TVA

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

April 26, 2013

10 CFR 50.73

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

Browns Ferry Nuclear Plant, Unit 3
Facility Operating License No. DPR-68
NRC Docket No. 50-296

Subject: **Licensee Event Report 50-296/2013-003-00**

The enclosed Licensee Event Report provides details of an automatic reactor shutdown due to an actuation of the reactor protection system from a turbine trip. The Tennessee Valley Authority is submitting this report in accordance with Title 10 of the Code of Federal Regulations (10 CFR) 50.73(a)(2)(iv)(A).

There are no new regulatory commitments contained in this letter. Should you have any questions concerning this submittal, please contact J. E. Emens, Jr., Nuclear Site Licensing Manager, at (256) 729-2636.

Respectfully,

K. J. Polson
Vice President

Enclosure: Licensee Event Report 50-296/2013-003-00 – Automatic Reactor Shutdown Due to an Actuation of the Reactor Protection System From a Turbine Trip

cc (w/ Enclosure):

NRC Regional Administrator - Region II
NRC Senior Resident Inspector - Browns Ferry Nuclear Plant

IE22
NRC

ENCLOSURE

Browns Ferry Nuclear Plant
Unit 3

Licensee Event Report 50-296/2013-003-00

Automatic Reactor Shutdown Due to an Actuation of the Reactor Protection System From a Turbine Trip

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See Enclosed

NRC FORM 366 (10-2010)	U.S. NUCLEAR REGULATORY COMMISSION	APPROVED BY OMB NO. 3150-0104 EXPIRES 10/31/2013
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 FOIA/Privacy Section (T-5 F53), U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, or by internet e-mail to infocollects.resource@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: Automatic Reactor Shutdown Due to an Actuation of the Reactor Protection System From a Turbine Trip

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 05000
02	25	2013	2013	- 003 -	00	04	26	2013	FACILITY NAME N/A	DOCKET NUMBER 05000

9. OPERATING MODE: 1

10. POWER LEVEL: 092

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

FACILITY NAME	TELEPHONE NUMBER (Include Area Code)
Eric Bates, Licensing Engineer	256-614-7180

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	SJ	PSF	X000	Y					

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 February 25, 2013, at approximately 1313 hours Central Standard Time, the Browns Ferry Nuclear Plant (BFN), Unit 3, reactor automatically scrammed due to an actuation of the Reactor Protection System from a turbine trip. The turbine tripped on low condenser vacuum due to a reactor feedwater piping separation. The Main Steam Isolation Valves were manually closed. There was one Safety Relief Valve that was manually operated to maintain reactor pressure due to the unavailability of the Main Turbine Bypass Valves upon loss of condenser vacuum. All systems responded as expected to the turbine trip. No Emergency Core Cooling System or Reactor Core Isolation Cooling (RCIC) system reactor water level initiation set points were reached. Reactor water level was controlled with the RCIC system and reactor pressure was controlled with the High Pressure Coolant Injection system.

The root causes for this event are: the feedwater long cycle valve design is incorrect for its specified application and station personnel do not consistently consider risk when making decisions.

The corrective actions to prevent recurrence are: issue and implement design change to replace BFN, Units 1, 2, and 3, feedwater long cycle valves and implemented a Strategic Performance Management process to reinforce and institutionalize conservative decision making principles at BFN.

NARRATIVE

I. Plant Operating Condition Before the Event

At the time of the event, Browns Ferry Nuclear Plant (BFN), Unit 3, was in Mode 1 at approximately 92 percent power.

II. Description of Events

A. Event:

On February 25, 2013, at approximately 1313 hours Central Standard Time (CST), the BFN, Unit 3, reactor automatically scrammed due to an actuation of the Reactor Protection System (RPS) [JC] from a turbine trip. The turbine tripped on low condenser vacuum due to a reactor feedwater [SJ] piping separation. The Main Steam Isolation Valves (MSIVs) [ISV] [SB] were manually closed. There was one Safety Relief Valve (SRV) that was manually operated to maintain reactor pressure due to the unavailability of the Main Turbine Bypass Valves [JI] upon loss of condenser vacuum. All systems responded as expected to the turbine trip. No Emergency Core Cooling System (ECCS) [BJ][BO][BM] or Reactor Core Isolation Cooling (RCIC) system [BN] reactor water level initiation set points were reached. Reactor water level was controlled with the RCIC system and reactor pressure was controlled with the High Pressure Coolant Injection (HPCI) system. All control rods fully inserted and electrical offsite power was in a normal shutdown configuration and Residual Heat Removal was aligned for suppression pool cooling.

