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**SOUTHERN COMPANY**
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June 13, 2002

Docket No. 50-321

HL-6248

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

Edwin I. Hatch Nuclear Plant - Unit 1
Licensee Event Report
Technical Specification Required Plant Shutdown Because of
High Unidentified Reactor Coolant System Leakage

Ladies and Gentlemen:

In accordance with the requirements of 10 CFR 50.73(a)(2)(i), Southern Nuclear Operating Company is submitting the enclosed Licensee Event Report (LER) concerning a Technical Specification required plant shutdown because of high unidentified reactor coolant system leakage.

Respectfully submitted,

Lewis Sumner

H. L. Sumner, Jr.

IFL/eb

Enclosure: LER 50-321/2002-002

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

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| NRC FORM 366 (7-2001) U.S. NUCLEAR REGULATORY COMMISSION | APPROVED BY OMB NO. 3150-0104 EXPIRES 7/31/2004 |
|---|---|
| **LICENSEE EVENT REPORT (LER)**<br>(See reverse for required number of digits/characters for each block) | Estimated burden per response to comply with this mandatory information collection request: 50 hrs. Reported lessons learned are incorporated into the licensing process and fed back to industry. Send comments regarding burden estimate to the Records Management Branch (T-6 E6), U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, or by internet e-mail to bjs1(@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 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. |

| 1. FACILITY NAME | 2. DOCKET NUMBER | 3. PAGE |
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4. TITLE
Technical Specification Required Plant Shutdown Because of High Unidentified Reactor Coolant System Leakage

| 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 |
| 04 | 19 | 2002 | 2002 | 002 | 0 | 06 | 13 | 2002 | FACILITY NAME | DOCKET NUMBER(S) 05000 |

| 9. OPERATING MODE | 1 |
|---|---|
| 10. POWER LEVEL | 18 |

11. THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check all that apply)

| | | | | | | | |
|---|---|---|---|---|---|---|---|
| | 20.2201(b) | | 20.2203(a)(3)(ii) | | 50.73(a)(2)(ii)(B) | | 50.73(a)(2)(ix)(A) |
| | 20.2201(d) | | 20.2203(a)(4) | | 50.73(a)(2)(iii) | | 50.73(a)(2)(x) |
| | 20.2203(a)(1) | | 50.36(c)(1)(i)(A) | | 50.73(a)(2)(iv)(A) | | 73.71(a)(4) |
| | 20.2203(a)(2)(i) | | 50.36(c)(1)(ii)(A) | | 50.73(a)(2)(v)(A) | | 73.71(a)(5) |
| | 20-2203(a)(2)(ii) | | 50.36(c)(2) | | 50.73(a)(2)(v)(B) | | OTHER |
| | 20-2203(a)(2)(iii) | | 50.46(a)(3)(ii) | | 50.73(a)(2)(v)(C) | | Specify in Abstract below or in NRC Form 366A |
| | 20.2203(a)(2)(iv) | X | 50.73(a)(2)(i)(A) | | 50.73(a)(2)(v)(D) | | |
| | 20.2203(a)(2)(v) | | 50.73(a)(2)(i)(B) | | 50.73(a)(2)(vii) | | |
| | 20.2203(a)(2)(vi) | | 50.73(a)(2)(i)(C) | | 50.73(a)(2)(viii)(A) | | |
| | 20.2203(a)(3)(i) | | 50.73(a)(2)(ii)(A) | | 50.73(a)(2)(viii)(B) | | |

12. LICENSEE CONTACT FOR THIS LER

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

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 | | RV | T020 | Yes | | | | | | |
| X | BF | RV | G202 | Yes | | | | | | |

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


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

On 4/19/02 at 1650 EST, Unit 1 was in the Run mode at a power level of approximately 525 CMWT (19 percent rated thermal power). Power was reduced in order to identify and repair the source(s) of leakage into the drywell floor drain sump. Limiting Condition for Operation (LCO) 3.4.4 establishes limits for both the increase in rate within a 24-hour period and the overall rate of unidentified leakage in the drywell. The leakage exceeded the allowable limits of Technical Specifications LCO 3.4.4. Therefore, at approximately 2100 EST a manual scram was inserted using the normal shutdown procedure, shutting down the unit as required by LCO 3.4.4, Required Action C.1. A drywell entry was made, and on 4/20/02 at approximately 0036 EST it was determined that the source of drywell leakage originated from a leaking Safety Relief Valve (SRV) 1B21F013J. The tailpipe vacuum breaker (1B21F037J) associated with this SRV had failed in the stuck open position allowing steam to vent directly into the drywell. By 0146 EST, the drywell unidentified leakage rate was within the Technical Specification limits.

