TVA

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

April 1, 2011

10 CFR 50.73

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

Browns Ferry Nuclear Plant, Unit 1
Facility Operating License No. DPR-33
NRC Docket No. 50-259

Subject: **Licensee Event Report 50-259/2010-005-01**

On February 4, 2011, the Tennessee Valley Authority (TVA) submitted Revision 0 to Licensee Event Report 50-259/2010-005 stating that TVA will provide additional details after completing the investigation and evaluation of the condition prohibited by Unit 1 Technical Specification 3.4.3 concerning Safety/Relief Valve operability.

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 - Unit 1 Safety/Relief Valves As-Found Setpoints Exceeded Technical Specification Lift Pressure Values

cc (w/ Enclosure):

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

IE22
NRR

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Enclosure
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Enclosure

Browns Ferry Nuclear Plant
Unit 1

Licensee Event Report - Unit 1 Safety/Relief Valves As-Found Setpoints Exceeded Technical Specification Lift Pressure Values

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SEE ATTACHED

NRC FORM 366 (10-2010)	U.S. NUCLEAR REGULATORY COMMISSION	APPROVED BY OMB NO. 3150-0104 EXPIRES 10/13/2013
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.

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4. TITLE
Safety/Relief Valves As-Found Setpoints Exceeded Technical Specification Lift Pressure Values

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
									N/A	05000
12	06	2010	2010 -	005 -	01	04	01	2011	N/A	05000

9. OPERATING MODE: 1

10. POWER LEVEL: 100

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	SB	RV	T020	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 December 6, 2010, the Tennessee Valley Authority (TVA) determined, as a result of offsite laboratory surveillance testing, 3 of 13 Browns Ferry Nuclear Plant - Unit 1 Safety/Relief Valves (S/RVs) mechanically actuated at pressures greater than the allowed 3 percent above their Technical Specification (TS) setpoint. Therefore, the S/RVs were inoperable for an indeterminate period during the previous Cycle 8 period of operation. Unit 1 TS Limiting Condition for Operation 3.4.3 requires the safety function of twelve (12) S/RVs to be operable in reactor modes 1, 2, and 3. With one or more required S/RVs inoperable, the unit is required to be placed in Mode 3 (hot shutdown) within 12 hours and in Mode 4 (cold shutdown) within 36 hours. Since 3 of 13 S/RVs actuated above their TS setpoint plus the 3 percent allowance, it is probable that Unit 1 operated longer than allowed by the TS.

There are two causes of the S/RVs exceeding the setpoint lift tolerance: (1) corrosion bonding between the disc and seat while the valve is in service, and (2) the thermal heat-up conditions used in the laboratory testing were not consistent among all 13 S/RVs.

TVA has revised administrative controls to ensure that the test facility provides a consistent heat-up rate for all 13 S/RVs tested pursuant to TS surveillance requirement 3.4.3.1. TVA will continue the application of the platinum coating to address corrosion bonding between the seat and disc.

NARRATIVE

I. PLANT CONDITIONS

At the time of discovery, Browns Ferry Nuclear Plant (BFN) - Unit 1 was at approximately 100 percent power (3458 MWT) and unaffected by the event since all 13 of the Reactor Coolant System Safety/Relief Valves [SB] (S/RVs) had been refurbished during the recently completed refueling outage.

II. DESCRIPTION OF EVENT

A. Event

On December 6, 2010, the Tennessee Valley Authority (TVA) determined, as a result of offsite laboratory surveillance testing, 3 of 13 BFN - Unit 1 S/RVs mechanically actuated at pressures greater than the allowed 3 percent above their Technical Specification (TS) setpoint. The S/RVs were thus inoperable for an indeterminate period during the previous Cycle 8 period of operation. Unit 1 TS Limiting Condition for Operation 3.4.3 requires the safety function of twelve (12) S/RVs to be operable in reactor modes 1, 2, and 3. With one or more required S/RVs inoperable, the unit is required to be placed in Mode 3 (hot shutdown) within 12 hours and in Mode 4 (cold shutdown) within 36 hours. Since 3 of 13 S/RVs actuated above their TS setpoint plus the 3 percent allowance, it is probable that Unit 1 operated longer than allowed by the TS.

TVA is submitting this report in accordance with 10 CFR 50.73(a)(2)(i)(B), as any operation or condition prohibited 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

December 2, 2008	During the Unit 1 Cycle 7 refueling outage, S/RV pilot cartridges tested to meet TS setpoint requirements were installed.
October 23, 2010, at 0900 hours	Operations personnel entered a planned Manual Scram in accordance with plant procedures to end Unit 1 Cycle 8 operation.
November 23, 2010	Unit 1 startup from the Cycle 8 refueling outage with refurbished S/RVs set within the TS setpoint requirements.
December 6, 2010, at 1052 hours	TVA documents that the as-found lift setpoint for 3 S/RVs exceeded the allowable TS value plus allowance during the Unit 1 Cycle 8 operating cycle in Problem Evaluation Report (PER) 294506.

