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

January 4, 2010

10 CFR 50.73

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

Browns Ferry Nuclear Plant, Unit 2
Facility Operating License No. DPR-52
NRC Docket No. 50-260

Subject: **Licensee Event Report 2009-002, Revision 1**

The enclosed Licensee Event Report (LER) discusses a leak in an American Society of Mechanical Engineers Code Class I reactor pressure boundary pipe. The Tennessee Valley Authority is submitting this revised LER to correct administrative errors within the enclosure.

The Tennessee Valley Authority is reporting this in accordance with 10 CFR 50.73, "Licensee Event Report System," paragraph (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 regulatory commitments contained in this letter. Should you have any questions concerning this submittal, please contact F. R. Godwin, Site Licensing and Industry Affairs Manager, at (256) 729-2636.

Respectfully,

Keith J. Polson

K. J. Polson
Site Vice President, BFN

cc: See page 2

IE22
NRR

Enclosure
cc (Enclosure):

NRC Regional Administrator - Region II

NRC Senior Resident Inspector - Browns Ferry Nuclear Plant

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
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4. TITLE: Leak In An ASME Class I Code Reactor Pressure Boundary Pipe

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
05	31	2009	2009-002-01			01	04	2010	FACILITY NAME None	DOCKET NUMBER N/A

9. OPERATING MODE	11. THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: *(Check all that apply)*			
4	☐ 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)
10. POWER LEVEL	☐ 20.2203(a)(2)(ii)	☐ 50.36(c)(1)(ii)(A)	☐ 50.73(a)(2)(iv)(A)	☐ 50.73(a)(2)(x)
0	☐ 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, BFN Licensing	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 type written lines*)

On May 31, 2009, at approximately 1210 hours Central Daylight Time (CDT), during the performance of the Unit 2, ASME Section XI System Leakage Test of the Reactor Pressure Vessel and Associated Piping, 2-SI-3.3.1.A, TVA identified a leak on the Residual Heat Removal Shutdown Cooling root valve that could not be isolated. Following the confirmation that the through-wall leak was associated with an ASME Class I pressure boundary the test was terminated. Operations maintained the reactor in Mode 4 in accordance with the Technical Requirements Manual Action Statement 3.4.3.A.1. A weld performed in February of 1992, on the valve body and the tapered plug contained a defect. The root pass was made using the gas-tungsten process. This process was the industry standard in 1992. TVA postulates that a small amount of moisture present during the weld operation caused a defect in the weld. The defect slowly propagated and failed.

FACILITY NAME (1)	DOCKET (2)	LER NUMBER (6)			PAGE (3)
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
Browns Ferry Nuclear Plant Unit 2	05000260	2009	-- 002	-- 01	2 of 4

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

I. PLANT CONDITION(S)

Unit 2 was in Mode 4, performing a pressure vessel hydrostatic test with reactor pressure approximately 525 psig. Units 1 and 3 were operating in Mode 1 at 100 percent RTP (3458 megawatts thermal). Units 1 and 3 were unaffected by the event.

II. DESCRIPTION OF EVENT

A. Event:

On May 31, 2009, at approximately 1210 hours Central Daylight Time (CDT), during the performance of the Unit 2 pressure vessel hydrostatic test, ASME Section XI System Leakage Test of the Reactor Pressure Vessel and Associated Piping, 2-SI-3.3.1.A, TVA identified a leak on the Residual Heat Removal (RHR) [BO] Shutdown Cooling root valve [RTV] that could not be isolated. Following the confirmation that the leak was a through wall leak associated with an ASME Class I pressure boundary the test was terminated. Operations maintained the reactor in Mode 4 in accordance with Technical Requirements Manual (TRM) Action Statement 3.4.3.A.1.

On May 31, 2009, TVA initially reported this to the NRC under 10 CFR 50.72(b)(3)(ii)(A) as a leak in a weld between a 0.75-inch diameter leak-off line attached to the RHR Shutdown Cooling Root Valve (2-SHV-074-049). However, following the removal of the insulation, additional inspections determined that the leak-off line had been previously removed and a tapered plug was installed in its place. A through-wall leak was found at the interface weld between the tapered plug and the valve body. On June 2, 2009, TVA provided NRC with a revision to the May 31, 2009, 10 CFR 50.72(b)(3)(ii)(A) report.

