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

February 12, 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 50-260/2009-004-01**

The enclosed Licensee Event Report (LER) provides details of a Technical Specifications shutdown due to a rise in unidentified drywell leakage and a subsequent automatic scram due to a failure in the reactor protection system. This LER was revised to correct LER Section 11 and to provide an updated Abstract and Narrative description of the event.

The Tennessee Valley Authority is submitting this report in accordance with 10 CFR 50.73(a)(2)(i)(A), as the completion of any nuclear plant shutdown required by Technical Specifications, and in accordance with 10 CFR 50.73(a)(2)(iv)(A), as any event or condition that resulted in manual or automatic actuation of the Reactor Protection System (RPS) including: reactor scram or reactor trip.

There are no new regulatory commitments contained in this letter. Revisions are identified by bars in the right-hand margin. Should you have any questions concerning this submittal, please contact F. R. Godwin, Site Licensing and Industry Affairs Manager, at (256) 729-2636.

Respectfully,

K. J. Polson
Vice President

cc: See page 2

IE22
NRR

Enclosure
cc (w/ 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	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 Nuclear Plant Unit 2	05000260	1 of 6

4. TITLE: Technical Specifications Shutdown Due to Rise in Unidentified Drywell Leakage

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 N/A	DOCKET NUMBER N/A
06	11	2009	2009	- 004 -	01	02	12	2010	FACILITY NAME N/A	DOCKET NUMBER N/A

9. OPERATING MODE: 1

10. POWER LEVEL: 12

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 Specify in Abstract below or in NRC Form 366A
☐ 20.2203(a)(2)(vi)	☐ 50.73(a)(2)(i)(B)	☐ 50.73(a)(2)(v)(D)	

12. LICENSEE CONTACT FOR THIS LER

NAME	TELEPHONE NUMBER *(Include Area Code)*
Mike Oliver, Licensing Engineer	256-729-7874

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
B	SB	RV	T020	Y	X	SB	CKV	U000	N

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)*

At 1200 hours Central Daylight Time (CDT) on June 11, 2009, Browns Ferry Nuclear Plant Unit 2 experienced a rise in drywell leakage during reactor startup. The four-hour unidentified leak rate from 0800 to 1200 hours CDT on June 10, 2009, was 0 gallons per minute (GPM), while the four-hour unidentified leak rate from 0800 to 1200 hours CDT on June 11, 2009 was 3.88 GPM. This increase in leakage exceeded the Technical Specifications (TS) Limiting Condition for Operation (LCO) 3.4.4 limit of a 2 GPM increase in unidentified leakage in a 24 hour period. At 1555 hours CDT on June 11, 2009, Unit 2 initiated a reactor shutdown via a manual reactor SCRAM to comply with TS LCO 3.4.4 Condition C to be in Mode 3 in 12 hours and to be in Mode 4 within 36 hours. Following verification that the procedure, 2-AOI-100-1, Reactor Scram, actions were completed, the reactor mode switch was placed in Shutdown. The increase in unidentified leakage was due to failure of a Main Steam Line B Safety Relief Valve (SRV) to fully close. As a result of this steam leakage, two main steam SRV tailpipe vacuum breakers, 2.5 inch and 10 inch, were cycling. This SRV failure and vacuum breaker cycling allowed steam to enter the drywell instead of going to the torus.

Additionally, upon reset of the manual reactor scram, Reactor Protection System (RPS) 'B' scram channel did not reset as expected. At 1609 hours CDT on June 11, 2009, RPS Channel 'A' actuated a full reactor scram due to Intermediate Range Monitor 'C' spiking high and the inability to reset RPS 'B' scram channel. This automatic scram was found to be the result of a loose scram relay/contactor terminal connection.

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

I. PLANT CONDITION(S)

Prior to the event, Browns Ferry Nuclear Plant (BFN) Units 1 and 3 were operating in Mode 1 at 100 percent thermal power (approximately 3458 megawatts thermal). BFN Units 1 and 3 were unaffected by the event. BFN Unit 2 was in Mode 1 at approximately twelve percent and in power ascension following a refueling outage.

