TVA

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

August 10, 2009

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 (LER) 50-260/2009-004-000: Technical Specification Shutdown**

The enclosed report provides details of a Technical Specification shutdown due to a rise in unidentified drywell leakage. TVA is reporting this in accordance with 10 CFR 50.73(a)(2)(i)(A) (the completion of any nuclear plant shutdown as required by the plant's Technical Specifications). There are no commitments contained in this letter.

Please direct any questions concerning this matter to Russ Godwin (256) 729-2636.

Respectfully,

R. G. West
Site Vice President

cc: See Page 2

IE22
NRR

Enclosure
cc (Enclosure):

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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)	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: Technical Specification 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	DOCKET NUMBER
									None	N/A
06	11	2009	2009 -	004 -	00	08	10	2009	FACILITY NAME None	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
☐ 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)*
Deborah Bentzinger, Licensing Engineer	256-729-7533

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 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 leak rate exceeded the Technical Specification limit of 2 GPM increase in unidentified leakage in a 24 hour period per Technical Specification 3.4.4. At 1555 hours CDT on June 11, 2009, Unit 2 initiated a reactor shutdown via a manual reactor SCRAM to comply with Technical Specification 3.4.4 Condition C to be in Mode 3 in 12 hours and Mode 4 within 36 hours. Following verification that the 2-AOI-100-1, Reactor Scram actions were completed the reactor mode switch was placed in shutdown. Operations reset the reactor scram at 1602 hours CDT. The Reactor Protection System (RPS) scram solenoid Group B Groups 2 and 3 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' (RPS spiking high and the inability to reset RPS 'B' scram solenoids Groups 2 and 3). The increase in unidentified leakage was due to failure of the Main Steam Line B Safety Relief Valve to fully close. The two main steam relief valve tailpipe vacuum breakers were cycling which allowed steam to enter the drywell instead of the drywell torus.

LICENSEE EVENT REPORT (LER)

FACILITY NAME (1)	DOCKET (2)	LER NUMBER (6)			PAGE (2)
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
Browns Ferry Nuclear Plant Unit 2	05000260	2009	-- 004	-- 00	2 of 5

NARRATIVE

I. PLANT CONDITION(S)

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

II. DESCRIPTION OF EVENT

A. Event:

At approximately 1200 hours Central Daylight Time (CDT) on June 11, 2009, Browns Ferry 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), the four hour unidentified leak rate from 0800 to 1200 hours CDT on June 11, 2009, increased to 3.88 GPM. This increase in leak rate exceeded the Technical Specification limit of a 2 GPM increase in unidentified leakage in a 24 hour period. In accordance with TS 3.4.4, at 1555 hours CDT on June 11, 2009, Unit 2 initiated a manual reactor SCRAM to comply with Technical Specification 3.4.4 Condition C to be in Mode 3 in 12 hours and Mode 4 within 36 hours.

At 1609 hours CDT on June 11, 2009, a full Reactor SCRAM occurred due to Intermediate Range Monitor 'C' spiking high and the inability to reset RPS 'B' scram solenoids groups 2 and 3.

During the event, all automatic functions resulting from the scram occurred as expected. All control rods [AA] inserted. No PCIS isolations were received.

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). The completion of any nuclear plant shutdown as required by the plant's Technical Specifications.

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

None.

C. Dates and Approximate Times of Major Occurrences:

June 11, 2009	1555 hours CDT	Unit 2 reactor manually scrammed.
June 11, 2009	1724 hours CDT	TVA made a four hour non-emergency report per 10 CFR 50.72(b)(2)(i)(B)

D. Other Systems or Secondary Functions Affected

None.

E. Method of Discovery

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

F. Operator Actions

Operations personnel initiated the manual scram as required by Technical Specification 3.4.4. Following the manual scram, operations entered AOI-100-1, Reactor Scram.

NARRATIVE

G. **Safety System Responses**

The RPS logic responded to the reactor scram. All control rods inserted. No PCIS isolations were received. RPS scram solenoid group B groups 2 and 3 did not reset after the manual scram. This was due to a loose terminal connection.

III. CAUSE OF THE EVENT

A. **Immediate Cause**

The immediate cause of the event was the failure of the Main Steam Line B Safety Relief Valve to fully close. Also, two main steam relief valve 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 this event.

The first root cause of this event was inadequate design of original manufacturer threaded main joint design on the Main Steam Safety Relief Valve [SB]. Destructive examination performed by TVA 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. This prevented the main body from cycling correctly. Additionally, the 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. Steam leakage was flowing down the tailpipe of the Main Steam Line B Safety Relief Valve, through the open vacuum breakers and into the drywell.

The second root cause was the failure to fully implement GE SIL 646. Organizational to Organizational interface deficiencies were identified as the underlying root cause to failure to fully implement GE SIL 646. Work orders were generated for valves outside of the SIL requirement and preventative maintenance work orders on remaining valves. A breakdown occurred during initial development of outage scope which included knowledge deficiencies related to outage scope communication issues. The Main Steam System Engineer generated the appropriate documentation to have GE SIL 646 implemented. The Valve Engineer identified the one main body to be replaced during refuel outages to coincide with past practices of changing only one main body during any given outage.

C. **Contributing Factors**

A contributing cause of the event was a packing leak on the Reactor Vessel Drain Valve. The packing leak was caused by ineffective maintenance.

NARRATIVE

IV. ANALYSIS OF THE EVENT

The steam leakage through the Main Steam Relief Valve stopped when reactor pressure decreased to approximately 850 psig. TVA initially thought the steam leakage was due to pilot valve leakage. This was based upon the observed discharge tailpipe indications and past experiences with pilot leakage. However, following the 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 is 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 MSRV is required to open which can result in the mechanical binding of the MSRV. 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 MSRV. The valve was installed at the 2-PCV-1-23 position since April 1999. GE SIL 646 recommends an inspection frequency between 6 and 10 years.

Typically the vacuum breakers do not cycle under normal plant conditions. The vacuum breakers were cycling due to the leaking 2-PCV-1-23. During a transient situation, the vacuum breakers could potentially cycle once and each time the relief valve opened and close off. Since this was a unique event, the vacuum breakers cycled continually during the time of the leaking SRV which was approximately 20 hours.

V. ASSESSMENT OF SAFETY CONSEQUENCES

The safety consequences of this event were not significant. The manual scram was not complicated. Operations reset the reactor scram at 1602 hours CDT. BFN analysis includes a manual scram of the reactor from low power operation. With the exception of the Group B PCIS failing 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 main steam relief valves [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 the event did not affect the health and safety of the public.

NARRATIVE

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 relief valve by observing Safety Relief Valve Tailpipe Flow or Main Steam Relief Valve Discharge Tailpipe Temperature. Operations attempted to close the Main Steam Relief Valve but still indicated partially open and a work order was initiated.

B. Corrective Actions to Prevent Recurrence [1]

The corrective actions to prevent recurrence is to complete an inspection and refurbishment of all affected main body valves installed on Units 2 and 3 at Browns Ferry outside of the recommended 6 - 10 year inspection frequency that do not have the interim fix of GE SIL 646 or permanent fix of flexible piston provided by Target Rock. TVA will implement GE SIL 646.

The second root cause corrective actions are to revise procedures to ensure appropriate preventative maintenance 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.

VII. ADDITIONAL INFORMATION

A. Failed Components

Failed components are the Main Steam Line B Safety Relief Valve, 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 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.

[1] TVA does not consider these corrective actions as regulatory requirements. TVA will track the completion of these actions in the Corrective Action Program.