated with Secondary Containment bypass leakage. The leakage acceptance criteria for Secondary Containment bypass are contained in the plant's Technical Specifications.

The primary cause of the failures for the inboard (2E51-F007) RCIC valve, and the outboard (2E51-F008) RCIC valve appears to be degradation over a period of time. Corrective actions for this event included repairing the valves by installing new wedges and performing another LLRT with successful results.

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 3/2/2005 at 1500 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 2E51-F008 and 2E51-F007 (EIIS Code BN) for penetration number 10. The plants Technical Requirements Manual (TRM) table T7.0-1 identifies primary containment penetrations and references applicable notes for the penetration barriers. Note 28 for Table T7.0-1 is used to identify penetrations that are required to meet the leakage requirements for Secondary Containment bypass penetrations. The present version of the TRM does not reference Note 28 for penetration number ten. However, Corporate Licensing has determined that the piping configuration for this penetration should be included as part of Secondary Containment bypass leakage. The Unit 2 Technical Specifications surveillance requirement (SR) 3.6.1.3.10 addresses the leakage restrictions for Secondary Containment bypass valves. Secondary Containment bypass valves 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 Secondary Containment bypass valves is 0.009 L_a (or approximately 544 ACCM) per SR 3.6.1.3.10.

Leakage through 2E51-F007 and 2E51-F008 was measured at 662 ACCM, which exceeded the allowable leakage established by the plant's Technical Specifications for Secondary Containment bypass leakage. These valves were disassembled and repaired. Both of the seats of these valves were observed to be relatively flat (i.e., the seating surface was larger than what is normally desirable) and this was considered to be caused by normal wear. The valves were repaired by installing new wedges.

CAUSE OF EVENT

The cause of the 2E51-F007 and 2E51-F008 valves leakage exceeding the allowable leakage established by the plant's Technical Specifications for Secondary Containment bypass leakage is considered to be most likely normal wear.

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 RCIC penetration number ten exceeded the allowable leakage established by the plant's Technical Specifications.

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

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).

Additionally, there is also some amount of "Secondary Containment bypass" leakage assumed to occur outside Secondary Containment where it is released without being treated by the SBGT system. Valves located in Primary Containment penetrations whose pipes lead outside the Secondary Containment are potential sources of such untreated leakage, so these valves are termed "Secondary Containment bypass valves." Since the atmospheres in such areas would not be treated by the SBGT system, the allowable leakage through these valves is specifically addressed by the Technical Specifications, and is limited to a total of 544 ACCM. The 662 ACCM leakage through 2E51-F007 and 2E51-F008 exceeded the allowable leakage established by the plant's Technical Specifications for Secondary Containment bypass leakage.

The allowable leakage for Secondary Containment bypass valves 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 actual measured leakage of the valves identified in this report 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.

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.

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

CORRECTIVE ACTIONS

Valves 2E51-F007 and 2E51-F008 were repaired by installing new wedges.

The information necessary to identify penetration ten as a potential bypass leakage path will be added to the Units 1 and 2 Technical Requirements Manual. This will be done by August 15, 2005.

ADDITIONAL INFORMATION

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

Failed Component Information:

Master Parts List Number: 2E51-F007&F008
Manufacturer: Powell
Model Number: 73600
Type: Valve, Shutoff
Manufacturer Code: P305

EIIS System Code: BN
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 the plant exceeded the Technical Specification limits for Secondary Containment bypasses leakage.

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