Also, Primary Containment Isolation System (PCIS) Groups 2, 3, 6, and 8 isolations were received due to low reactor water level. The PCIS consists of isolation valves that will automatically close as necessary to protect against the release of fission products, as well as, to conserve reactor coolant. Upon receipt of these isolations, the required components actuated, with the exception of one Group 6 valve, Drywell Continuous Air Monitor (CAM) Inboard Return Isolation Valve. The Drywell CAM Inboard Return Isolation Valve did not have indication following the isolation and was not able to be verified locally.

Due to the Drywell CAM Inboard Return Isolation Valve not actuating, BFN, Unit 3, Technical Specification (TS) Limiting Condition for Operation (LCO) 3.6.1.3 was entered. The BFN, Unit 3, TS LCO 3.6.1.3 requires that each Primary Containment Isolation Valve (PCIV) be Operable in reactor Modes 1, 2, and 3, and when the associated instrumentation is required to be Operable per LCO 3.3.6.1, "Primary Containment Isolation Instrumentation." On February 25, 2013, at approximately 1313 hours CST, TS 3.6.1.3 Condition A was entered due to one or more penetration flow paths with one PCIV inoperable. The TS 3.6.1.3 Required Action A.1 requires the affected penetration flow path to be isolated by use of at least one closed and de-activated automatic valve, closed manual valve, blind flange, or check valve with flow through the valve secured within four hours except for the main steam line which is required within eight hours. The TS 3.6.1.3 Required Action A.2 requires verification, once per 31 days, that the affected penetration flow path is isolated. Indication was subsequently restored following restoration of containment isolation signals, and the Drywell CAM Inboard Return Isolation Valve was manually isolated on February 25, 2013, at approximately 1422 hours CST with positive

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indication of isolation. On February 25, 2013, at approximately 1514 hours CST, the isolation valves were deactivated satisfying TS LCO 3.6.1.3 Required Actions.

On February 25, 2013, at approximately 1415 hours CST, suppression pool water level exceeded -1.0 inches. Suppression pool water level could not be maintained within normal bands due to running HPCI system for pressure control and suppression pool water level was not able to be pumped down due to the reactor feedwater piping separation.

The BFN, Unit 3, TS LCO 3.6.2.2 requires that suppression pool water level shall be greater than or equal to -6.25 inches with and -7.25 inches without differential pressure control and less than or equal to -1.0 inches in reactor Modes 1, 2, and 3. On February 25, 2013, at approximately 1415 hours CST, TS 3.6.2.2 Condition A was entered due to suppression pool water level not within limits. The TS 3.6.2.2 Required Action A.1 requires suppression pool water level to be restored within limits within two hours. On February 25, 2013, at approximately 1615 hours CST, TS 3.6.2.2 Condition B was entered due to required action and completion time not being met for TS 3.6.2.2 Condition A. The TS 3.6.2.2 Required Action B.1 requires the unit to be in Mode 3 (Hot Shutdown) within 12 hours and in Mode 4 (Cold Shutdown) within 36 hours. The BFN, Unit 3, entered Mode 3 on February 25, 2013, at approximately 1313 hours CST and Mode 4 on February 25, 2013, at approximately 2141 hours CST.

B. Status of structures, components, or systems that were inoperable at the start of the event and that contributed to the event:

The reactor feedwater long cycle valve return line connection to the miscellaneous drain header caused a rapid loss of condenser vacuum which caused the turbine trip.

C. Dates and approximate times of occurrences:

February 25, 2013, at 1313 hours CST	Reactor automatically scrammed due to actuation of the RPS from a turbine trip. Also, TS 3.6.1.3 Condition A was entered due to one or more penetration flow paths with one PCIV inoperable. The BFN, Unit 3, entered Mode 3.
February 25, 2013, at 1324 hours CST	The RCIC system was initiated to control reactor water level.
February 25, 2013, at 1326 hours CST	The HPCI system was initiated to control reactor pressure.
February 25, 2013, at 1415 hours CST	The TS 3.6.2.2 Condition A was entered due to suppression pool water level not within limits.

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February 25, 2013, at 1422 hours CST	Indication was subsequently restored following restoration of containment isolation signals, and the Drywell CAM Inboard Return Isolation Valve was manually isolated.
February 25, 2013, at 1514 hours CST	Isolation valves were deactivated to satisfy TS LCO 3.6.1.3 Required Actions.
February 25, 2013, at 1615 hours CST	The TS 3.6.2.2 Condition B was entered due to required action and completion time not being met for TS 3.6.2.2 Condition A.
February 25, 2013, at 1649 hours CST	The BFN reported event to the NRC.
February 25, 2013, at 2141 hours CST	The BFN, Unit 3, entered Mode 4.