II. DESCRIPTION OF EVENT

A. Event:

During reactor startup from the BFN Unit 2 Spring refueling outage, a failure of a Main Steam (MS) Line B Safety Relief Valve (SRV) [SB] to fully close was revealed. Steam leakage through this SRV stopped when reactor pressure decreased to approximately 850 psig. The Tennessee Valley Authority (TVA) initially thought the steam leakage was due to pilot valve leakage because of observed discharge tailpipe indications and past experiences with pilot leakage. However, following destructive testing, it was determined to be steam leaking by the main valve body. As a result of this steam leakage, two MS SRV tailpipe vacuum breakers, 2.5 inch and 10 inch, were cycling. This SRV failure and vacuum breaker cycling allowed steam to enter the drywell instead of going to the torus.

At approximately 1200 hours Central Daylight Time (CDT) on June 11, 2009, BFN Unit 2 experienced an increase in drywell leakage during reactor startup. The four-hour unidentified leakage from 0800 to 1200 hours CDT on June 10, 2009, was 0 gallons per minute (GPM), the four-hour unidentified leakage from 0800 to 1200 hours CDT on June 11, 2009, increased to 3.88 GPM. This increase in Reactor Coolant System (RCS) operational leakage exceeded the Technical Specifications (TS) 3.4.4, RCS Operational Leakage, limit of a 2 GPM increase in unidentified leakage within the previous 24 hour period.

Therefore, the TS Limiting Condition for Operation (LCO) 3.4.4 was not met and at 1555 hours CDT on June 11, 2009, Unit 2 Operations personnel initiated a manual reactor scram to comply with TS 3.4.4 LCO Condition C, to be in Mode 3 in 12 hours and to be in Mode 4 within 36 hours.

During the reactor shutdown, all automatic functions resulting from the manual scram occurred as expected. All control rods [AA] inserted. No primary containment isolation system (PCIS) [JE] isolations were received.

Subsequently, at 1609 hours CDT on June 11, 2009, a full reactor scram occurred due to Intermediate Range Monitor 'C' spiking high concurrent with the inability to reset Reactor Protection System [JC] (RPS) 'B' scram channel.

Following verification that the 2-AOI-100-1, Reactor Scram, actions were completed, the reactor mode switch was placed in shutdown.

TVA is submitting this report in accordance with 10 CFR 50.73(a)(2)(i)(A), as the completion of any nuclear plant shutdown required by Technical Specifications, and in accordance with 10 CFR 50.73(a)(2)(iv)(A), as any event or condition that resulted in manual or automatic actuation of the RPS including: reactor scram or reactor trip.

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

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

None.

C. **Dates and Approximate Times of Major Occurrences:**

June 11, 2009, at 1555 hours CDT	Unit 2 reactor manually scrammed.
June 11, 2009, at 1609 hours CDT	Unit 2 full reactor scram occurred due to Intermediate Range Monitor 'C' spiking concurrent with inability to reset RPS 'B' scram channel.
June 11, 2009, at 1724 hours CDT	Operations made an Emergency Notification System report in accordance with 10 CFR 50.72(b)(2)(i)(B).

D. **Other Systems or Secondary Functions Affected**

None.

E. **Method of Discovery**

The annunciator for Drywell Floor Drain Sump Pump Excessive Operation was received in the Main Control Room.

F. **Operator Actions**

Operations personnel completed the shutdown as required by Technical Specifications 3.4.4 and entered 2-AOI-100-1, Reactor Scram.

G. **Safety System Responses**

The RPS logic responded to the manual reactor scram. All control rods inserted. No PCIS isolations were received.

RPS 'B' scram channel did not reset after the manual scram.

III. CAUSE OF THE EVENT

A. **Immediate Cause**

The immediate cause of the excessive RCS operational leakage was the failure of a MS Line B SRV to fully close. Also, two MS SRV tailpipe vacuum breakers, 2.5 inch and 10 inch, were cycling. This cycling allowed steam to enter the drywell instead of going to the torus.

B. **Root Cause**

There are two root causes for the excessive RCS operational leakage.

