TVA

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

April 1, 2011

10 CFR 21
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 50-259/2010-003, Revision 1**

Reference: Letter from TVA to NRC, "Licensee Event Report 50-259/2010-003, Revision 0," dated December 22, 2010

As indicated in the referenced letter, TVA completed the investigation and evaluation and the enclosed Licensee Event Report (LER) revision provides details of a failure of a low pressure coolant injection flow control valve. The Tennessee Valley Authority (TVA) is submitting this revised report in accordance with 10 CFR 21.2(c), reporting of defects and noncompliance; 10 CFR 50.73(a)(2)(i)(B), any operation or condition prohibited by the plant's Technical Specifications; and 10 CFR 50.73(a)(2)(v)(D), any event or condition that could have prevented fulfillment of the safety function or structures or systems that are needed to mitigate the consequences of an accident.

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 - Failure of a Low Pressure Coolant Injection Flow Control Valve

IE22
NRR

cc (w/ Enclosure):

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

ENCLOSURE

Browns Ferry Nuclear Plant
Unit 1

Licensee Event Report - Failure of a Low Pressure Coolant Injection Flow Control Valve

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
Failure of a Low Pressure Coolant Injection Flow Control Valve

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
10	23	2010	2010 -	003 -	01	04	01	2011	FACILITY NAME N/A	DOCKET NUMBER N/A

9. OPERATING MODE	11. THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: *(Check all that apply)*			
3	☐ 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)
10. POWER LEVEL	☐ 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)
000	☐ 20.2203(a)(2)(v)	☐ 50.73(a)(2)(i)(A)	☐ 50.73(a)(2)(v)(C)	☒ OTHER - Part 21
	☐ 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

FACILITY 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	BO	FCV	W030	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 October 23, 2010, during a refueling outage for Browns Ferry Nuclear Plant (BFN) Unit 1, the Tennessee Valley Authority (TVA) discovered that a Residual Heat Removal (RHR) Loop II low pressure coolant injection (LPCI) flow control valve failed to open while attempting to establish shutdown cooling (SDC) while in Mode 3. Operations personnel declared RHR Loop II inoperable for ECCS and placed RHR Loop I in service for SDC.

Unit 1 Technical Specification (TS) limiting condition for operation (LCO) 3.5.1, Emergency Core Cooling System (ECCS) - Operating, requires both RHR loops of LPCI to be operable in reactor Modes 1, 2, and 3. Investigation of the valve failure to open determined that the root cause was a manufacturer's defect resulting in undersized disc skirt threads at disc connection. Based on causal analysis information, the stem-to-disc separation occurred sometime before November 2008. Thus, RHR Loop II was inoperable for a period longer than the 7 days allowed by TS 3.5.1.

This report constitutes a Part 21 notification.

NARRATIVE

I. PLANT CONDITION(S)

At the time of discovery, Browns Ferry Nuclear Plant (BFN) Unit 1 was at 0 percent power (Mode 3, Hot Shutdown) and in a refueling outage.

II. DESCRIPTION OF EVENT

A. Event:

On October 23, 2010, the BFN Unit 1 Residual Heat Removal (RHR) [BO] Loop II low pressure coolant injection (LPCI) flow control valve [FCV], 1-FCV-074-066, failed to open while attempting to place RHR Loop II in shutdown cooling (SDC). Control Room lights indicated the valve to be open, but no flow was indicated for RHR Loop II with the associated 1B RHR pump in service. RHR Loop I was then successfully placed in service for SDC.

Investigation of the event determined that the 1-FCV-074-066 disc had become separated from the skirt/stem and wedged into the seat, preventing immediate SDC flow.

Unit 1 Technical Specification (TS) limiting condition for operation (LCO) 3.5.1 requires both RHR loops of LPCI to be operable in reactor Modes 1, 2, and 3. At the time of discovery, the reactor was in Mode 3. Operations personnel declared RHR Loop II inoperable for the LPCI function of the Emergency Core Cooling System (ECCS) and entered TS LCO 3.5.1 Condition A with a required action to restore RHR Loop II to operable status within 7 days. With RHR Loop I operable, two RHR SDC subsystems remained operable and TS LCO 3.4.7 for RHR Shutdown Cooling System - Hot Shutdown was satisfied.

