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SOUTHERN COMPANY

December 9, 2011

Docket Nos.: 50-366

NL-11-2143

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

Edwin I. Hatch Nuclear Plant – Unit 2
Licensee Event Report 2011-001-01, Revision 1
Primary Containment Isolation Penetration Exceeded Overall Allowable Technical Specification Leakage Limits

Ladies and Gentlemen:

In accordance with the requirements of 10 CFR 50.73(a)(2)(ii)(A) and 10CFR 50.73(a)(2)(v)(C), Southern Nuclear Operating Company (SNC) is submitting a revision to the enclosed licensee event report (LER) concerning a primary containment isolation penetration exceeding overall allowable Technical Specification leakage limits.

This letter contains no NRC commitments. If you have any questions, please contact Doug McKinney at (205) 992-5982.

Respectfully submitted,

Mark J Ajluni

M. J. Ajluni
Nuclear Licensing Director

MJA/SBT/lac

Enclosure: LER 2011-001-01

IE22
NRR

cc: Southern Nuclear Operating Company
Mr. S.E. Kuczynski, Chairman, President & CEO
Mr. D. G. Bost, Chief Nuclear Officer
Ms. P. M. Marino, Vice President – Engineering
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U. S. Nuclear Regulatory Commission
Mr. V. M. McCree, Regional Administrator
Mr. P. G. Boyle, NRR Project Manager
Mr. E. D. Morris, Senior Resident Inspector – Hatch

**Enclosure**

**Edwin I. Hatch Nuclear Plant**
**Licensee Event Report 2011-001-01, Revision 1**
Primary Containment Isolation Penetration Exceeded Overall Allowable Technical Specification Leakage Limits

NRC FORM 366 (9-2007) U.S. NUCLEAR REGULATORY COMMISSION

APPROVED BY OMB: NO. 3150-0104 EXPIRES: 10/31/2013

Estimated burden per response to comply with this mandatory collection request: 80 hours. Reported lessons learned are incorporated into the licensing process and fed back to industry. Send comments regarding burden estimate to the Records and FOIA/Privacy Service Branch (T-5 F53), U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, or by internet e-mail to infocollects.resources@nrc.gov, and to the Desk Officer, Office of Information and Regulatory Affairs, NEOB-10202, (3150-0104), Office of Management and Budget, Washington, DC 20503. If a means used to impose an information collection does not display a currently valid OMB control number, the NRC may not conduct or sponsor, and a person is not required to respond to, the information collection.

# LICENSEE EVENT REPORT (LER)

| 1. FACILITY NAME | 2. DOCKET NUMBER | 3. PAGE |
|---|---|---|
| Edwin I. Hatch Nuclear Plant Unit 2 | 05000 366 | 1 OF 6 |

**4. TITLE**
Primary Containment Isolation Penetration Exceeded Overall Allowable Technical Specification Leakage Limits

| 5. EVENT DATE | | | 6. LER NUMBER | | | 7. REPORT DATE | | | 8. OTHER FACILITIES INVOLVED | |
|---|---|---|---|---|---|---|---|---|---|---|
| MONTH | DAY | YEAR | YEAR | SEQUENTIAL NUMBER | REV NO. | MONTH | DAY | YEAR | FACILITY NAME | DOCKET NUMBER |
| 04 | 16 | 2011 | 2011 | - 001 - | 1 | 12 | 09 | 2011 | FACILITY NAME | DOCKET NUMBER |