The cause of SRV 1B21F013J leaking was component failure. The valve's main disc was stuck in the partially open position. The actuator piston had become cocked on the main disc's stem causing it to bind in the valve body. The cause of the vacuum breaker (1B21F037J) failing was determined to be the result of leakage past the main seat of SRV 1B21F013J. This leakage caused the vacuum breaker to rapidly cycle open and closed which caused the bolts holding the disc to the swing arm to break. The safety relief valve and the vacuum breaker were replaced and functionally tested.

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

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 4/19/02 at 1650 EST, Unit 1 was in the Run mode at a power level of approximately 525 CMWT (19 percent rated thermal power). Power was reduced in order to identify and repair the source(s) of leakage into the drywell floor drain sump (EIIS Code IJ). Drywell floor drain sump in-leakage, that is, unidentified leakage into the drywell, had increased from approximately 0.1 gpm at 0743 EST on 04/19/02 to approximately 6.85 gpm at 1647 EST on 04/19/02.

The Unit 1 Technical Specifications Limiting Condition for Operation (LCO) 3.4.4 establishes limits for Reactor Coolant System (RCS) leakage. A limit is established for the increase in the rate of leakage within a 24-hour period (2-gpm maximum), and a limit is established for the overall rate of unidentified leakage (5-gpm maximum). The leakage that occurred during this event exceeded both of these unidentified RCS leakage rate limits defined by the Technical Specifications. Therefore, at approximately 2100 EST a manual scram was inserted using the normal shutdown procedure, shutting down the unit as required by LCO 3.4.4, Required Action C.1.

After entering the drywell, plant personnel discovered (on 4/20/02 at approximately 0036 EST) that the source of drywell leakage was a leaking Safety Relief Valve (SRV) 1B21F013J. The tailpipe vacuum breaker (1B21F037J) (EIIS Code BF) associated with this SRV had failed in the stuck open position allowing steam to vent directly into the drywell. The leakage from the SRV did not meet the Technical Specifications definition of pressure boundary leakage, that is, leakage through a nonisolable fault in the reactor coolant system. Therefore, a notification of unusual event (NUE) was not declared because the Emergency Action Level (EAL) for Plant Hatch's Emergency Plan is based on confirming a reactor coolant pressure boundary leak.

By 0146 EST, the Reactor Coolant System operational leakage rate was within the Technical Specification limits. The safety relief valve and the vacuum breaker were replaced and functionally tested 4/22/02.

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

## CAUSE OF EVENT

The cause of SRV 1B21F013J leaking was component failure. The main disc of this valve has an integral stem that is partially threaded. The stem portion of the main disc passes through the body of the valve into another chamber (cylinder) where a piston is threaded onto the stem. A washer (which is bent after installation) and a jam nut are threaded onto the stem locking the piston onto the stem. An inspection of the failed valve determined that the piston was cocked on the stem and was in contact with the cylinder portion of the valve causing it to be stuck in the partially open position. Additionally the jam nut was found to be loose. This loose jam nut was considered to be a contributor to this event. A review of the historical SRV valve inspection reports determined that of 35 main valve bodies inspected since 1992, four had been found with the jam nut not fully tight. All of the valves found with loose jam nuts were in service for more than twelve years. Since a loose jam nut was considered to be a contributor to this event, it was concluded that age was a factor in the failure of this valve. The cause of the vacuum breaker (1B21F037J) failing in the open position was determined to be leakage past SRV 1B21F013J. This leakage resulted in the vacuum breaker rapidly cycling open and closed which caused the bolts holding the disc to the swing arm to break.

## REPORTABILITY ANALYSIS AND SAFETY ASSESSMENT

This report is required by 10 CFR 50.73 (a)(2)(i) because this condition resulted in a nuclear plant shutdown required by the plant's Technical Specifications. The Unit 1 Technical Specifications Limiting Condition for Operation (LCO) 3.4.4 establishes limits for Reactor Coolant System (RCS) leakage. A limit is established for the increase in the rate of leakage within a 24-hour period (2-gpm maximum) and a limit is established for the overall rate of unidentified leakage (5-gpm maximum). The leakage that occurred during this event exceeded both of these unidentified RCS leakage rate limits defined by the Technical Specifications.

