TVA

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

September 30, 2013

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 3**

References:
1. Letter from TVA to NRC, "Licensee Event Report 50-259/2010-003, Revision 0," dated December 22, 2010
2. Letter from TVA to NRC, "Licensee Event Report 50-259/2010-003, Revision 1," dated April 1, 2011
3. Letter from TVA to NRC, "Licensee Event Report 50-259/2010-003, Revision 2," dated February 10, 2012

By letter dated December 22, 2010 (Reference 1), the Tennessee Valley Authority (TVA) submitted a Licensee Event Report (LER) containing details of a failure of a low pressure coolant injection flow control valve. The LER indicated that the investigation and evaluation for the event were being completed, and, upon completion of these actions, a revision to the LER would be submitted. A revised LER, which included the results of TVA causal analysis, was submitted on April 1, 2011 (Reference 2). TVA submitted Revision 2 to LER 50-259/2010-003 (Reference 3) on February 10, 2012, after an additional investigation and evaluation of the valve failure was performed.

An evaluation was performed on past concurrent inoperabilities of safety systems. The TVA is submitting Revision 3 to provide updated information on the collective inoperabilities and correct minor editorial errors.

TVA is submitting this revised report in accordance with Title 10 of the Code of Federal Regulations (10 CFR) 21.2(c), as a report of a potential defect, 10 CFR 50.73(a)(2)(i)(B), any operation or condition prohibited by the plant's Technical Specifications, 10 CFR 50.73(a)(2)(v)(A), any event or condition that could have prevented the fulfillment of the safety function of structures or systems that are needed to shut down the reactor and maintain it in a safe shutdown condition, 10 CFR 50.73(a)(2)(v)(B), any event or condition that could have prevented the fulfillment of the safety function of structures or systems that are needed to remove decay heat, and 10 CFR 50.73(a)(2)(v)(D), any event

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or condition that could have prevented the fulfillment of the safety function of 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,

S. BONO
FOR K. POLSON
9/27/13

K. J. Polson
Vice President

Enclosure: Licensee Event Report 50-259/2010-003-03 - Failure of a Low Pressure Coolant Injection Flow Control Valve

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 50-259/2010-003-03**

**Failure of a Low Pressure Coolant Injection Flow Control Valve**

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**See Enclosed**

NRC FORM 366 U.S. NUCLEAR REGULATORY COMMISSION
(10-2010)

# LICENSEE EVENT REPORT (LER)



| 1. FACILITY NAME | 2. DOCKET NUMBER | 3. PAGE |
|---|---|---|
| Browns Ferry Nuclear Plant, Unit 1 | 05000259 | 1 of 13 |

**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 - | 03 | 09 | 30 | 2013 | FACILITY NAME: N/A | DOCKET NUMBER: N/A |

**9. OPERATING MODE:** 3

**10. POWER LEVEL:** 000

**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 - 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)* |
|---|---|
| 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 | 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 condition is reportable as both operations prohibited by TS and as an event or condition that could have prevented fulfillment of a safety function.

This report constitutes a Part 21 notification.

NARRATIVE

## . 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-74-66, 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-74-66 disc had become separated from the skirt/stem and wedged into the seat, preventing SDC flow.

Investigation of the valve failure to open determined that the direct root cause was a manufacturing defect, undersized disc skirt threads at the disc connection. A brief history of cause-related events for this valve indicates that the original disc-skirt/disc assembly with the defect was installed during construction of BFN Unit 1 in the 1968-69 timeframe. Opportunities to detect the defect prior to the failure included a design change that installed a V-notch disc with linear flow characteristics in 1983 to mitigate flow-induced vibration problems. Additionally, testing opportunities to identify anomalies that could have resulted in valve degradation identification were reduced in 1997 (Units 2 and 3) and 2004 (Unit 1), when, in accordance with Supplement 1 to Generic Letter (GL) 89-10, RHR System Loop I/II Outboard Injection Valves, 1/2/3-FCV-74-52/66, respectively, were determined to be "passive" based on operating in their safety position during normal alignment and were removed from the GL 89-10 program.

Recent valve maintenance history indicated that 1-FCV-74-66 had been refurbished in 2006, prior to the return to service of the Unit 1 RHR System for Unit 1 restart after an extended outage. Based on causal analysis information and MOV performance data taken, the valve stem-to-disc separation occurred sometime before November 2008. Thus, RHR Loop II was inoperable for a significant period of time.