**9. OPERATING 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)
- ☐ 20.2201(d)
- ☐ 20.2203(a)(1)
- ☐ 20.2203(a)(2)(i)
- ☐ 20.2203(a)(2)(ii)
- ☐ 20.2203(a)(2)(iii)
- ☐ 20.2203(a)(2)(iv)
- ☐ 20.2203(a)(2)(v)
- ☐ 20.2203(a)(2)(vi)
- ☐ 20.2203(a)(3)(i)
- ☐ 20.2203(a)(3)(ii)
- ☐ 20.2203(a)(4)
- ☐ 50.36(c)(1)(i)(A)
- ☐ 50.36(c)(1)(ii)(A)
- ☐ 50.36(c)(2)
- ☐ 50.46(a)(3)(ii)
- ☐ 50.73(a)(2)(i)(A)
- ☐ 50.73(a)(2)(i)(B)
- ☐ 50.73(a)(2)(i)(C)
- ☒ 50.73(a)(2)(ii)(A)
- ☐ 50.73(a)(2)(ii)(B)
- ☐ 50.73(a)(2)(iii)
- ☐ 50.73(a)(2)(iv)(A)
- ☐ 50.73(a)(2)(v)(A)
- ☐ 50.73(a)(2)(v)(B)
- ☒ 50.73(a)(2)(v)(C)
- ☐ 50.73(a)(2)(v)(D)
- ☐ 50.73(a)(2)(vii)
- ☐ 50.73(a)(2)(viii)(A)
- ☐ 50.73(a)(2)(viii)(B)
- ☐ 50.73(a)(2)(ix)(A)
- ☐ 50.73(a)(2)(x)
- ☐ 73.71(a)(4)
- ☐ 73.71(a)(5)
- ☐ OTHER

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 / Steven Tipps – Principal Engineer – Licensing | 912-537-5880 |

**13. COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT**

| CAUSE | SYSTEM | COMPONENT | MANU-FACTURER | REPORTABLE TO EPIX | CAUSE | SYSTEM | COMPONENT | MANU-FACTURER | REPORTABLE TO EPIX |
|---|---|---|---|---|---|---|---|---|---|
| X | BB | SHV | F130 | Y | | | | | |

**14. SUPPLEMENTAL REPORT EXPECTED**

☐ YES *(If yes, complete 15. EXPECTED SUBMISSION DATE)* ☒ NO

**15. EXPECTED SUBMISSION DATE**

| MONTH | DAY | YEAR |
|---|---|---|
| | | |


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

On April 16, 2011, during the Hatch Nuclear Plant outage, a local leak rate test (LLRT) was performed on torus purge supply primary containment isolation valve (PCIV) 2T48-F324 which is associated with primary containment penetration 2T23-X205. At that time, plant engineers and technicians were performing an LLRT for penetration 2T23-X205 when it was discovered that both PCIVs had failed their LLRTs for this penetration. This resulted in the penetration leakage exceeding the overall allowable leakage ($L_a$) required by the Technical Specifications (TS) for primary containment. This is considered a safety system functional failure of the primary containment function for this penetration.

The primary cause for the excessive leakage for valve 2T48-F324 was attributed to "over-travel" of the valve disc which reduced the seat contact with the disc. This was caused by inadequate procedural guidance for adjusting the valve travel. The excessive leakage for 2T48-F309 was attributed to the valve disc failing to "center" on the seat and wear on the actuator and its linkage. The necessary repairs and adjustments were made and the valves subsequently passed their respective LLRTs, which restored penetration 2T23-X205 to within its TS leakage limits. Additional procedural guidance is being provided to better control "travel" adjustments and in performance of valve inspections.

## PLANT AND SYSTEM IDENTIFICATION:

General Electric - Boiling Water Reactor
Energy Industry Identification System codes appear in the text as [NN].

## DESCRIPTION OF EVENT:

On April 16, 2011, during the Hatch Nuclear Plant 2R21 refueling outage, a local leak rate test (LLRT) was performed on torus purge supply primary containment isolation valve (PCIV) 2T48-F324[BB] which is associated with primary containment penetration 2T23-X205. At that time, plant engineers and technicians were performing a local leak rate test (LLRT) for penetration 2T23-X205, when it was discovered that both PCIVs (2T48-F309 and 2T48-F324)[BB] had failed their LLRTs for this penetration. This resulted in the penetration leakage exceeding the overall allowable leakage ($L_a$) requirements in the Technical Specifications for primary containment. This is considered a safety system functional failure for the primary containment function for this penetration.