D. Manufacturer and model number (or other identification) of each component that failed during the event:

A section of reactor feedwater piping, BFN-3-MISC-003, separated resulting in a loss of condenser vacuum.

E. Other systems or secondary functions affected:

There were no other systems or secondary functions affected.

F. Method of discovery of each component or system failure or procedural error:

Operations personnel observed a scram turbine generator load reject annunciator in the control room resulting in a reactor scram. The reactor feedwater piping separation was discovered by Operations personnel via a remote camera.

G. The failure mode, mechanism, and effect of each failed component, if known:

The failure mode of the feedwater long cycle return line connection to the miscellaneous drain header was due to excessive vibration. The excessive vibration was a result of pipe movement due to seat leakage of one or more of the feedwater long cycle valves. The valve leakage caused flashing in the drain header which caused subsequent failure of the miscellaneous drain header pipe wall. This failure caused a loss of vacuum on the turbine which caused the turbine to trip. When the turbine tripped on low condenser vacuum, an automatic reactor scram occurred.

H. Operator actions:

Operations personnel responded in accordance with Emergency Operating Instructions on Low Reactor Water Level. Also, Operations personnel responded in accordance with the Abnormal Operating Instructions for the automatic scram.

I. Automatically and manually initiated safety system responses:

The BFN, Unit 3, reactor automatically scrammed due to the actuation of the RPS from a turbine trip. The MSIVs were manually closed. There was one SRV that was

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manually operated to maintain reactor pressure due to the unavailability of the Main Turbine Bypass Valves upon loss of condenser vacuum. Reactor water level was controlled with the RCIC system and reactor pressure was controlled with the HPCI system.

III. Cause of the event

A. The cause of each component or system failure or personnel error, if known:

Direct Cause

The direct cause of the event was the failure of the feedwater long cycle return line connection to the miscellaneous drain header due to cyclic fatigue. The cyclic fatigue was a result of pipe movement due to seat leakage of one or more of the feedwater long cycle valves. The pipe movement caused the material stress on the pipe connection to be much greater than the endurance level of 10,000 pounds per square inch (psi) for the carbon steel material.

Root Cause

Valve design used for flow control valves 3-FCV-003-0071, -0072, and -0073 is incorrect for the specified application. This led to valve leakage which caused flashing in the drain header with subsequent failure of the miscellaneous drain header pipe wall due to excessive vibration. The wall failure caused a loss of condenser vacuum with a subsequent reactor scram. Currently, the valves used in this application are solid disc gate valves. These valves are not ideally suited for severe (high differential pressure) applications.

B. The cause(s) and circumstances for each human performance related root cause:

Root Cause

Station personnel do not consistently consider risk when making decisions. Work orders to replace the BFN, Unit 3, feedwater long cycle valves have been cancelled with little or no justification. This exposed BFN, Unit 3, to continued risk of failed piping downstream of the feedwater long cycle valves providing the potential for degraded, or loss of, condenser vacuum. Work orders written in 2007 to replace valves with a more suitable valve were never worked and were subsequently cancelled in 2010.

IV. Analysis of the event:

The Tennessee Valley Authority (TVA) is submitting this report in accordance with Title 10 of the Code of Federal Regulations (10 CFR) 50.73(a)(2)(iv)(A) as any event or condition that resulted in manual or automatic actuation of any of the systems listed in 10 CFR 50.73(a)(2)(iv)(B), including: RPS which includes a reactor scram or a reactor trip, general containment isolation signals affecting containment isolation valves in more than one system or MSIVs, HPCI, and RCIC.

The BFN, Unit 3, scrammed due to the 8 inch reactor feedwater long cycle return pipe breaking at the connection with the 24 inch miscellaneous drain header which caused a rapid loss of

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condenser vacuum. Once BFN, Unit 3, was shut down, a walkdown was performed to assess the 8 inch reactor feedwater long cycle return pipe for any additional damage and ensure the remainder of the piping system was per design requirements. During the walkdown, it was verified that the support spacing met code requirements for the 8 inch piping. Although no significant damage was noted during the walkdown, it was determined that the piping had been experiencing a significant amount of movement during the last operating cycle based on rub marks noted on the piping at individual support locations. It was determined that the excessive pipe movement was due to leakage past the normally closed feedwater long cycle flow control valve 3-FCV-003-0073. Also, normally closed feedwater long cycle flow control valves 3-FCV-003-0071 and 3-FCV-003-0072 were potentially leaking past their seats. When these normally closed valves leak past their seats, high pressure and temperature water was exposed to a low pressure (condenser vacuum) normally stagnant system which in turn caused the excessive pipe movement. The analysis used for this condition determined that material fatigue was the most probable cause of the failure of the connection with the 24 inch miscellaneous drain header due to the material stress being much greater than the endurance limit of 10,000 psi for the carbon steel material.

