NRC Form 366
(9-83)

U.S. NUCLEAR REGULATORY COMMISSION
APPROVED OMB NO. 3150-0104
EXPIRES: 8/31/88

# LICENSEE EVENT REPORT (LER)

| FACILITY NAME (1) | DOCKET NUMBER (2) | PAGE (3) |
|---|---|---|
| Browns Ferry Unit 2 | 0 5 0 0 0 2 6 0 | 1 OF 0 2 |

TITLE (4): Incorrect Performance of Local Leak Rate Test

| 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 NAMES | DOCKET NUMBER(S) |
| 0 5 | 3 0 | 8 6 | 8 6 | 0 0 9 | 0 0 | 0 6 | 2 7 | 8 6 | | 0 5 0 0 0 |
| | | | | | | | | | | 0 5 0 0 0 |

OPERATING MODE (9): N

POWER LEVEL (10): 0 0 0

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

| | | | | | | | |
|---|---|---|---|---|---|---|---|
| | 20.402(b) | | 20.405(c) | | 50.73(a)(2)(iv) | | 73.71(b) |
| | 20.405(a)(1)(i) | | 50.36(c)(1) | | 50.73(a)(2)(v) | | 73.71(c) |
| | 20.405(a)(1)(ii) | | 50.36(c)(2) | | 50.73(a)(2)(vii) | | OTHER (Specify in Abstract below and in Text, NRC Form 366A) |
| | 20.405(a)(1)(iii) | X | 50.73(a)(2)(i) | | 50.73(a)(2)(viii)(A) | | |
| | 20.405(a)(1)(iv) | | 50.73(a)(2)(ii) | | 50.73(a)(2)(viii)(B) | | |
| | 20.405(a)(1)(v) | | 50.73(a)(2)(iii) | | 50.73(a)(2)(x) | | |

LICENSEE CONTACT FOR THIS LER (12)

| NAME | TELEPHONE NUMBER (AREA CODE) |
|---|---|
| Alan W. Gordon, Compliance Engineer | 2 0 5 7 2 9 - 2 5 3 7 |

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

| CAUSE | SYSTEM | COMPONENT | MANUFAC-TURER | REPORTABLE TO NPRDS | CAUSE | SYSTEM | COMPONENT | MANUFAC-TURER | REPORTABLE TO NPRDS |
|---|---|---|---|---|---|---|---|---|---|
| | | | | | | | | | |
| | | | | | | | | | |

SUPPLEMENTAL REPORT EXPECTED (14)

[ ] YES (If yes, complete EXPECTED SUBMISSION DATE) [X] NO

EXPECTED SUBMISSION DATE (15): MONTH | DAY | YEAR

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


While conducting a local leak rate test (LLRT) on unit 1 residual heat removal system valve 1-FCV- 74-48 on May 30, 1986, it was determined that the test procedure was in error. The procedure specified a test valve alignment which did not provide a flow path downstream of the valve seat being tested. The testing was stopped, but since the procedure had been used earlier for the same valve in unit 2, the LLRT test results on 2-FCV-74-48 may be clouded. Because maintenance has been performed on 2-FCV-74-48 since the test, no valid "as found" leakage, for this particular valve required for inclusion in the overall integrated leak rate test report, will be available. Inadequate technical review of the test procedure when it was expanded in September 1984 is the cause of this condition. The procedure has been revised to correct all identified deficiencies.




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| FACILITY NAME (1) | DOCKET NUMBER (2) | LER NUMBER (6) YEAR | SEQUENTIAL NUMBER | REVISION NUMBER | PAGE (3) |
|---|---|---|---|---|---|
| Browns Ferry Unit 2 | 0 5 0 0 0 2 6 0 | 8 6 | — 0 0 9 — | 0 0 | 0 2 OF 0 2 |

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

Units 1 and 2 are defueled and in refueling outages. Unit 3 was in an extended maintenance outage when the situation was discovered. This situation affects only unit 2.

On May 30, 1986, while conducting a local leak rate test (LLRT) of unit 1 residual heat removal (RHR) (BO) system inboard isolation valve 1-FCV-74-48, it was discovered that Surveillance Instruction (SI) 4.7.A.2.g-3, "Containment Isolation Valve Leak Rate Tests," was in error. The same procedure had previously been used in January 1985 for the FCV 74-48 valve in unit 2. The local pressurization test, required by 10 CFR 50 Appendix J, was correct in that the valve was tested in the flow direction required to perform its safety function. However, the procedure specified a valve lineup which did not provide a test flow path downstream of the valve. Thus, leakage was measured through several closed test and isolation valves in addition to the valve being tested. Therefore, an accurate measurement of the unit 2 valve was not made. Maintenance has since been performed on the valve, consequently no valid "as found" leakage for this valve is available for inclusion in the overall integrated leak rate test report. The erroneous procedure was not used on unit 3. The problem was identified when the procedure was to be used a second time to test 1-FCV 74-48. Upon discovery of the incorrect valve lineup, all work was stopped, and procedure revisions initiated.

The valve lineup was incorrectly specified under a general revision of SI 4.7.A.2.g-3 in September 1984 which added alignments for leak testing numerous primary containment isolation valves. This revision apparently did not receive adequate technical review, due to a lack of qualified (NDE level II) personnel on site at the time. The procedure has been technically reviewed and revised to correct the valve lineup for FCV 74-48, along with other identified items. No further corrective action is planned.

2-FCV-74-48 is the RHR shutdown cooling inboard isolation valve. The outboard valve 2-FCV-74-47 was leak tested at the same time (January 1985) and met its requirements. Therefore, it is unlikely that the deficient test data has significant effects on the overall integrated leak test report data.

Responsible Plant Section - Engineering

Previous Events - BFRO 50-259/85008; 50-260/83005

TENNESSEE VALLEY AUTHORITY
Browns Ferry Nuclear Plant
P.O. Box 2000
Decatur, Alabama 35602

June 27, 1986

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

Dear Sir:

TENNESSEE VALLEY AUTHORITY - BROWNS FERRY NUCLEAR PLANT UNIT 2 - DOCKET NO. 50-260 - FACILITY OPERATING LICENSE DPR-52 - REPORTABLE OCCURRENCE REPORT BFRO-50-260/86009

The enclosed report provides details concerning incorrect performance of local leak rate test. This report is submitted in accordance to 10 CFR 50.73 (a)(2)(i).

Very truly yours,

TENNESSEE VALLEY AUTHORITY

EA Grimm
for Robert L. Lewis
Plant Manager
Browns Ferry Nuclear Plant

Enclosures
cc (Enclosures):
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U.S. Nuclear Regulatory Commission
Office of Inspection and Enforcement
Region II
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Atlanta, Georgia 30303

INPO Records Center
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NRC Resident Inspector, Browns Ferry Nuclear Plant

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