Tennessee Valley Authority, Post Office Box 2000, Decatur, Alabama 35609-2000

January 22, 2009

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Mail Stop: OWFN, P1-35
Washington, D. C. 20555-0001

10 CFR 50.73

Dear Sir:

TENNESSEE VALLEY AUTHORITY - BROWNS FERRY NUCLEAR PLANT (BFN) - UNIT 1 - DOCKET 50-259 - FACILITY OPERATING LICENSE DPR - 33 - LICENSEE EVENT REPORT (LER) 50-259/2008-002

The enclosed report provides details of an ASME Code Class 1 Boundary Leak on an Instrument Line Connected to the Reactor Vessel. TVA is submitting this report in accordance with 10 CFR 50.73(a)(2)(ii)(A) as an event or condition that resulted in the nuclear power plant, including principal safety barriers, being seriously degraded. There are no commitments contained in this letter.

Sincerely,

R. G. West
Site Vice President, BFN

cc: See page 2

IE22
NRR

Enclosure
cc (Enclosure):

Ms. Eva Brown, Project Manager
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NRC FORM 366 (9-2007) U.S. NUCLEAR REGULATORY COMMISSION	APPROVED BY OMB NO. 3150-0104 EXPIRES 08/31/2010
LICENSEE EVENT REPORT (LER) (See reverse for required number of digits/characters for each block)	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 F52), U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, or by internet e-mail to infocollects@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.

1. FACILITY NAME	2. DOCKET NUMBER	3. PAGE
Browns Ferry Unit 1	05000259	1 of 4

4. TITLE: ASME Code Class 1 Pressure Boundary Leak On An Instrument Line Connected to the Reactor Vessel

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: None	DOCKET NUMBER: N/A
11	23	2008	2008	- 002 -	00	01	22	2009	FACILITY NAME: None	DOCKET NUMBER: N/A

9. OPERATING MODE: 4

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)	☒ 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

NAME	TELEPHONE NUMBER (Include Area Code)
Steve Austin, Licensing Engineer	256-729-2070

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

14. SUPPLEMENTAL REPORT EXPECTED		15. EXPECTED SUBMISSION DATE	MONTH	DAY	YEAR
☐ YES (If yes, complete 15. EXPECTED SUBMISSION DATE)	☒ NO		N/A	N/A	N/A

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

On November 23, 2008, at approximately 1200 hours Central Standard Time (CST), during the performance of the Unit 1 vessel hydrostatic test, ASME Section XI System Leakage Test of the Reactor Pressure Vessel and Associated Piping, 1-SI-3.3.1.A, a reactor pressure boundary leak was discovered on an unisolatable instrument line connected to the reactor vessel. This instrument line is an ASME Code Class 1 equivalent component, 2-inch pipe, near pressure vessel nozzle N11B. BFN entered Technical Requirements Manual (TRM) Section 3.4.3 - Structural Integrity, which requires the integrity of ASME Code Class 1, 2, and 3 equivalent components shall be maintained through the life of the plant. Unit 1 was in mode 4 at the time of discovery; it remained in mode 4 until the repairs were completed. The root cause of the event was residual stress introduced to the safe end inside diameter during initial fabrication. The N11B safe end was examined ultrasonically (UT). The through wall leak was repaired by weld overlay.

FACILITY NAME (1)	DOCKET (2)	LER NUMBER (6)			PAGE (3)
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
Browns Ferry Nuclear Plant Unit 1	05000259	2008	-- 002	-- 00	2 of 4

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

I. PLANT CONDITION(S)

Unit 1 was in Mode 4, approximately 1055 psig. Units 2 and 3 were at 100 percent power (3458 Megawatts thermal) and unaffected by the event.

II. DESCRIPTION OF EVENT

A. Event:

On November 23, 2008, at approximately 1200 hours Central Standard Time (CST), during the performance of the Unit 1 vessel hydrostatic test, ASME Section XI System Leakage Test of the Reactor Pressure Vessel and Associated Piping, 1-SI-3.3.1.A, a leak was discovered on an unisolatable instrument line connected to the reactor vessel. This instrument line is an ASME Code Class 1 equivalent component, 2-inch pipe, near pressure vessel nozzle N11B. BFN entered Technical Requirements Manual (TRM) Section 3.4.3 - Structural Integrity, which requires the integrity of ASME Code Class 1, 2, and 3 equivalent components shall be maintained through the life of the plant.

Unit 1 was in mode 4 at the time of discovery; it remained in mode 4 until the repairs were completed.

TVA is submitting this report in accordance with 10 CFR 50.73(a)(2)(ii)(A) as any event or condition that resulted in the nuclear power plant, including principal safety barriers, being seriously degraded.

B. Inoperable Structures, Components, or Systems that Contributed to the Event:

None.

C. Dates and Approximate Times of Major Occurrences:

November 23, 2008	1200 hours CST	Identified Code Class 1 pressure boundary leak.
November 23, 2008	1600 hours CST	Unit 1 is depressurized.
November 23, 2008	1813 hours CST	TVA makes an eight-hour non-emergency notification to NRC in accordance with 10 CFR 50.72 (b)(3)(ii)(A).