D. Other Systems or Secondary Functions Affected

None

E. Method of Discovery

The out-of-tolerance lift setpoints were identified during the performance of Surveillance Procedure 0-SR-3.4.3.1.B, “Bench Test Relief Valves As-Found,” at the test facilities of Wyle Laboratories, Huntsville, Alabama.

F. Operator Actions

None

NARRATIVE

G. **Safety System Responses**

None

III. CAUSE OF THE EVENT

A. **Immediate Cause**

The immediate cause for this reportable condition is undetectable, out-of-tolerance, high-lift setpoints on 3 of 13 S/RVs, which existed for longer than allowed by the TS.

B. **Root Cause**

There are two causes of the S/RVs exceeding the setpoint lift tolerance: (1) corrosion bonding between the disc and seat while the valve is in service, and (2) the thermal heat-up conditions used in the laboratory testing were not consistent among all 13 S/RVs.

C. **Contributing Factors**

None

IV. ANALYSIS OF THE EVENT

The condition being reported is the operation of Unit 1 in a manner prohibited by TS. The as-found S/RV lift setpoints following Unit 1 Cycle 8 operation are summarized in the following table.

Unit 1 Cycle 8 As-Found Lift Setpoints(1)

S/RV Unique Identification Number	Pilot Valve Serial Number	S/RV TS Setpoint	1st Test / Dev.	2nd Test / Dev.	3rd Test / Dev.
1-PCV-001-0004	1068	1155	1191 / 3.1%	1155 / 0.0%	1160 / 0.4%
1-PCV-001-0005	1075	1145	1158 / 1.1%	1159 / 1.2%	1166 / 1.8%
1-PCV-001-0018	1018	1145	1160 / 1.3%	1136 / -0.8%	1151 / 0.5%
1-PCV-001-0019	1059	1135	1103 / -2.8%	1102 / -2.9%	1107 / -2.5%
1-PCV-001-0022	1028	1145	1153 / 0.7%	1150 / 0.4%	1147 / 0.2%
1-PCV-001-0023	1033	1135	1312 / 15.6%	1150 / 1.3%	1148 / 1.1%
1-PCV-001-0030	1076	1145	1150 / 0.4%	1142 / -0.3%	1143 / -0.2%
1-PCV-001-0031	1267	1135	1137 / 0.2%	1143 / 0.7%	1147 / 1.1%
1-PCV-001-0034	1020	1135	1133 / -0.2%	1133 / -0.2%	1132 / -0.3%
1-PCV-001-0041	1085	1155	1167 / 1.0%	1143 / -1.0%	1142 / -1.1%
1-PCV-001-0042	1027	1155	1239 / 7.3%	1177 / 1.9%	1178 / 2.0%
1-PCV-001-0179	1032	1155	1166 / 1.0%	1164 / 0.8%	1167 / 1.0%
1-PCV-001-0180	1016	1155	1150 / -0.4%	1150 / -0.4%	1159 / 0.3%

(1) The shaded values indicate test results outside the TS required 3 percent tolerance

The BFN S/RVs are Target Rock Model 7567F two-stage S/RVs The valve is a leak tolerant valve; however, it exhibits significant in-service setpoint drift because of corrosion bonding between the valve seat and pilot disc. The pilot valve seats are constructed from erosion and wear resistant Stellite 6B. The Stellite alloy develops a hard, metal-oxide corrosion layer on the pilot disc. When placed in an operating environment typical of a boiling water reactor, the steam-exposed surfaces can oxidize, forming a surface corrosion film. This corrosion film forms a bond between the valve seat and disc. The bond adds to the resistance of the setpoint adjustment spring pressure necessary to open the valve and increases the pressure required to actuate the valve.

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As corrective action and recurrence control from previous similar events, the pilot discs installed for Unit 1 Cycle 8 operation were coated with platinum to mitigate corrosion bonding between the pilot disc and seat. The results from Unit 1 Cycle 8 operation were 3 S/RVs failed to meet setpoint tolerances, compared to previous testing when 10 S/RVs failed to meet setpoint tolerances after Unit 1 Cycle 7 operation. The results indicate that platinum coating has significantly decreased the failure rate of the S/RVs.

A different heat-up rate applied during testing of the first two S/RVs contributed to lift setpoint values above acceptance criteria, when compared to the other eleven S/RVs tested. The third S/RV that failed was 3.1 percent above the setpoint versus the 3.0 percent acceptance criteria, which indicates that platinum coating may not completely prevent bonding of the valve disc to the seat.

V. ASSESSMENT OF SAFETY CONSEQUENCES

The reactor vendor, General Electric Hitachi Nuclear Energy, has completed an analysis of the safety consequences of this event by comparing it to the safety consequences resulting from the Unit 1 Cycle 7 S/RV test failures. The safety consequences of this event were not significant based on supporting overpressure protection and Anticipated Transients Without Scram (ATWS) evaluations.