1-FCV-74-66 is in a portion of the system that is necessary for execution of the Low Pressure Coolant Injection (LPCI) mode for that loop and if closed will block flow to the reactor. This blocked flow condition on Division II RHR, coupled with the strategy for 10 CFR 50 Appendix R postulated fires in the Division I RHR areas, would have led to no shutdown cooling flow in an Appendix R fire event. As such, the ability to achieve and maintain a safe shutdown condition, by removal of residual heat after an Appendix R fire, was lost.

NARRATIVE

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.

TVA is submitting this report in accordance with 10 CFR 21.2(c), reporting of defects and non-compliance, and 10 CFR 50.73(a)(2)(i)(B), any operation or condition prohibited by the plant's Technical Specifications. TVA is also submitting this report in accordance with 10 CFR 50.73(a)(2)(v), any event or condition that could have prevented fulfillment of the safety function or structures or systems that are needed to: (A) shut down the reactor and maintain it in a safe shutdown condition, (B) remove residual heat, and (D) 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 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-74-66, failed to pass flow with the valve operator in the open position. 1-FCV-74-66 is a motor-operated valve (MOV) physically located in the RHR LPCI Loop II flow path and is normally in the full-open position. 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.

NARRATIVE

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 of the original installation of BFN Unit 1 LPCI flow control valve 1-FCV-74-66. |
| 1983 | Manufacturing defect not recognized during 1-FCV-74-66 fit-up reassembly. |
| 1997 | The 2/3-FCV-74-52/66 valves were classified as "passive" and removed from the GL 89-10 scope. |
| 2004 | The 1-FCV-74-52/66 valves were classified as "passive" and not included in the GL 89-10 scope. |
| 2006 | 1-FCV-74-66 was refurbished prior to Unit 1 restart after an extended outage. |
| Before November 2008 | 1-FCV-74-66 stem-to-disc separation occurred based on MOV performance data (i.e, no unseating force). |
| 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-74-66 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. |
| 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. |

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.

NARRATIVE

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**

Root cause evaluations identified three root causes.

**Root Cause - Valve Failure**

1. An undersized thread barrel (manufacturing defect), when subjected to system differential pressure greater than the capacity of the reduced thread engagement, caused skirt/disc separation in 1-FCV-74-66.

**Root Causes - Failure to Detect Valve Failure**

2. Lack of requirement for verification of thread dimensions during reassembly of 1-FCV-74-66 using a new disc with the old disc-skirt in 1983 resulted in failure to identify and correct the undersized thread barrel leading to the valve failure.
3. Misapplication of criteria for determination of active/passive function of 1-FCV-74-66 resulted in inappropriate classification and removal from the GL 89-10 program. This resulted in missed opportunities to identify and correct the valve failure.

## IV. ANALYSIS OF THE EVENT

The condition being reported is a defect in a basic component, 1-FCV-74-66, and the operation of Unit 1 in a manner prohibited by TS and the loss of a safety function as a result of this defect.

The RHR system consists of two essentially complete and independent loops identified as Loop I and Loop II. The RHR system is a multipurpose system designed to remove stored and decay heat from the reactor and containment during normal, shutdown, and accident conditions.

The RHR system consists of five modes of operation:

1. LPCI,
2. Containment Spray Cooling (CSC) and Suppression Pool Cooling (SPC),
3. Standby Cooling,
4. SDC, and
5. Supplemental Fuel Pool Cooling.

The RHR System Loop I/II Outboard Injection Valves, 1/2/3-FCV-74-52/66, respectively, have an active safety function to open in order to maintain a flow path for LPCI injection during normal operation. Also, the subject valves are throttled in SDC and vessel make up and are fully closed when used in CSC and SPC modes of RHR operation.

**NARRATIVE**

FCV-74-66 and FCV-74-52 are left open during normal plant operations to keep the RHR Loop injection lines filled with water. During SDC operations, the valve may be throttled by operator action (as cooling needs require). 1-FCV-74-66 and FCV-74-52 will automatically open on a LPCI signal (if closed) and remain interlocked open for five minutes before the valve can be throttled. The valve receives no automatic closure signal and is closed by operator action (RHR System Operating Instructions 1/2/3-OI-74).