Within one hour of the determination of RHR Loop II inoperability, Unit 1 entered Mode 4, Cold Shutdown. Since TS LCO 3.5.1, ECCS - Operating, is not applicable in Mode 4, Operations personnel exited TS LCO 3.5.1 Condition A. TS LCO 3.5.2, ECCS - Shutdown, became applicable and requires two low pressure ECCS injection/spray subsystems to be operable. At that time, Unit 1 had both Core Spray (CS) [BM] subsystems operable and one RHR subsystem operable for ECCS. Note that RHR Loop I was operable for LPCI in accordance with TS LCO 3.5.2, which states that one LPCI subsystem may be considered operable during alignment and operation for decay heat removal if capable of being realigned and not otherwise inoperable.

Investigation of the valve failure to open determined that the root cause was a manufacturing defect, undersized disc skirt threads at the disc connection. Based on causal analysis information, the valve stem-to-disc separation occurred sometime before November 2008. Thus, RHR Loop II was inoperable for a period longer than the 7 days allowed by TS 3.5.1.

TVA is submitting this report in accordance with 10 CFR 21.2(c), reporting of defects and non-compliance; 10 CFR 50.73(a)(2)(i)(B), any operation or condition prohibited by the plant's Technical Specifications; and 10 CFR 50.73(a)(2)(v)(D), any event or condition that could have prevented fulfillment of the safety function or structures or systems that are needed to mitigate the consequences of an accident. The past inoperability is based on the inability of RHR Loop II to provide LPCI within its analyzed time. The exact date at which RHR Loop II would have failed to meet its analyzed

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LPCI time is difficult to determine with certainty. However, violations of TS LCOs 3.5.1 and 3.5.2 most likely occurred since March 13, 2009, based on other low pressure ECCS system inoperability due to maintenance and testing. Additionally, since that time, because the degraded condition was not recognized, LCO 3.0.4 was not met due to mode change. Based on NUREG-1022 guidance of event date reporting, for reporting purposes, the discovery date will be retained as the event date.

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

BFN Unit 1 RHR Loop II LPCI flow control valve, 1-FCV-074-066, failed to pass flow with the valve operator in the open position. 1-FCV-074-066 is physically located in the RHR LPCI Loop II flow path and is normally in the full-open position for the passive safety-related function of the valve. This is the outboard valve in the injection path, used to throttle SDC flow and closed to divert flow from the LPCI flow path when containment cooling is desired. There were six of these type valves installed during initial construction of the Browns Ferry Nuclear Plant, two per unit with one in each loop of LPCI. The component is a Walworth Company 24-inch No. 5297PS, 600-lb MSS SP-66 Rating cast carbon steel, butt welded, pressure-seal angle globe valve operated by a Limitorque SMB-5T-350 motor operator.

The disc skirt is threaded into the back of the disc and secured by tack welds; this assembly surrounds the bottom section of the stem that includes a larger diameter shoulder at the base. The shouldered stem base below the skirt transfers opening thrust to the disc through the skirt/disc connection, and closing thrust directly to the disc.

C. Dates and Approximate Times of Major Occurrences:

1968-1969 timeframe	Timeframe of the original installation of BFN Unit 1 LPCI flow control valve 1-FCV-074-066.
June 2006	1-FCV-074-066 was refurbished prior to Unit 1 restart after an extended outage.
Before November 2008	1-FCV-074-066 stem-to-disc separation occurred based on motor-operated valve performance data.
March 13, 2009, at 0553 hours	During the Unit 1 Cycle 7 refueling outage, RHR Loop II was in service for SDC. When SDC was secured per RHR System Operating Instruction (OI) 1-OI-74, 1-FCV-074-066 was closed (this was the last confirmed successful operation of the valve).
October 23, 2010, at 1417 hours	During the Unit 1 Cycle 8 refueling outage, Operations personnel attempted to place RHR Loop II in service for SDC in accordance with 1-OI-74. Flow could not be confirmed. The issue was entered in the TVA Corrective Action Program and Problem Evaluation Report, PER 271338, was subsequently initiated.

