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

Southern Nuclear Operating Company, Inc.
Plant Edwin I. Hatch
11028 Hatch Parkway, North
Baxley, Georgia 31513

Tel 912.537.5859
Fax 912.366.2077

April 11, 2007

SOUTHERN COMPANY
Energy to Serve Your World™

Docket No.: 50-366

NL-07-0707

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

Edwin I. Hatch Nuclear Plant – Unit 2
Licensee Event Report 2-2007-001
Main Steam Isolation Valves Fail Local Leak Rate Testing
Due to Out-of-Specification Internal Tolerances

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 main steam isolation valves that failed local leak rate testing due to out-of-specification internal tolerances.

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

Sincerely,

Dennis Madison

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

DRM/CLT/daj

Enclosure: LER 2-2007-001

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)

1. FACILITY NAME	2. DOCKET NUMBER	3. PAGE
Edwin I. Hatch Nuclear Plant Unit 2	05000366	1 OF 5

4. TITLE
Main Steam Isolation Valves Fail Local Leak Rate Testing Due to Out of Specification Internal Tolerances

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
02	12	2007	2007	001	00	04	11	2007	FACILITY NAME	DOCKET NUMBER(S) 05000

9. OPERATING MODE: 5

10. POWER LEVEL: 000

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 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*)		NO				

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

On February 12, 2007 Hatch Unit 2 was in the refueling mode with fuel in the vessel and the reactor cavity flooded for refueling operations. Personnel were performing local leak rate testing (LLRT) on the Main Steam Isolation Valves (MSIVs). These tests determined, for both the 'B' penetration (2B21-F022B (inboard) and 2B21-F028B (outboard)) and the 'C' penetration (2B21-F022C (inboard) and 2B21-F028C (outboard)), that the minimum pathway leakage for each of these penetrations exceeded the values specified in the plant's Technical Specifications. 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 minimum pathway leakage for the 'B' penetration was 294 scfh, and for the 'C' penetration was 280 scfh. The total minimum pathway leakage through all four main steam lines was 574 scfh.

The most likely direct causes of the MSIV minimum pathway leakages exceeding the values specified in the Technical Specifications were out-of-specification internal valve tolerances and dimensions. It was determined that the probability of the MSIVs seating leak-tight will be reduced if the clearances between the in-body valve guides and the main disc are too large, or if anomalies in the seating areas for these valves exist, such as incorrect seating angle, width of the seating surface, or high or low spots in the seating surfaces. Internal inspection of the MSIVs determined that one or more of these conditions existed in each of the four referenced MSIVs. Each of the anomalies identified were corrected by machining or by installing an oversize disc that reduced the clearances in the valve. Review of the plant's maintenance practices determined that checking these clearances was not part of the plant's normal maintenance activities. The applicable plant procedure has been revised to address this issue.

1. FACILITY NAME	2. DOCKET	6. LER NUMBER			3. PAGE
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
Edwin I. Hatch Nuclear Plant - Unit 2	05000366	2007	-- 001	-- 00	2 OF 5

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 February 12, 2007 Hatch Unit 2 was in the refueling mode for its 19th Refueling Outage 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 the Main Steam Isolation Valves (MSIVs, EIIS Code SB). These tests determined, for both the 'B' penetration (2B21-F022B (inboard) and 2B21-F028B (outboard)) and the 'C' penetration (2B21-F022C (inboard) and 2B21-F028C (outboard)), that the minimum pathway leakage for each of these penetrations exceeded the values specified in the plant's Technical Specifications surveillance requirement (SR) 3.6.1.3.11. This SR addresses the leakage restrictions through the MSIVs. 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 (MSLs) is 250 standard cubic feet per hour (scfh). The as-found measured LLRT minimum pathway leakage for the 'B' penetration was 294 scfh, and for the 'C' penetration was 280 scfh. Additionally, outboard MSIVs 2B21-F028A and 2B21-F028D, for the 'A' and 'D' penetrations respectively, were found to exceed the leakage limits specified for an individual valve; however, the inboard MSIVs for these penetrations met their Technical Specification leakage limits. The total minimum pathway leakage through all four MSLs was 574 scfh.