Vessel overpressure analyses were performed for the limiting Unit 1 Cycle 7 and Cycle 8 transient events to demonstrate compliance with the American Society of Mechanical Engineers (ASME) overpressure limit, which is 1375 psig for maximum vessel (bottom) pressure during the design basis event. For Unit 1, the design basis event is the main steam isolation valve closure with high neutron flux scram (MSIVF) at the end of cycle exposure. The MSIVF event was evaluated at both increased core flow and maximum extended load line limit analysis (MELLLA) core flow conditions. A comparison of the MSIVF events between two cycles (Cycles 7 and 8) demonstrates that the initial conditions and peak-pressure results are the same or very similar.

The Cycle 7 specific analysis of the MSIVF event was performed consistent with the Unit 1 Cycle 7 analysis, with the exception that the Cycle 7 as-found S/RV opening setpoints were modeled and credit was taken for all 13 S/RVs. The peak dome pressure and peak vessel (bottom) pressure results were compared to the Unit 1 Cycle 7 and Cycle 8 limiting values. The peak vessel pressure (1332 psig) remains below the ASME upset code limit (1375 psig). The peak dome pressure (1304 psig) remains below the TS safety limit (1325 psig).

Given that the MSIVF event peak pressure results are virtually the same for Cycles 7 and 8, if Cycle 8 had the same as-found S/RV opening pressures as Cycle 7, then the peak pressure results would be similar to those calculated previously. However, the as-found S/RV opening pressures for Cycle 8 are lower than the as-found S/RV opening pressures for Cycle 7. A comparison of the Cycle 8 as-found valve data to the Cycle 7 as-found valve data shows that the same or more valves would be open for Cycle 8 than for Cycle 7 across the entire range of opening pressures. Therefore, the peak pressure results for Cycle 8 as-found S/RV setpoints would be lower than the Cycle 7 as-found S/RV setpoints, and the ASME upset code pressure limit is met.

ATWS overpressure analyses were performed for the limiting Unit 1 event to demonstrate compliance with the ASME Service Level C Limit (1500 psig). For Unit 1, the limiting event is the failure of the pressure regulator in open position (PRFO) with a failure to scram at the beginning of cycle exposure. The PRFO event was evaluated at the limiting MELLLA core flow conditions. The plant-specific analysis of the PRFO event was performed consistent with the ATWS analysis with the exception that the Cycle 7 as-found S/RV opening setpoints were modeled and credit was taken for all 13 S/RVs.

NARRATIVE

The peak vessel (bottom) pressure of 1463 psig remains below the ASME Service Level C Limit (1500 psig). Because of the similarities between Cycles 7 and 8, and because there are the same or more Cycle 8 as-found S/RVs open at a given pressure (assuming credit for all 13 S/RVs) for all pressures than for the Cycle 7 as-found S/RVs, the ATWS analyses bounds the expected response for Cycle 8.

The safety and power generation design basis are met by the three functional modes of the S/RVs.

- Overpressure relief operation to limit and control reactor pressure.

  This functional mode consists of a remote manual mode and S/RV automatic actuation logic. The remote manual mode allows all 13 S/RVs to be electrically opened using hand switches in the Main Control Room. The S/RV automatic actuation logic allows an electrical open signal generated from an electrical logic bus which is designed to back-up the mechanical setpoint of the S/RVs.

- Overpressure safety operation to limit reactor pressure.

  The valves are opened (mechanical self-actuation) to prevent exceeding the design allowable stress limits on the reactor vessel and associated piping.

- Automatic depressurization system operation allows 6 of 13 S/RVs to be available to open automatically with an electrical signal as part of emergency core cooling system.

The pilot cartridge, within the design of the Target Rock Model 7567F, provides the mechanical and electrical control of the S/RV. The pilot cartridge contains the components which determine mechanical setpoint, pneumatic interface, and electrical control air solenoid.

Therefore, TVA has concluded that there was no significant reduction in the protection of the public by this event.

VI. CORRECTIVE ACTIONS

A. Immediate Corrective Actions

All 13 S/RV pilot cartridges were replaced during the Unit 1 Cycle 8 refueling outage. Prior to installation in the unit, each of the replacement cartridges with platinum coated pilot discs demonstrated a lift setpoint within the as-left TS requirements (i.e., plus or minus one percent following testing) during bench testing.

B. Corrective Actions to Prevent Recurrence

TVA has revised administrative controls to ensure that the test facility provides a consistent heat-up rate for all 13 S/RVs tested persuant to TS surveillance requirement 3.4.3.1. TVA will continue the application of the platinum coating to address corrosion bonding between the seat and disc.

VII. ADDITIONAL INFORMATION

A. Failed Components

None

B. Previous Similar Events

TVA has submitted previous reports on similar events at BFN. LERs 50-259/2008-003-00, 50-260/2009-003-00, and 50-296/2010-001-00 are the most recent LERs. The previous LER for Unit 1 reported probable inoperability of 10 of 13 S/RVs during Cycle 7 operation.

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NARRATIVE

C. **Additional Information**

The corrective action documents for this report are PER 294506 and PER 318643.

D. **Safety System Functional Failure Consideration**

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

E. **Scram With Complications Consideration**

This event did not include a reactor scram.

VIII. COMMITMENTS

None