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October 23, 2010, at 1433 hours	Operations personnel placed RHR Loop I in SDC in accordance with 1-OI-74.
October 23, 2010, at 1505 hours	Unit 1 entered Mode 4.
March 14, 2011	TVA determines that the root cause of the valve failure is reportable in accordance with 10 CFR 21.

D. Other Systems or Secondary Functions Affected:

None

E. Method of Discovery:

The valve failure was discovered during the performance of BFN Unit 1 system operating instruction 1-OI-74, "Residual Heat Removal System," Section 8.12.2, "Initiation/Operation of RHR Loop II in Shutdown Cooling."

F. Operator Actions:

Operations personnel declared RHR Loop II inoperable for ECCS and placed RHR Loop I in service for SDC.

G. Safety System Responses:

None

III. CAUSE OF THE EVENT

A. Immediate Cause:

The immediate cause for this condition was separation of the valve disc from the stem/skirt, with the disc wedged into the seat in the closed position.

B. Root Cause:

The root cause of this event was a manufacturing defect, undersized disc skirt threads at the disc connection. Original manufacturer's documentation specified design requirements for the threaded connection which were not met in the disc skirt that was removed from the valve. The disc skirt was part of an original assembly installed during construction of BFN Unit 1 (installed in 1968-69 timeframe). The safety-related 24-inch globe valve was manufactured by the Walworth Company which has since been acquired by Crane Nuclear, Inc. Discussions with the vendor indicated that no historic inspection documentation of the valve internals is available. TVA determined that the disc skirt threaded connection dimensions were not as denoted on the vendor drawing. No receipt inspection of an assembly of this nature and classification would have been required, and the manufacturer provided certification documentation to the quality level specified. A review of the valve maintenance history indicates the valve skirt was part of the original assembly and has not been replaced with other parts by substitution.

IV. ANALYSIS OF THE EVENT

The condition being reported is the operation of Unit 1 in a manner prohibited by TS, the loss of a safety function, and a defect in a basic component.

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BFN Unit 1 RHR Loop II LPCI flow control valve, 1-FCV-074-066, is normally in the open position for the passive safety-related function of the valve. The valve is used to throttle

SDC flow and is closed to divert flow from the LPCI flow path when containment cooling is desired. It is the outboard valve in the injection path.

On October 23, 2010, BFN Unit 1, 1B RHR pump was started to provide SDC of the reactor core in support a refueling outage. After 110 seconds of observing no flow to the reactor, the pump was secured. Subsequent investigation discovered the 1-FCV-074-066 disc had become separated from the stem and disc skirt and lodged into the valve seat preventing flow to the reactor. Operations personnel secured RHR Loop II SDC and established SDC using RHR Loop I in accordance with 1-OI-74. Following preliminary investigations, the failed valve was reworked and tested, and RHR Loop II was returned to service.

The TVA investigation determined the root cause of the stem-to-disc separation was a manufacturer's defect, undersized disc skirt threads at skirt/disc connection. The failure mechanism was opening thrust exceeding the strength of the threaded connection between the disc skirt and disc. The over thrust condition was in the axial direction in which back pressure on the top of the valve disc exceeded the capacity of the threaded connection, allowing the valve skirt and stem to pull-out of the valve disc. This resulted from pressure entrapment between the inboard and outboard injection valves during surveillance testing resulting in failure of the valve disc to lift off of the valve seat when the valve operator was given an open signal. If the threaded connection met design specifications, it could withstand system back pressure.

The disc could not be removed from the body by conventional means (e.g., chain falls) with the operator removed. A combination of hydraulic jacks and heating the valve body to prevent galling the seats freed the disc. It should be noted that the valve operator was stroked one time after securing from attempting shutdown cooling and at least three additional times before the valve was disassembled for inspection. The impact force of the valve stem in the closed direction lodged the disc into the seat. The required unseating force after one stem stroke would prevent removal of the disc from the seat by conventional means with the operator removed.