The inboard PCIV for the torus purge supply system, 2T48-F309 failed to meet its LLRT acceptance criteria for this penetration when it failed to pressurize the penetration on April 5, 2011, thereby exceeding the overall allowable leakage ($L_a$) required by the Technical Specifications for primary containment. Following repair of valve 2T48-F309, the penetration was retested on April 16, 2011, at which time outboard PCIV 2T48-F324 was found to be leaking to such a degree that it too failed to meet its LLRT acceptance criteria for this penetration when it failed to pressurize the penetration. This failure coupled with that of 2T48-F309 caused the penetration leakage to exceed the overall allowable leakage ($L_a$) required by the Technical Specifications for primary containment. The valves are 18 inch Fisher model 9220 butterfly valves with an air operated Bettis model 733B-SR Robotarm actuator. The valves are wafer style that utilizes an elastomer T-ring seat to form a seal against its disc when closed and are located in the Torus bay #6 area.

## CAUSE OF EVENT:

Maintenance personnel attributed the cause of the valve leakage for 2T48-F309 to the fact that the valve disc was not staying centered within the valve body, resulting in a gap between the disc and the seat at the top of the valve. This was caused by workmanship issues in 2009 when neither the valve vendor nor plant Maintenance personnel ensured the valve disc was properly centered as part of the valve refurbishment and setup during that outage. The valve adjustments made during the 2009 refueling outage were sufficient to allow the valve to pass its LLRT with an "as left" leakage of 639 sccm, but the adjustments were made external to the valve which did not ensure the disc was properly centered. There was also inadequate procedural guidance for identifying or replacing worn components for the inboard isolation valve 2T48-F309 body. The necessary adjustments were made to the valve during the 2011 outage which included properly centering the valve disc to ensure the required contact was made between the valve disc and seat and for proper setup of the valve.

Additional investigation revealed that the corrective actions for previous failures to 2T48-F309 were less than adequate in repairing the valve to ensure the ongoing reliability of the valve. This valve has failed to previously meet its LLRT acceptance criteria since the refueling outage in 2005, but the penetration minimum pathway leakage was not previously exceeded for this penetration until the current 2011 outage. Based on a review of work packages from previous outages, the replacement of the T-ring and adjustment of valve seat and disc was apparently done to gain a better seal as the corrective action of choice to correct excessive leakage for this valve. Because of recent failures and resulting Maintenance observations during repairs and adjustments, engineering judgment indicates that a contributor to this and previous failures is continuing actuator wear. However, the more detailed work on the valve along with ensuring its proper setup during the 2011 outage should ensure reliable operation during the current operating cycle following the 2011 refueling outage. Following the repair and adjustments valve 2T48-F309 successfully passed its LLRT with an "as left" leakage of 33 sccm against an acceptance criteria of 4,850 sccm.

Similarly, LLRT test pressure could not be reached for valve 2T48-F324 due to the size of the leakage past the valve seat which was determined to be caused by indications found on the valve seat at the 10 and 6 o'clock positions. The valve disc was over-traveling, preventing a good seal from developing between the disc and seat that apparently caused the damage. During the replacement of the valve T-ring, Maintenance mechanics noticed that the disc moved when the valve's retaining ring / internal stop was removed during T-ring replacement. This is evidence that the internal stops of the valve were being utilized for controlling valve travel rather than the actuator stops which allowed the disc to "long travel". Review of the valve's vendor manual and site procedures recommend utilization of the actuator stops rather than valve internal stops during its set up. This is done to avoid potentially overstressing internal valve components and potentially causing damage if the actuator is not adjusted appropriately. Additionally, procedure 52PM-MNT-011-0, "Bettis Robotarm Valve Actuator Inspection", contains cautions concerning potential damage that can occur when allowing the valve disc to seat against the internal valve stop rather than its actuator stop. That is also the basis for procedure 52PM-T48-013-0, "Purge and Vent Valve T-Ring Replacement", specifying that after closure of the valve with its actuator, the valve's internal stop should be backed off to govern the travel of the valve.