D. Other Systems or Secondary Functions Affected

None.

E. Method of Discovery

The leak was identified by visual examination (VT-2) during a scheduled performance of 1-SI-3.3.1.A.

F. Operator Actions

None.

G. Safety System Responses

None.

FACILITY NAME (1)	DOCKET (2)	LER NUMBER (6)			PAGE (3)
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
Browns Ferry Nuclear Plant Unit 1	05000259	2008	-- 002	-- 00	3 of 4

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

III. CAUSE OF THE EVENT

A. Immediate Cause

The immediate cause of the event was a through wall leak in an 2 inch line nozzle near the safe end/instrument line interface weld very close to the heat affected zone (HAZ).

B. Root Cause

The root cause of the event was residual stress introduced to the safe end inside diameter during initial fabrication. The vessel manufacturer fabricated the safe end from a stainless steel forging. The forging was butt welded to the instrument nozzle. Then the inside diameter of the safe end was machined in place. This fabrication method resulted in high residual stresses on the inside diameter surface of the safe end. The butt welding operation typically results in higher heat input than socket or fillet weld. The higher heat input associated with the weld results in a larger HAZ and a sensitized microstructure that is conductive to IGSCC. In addition, the water in the area of the instrument penetrations contains oxidants, which can create an aggressive environment for the growth of IGSCC. The combination of the water chemistry, sensitized microstructure, and fabrication methodology promoted the growth of IGSCC and the eventual through wall leak.

C. Contributing Factors

None.

IV. ANALYSIS OF THE EVENT

There are three key contributors required to promote IGSCC in 304 type stainless steel pipes: weld sensitized microstructure, an oxygenated environment, and tensile stress. A sensitized microstructure results from heating and cooling the material at various time and temperature combinations that form chromium carbides at the grain boundaries. These carbides deplete the surrounding area of chromium, providing a continuous path of lower corrosion resistance along the grain boundaries for the propagation of cracks. Welding can create this condition. The HAZ of the weld is susceptible to this carbide depletion. The through wall crack identified by the inspection was close to the HAZ.

A BWR environment contains dissolved oxygen that in the correct amounts promotes IGSCC growth. The water in the location of the safe end is high in oxidants, making the location of the N11B safe end susceptible to IGSCC growth. Residual stresses from the welding, grinding, and machining process, all contribute to the overall tensile stress in the safe end. The manufacturing and installation process for the N11B safe end created stresses that, when combined with the weld residual stress that when combined with the weld residual stress, the oxygenated environment and the sensitized weld HAZ, exceeded the yield strength of material. Once cracking initiated, it propagated through IGSCC susceptible metal and became a through wall leak.

V. ASSESSMENT OF SAFETY CONSEQUENCES

The safety consequences of this event were not significant. Plant Technical Specifications (TSs) require monitoring of reactor coolant leakage. When leakage limits are met, the TSs requires the reactor be placed in mode 4. During the previous operating cycle, reactor coolant leakage was less than the TS limits.

FACILITY NAME (1)	DOCKET (2)	LER NUMBER (6)			PAGE (3)
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
Browns Ferry Nuclear Plant Unit 1	05000259	2008	-- 002	-- 00	4 of 4

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

The visual inspection during the performance of 1-SI-3.3.1.A identified the through wall leak at the instrument line connection to the reactor vessel. BFN entered Technical Requirements Manual (TRM) 3.4.3, Structural Integrity, Condition A, which requires immediate restoration of the structural integrity of the affected component or maintain the reactor in mode 4 of the reactor coolant system less than 50 degrees F above the minimum temperature requires for nondestructive testing considerations, until each indication has been investigated and evaluated. Until repairs were completed, BFN maintained the reactor in accordance with these requirements. Therefore, the event did not adversely affect the safety of the public and plant personnel.

VI. CORRECTIVE ACTIONS

A. Immediate Corrective Actions

Once TVA determined there was an un-isolatable leak in the ASME Class 1 reactor pressure boundary, the reactor was depressurized to the pre-test pressure and mode 4 was maintained.

B. Corrective Actions to Prevent Recurrence [1]

To determine the extent of the through wall crack, the N11B safe end was examined ultrasonically (UT). The through wall leak was repaired by weld overlay.

BFN ultrasonically examined the remaining Unit 1 small-bore (less than 4 inches in diameter) instrument nozzle safe ends and the core delta pressure - Standby Liquid Control [BR] line nozzle safe end. The examinations did not identify any other recordable indications.

VII. ADDITIONAL INFORMATION

A. Failed Components

None.

B. Previous LERs on Similar Events

None.

C. Additional Information

Corrective action document for this report is PER 157918.

D. Safety System Functional Failure Consideration:

This event is not a safety system functional failure according to NEI 99-02.

E. Scram With Complications Consideration:

This event was not a complicated scram according to NEI 99-02.

VIII. COMMITMENTS

None.

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[1] TVA does not consider the corrective action a regulatory requirement. The completion of the action will be tracked in TVA's Corrective Action Program.