The disc skirt was part of an original assembly installed during construction of BFN Unit 1 (installed in 1968-69 timeframe). The safety-related 24-inch globe valve was manufactured by the Walworth Company which has since been acquired by Crane Nuclear, Inc. Discussions with the vendor indicated that no historic inspection documentation of the valve internals is available. TVA determined that the disc skirt threaded connection dimensions were not as denoted on the vendor drawing. A review of the valve maintenance history indicates the valve skirt was part of the original assembly and has not been replaced with other parts by substitution. Thus, the defective undersized disc skirt threads may be present in each of the BFN Units 1, 2, and 3 FCV-074-066/-052 valves. However, all these valves (other than 1-FCV-074-066) were inspected and found to be intact.

To address extent of condition of the similar valve on Unit 1 RHR Loop I and the similar valves on Units 2 and 3 RHR Loops I and II, a functional evaluation (FE) was prepared assuming that the non-conforming condition is applicable and exists on each of the valves. The FE determined that, provided the pressure just downstream of the RHR LPCI outboard

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injection valve (between the inboard and outboard injection valves) is maintained less than the calculated thread capacity when opening the valve, the valves will function as designed and the Unit 1, 2, and 3 RHR Loops I and II safety functions are capable of performing their intended functions. Assuming the threads are undersized in these valves, the over thrusting due to trapped pressure can be managed, thus avoiding future separation until the threads can be verified to be acceptable or corrections are made.

V. ASSESSMENT OF SAFETY CONSEQUENCES

The applicable safety-related basis for the RHR system is to provide a flow path for transmission of water supply to the reactor for a mission time up to 30 days for core cooling following initiation. Unit 1 TS LCO 3.5.1 requires both RHR loops of LPCI to be operable for ECCS in reactor Modes 1, 2, and 3. Thus, RHR Loop II was inoperable for a period longer than the 7 days allowed by TS 3.5.1.

For Design Basis Accidents, in the event of the failure of 1-FCV-074-066, the remaining ECCS subsystems (i.e., LPCI associated with RHR Loop I, two CS subsystems, High Pressure Coolant Injection (HPCI) [BJ] and the Automatic Depressurization System (ADS) [SB]) would be able to fulfill the ECCS safety function associated with RHR Loop II. ADS would be manually actuated in accordance with Emergency Operating Instructions. Long term decay heat removal would be available using RHR Suppression Pool Cooling.

The last confirmed successful operation of 1-FCV-074-066 was on March 13, 2009, during the Unit 1 Cycle 7 refueling outage when RHR Loop II was in service for SDC. However, motor-operated valve performance data indicates that the stem-to-disc separation occurred some time before November 2008. Since either time, it is recognized that one or more of the remaining low pressure ECCS subsystems were inoperable for maintenance or testing. Therefore, TVA is also reporting the failure of 1-FCV-074-066 in accordance with 10 CFR 50.73(a)(2)(v)(D), any event or condition that could have prevented fulfillment of the safety function or structures or systems that are needed to mitigate the consequences of an accident.

For 10 CFR 50 Appendix R considerations, based on the results of testing and analyses, TVA has determined that RHR Loop II would have been able to fulfill its fire safe shutdown function by system pressure and vibration causing the release of the valve disc from the seat allowing makeup flow. These results indicate that the valve disc would have been released from its wedged position within seven minutes such that the required injection flow could be established. The seven-minute time period is within the time required for injection using RHR Loop II to ensure that 10 CFR 50 Appendix R Fire Safe Shutdown requirements are satisfied. This time period also fully complies with the 10 CFR 50 Appendix R Safe Shutdown Instructions, which the operator would be using.