Following the repair and adjustment of 2T48-F324 a successful LLRT was performed with an "as left" leakage of 33 sccm against leakage acceptance criteria of 4,850 sccm.

The successful LLRTs performed for 2T48-F309 and 2T48-F324 confirmed that the "as left" leakage for penetration 2T23-X205 met the leakage acceptance criteria that is considered for input into the total primary containment leakage to ensure the Technical Specification allowable leakage was satisfied for this penetration.

## REPORTABILITY ANALYSIS AND SAFETY ASSESSMENT:

This event is reportable per 10 CFR 50.73(a)(2)(ii)(A), because a condition was identified which resulted in the degradation of one of the plant's principal safety barriers. Specifically, the primary containment pathway through penetration 2T23-X205, containing torus purge supply

system PCIVs 2T48-F309 and 2T48-F324 exceeded the allowable leakage established by the plant's Technical Specifications. Additionally, the failure of the penetration isolation constituted a functional failure of the PCIV function for these valves and directly impacted the capability of controlling the release of radioactive material through this penetration. However, the resulting leakage would be within the secondary containment which serves as the alternate structure that can perform the safety function to ultimately control the release of radioactive material. The event must be reported regardless of whether or not an alternate safety system could have been used to perform the safety function. For this reason the event is also reportable in accordance with 10CFR50.73(a)(2)(v)(C).

The function of the PCIVs, in combination with other accident mitigation systems, is to limit fission product release during and following postulated Design Basis Accidents (DBAs) to within limits. Primary containment isolation ensures that the release of radioactive material to the environment will be consistent with the assumptions used in the analyses for a DBA. The OPERABILITY requirements for PCIVs help ensure that an adequate primary containment boundary is maintained during and after an accident by minimizing potential paths to the environment. Therefore, the OPERABILITY requirements provide assurance that the primary containment function assumed in the safety analyses will be maintained. Two barriers in series are provided for each penetration so that no single credible failure or malfunction of an active component can result in a loss of isolation or leakage that exceeds limits assumed in the safety analyses. The DBAs that result in a release of radioactive material for which the consequences are mitigated by PCIVs are a loss of coolant accident (LOCA) and a main steam line break (MSLB). In the analysis for each of these accidents, it is assumed that PCIVs are either closed or close within the required isolation times following event initiation. This ensures that potential paths to the environment through PCIVs (including primary containment purge valves) are minimized.

The single failure criterion required to be imposed in the conduct of unit safety analyses was considered in the original design of the primary containment purge valves. Two valves in series on each purge line provide assurance that both the supply and exhaust lines could be isolated even if a single failure occurred. 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. PCIVs form a part of the primary containment boundary. The PCIV safety function is related to minimizing the loss of reactor coolant inventory and establishing the primary containment boundary during a DBA. 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 as $L_a$ 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 60,432 accm or 254,937 sccm, most of which is assumed to occur within the Secondary Containment where it will be treated by the Standby Gas Treatment System (SGT) before being released at an elevated point through the Main Stack.

Even though the total leakage through penetration 2T23-X205 would have exceeded $L_a$, the leakage would still be released into the Secondary Containment and would be treated prior to the elevated release to the environment. 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 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. A leaking fuel bundle was detected on 11/02/2009 during the previous operating cycle and was removed from the core during an April 2010 mid-cycle outage. The leaker was examined and determined to have one leaking fuel rod caused by a debris fret. The expected release from this damaged fuel rod would be minimal since much of the noble gases and halogens would have already escaped from the fuel rod. Since the leakage would be released to the Secondary Containment and the radioactive materials postulated to be present in the reactor coolant would be those already present due to normal operation with small contributions to crud released from the reactor coolant system, the leakage would be treated by the SGT system prior to the elevated release. For this reason the release would be within 10CFR100 offsite dose limits and would be considered of low safety significance.