The reactor coolant system includes systems and components that contain or transport the coolant to or from the reactor core. The pressure-retaining components of the reactor coolant system and the portions of connecting systems out to and including the isolation valves define the reactor coolant pressure boundary. During plant life, the joint and valve interfaces can produce varying amounts of reactor coolant leakage, through either normal operational wear or mechanical deterioration. Limits on reactor coolant system operational leakage are required to ensure appropriate action is taken before the integrity of the reactor coolant pressure boundary is impaired. The Technical Specifications specify the types and limits of leakage.

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

The unidentified leakage flow limit allows time for corrective action before the reactor coolant pressure boundary can be compromised significantly. The 5-gpm limit is a small fraction of the calculated flow from a critical crack in the primary system piping. A critical crack is one large enough to propagate rapidly, ultimately leading to failure of the affected component. Crack behavior from experimental programs shows that leakage rates of hundreds of gallons per minute will precede crack instability (see Unit 1 Final Safety Analysis Report, section 4.10, "Nuclear System Leakage Detection and Leakage Rate Limits," and Unit 1 Technical Specifications Bases B 3.4.4, "RCS Operational Leakage").

In this event, unidentified leakage into the drywell had increased from approximately 0.1 gpm at 0743 EST on 04/19/02 to approximately 6.85 gpm at 1647 EST on 04/19/02. After entering the drywell, plant personnel discovered that the source of drywell leakage was a leaking SRV 1B21F013J. The tailpipe vacuum breaker (1B21F037J) associated with this SRV had failed in the stuck open position allowing steam to vent directly into the drywell.

At the time the unit was shut down, the unidentified leakage rate was 6.85 gpm and exceeded the Technical Specifications-allowed limit of 5-gpm. The source of the leakage was determined to be a leaking SRV and not a crack. Therefore, the leakage source was not an unstable crack that would have resulted in catastrophic failure of a line. However, a worst-case scenario of a fully stuck open relief valve is addressed in section 15.2.8.1 of the FSAR, "Inadvertent Opening of an SRV (Event 22)," which analyzes the effects of the complete opening of a Safety Relief Valve and clearly bounds any amount of valve leakage. The effects of an inadvertent opening of a Safety Relief Valve are described in the Final Safety Analysis Report as "mild" with respect to depressurization and "inconsequential" with respect to offsite doses. Concerning containment issues with the stuck open vacuum breaker (venting the steam into the drywell instead of the torus), section 15.3.3, "LOCA (RADIOLOGICAL CONSEQUENCES) (EVENT 32)" clearly bounds the amount of reactor coolant leakage being released into the drywell. This event assumes that a recirculation line is instantaneously severed with coolant discharged from both ends of the break which clearly exceeds the amount of leakage that can result from a stuck open vacuum breaker on an SRV discharge line where the SRV is stuck open.

Based upon the preceding analysis, it is concluded this event had no adverse impact on nuclear safety.

## CORRECTIVE ACTIONS

The particular valves that failed in this event were replaced during the shutdown per Maintenance Work Order 1-02-1609 (for valve 1B21F013J)) and Maintenance Work Order 1-02-1644 (for valve 1B21F037J.

A recommended scope of SRV repair work to be performed during the next Unit 2 refueling outage will be developed. This recommended scope of SRV repair work will be incorporated into the schedule for the next Unit 2 refueling outage.

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

ADDITIONAL INFORMATION

No systems other than those already mentioned in this report were affected by this event.
Failed Component Information:

Master Parts List Number: 1B21F013J
Type: Safety Relief Valve
Manufacturer: Target Rock
Vendor Code (VPN): 438428
Model: 7567F
EIIS System Code:
EIIS Component Code: RV
Root Cause Code: X
Reportable to EPIX: Yes

Master Parts List Number: 1B21F037J
Type: Vacuum Breaker
Manufacturer: GPE CONTROLS
Vendor Code: 429700
Model: LD24425
EIIS System Code: BF
EIIS Component Code: RV
Root Cause Code: X
Reportable to EPIX: Yes

A previous similar event occurred in July of 1999 when an SRV experienced a failure while being tested at Wyle Labs. This valve stuck open because the nut on the pilot side of the piston came off the disc stem and the piston experienced some degree of cocking on the stem. It should be noted that this failure did not occur while the valve was in service in the plant, and the valve stroked successfully three times before failing on the fourth actuation. Corrective actions for the event included committing to an inspection regime for future SRVs sent to Wyle.

Corrective actions for this previous event could not have prevented this event because the component that failed during this event had not yet been sent to Wyle under the normal recertification schedule to be inspected under the augmented inspection program.