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.

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

None.

C. Dates and Approximate Times of Major Occurrences:

May 31, 2009	1210 hours CDT	Identified leak.
May 31, 2009	1639 hours CDT	TVA made an 8-hour non-emergency report per 10 CFR 50.72(b)(3)(ii)(A).
June 2, 2009	1739 hours CDT	TVA provides an update to May 31, 2009, 8-hour non-emergency report.

D. Other Systems or Secondary Functions Affected

None.

E. Method of Discovery

TVA identified the leak through visual inspection during a scheduled performance of 2-SI-3.3.1.A.

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

F. **Operator Actions**

None.

G. **Safety System Responses**

None.

III. CAUSE OF THE EVENT

A. **Immediate Cause**

The immediate cause for the through wall leak was a flaw in the interface weld between the tapered plug and the 2-SHV-074-049 body.

B. **Root Cause**

A weld, performed in February of 1992, on the valve body and the tapered plug contained a defect. The root pass was made using the gas-tungsten process. This process was the industry standard in 1992. TVA postulates that a small amount of moisture that was present during the weld process caused a defect in the weld that was not identified during non-destructive examination (NDE). The flaw slowly propagated and resulted in the failure. The leak was detected on May 31, 2009, during the performance of 2-SI-3.3.1.A.

C. **Contributing Factors**

None.

IV. ANALYSIS OF THE EVENT

In February of 1992, TVA shut down Unit 2 due to excessive leakage in the drywell. The follow-up inspection identified a leak in the weld between a test pipe nipple and 2-SHV-074-049 valve body. The test nipple was removed and a tapered plug was installed in the valve body. During the repair, water intrusion between the valve and the tapered plug prevented proper moisture-free weld site preparation. To combat the moisture issue, TVA installed an o-ring on the tapered plug. The o-ring was intended to reduce the leakage long enough to allow the completion root pass. TVA completed the root pass of the weld using the gas-tungsten process. This process was the industry standard at the time the repair was made. Following successful inspection of the weld root pass, the weld was completed and final inspections performed.

On May 2, 2009, TVA repaired the weld made in February 1992. After excavating the original weld, a root pass was performed using the shielded-metal arc process. This process is less susceptible to moisture intrusion than the gas-tungsten and is now the industry standard. The completion of the weld activity was followed by successful completion of 2-SI-3.3.1.A.

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 exceeded, the TSs require the reactor be placed in Mode 4. During the previous operating cycle, reactor coolant leakage had been less than the TS limits.

The visual inspection, performed as part of 2-SI-3.3.1.A, identified the through-wall leak at the tapered plug and the valve body welded interface. BFN entered TRM Condition 3.4.3, Structural Integrity, Condition A.1, which requires either immediate restoration of structural integrity of the affected component or maintaining the reactor in Mode 4, or the reactor coolant system less than 50 degrees F above the minimum temperature required by nondestructive testing considerations, until each indication has been investigated and evaluated. Until completion of repairs, BFN maintained the reactor in accordance with these requirements. Therefore, TVA concludes that this event did not affect the health and safety of the public.

VI. CORRECTIVE ACTIONS

A. Immediate Corrective Actions

Once it was determined that the leak was part of the ASME Class I pressure boundary, Operations maintained the reactor in Mode 4.

B. Corrective Actions to Prevent Recurrence [1]

TVA repaired the weld. The completion of the weld repair was followed by successful completion of 2-SI-3.3.1.A.

VII. ADDITIONAL INFORMATION

A. Failed Components

None.

B. Previous LERs on Similar Events

None.

C. Additional Information

Corrective action document for this report is Problem Evaluation Report 172551.

D. Safety System Functional Failure Consideration:

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

E. Scram With Complications Consideration:

This event did not result in a complicated scram according to NEI 99-02.

VIII. COMMITMENTS

None.

[1] TVA does not consider this corrective action a regulatory requirement. TVA will track the completion of this action in the Corrective Action Program.