## CORRECTIVE ACTIONS:

The necessary adjustments were made to valve 2T48-F309 to ensure the required contact was made between the valve disc and seat. The valve subsequently passed its LLRT following repairs and adjustments that addressed the identified workmanship issues with the valve that occurred in the 2009 refueling outage. The more detailed work on the valve along with ensuring its proper setup during the 2011 outage should ensure reliable operation during the current operating cycle following the 2011 refueling outage. The current plans are to obtain the needed parts and replace the worn parts or entire valve if necessary during the next scheduled refueling outage.

The necessary adjustments and repairs were made to 2T48-F324 to correct the valve travel and to ensure good contact between the valve disc and seat. The valve subsequently passed its LLRT. PM procedure 52PM-T48-013-0 will be revised to ensure the needed valve travel adjustments in the future are properly made for 2T48-F324 following a failed LLRT with the valve pulled from the pipeline. A caution similar to 52PM-MNT-011 section 5.1.1 and caution at 7.2.5.4 will also be included regarding the likelihood of valve internal damage if actuator stops adjustments are made inappropriately.

## ADDITIONAL INFORMATION:

Similar LLRT results of the other PCIVs in the 2T48 system were reviewed to determine if the condition and causes extended to those components. Valve 2T48-F310 was the only other similar valve that failed its "as found" LLRT during this outage in the same manner in that it

would not pressurize. The failure of the 2T48-F310 valve was attributed to T-ring wear. The leakage contribution could be measured through the corresponding PCIV (2T48-F328) in this penetration and the "as found" would not have resulted in $L_a$ being exceeded. No other 2T48 valves failed their "as found" LLRTs during the testing performed in the 2011 refueling outage or exhibited similar symptoms that would indicate the presence of adjustment issues. Valve 2T48-F310 was subsequently repaired with a 0 sccm "as left" leakage.

Unit 1 LLRT results were reviewed for the corresponding Unit 1 valves and no penetrations containing the similar 1T48 valves failed their respective LLRTs during that outage that would cause $L_a$ to be exceeded for primary containment. However, one Unit 1 penetration containing valves 1T48-F310 and 1T48-F328A did fail to meet the leakage acceptance criteria for its penetration, but the minimum pathway leakage for the penetration could be measured. The contribution of the leakage from this penetration along with the other minimum pathway leakages for primary containment did not exceed $L_a$. The valves were repaired and adjusted and successfully met their respective acceptance criteria in the "as left" LLRT.

Failed Components Information:

Master Parts List Number: 2T48-F309 / 2T48-F324
Manufacturer: Fisher Controls
Model Number: 18 in Fisher model 9220 butterfly valve
Type: Valve, Shutoff
Manufacturer Code: F130

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

Previous Similar Events:

One Unit 1 penetration containing valves 1T48-F310 and 1T48-F328A did fail to meet the leakage acceptance criteria for its penetration in the 2010 refueling outage, but the minimum pathway leakage for the penetration could still be measured. The cause for 1T48-F310 failing to pressurize was attributed to a different cause than those identified in this LER. There was not a recent previous similar event that addressed the causes for excessive leakage reported in this Unit 2 LER. The contribution of the leakage from this penetration along with the other minimum pathway leakages for primary containment did not exceed $L_a$.

LER 2-2009-002 documents a similar event for the 'A' feedwater line in which both inboard and outboard isolation valves failed the LLRT testing resulting in failure of the affected penetration for bypass leakage. The main feedwater check valves are a different type valve and the causes for their LLRT failure were also different from those experienced with the 2T48 valves reported in this LER. The resulting corrective actions from that LER would have no bearing on the work performed on the 2T48 valves and would not have prevented their LLRT failure.

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