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SOUTHERN COMPANY
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April 21, 2000

Docket No. 50-366 HL-5927

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

Edwin I. Hatch Nuclear Plant - Unit 2
Licensee Event Report
Excessive Leakage Identified on Secondary Containment Bypass Valves

Ladies and Gentlemen:

In accordance with the requirements of 10 CFR 50.73(a)(2)(ii), Southern Nuclear Operating Company is submitting the enclosed Licensee Event Report (LER) concerning excessive leakage identified on secondary containment bypass valves.

Respectfully submitted,

Lewis Sumner

H. L. Sumner, Jr.

OCV/eb

Enclosure: LER 50-366/2000-004

cc: Southern Nuclear Operating Company
Mr. P. H. Wells, Nuclear Plant General Manager
SNC Document Management (R-Type A02.001)

U.S. Nuclear Regulatory Commission, Washington, D.C.
Mr. L. N. Olshan, Project Manager - Hatch

U.S. Nuclear Regulatory Commission, Region II
Mr. L. A. Reyes, Regional Administrator
Mr. J. T. Munday, Senior Resident Inspector - Hatch

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NRC FORM 366 U.S. NUCLEAR REGULATORY COMMISSION
(06-1998)

LICENSEE EVENT REPORT (LER)

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

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TITLE (4)
Excessive Leakage Identified on Secondary Containment Bypass Valves

EVENT DATE (5)			LER NUMBER (6)			REPORT DATE (7)			OTHER FACILITIES INVOLVED (8)	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	MONTH	DAY	YEAR	FACILITY NAME	DOCKET NUMBER(S) 05000
03	27	2000	2000	004	00	04	21	2000		DOCKET NUMBER(S) 05000

OPERATING MODE (9): 5
POWER LEVEL (10): 0

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more) (11)

	20.2201(b)		20.2203(a)(2)(v)		50.73(a)(2)(i)		50.73(a)(2)(vii)
	20.2203(a)(1)		20.2203(a)(3)(i)	X	50.73(a)(2)(ii)		50.73(a)(2)(ix)
	20.2203(a)(2)(i)		20.2203(a)(3)(ii)		50.73(a)(2)(iii)		73.71
	20.2203(a)(2)(ii)		20.2203(a)(4)		50.73(a)(2)(iv)		OTHER
	20.2203(a)(2)(iii)		50.36(c)(1)		50.73(a)(2)(v)		Specify in Abstract below or in NRC Form 366A
	20.2203(a)(2)(iv)		50.36(c)(2)		50.73(a)(2)(vii)		

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER (Include Area Code)
Steven B. Tipps, Nuclear Safety and Compliance Manager, Hatch	(912) 367-7851

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

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

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

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

On 3/27/2000 at 0800 EST, Unit 2 was in the Refuel mode with the reactor vessel head in place with the head bolts not yet tensioned. At that time, the plant engineer responsible for local leak rate testing (LLRT) on primary containment isolation valves (PCIVs) determined that two secondary containment bypass valves, 2G11-F003 and 2G11-F004, had failed their associated LLRT. Both valves are located in the same penetration which exits the drywell floor drain sump. The valves are classified as secondary containment bypass valves as the drain piping is routed to the Radwaste Building which is not inside the secondary containment and, consequently, not served by the standby gas treatment system. The cause of the failure of valve 2G11-F004 was foreign material intrusion. A nylon tie-wrap and a piece of paper were found on the valve seat upon disassembly. The cause of the failure of 2G11-F003 could not be conclusively determined. The possibility could not be ruled out that foreign material also caused the failure of 2G11-F003. Corrective actions for this event included removing the foreign material from 2G11-F004, adjusting the pneumatic operator on 2G11-F003, re-testing the valves, and performing cleaning and inspections to ensure that measures taken for foreign material exclusion were as effective as possible to preclude further intrusion of objects which could affect operation of the PCIVs.

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 3/27/2000 at 0800 EST, Unit 2 was in the Refuel mode with the reactor vessel head in place but the head bolts not yet tensioned. At that time, the engineer responsible for local leak rate testing (LLRT) reported to licensed personnel that the Unit 2 drywell penetration leading from the drywell floor drain sump had failed its LLRT requirements. This penetration contains two valves, 2G11-F003 and 2G11-F004, and 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 special leak rate leak rate limitations per Unit 2 Technical Specifications surveillance requirement 3.6.1.3.10.