However, in the event 1-FCV-074-066 and RHR Loop II are unable to fulfill the fire safe shutdown makeup function, alternate flow paths for makeup water needed for fire safe shutdown would be available for each applicable Fire Area (FA). Although the 10 CFR 50 Appendix R Safe Shutdown Instructions do not direct the operator to use these alternate flow paths in the event of a component failure not caused by fire damage, these alternate flow paths for makeup would be available using the CS System or the Condensate System [KA] (for FAs other than FA 25). In addition, for some of the affected FAs, including FA 25, HPCI and/or Reactor Core Isolation Cooling [BN]) would be available.

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Therefore, safety margin was maintained and this event was of very low safety significance.

VI. CORRECTIVE ACTIONS

A. Immediate Corrective Actions:

1. Valve 1-FCV-074-066 was repaired to vendor specifications during the refueling outage and is fully functional. The valve was re-assembled with the proper mating threads, the skirt keyed to the stem, and the skirt welded to the disc, which returns the valve to the correct configuration.

2. A functional evaluation (FE) was prepared assuming that the non-conforming condition is applicable to address extent of condition of the similar valve on Unit 1 RHR Loop I and the similar valves on Units 2 and 3 RHR Loops I and II. The FE determined that, provided the pressure just downstream of the RHR LPCI outboard injection valve (between the inboard and outboard injection valves) is maintained less than the calculated thread capacity when opening the valve, the valves will function as designed and the Unit 1, 2, and 3 RHR Loops I and II safety functions are capable of performing their intended functions.

3. Metallurgical analyses of the removed skirt, yoke nut, and stem for valve 1-FCV-074-066 were sent to Westinghouse Electric Company LLC to validate suspected failure modes and estimate the time of failure.

B. Corrective Actions to Prevent Recurrence:

TVA will verify that the similar valve on Unit 1 RHR Loop I and Units 2 and 3 RHR Loops I and II will be disassembled to inspect and rework, as required, to ensure the valves have the correct design configuration (See Section F.vii for the action schedule).

VII. ADDITIONAL INFORMATION

A. Failed Components:

The RHR Loop II LPCI flow control valve, 1-FCV-074-066, was manufactured by the Walworth Company. The valve is a 24-inch No. 5297PS, 600-lb MSS SP-66 Rating cast carbon steel, butt welded, pressure-seal angle globe valve operated by a Limitorque SMB-5T-350 motor operator.

General Electric (GE) provided the valve as the Nuclear Steam Supply System Supplier via TVA/GE Contract No. 66C60-90744 (Units 1 and 2) and 67C60-91750 (Unit 3) as meeting all requirements of GE Co. Specification No. 21A1047, Miscellaneous Gate and Globe Valves, and GE Purchase Specification 21A1047AS, Globe Valves - Motor Operated (GE Parts List No. 10-154)

B. Previous LERS or Similar Events:

TVA BFN Abnormal Occurrence Report (LER) No. BFAO-50-260/7432W, event date of December 4, 1974, details a similar, but different failure of 2-FCV-074-066. In that event, flow-induced vibration caused the failure of small tack welds, intended to prevent rotation between the valve disc and the stem guide ring. The tack weld

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failure allowed the disc to unscrew from the stem guide ring and become wedged in the seat opening. Corrective actions for this event included the addition of a larger, stronger retaining weld to prevent separation of the parts. The undersized threads were not identified as a causal factor for that event. Mating threads on the disc and disc guide were cleaned, inspected, and found satisfactory.

C. Additional Information:

The corrective action document for this report is PER 271338. PER 279911 is an associated corrective action document used to document and track the update of the design output of the approved design changes. PER 303097 provides the functional evaluation of the other similar valves possibly containing the defect.

D. Safety System Functional Failure Consideration:

Because of the defect, the fulfillment of a safety function (i.e., LPCI injection) could have been prevented; therefore, in accordance with NEI 99-02 guidance, this event is considered a safety system functional failure.

E. Scram With Complications Consideration:

This event did not include a reactor scram.

F. 10 CFR Part 21 Reporting Requirements:

The following information is provided at this time to meet the requirements of 10 CFR Part 21.21(d)(4)(i)-(viii).

(i) Name and address of the individual or individuals informing the Commission.