The first root cause of this event was identified as an inadequate original manufacturer-threaded main joint design that develops a fretting condition after years of service of a main body valve on a Target Rock Two or Three Stage SRV. Destructive examination at Wyle Laboratories identified that mating threads on the main valve piston-to-main valve stem were damaged to the point that the shaft appeared to be cocked approximately 1/4 inch, which would prevent the main body from cycling correctly. General Electric (GE) Service Information Letter (SIL) 646 documents this same condition, which previously occurred at Plant Hatch. GE SIL 646 is applicable to Target Rock Two or Three Stage SRVs, and BFN has Target Rock Two Stage

FACILITY NAME (1)	DOCKET (2)	LER NUMBER (6)			PAGE (3)
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
Browns Ferry Nuclear Plant Unit 2	05000260	2009	-- 004	-- 01	4 of 6

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

SRVs. The design detail dimensioning tolerance for the parts, when originally manufactured and assembled, permitted the lead thread of the piston to prematurely contact the under-cut area between the load bearing shoulder and the final thread on the stem of the main valve disc. In this condition, the torque (or applied preload) between the jam nut and the piston was lost during certification testing of the main valve body. Subsequent vibration of the loosened piston during normal plant operations allows the piston to fret the threads of the stem which can result in mechanical binding of the SRV.

The SRV steam leakage into the drywell occurred because the associated 2.5 inch vacuum breaker was found stuck open and the 10 inch vacuum breaker was found open with the spring mechanism found to be weak. These conditions were determined to be caused by excessive cycling due to the leaking SRV. Steam leakage was flowing down the tailpipe of the MS Line B SRV, through the open vacuum breakers, and into the drywell.

The second root cause of this event deals primarily with the failure to fully implement GE SIL 646 at BFN. Organizational to Organizational Interface Deficiencies were identified as the underlying root cause for failure to fully implement GE SIL 646. The GE SIL 646 recommended actions consisted of inspections and, as needed, interim modifications, which Target Rock would provide, over the next three refueling outages with follow-up inspections in the succeeding 6 to 10 years of service. For installed SRVs and spares, the population of valves to be inspected was recommended to be approximately one-third of the SRVs each outage. The inspection of other SRVs not installed or spares was also prescribed.

Per the cause analysis, previous implementation of GE SIL 646 at BFN consisted of two main parts: generation of work orders for valves outside of the SIL requirement and preventative maintenance (PM) work orders generated on the remaining valves. Review of both parts of the implementation revealed that a breakdown occurred during initial development of the refueling outage scope. The MS System Engineer generated the appropriate documentation to have GE SIL 646 implemented; however, the Valve Engineer identified only one main body to be replaced during refuel outages to coincide with past practices of changing only one main body in any given outage. Therefore, refuel outage work scope development and control were deficient.

The root cause for the failure of the RPS 'B' scram channel to not reset was found to be a loose scram relay/contactor terminal connection. On June 11th at 1602 CDT, the RPS 'B' scram channel did not reset as expected. Investigation found that the 5A-K14H scram relay/contactor was not re-energized due to a loose connection block. The connection was tightened and the relay/contactor operated correctly. This relay had been recently operated in support of the BFN Unit 2 refuel low power startup. Failure analysis has identified that the friction connection (pressure pad) of the power feed connection block was loose with the likely cause of less than adequate tightening during coil replacement two weeks prior to the event.

C. <u>Contributing Factors</u>

A significant component of the unidentified RCS operational leakage was a packing leak on the Reactor Vessel Drain Valve. The packing leak was caused by ineffective maintenance.

IV. ANALYSIS OF THE EVENT

During reactor startup from the BFN Unit 2 Spring refueling outage, a failure of a MS Line B SRV to fully close was revealed. Steam leakage through this SRV stopped when reactor pressure decreased to approximately 850 psig. TVA initially thought the steam leakage was due to pilot valve leakage because of observed discharge tailpipe indications and past experiences with pilot

FACILITY NAME (1)	DOCKET (2)	LER NUMBER (6)			PAGE (3)
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
Browns Ferry Nuclear Plant Unit 2	05000260	2009	-- 004	-- 01	5 of 6

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

leakage. However, following destructive testing, it was determined to be steam leaking by the main valve body.

The GE SIL 646 failure mode is described as loss of torque at the main valve piston-to-main valve stem threaded joint due to deformation of the leading edge of the piston threads. The failure mode occurs when the leading thread edge of the piston prematurely contacts the under-cut area between the load bearing shoulder and final thread on the stem of the main valve disc. In this condition, the torque (or applied preload) between the jam nut and the piston was lost during certification testing of the main valve body at a limited steam supply test facility. The loss of torque condition is undetectable without disassembly of the certified main valve body. When the main valve body is installed on the steam line header, the steam flow-induced vibration allows the piston to fret the threads of the stem. If the main valve body is subjected to this degradation process long enough, the entire threaded joint is compromised. The alignment between the piston and cylinder cannot be maintained when the SRV is required to open which can result in the mechanical binding of the SRV. The mechanism by which the main valve body opens is identical for both the mechanical and electrical opening mode. Therefore, the condition that resulted in the failure of 2-PCV-1-23 is applicable to both the mechanical and electrical operating modes of the SRV. The valve was installed at the 2-PCV-1-23 position since April 1999 and had not been inspected per the GE SIL 646 recommendations.