The feedwater long cycle valves are solid disc gate valves. These valves have continually been identified as leaking past their seats since installation and have been repaired numerous times. The repair to these valves was not a permanent fix because the valves continued to leak. The leakage from these valves has caused piping erosion that has lead to through wall leaks. Additionally, the leakage has caused failure of the feedwater long cycle piping at welding joints and at the wall of the miscellaneous drain header.

V. Assessment of Safety Consequences

The RPS provides timely protection against the onset and consequences of conditions that threaten the integrity of the fuel barrier and the nuclear system process barrier, i.e. fuel cladding and reactor coolant system pressure boundary respectively. The RPS is designed such that no single failure can prevent a reactor scram, and the RPS is designed to automatically shutdown the reactor based on parameters which deviate from normal.

A. Availability of systems or components that could have performed the same function as the components and systems that failed during the event:

The MSIVs were manually closed. There was one SRV that was manually operated to maintain reactor pressure due to the unavailability of the Main Turbine Bypass Valves due to the loss of condenser vacuum. All systems responded as expected to the turbine trip. No ECCS or RCIC system reactor water level initiation set points were reached. Reactor water level was controlled by the RCIC system and reactor pressure was controlled with the HPCI system.

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B. For events that occurred when the reactor was shut down, availability of systems or components needed to shutdown the reactor and maintain safe shutdown conditions, remove residual heat, control the release of radioactive material, or mitigate the consequences of an accident:

All safety systems remained available during this event and operated as designed.

C. For failure that rendered a train of a safety system inoperable, an estimate of the elapsed time from discovery of the failure until the train was returned to service:

There were no safety systems rendered inoperable as a result of this event.

Therefore, TVA concluded that there was minimal safety significance for this event.

VI. Corrective Actions

Corrective Actions are being managed by TVA's corrective action program under Problem Evaluation Reports (PERs) 516455 and 687732.

Immediate Corrective Actions

A Temporary Alteration Control Form was initiated to cut and cap the feedwater long cycle lines downstream of each feedwater long cycle valve and upstream of the miscellaneous drain header.

Corrective Actions to Prevent Recurrence

1. Issue and implement design change to replace BFN, Units 1, 2, and 3, feedwater long cycle valves.
2. Implemented a Strategic Performance Management process to reinforce and institutionalize conservative decision making principles at BFN.
3. Establish initial and continuing training requirements, and develop and deliver training to provide expected behaviors for leaders and craft that support their roles and responsibilities.

VII. Additional Information:

A. Previous similar events at the same plant:

A search of BFN Licensee Event Reports (LERs) for Units 1, 2, and 3 for the last several years identified LER 50-296/2013-002-00, Manual Actuation of Reactor Core Isolation Cooling System During Reactor Shutdown. This LER identified a similar condition concerning the reactor feedwater piping separation on the feedwater long cycle lines. The analysis for that LER was ongoing at the time of this event; therefore, the corrective actions for that LER would not have prevented this event from occurring.

A search was performed on the BFN corrective action program. Similar PERs related to the condition which caused the event reported in this LER are PERs 41131 and 52947.

LICENSEE EVENT REPORT (LER)
CONTINUATION SHEET

FACILITY NAME (1)	DOCKET (2)	LER NUMBER (6)			PAGE (3)
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
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PER 41131 states, "Chemistry identified a threat to the integrity of the welded pipe connection of the 8 inch long cycle return line at the miscellaneous drain header and potential condenser vacuum issues with movement observed from the line giving a strong indication of long cycle valve leakage with subsequent flashing occurring in the long cycle return line." The PER 41131 was closed to PER 52947 which identified that excessive movement of the long cycle piping caused the weld to the miscellaneous drain header to fatigue. The piping movement was caused by flashing in the line due to leakage through one or more of the feedwater long cycle valves. However, the corrective actions did not address the risk of failing to repair and maintain the valves that were the cause of the pipe movement which resulted ultimately in the failure of the 8 inch long cycle feedwater return line at the tie into the 24 inch miscellaneous drain header.

As a result of not addressing the actual failure mode and recognizing the risk associated with the long cycle return valves, BFN missed an opportunity to utilize available Operating Experience to prevent this event.

B. Additional Information:

There is no additional information.

C. Safety System Functional Failure Consideration:

In accordance with Nuclear Energy Institute (NEI) 99-02, this condition is not considered a safety system functional failure.

D. Scram with Complications Consideration:

In accordance with NEI 99-02, this event is considered an unplanned scram with complications.

VIII. COMMITMENTS

There are no commitments.