D. R. Madison (Dennis)
Vice President - Hatch

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SOUTHERN COMPANY
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April 30, 2007

Docket No.: 50-366

NL-07-0873

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

Edwin I. Hatch Nuclear Plant – Unit 2
Licensee Event Report 2-2007-003
Excessive Leakage on
Secondary Containment Bypass Valves Due to Foreign Material

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 concerning secondary containment bypass valves that failed local leak rate testing due to foreign material intrusion.

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

Sincerely,

D. R. Madison
Vice President – Hatch
Edwin I. Hatch Nuclear Plant
11028 Hatch Parkway North
Baxley, GA 31513

DRM/CLT/daj

Enclosure: LER 2-2007-003

cc: Southern Nuclear Operating Company
Mr. J. T. Gasser, Executive Vice President
Mr. D. H. Jones, Vice President – Engineering
RTYPE: CHA02.004

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

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
Excessive Leakage on Secondary Containment Bypass Valves Due to Foreign Material

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	01	2007	2007	003	00	04	30	2007	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)							
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)
		20-2203(a)(2)(iii)		50.36(c)(2)		50.73(a)(2)(v)(A)		73.71(a)(4)
000		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 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

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	MANUFACTURER	REPORTABLE TO EPIX		CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO EPIX
X	WK	SHV	C635	No		X	WK	SHV	C635	No

14. SUPPLEMENTAL REPORT EXPECTED			15. EXPECTED SUBMISSION DATE	MONTH	DAY	YEAR
	YES (If yes, complete 15. EXPECTED SUBMISSION DATE)	NO				

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

On 3/1/2007, Unit 2 was in the Refuel mode with fuel in the reactor vessel. Personnel were performing local leak rate testing (LLRT) on the drywell floor drain sump isolation valves. These tests determined that the minimum pathway leakage associated with containment penetration isolation valves 2G11-F003 and 2G11-F004 had exceeded the associated LLRT acceptance criteria. The valves are classified as secondary containment bypass valves since the drain piping is routed to the Radwaste Building which is outside the secondary containment boundary and is, consequently, not served by the standby gas treatment system. The cause of the valve test failures was foreign material intrusion.

Corrective actions for this event included removing the foreign material, cleaning and inspecting the valves, adjusting the pneumatic operator on 2G11-F003, and re-testing the valves. Cleaning and inspection of the drywell floor drain sumps was conducted to provide further assurance that measures taken for foreign material exclusion were effective. An outage repetitive task was also created to ensure cleaning and inspection of the drywell floor drain sumps is performed during each refueling outage.

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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 3/1/2007, Unit 2 was in the Refuel mode with fuel in the reactor vessel. At that time, engineers and technicians were performing local leak rate testing (LLRT) on the drywell floor drain sump primary containment isolation valves (PCIVs) 2G11-F003 and 2G11-F004. These tests determined, for the containment penetration serviced by those valves, that the minimum pathway leakage exceeded the LLRT acceptance criteria for that pathway. This penetration leads from the drywell floor drain sump through the secondary containment to the Radwaste Building (EIIS Code NE). Because the piping terminates in an area not served by the standby gas treatment (SGT, EIIS Code BH) system, these valves are designated as "secondary containment bypass valves." These valves are the subject of a special leak rate limitation per Unit 2 Technical Specifications.

When the LLRT was performed on 2G11-F003, the as-found leakage rate was 5359 actual cubic centimeters per minute (accm). When valve 2G11-F004 was tested, the as-found leakage rate was greater than 30,000 accm. The total leakage limit for all secondary containment bypass valves is 544 accm. Therefore, the minimum pathway leakage through this penetration of 5359 accm exceeded the allowable value for all valves tested under this surveillance requirement.

The valves were disassembled and repaired. As-left LLRT testing was performed on 3/7/2007 with each valve leaking 20 accm after repairs. The sump was cleaned at the conclusion of the outage to remove foreign material that potentially could be drawn into these valves and hinder their ability to perform their isolation function.

CAUSE OF EVENT

The root cause analysis for this event concluded that the primary cause of the LLRT failure for these valves was due to foreign material found in the valves.

REPORTABILITY ANALYSIS AND SAFETY ASSESSMENT

This event is reportable per 10 CFR 50.73 (a)(2)(ii)(A) because an event occurred which resulted in one of the plant's principal safety barriers being degraded. Specifically, the primary containment isolation function involving secondary containment bypass valves was found to not satisfy the leakage requirements of the 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 sub 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 60,431 accm, most of which is assumed to occur within the secondary containment where it would be treated by the SGT system before being released at an elevated point through the main stack (EIIS Code VL). However, some small amount of leakage is assumed to occur outside secondary containment where it is released without being treated by the SGT 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 filtered by the SGT system, the allowable leakage through these valves is specifically addressed by the Technical Specifications, and is limited to a total of 544 accm. The leakage rates measured in this event were greater than this amount.

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 10CFR100 during a postulated DBA that assumes fuel damage per NRC Regulatory Guide 1.3.

The Final Safety Analysis Report (FSAR) for Plant Hatch Unit 2 designates the 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

Both valves 2G11-F003 and 2G11-F004 were disassembled and repaired. Post-maintenance LLRT testing was performed to demonstrate that the as-left leakage characteristics were within applicable LLRT acceptance criteria.

The drywell floor drain sump was cleaned and inspected to remove foreign material. Floor drain system piping was flushed to radwaste by performing a loop check on the drywell floor drain instrumentation.

An outage repetitive task was created to ensure cleaning and inspection of the drywell floor drain sumps is performed during each refueling outage to preclude introduction of objects into the drywell floor drain sump that could affect operation of the PCIVs.

ADDITIONAL INFORMATION

Other Systems Affected: No other systems were affected by this event.

Failed Components Information:

Master Parts List Number: 2G11-F003
Manufacturer: Copse Vulcan
Model Number: 0-100-16
Type: Valve, Cutoff
Manufacturer Code: C635

EIIS System Code: WK
Reportable to EPIX: No
Root Cause Code: X
EIIS Component Code: SHV

Master Parts List Number: 2G11-F004
Manufacturer: Copse Vulcan
Model Number: 0-100-16
Type: Valve, Cutoff
Manufacturer Code: C635

EIIS System Code: WK
Reportable to EPIX: No
Root Cause Code: X
EIIS Component Code: SHV

Commitment Information: There are no new permanent regulatory commitments in this LER.

Previous Similar Events:
LER 2-2005-002 documents a similar event for valves in pathways that potentially could bypass secondary containment where both the inboard and outboard valves on the line failed the LLRT testing. The valves in that event are in a different system than the valves reported by this LER. Corrective actions for that event repaired the applicable valves.