With respect to the vacuum breaker cycling, typically, these vacuum breakers do not cycle under normal plant conditions. The vacuum breakers were cycling due to the leaking 2-PCV-1-23. Normally, during a transient situation, the vacuum breakers could potentially cycle once and each time the SRV opened and closed. Since this was a unique event, the vacuum breakers cycled continually during the time of the leaking SRV or approximately 20 hours.

V. ASSESSMENT OF SAFETY CONSEQUENCES

The safety consequences of this event were not significant. The manual scram was not complicated. The operational impact was manageable during these valve failures. Operations reset the reactor scram at 1602 hours CDT.

Two cases identified in the industry where the SRV failure mechanism has occurred resulted in valves that either did not open or only partially opened. BFN Updated Final Safety Analysis Report (UFSAR) Section 14.5, Analysis of Abnormal Operational Transients – Uprated, includes various analyses, which result in a scram of the reactor from low power operation. Included are analyses for transients and accidents with only 12 of the 13 SRVs available for opening and for the inadvertent opening of a MSRV at limiting conditions, beginning of core life at rated core flow conditions. These analyses bound this actual condition or event. Further, based on the infrequent occurrence of this type of valve failure and the lack of a history of the GE SIL 646 failure mechanism from past inspections of BFN MS SRVs, there is a high confidence that the installed SRVs will perform their safety function.

With the exception of the RPS failure to reset, all safety systems operated as required during the manual scram. As expected, there were no PCIS Group 2, 3, 6, or 8 isolations. Although the Emergency Core Cooling Systems were available, none were required. No MS SRVs [SB] actuated. The turbine bypass valves [JI] maintained reactor pressure. The main condenser remained available for heat rejection. Reactor water level was recovered and maintained by the reactor feed water [SJ] and condensate [SG] systems. Therefore, TVA concludes that there was no significant reduction in the protection of the public by this event.

FACILITY NAME (1)	DOCKET (2)	LER NUMBER (6)			PAGE (3)
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
Browns Ferry Nuclear Plant Unit 2	05000260	2009	-- 004	-- 01	6 of 6

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

VI. CORRECTIVE ACTIONS

A. Immediate Corrective Actions

Operations performed the immediate actions of operating procedure "Relief Valve Stuck Open." The immediate actions were to identify the stuck open SRV by observing Safety Relief Valve Tailpipe Flow or Main Steam Relief Valve Discharge Tailpipe Temperature. Operations attempted to close the MS SRV, but it still indicated partially open and a work order was initiated.

B. Corrective Actions to Prevent Recurrence - The corrective actions to prevent recurrence are being managed by BFN's corrective action program.

The corrective actions to prevent recurrence are to complete an inspection and refurbishment of all affected main body valves installed on Units 2 and 3 at BFN in accordance with the GE SIL 646 recommended action. TVA will fully implement these recommendations.

The second root cause corrective actions are to revise procedures to ensure appropriate PM work orders are scheduled and to require additional rigor and documentation to initial outage scoping. A training needs analysis will be performed to determine training needs with regards to engineering responsibility for outage scope.

For the extent of condition evaluation, corrective actions also include performance of a review of previous, associated corrective action documents to determine if applicable GE SILs and GE Technical Information Letters (TILs) were appropriately implemented at BFN.

To address the relay/contactor loose connection, periodic verification of coil power termination tightness was added for each relay/contactor being inspected in the procedures.

VII. ADDITIONAL INFORMATION

A. Failed Components

Failed components are a MS Line 'B' SRV, associated vacuum breakers, and Reactor Vessel Drain Valve stem packing.

B. PREVIOUS LERS ON SIMILAR EVENTS

None.

C. Additional Information

Corrective action documents for this report are Problem Evaluation Reports 173480, 174037, and 174044.

D. Safety System Functional Failure Consideration:

This event is a not a safety system functional failure in accordance with NEI 99-02.

E. Scram With Complications Consideration:

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

VIII. COMMITMENTS

None.