November 21, 2007

U.S. Nuclear Regulatory Commission 10 CFR 50.73
ATTN: Document Control Desk
Mail Stop: OWFN, P1-35
Washington, D. C. 20555-0001

Dear Sir:

TENNESSEE VALLEY AUTHORITY - BROWNS FERRY NUCLEAR PLANT (BFN) - UNIT 3 - DOCKET 50-296 - FACILITY OPERATING LICENSE DPR - 68 - LICENSEE EVENT REPORT (LER) 50-296/2007-003-00

The enclosed report provides details of a leak in an ASME Class I Code Reactor Pressure Boundary pipe. TVA is reporting this 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,

Original signed by:

Brian O'Grady

cc: See page 2

Enclosure
cc (Enclosure):

Ms. Eva Brown, Project Manager
U.S. Nuclear Regulatory Commission
(MS 08G9)
One White Flint, North
11555 Rockville Pike
Rockville, Maryland 20852-2739

Branch Chief
U.S. Nuclear Regulatory Commission
Region II
Sam Nunn Atlanta Federal Center
61 Forsyth Street, SW, Suite 23T85
Atlanta, Georgia 30303-8931

NRC Resident Inspector
Browns Ferry Nuclear Plant
10833 Shaw Road
Athens, Alabama 35611-6970

JEE:SWA:BAB
Enclosure
cc (Enclosure):

G.P. Arent, EQB 1B-WBN
W. R. Campbell, Jr. LP 3R-C
J. R. Douet, LP 3R-C
R. G. Jones, POB 2C-BFN
R. F. Marks, Jr., PAB 1C-BFN
D. C. Matherly, BFT 2A-BFN
L. E. Nicholson, BR 4X-C
B. A. Wetzel, BR 4X-C
S. A. Vance, WT 6A-K
E. J. Vigluicci, ET 11A-K
INPO:LEREvents@inpo.org
NSRB Support, LP 5M-C
EDMS WT CA–K,

S:lic/submit/lers/296 2007 003.doc

NRC FORM 366 (6-2004)	U.S. NUCLEAR REGULATORY COMMISSION	APPROVED BY OMB NO. 3150-0104 EXPIRES 06/30/2007
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:: 50 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 3	05000296	1 of 4

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	DOCKET NUMBER
									None	N/A
09	22	2007	2007	003	00	11	21	2007	FACILITY NAME None	DOCKET NUMBER N/A

9. OPERATING MODE: 3

10. POWER LEVEL: 0

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)	X	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, Licensing and Industry Affairs	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 September 22, 2007, at approximately 1000 hours Central Daylight Time (CDT), with BFN Unit 3 in Mode 3, an entry into the primary containment (drywell) identified an American Society of Mechanical Engineers (ASME) Class I reactor pressure boundary leak that could not be isolated. During the drywell entry, TVA identified a one inch test line associated with a Residual Heat Removal (RHR) system testable check valve had a through wall leak in a welded connection. At 1245 hours CDT, following confirmation that the leak was part of the ASME Class I pressure boundary; operations placed the reactor in Mode 4 per the Technical Specifications.

This report is submitted 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.

FACILITY NAME (1)	DOCKET (2)	LER NUMBER (6)			PAGE (3)
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
Browns Ferry Nuclear Plant Unit 3	05000296	2007	-- 003	-- 00	2 of 4

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

I. PLANT CONDITION(S)

Unit 3 was in Mode 3 approximately 960 psig. Units 1 and 2 were operating in Mode 1 at 100 percent RTP (3458 megawatts thermal). Units 1 and 2 were unaffected by the event.