As a result of these MSIV LLRT failures, an event recovery team was assembled, including a representative from the valve manufacturer, to determine the causes for each of the valve failures and to ensure that adequate corrective actions were taken to restore the valves to a condition that would provide reliable service. A fault tree was constructed to determine the most likely cause of the LLRT failures. As-found Air Operated Valve (AOV) diagnostics and various tests were performed to determine if the MSIV actuators were a likely cause or contributor to the MSIV failures. It was concluded that the most likely direct causes of the MSIV failures were out-of-specification internal valve tolerances and dimensions. The as-found conditions of the valves were determined by performing internal valve inspections that were focused on areas of potential leakage. MSIVs have a main disc that has a seat in the main valve body and a stem disc that has a "pilot" seat in the main disc. This design establishes four seating surfaces where an anomaly could cause internal valve leakage. Additionally, other plants were contacted that use Rockwell International MSIVs to gain insight from their experience. From discussions with these plants and the vendor representative, it was determined that if the clearances between the in-body valve guides and the main disc are too large the probability of MSIVs seating leak-tight is reduced.

A review of the plant's maintenance practices determined that checking these clearances was not part of the normal maintenance activities. The diametral clearances between the valve body guides and the main disc need to be checked and maintained within limits in order to assure leak-tightness of the valves. These clearances are small and difficult to obtain when profiling the valves to determine the actual as-found conditions. Additionally, it was observed that the vendor manuals reviewed did not contain any specific recommendations to check these dimensions when performing maintenance.

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

Each MSIV that failed was subsequently disassembled and as-found data recorded. Anomalies that were found for these valves included a main disc valve seating angle that was 46 degrees instead of 45 degrees, high spots in the stem disc seat, in-body valve seats that were wider than those specified by the valve manufacturer, and clearances between the valve body guides and the main disc that exceeded limits. Each of the anomalies identified was corrected by machining or by installing an oversize disc that reduced the clearances in the valve.

CAUSE OF EVENT

The most likely direct causes of the MSIV failures were from out-of-specification internal valve tolerances and dimensions.

The problems found with valve seats that were too wide or cut at the wrong angle were considered to be procedural compliance issues. The plant procedures specified the correct values but the as-found conditions did not comply with these requirements. The as-found conditions were not considered to be caused by valve operation.

The problem with diametral clearances exceeding applicable limits was considered to be caused by a lack of procedural guidance to require checking these clearances against applicable limits and require restoring clearances to within limits when necessary.

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 'B' and ‘C’ MSL minimum pathway leakages 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 Actual Cubic Centimeters per Minute (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).

LICENSEE EVENT REPORT (LER)

1. FACILITY NAME	2. DOCKET	6. LER NUMBER			3. PAGE
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
Edwin I. Hatch Nuclear Plant - Unit 2	05000366	2007	-- 001	-- 00	4 OF 5

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

The MSLs lead outside of secondary containment and have their own specific limits for leakage established in the plant's Technical Specifications of 250 scfh maximum pathway leakage for all four MSLs. The leakage rates measured in this event were greater than this amount. The allowable leakage for the MSLs has been factored into the plant's safety analysis.

Primary Containment leakage criteria were 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 rates determined for the MSIVs based on the results of the LLRT 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. Realistically, therefore, the 10 CFR 100 off-site dose limits would likely not have been exceeded had an actual event occurred.

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

CORRECTIVE ACTIONS

Immediate actions were taken to correct each of the anomalies identified, by machining or by installing an oversize disc that reduced the clearances in the valve. The as-left LLRT testing was performed with the following results: 2B21-F022B leaked 0 scfh, 2B21-F022C leaked 0.03 scfh, 2B21-F028A leaked 0.19 scfh, 2B21-F028B leaked 0.19 scfh, 2B21-F028C leaked 0.19 scfh, and 2B21-F028D leaked 0.37 scfh

The applicable plant procedure has been revised to capture the required tolerances and dimensions necessary to improve long term valve reliability.

The Maintenance individuals supervising contractor personnel for MSIV repair activities were counseled on the importance of procedure compliance. The consequences of the failure to meet expectations were emphasized.

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

ADDITIONAL INFORMATION

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

Failed Components Information:
Master Parts List Number: 2B21 -F022B, F028B,
2B21- F022C, F028C
EIIS System Code: SB
Manufacturer: Rockwell International
Reportable to EPIX: Yes
Model Number: 1612 JM MNTY
Root Cause Code: X
Type: Valve, Shutoff
EIIS Component Code: SHV
Manufacturer Code: R344

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

Previous Similar Events:
LER 2-2005-001 documents a similar event for the MSIVs where both the inboard and outboard valves on an MSL failed the LLRT testing. Corrective actions for that event did not take into account the potential impact of not maintaining the internal tolerances and dimensions identified in this event.