When the LLRT was performed on 2G11-F003, the valve failed to pressurize at a flow rate of greater than 30,000 actual cubic centimeters per minute (ACCM). Hence, the actual leakage rate could not be measured. When valve 2G11-F004 was tested, the initial leakage rate was 7520 ACCM as compared to a leakage limit of 544 ACCM for all secondary containment bypass valves. Therefore, the leakage through this penetration was 7520 ACCM, which exceeded the allowable for all valves tested under this surveillance requirement.

CAUSES OF EVENT

The cause of the failure of valve 2G11-F004 was determined to be foreign material that came to rest on the valve seat, preventing the valve from getting to the full-closed position. This material consisted of one nylon tie-wrap and a piece of paper. Investigators were not able to determine the source of the material because these are common materials used in many different kinds of work in areas whose drainage passes through these valves. When the material was removed and the valve was re-tested after assembly, it showed no leakage.

The cause of the failure of 2G11-F003 could not be conclusively determined. The pneumatic valve actuator was adjusted to increase the spring force from 23.5 lb_f to 28 lb_f, and this adjustment involved stroking the valve several times. Upon re-test, the valve showed zero leakage. SNC valve technicians do not believe that increasing the closing force accounted for the dramatic change in performance; therefore, it is considered more likely that a piece of foreign material was under the valve seat and was moved by stroking the valve. It is probable that the material originated in the drywell and entered the sump through the floor drain system. It is possible, though less likely, that the material could have entered the piping through a common connection with the reactor building sump and moved to the valve area. This is considered less likely, however, because check valves in the drywell floor drain piping prevent flow from moving toward the drywell

After both valves were initially tested with the results as previously stated, valve 2G11-F003 was stroked several times in the process of improving the bench set on the spring in the pneumatic actuator. Subsequently, valve 2G11-F004 was re-tested and showed significantly worse performance than was observed during the initial as-found test. Therefore, it is believed that the foreign material was initially under the valve seat of 2G11-F003, and during valve maintenance it migrated to 2G11-F004 where it was found upon disassembly.

REPORTABILITY ANALYSIS AND SAFETY ASSESSMENT

This event is reportable per 10 CFR 50.73 (a)(2)(ii) 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 not to satisfy the leakage requirements of Technical Specifications surveillance requirement 3.6.1.3.10.

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 36,244 ACCM, most of which is assumed to occur within the secondary containment where it will 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

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

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 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. If it is conservatively assumed that all reactor coolant released into the primary containment drains through valves 2G11-F003 and 2G11-F004 into the Unit 2 Radwaste Building where is it released to the environment (neglecting plateout), this would represent an unfiltered, ground level release of all the radioactive materials present in the coolant at the time of the accident. Under these extremely conservative assumptions, calculations performed by the Architect/Engineer show that the dose rate at the site boundary would still be very much less than the limits prescribed by 10 CFR 100.

CORRECTIVE ACTIONS

1. Both valves 2G11-F003 and 2G11-F004 have been given appropriate maintenance and testing and have passed LLRT. This action is complete.

2. The drywell floor drain sump has been cleaned and inspected to ensure no foreign materials remained in it which could hinder operation of the PCIVs. This cleaning included the "subpile" room beneath the reactor. This action is complete.

3. Check valves 2G11-F001A and 2G11-F001B, which are upstream of the PCIVs in their respective lines, have been breached and inspected to ensure that no foreign materials remain inside these valves. No foreign materials were found. This action is complete.

4. The drywell was cleaned and inspected prior to the end of the refueling outage. This action is complete.

5. In future drywell entries when work is performed which poses the risk of generating foreign material that could impede operation of the floor drain sump valves, a fine mesh will be placed over drywell floor drain gratings.

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ADDITIONAL INFORMATION

1. Other Systems Affected: No systems were affected by this event other than those which have already been mentioned in this report.

2. Failed Components Information: No failed components contributed to or resulted from this event.

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

4. Previous Similar Events: Two events have been reported since 1995 involving similar failures of secondary containment bypass valves. These events are described in LER 50-366/1995-004, dated 11/08/1995, and 50-366/1997-003, dated 04/22/1997. The corrective actions for the 1995 event included refurbishing 2G11-F003 and replacing 2G11-F004 with a new valve of identical design. The corrective actions for the 1997 event included a major design change in which gate valves were replaced with the current globe valves and the piping configuration was altered to provide a better orientation for the valve operators. These corrective actions would not have prevented this event because the valves could not provide satisfactory isolation performance with foreign material on the valve seats regardless of valve design or orientation.