II. DESCRIPTION OF EVENT

A. Event:

On September 22, 2007, at approximately 1000 hours Central Daylight Time (CDT), with BFN Unit 3 in Mode 3, an entry into the primary containment (drywell) identified an American Society of Mechanical Engineers (ASME) Class I reactor pressure boundary leak that could not be isolated. A one inch test line associated with a Residual Heat Removal (RHR) [BO] system testable check valve [CKV] had a through wall leak in a welded connection. BFN engineering personnel confirmed the leak was part of the American Society of Mechanical Engineers (ASME) Class I reactor pressure boundary. At 1245 hours CDT, following confirmation that the leak was part of the ASME Class I pressure boundary; operations placed the reactor in Mode 4 per the Technical Specifications.

This report is submitted 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:

September 22, 2007, at 0900 hours CDT	Unit 3 reactor was manually scrammed from 24 percent power, entering Mode 3.
September 22, 2007 at 1000 hours CDT	Identified reactor pressure boundary leak.
September 22, 2007, at 1245 hours CDT	Operations placed Unit 3 in Mode 4.
September 22, 2007, at 1535 hours CDT	TVA made an 8 hour non-emergency report per 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 inspection.

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 3	05000296	2007	-- 003	-- 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 for the through wall leak is the test connection experienced vibration induced failure.

B. Root Cause

The test connection set was not properly supported during Unit 3 operation. The U-bolt that clamped the test connection to the engineered pipe support was absent from the support. The reason for the missing U-bolt from the pipe support can only be postulated. The last work activity performed on the pipe support was in March of 2004, when the U-bolt clamp was removed to support a weld repair activity. Documentation reviewed as part of the investigation indicates the U-bolt clamp was replaced following the repair activity. TVA postulates that the U-bolt could have been either improperly installed, or the method for installation is inadequate.

C. Contributing Factors

The as found configuration contributed in the failure of the weld joint. A short nipple (approximately 1 inch in length) between an elbow and the test connection inboard isolation valve resulted in inadequate space for a two to one taper socket weld and a heat affected zone overlap, thus weakening the weld joint.

IV. ANALYSIS OF THE EVENT

Small bore piping in the Unit 3 drywell, including the test connections on the RHR system, is susceptible to Reactor Recirculation System vibration. The small bore piping has been evaluated and either removed or supported such that the vibration induced fatigue is eliminated.

In this event, the missing pipe support component (U-bolt clamp) rendered the pipe support ineffective. Although weld issues contributed in the failure, TVA postulates the as found test connection configuration would not have failed if properly supported.

V. ASSESSMENT OF SAFETY CONSEQUENCES

The safety consequences of this event were not significant. The TSs allows no reactor pressure boundary leaks in Mode 1. Increasing trend in unidentified primary containment leakage lead TVA to voluntarily shut down Unit 3 and enter a midcycle outage to perform a visual inspection in the primary containment to identify the leak path. The drywell unidentified leakage was well within the TS limits. The visual inspection identified a through wall leak in the inboard test connection for the RHR system testable check valve. Once it was determined the leak was in an un-isolable portion of the ASME Class I boundary, the Unit 3 reactor was taken to Mode 4 per the TSs.

The manual reactor scram was not complicated. No safety system operation was required as a result of the scram. Reactor water level following the scram was recovered and maintained by the reactor feed pumps. Therefore, TVA concludes that the health and safety of the public was not affected by this event.

VI. CORRECTIVE ACTIONS

A. Immediate Corrective Actions

Once it was determined that the leak was part of the ASME Class I reactor pressure boundary, per the TSs, operations placed Unit 3 in Mode 4.

FACILITY NAME (1)	DOCKET (2)	LER NUMBER (6)			PAGE (3)
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
Browns Ferry Nuclear Plant Unit 3	05000296	2007	-- 003	-- 00	4 of 4

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

B. Corrective Actions to Prevent Recurrence (1)

The 1-inch test line was removed and a cap was welded in its place.

The small bore pipe attached to the RHR system inside the primary containment was visually inspected for absent U-bolts.

TVA will perform a training needs analysis on the removal and installation of pipe support members. Based on the results of the analysis the appropriate personnel may receive training.

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 130777.

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. The completion of this action will be tracked in TVA's Corrective Action Program.