K. J. Polson
Vice President
Tennessee Valley Authority
Browns Ferry Nuclear Plant
Post Office Box 2000
Decatur, Alabama 35609-2000

(ii) Identification of the facility, the activity, or the basic component supplied for such facility or such activity within the United States which fails to comply or contains a defect.

Facility: Browns Ferry Nuclear Plant

Basic component which contains a defect: 24-inch, 600-lb MSS SP-66 Rating cast carbon steel, butt welded, pressure-seal angle globe valve operated by a Limitorque SMB-5T-350 motor operator

(iii) Identification of the firm constructing the facility or supplying the basic component which fails to comply or contains a defect.

Basic component supplier: General Electric (GE) provided the valve as the Nuclear Steam Supply System Supplier via TVA/GE Contract No. 66C60-90744 (Units 1 and 2) and 67C60-91750 (Unit 3) as meeting all requirements of GE Co. Specification No. 21A1047 and GE Purchase Specification 21A1047AS, Rev. 5 - Globe Valves - Motor Operated (GE Parts List No. 10-154)

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(iv) Nature of the defect or failure to comply and the safety hazard which is created or could be created by such defect or failure to comply.

Nature of the defect : The root cause analysis identified the failure mechanism was due to opening thrust exceeding the threaded connection between the disc skirt and disc due to a manufacturing defect in the threads of the disc skirt.

The disc skirt was part of an original assembly installed during construction of BFN Unit 1 (installed in 1968-69 timeframe). The 24-inch globe valve was manufactured by the Walworth Company, which has since been acquired by Crane Nuclear, Inc. Discussions with the vendor indicated that no historic inspection documentation of the valve internals is available. TVA determined that the disc skirt threaded connection dimensions were not as denoted on the vendor drawing. A review of the valve maintenance history indicates the valve skirt was part of the original assembly and has not been replaced with other parts by substitution.

Safety hazard which could be created by such defect: With the BFN-1-FCV-074-066 valve stem-to-disc separation, had this condition existed during an accident condition, only one RHR system loop would have been available for LPCI injection.

(v) The date on which the information of such defect or failure to comply was obtained.

BFN Site Engineering completed the Part 21 evaluation on March 14, 2011.

(vi) In the case of a basic component which contains a defect or fails to comply, the number and location of all such components in use at, supplied for, or being supplied for one or more facilities or activities subject to the regulations in this part.

Number and location of all such components in use at BFN: The extent of condition is limited to the BFN Units 1, 2, and 3 RHR Loops I and II Outboard LPCI valves. By TVA unique identifier, these valves are:

BFN-1/2/3-FCV-074-052/-066

(vii) The corrective action which has been, is being, or will be taken; the name of the individual or organization responsible for the action; and the length of time that has been or will be taken to complete the action.

In addition to the immediate corrective actions taken (See Section VI of this LER), the following preventative corrective actions are planned.

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Corrective Action:

No.	Action	Responsible	Expected Completion Date
CA-1	Repair and re-configure 1-FCV-074-066 to design configuration.	TVA-Site Engineering-Components	Complete
CA-2	Verify 1-FCV-074-052 work is completed to restore the valve to design configuration, as required.	TVA-Site Engineering-Components	December 21, 2012
CA-3	Verify 2-FCV-074-052 work is completed to restore the valve to design configuration, as required.	TVA-Site Engineering-Components	April 28, 2011
CA-4	Verify 2-FCV-074-066 work is completed to restore the valve to design configuration, as required.	TVA-Site Engineering-Components	April 28, 2011
CA-5	Verify 3-FCV-074-052 work is completed to restore the valve to design configuration, as required.	TVA-Site Engineering-Components	June 15, 2012
CA-6	Verify 3-FCV-074-066 work is completed to restore the valve to design configuration, as required.	TVA-Site Engineering-Components	June 15, 2012

(viii) Any advice related to the defect or failure to comply about the facility, activity, or basic component that has been, is being, or will be given to purchasers or licensees.

None

VIII. COMMITMENTS

None