1-FCV-74-66 is in a portion of the Division II RHR system that is part of a flow path necessary for vessel injection and has the ability to prevent flow from reaching the reactor if closed or blocked. When the valve disc separation eventually led to the blocked flow condition identified on October 23, 2010, it would have rendered the Appendix R strategy for fires in the Division I RHR areas unusable. Had the valve remained closed during a fire event, the plant operators would have had to exit from the Safe Shutdown Instructions (SSIs) procedures and either re-energize components in the fire affected areas or implement alternate plans for establishing flow from other low pressure sources.

Under all other (non-Appendix R) conditions the redundant loop of RHR would have been available to provide the LPCI function needed for adequate core cooling.

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-74-66 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.

NARRATIVE

The disc skirt was part of an original assembly installed during construction of BFN Unit 1 (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 have been present in each of the 1/2/3-FCV-74-52/66 valves.

To address the extent of condition, TVA evaluated all globe valves with safety related functions which could block flow and the disc-skirt/disc separation could go undetected. Valves 2-FCV-74-66, 3-FCV-74-66, and 1,2,3-FCV-74-52 were assumed to have incorrect thread engagement. This condition was addressed by installing gussets to preclude the need for correct thread engagement.

As a result of TVA investigation, it was identified that the following occurrences represent missed opportunities to identify the manufacturing defect prior to valve failure.

- Flow-induced vibration problems with this valve led to a design change in 1983 that replaced the existing valve disc with a new V-notch disc with linear flow characteristics. Since the skirt was not replaced during this effort, any measurements taken at the time would have been limited to the internal threads of the new disc, and the undersized skirt threads would have again gone undetected. Given that the 1983 design change had access to thread measurements of the new disc and existing skirt, it would have been reasonable to measure the skirt threads to ensure the new disc threads were an appropriate match.

- In the mid 1990s, when TVA was developing the administrative processes for the restart of Browns Ferry Units 2 and 3, an area that required significant consideration was compliance with Generic Letter (GL) 89-10 (Motor Operated Valve Testing and Surveillance). This program established the requirements to ensure MOVs meet design requirements and to verify that MOVs continue to function as designed through periodic testing. TVA initially included the FCV-74-52 and FCV-74-66 valves, along with numerous other safety-related valves, in the GL 89-10 program when it was first established at Browns Ferry. In 1994, TVA contracted an engineering firm familiar with the requirements of the GL 89-10 program to provide an assessment of the GL 89-10 valve requirements as they applied to Units 2 and 3. The assessment was focused on identifying those valves required to have an active safety function for design basis events with the purpose of removing those determined not to have an active safety function from the program. TVA submitted letters to NRC on January 6, 1997, for Units 2 and 3, and May 5, 2004, for Unit 1, which excluded these valves from the GL 89-10 program. Misapplied criteria for determination of active/passive function of 1-FCV-74-66 resulted in inappropriate classification and not included in the GL 89-10 program. This contributed to the untimely identification of the valve failure. Had the 1-FCV-74-66 valve been included in the GL 89-10 program, detection of the valve disc separation may have been more timely.

NARRATIVE

## 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-74-66, 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 SPC.

The last confirmed successful operation of 1-FCV-74-66 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-74-66 in accordance with 10 CFR 50.73(a)(2)(v), any event or condition that could have prevented fulfillment of the safety function or structures or systems that are needed to: (A) shut down the reactor and maintain it in a safe shutdown condition, (B) remove residual heat, and (D) 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 Appendix R Fire Safe Shutdown requirements are satisfied. This time period also fully complies with the Appendix R SSIs, which the operator would be using.

However, in the event 1-FCV-74-66 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 Appendix R SSIs 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.

| FACILITY NAME (1) | DOCKET (2) | LER NUMBER (6) | | | PAGE (3) |
|---|---|---|---|---|---|
| | | YEAR | SEQUENTIAL NUMBER | REVISION NUMBER | |
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NARRATIVE

An evaluation was performed for the past concurrent inoperabilities of safety systems for the period of the time period that Loop II LPCI was inoperable. The following conditions were considered:

- LPCI Loop I (RHR Pump 1C) inoperable from November 2008, to November 16, 2010 (Available)
- Loop I RHR Pumps 1A and 1C inoperable from March 20, 2009, at 2055 hours to March 22, 2009, at 1414 hours (Available)
- Unit 1/2 A Emergency Diesel Generator inoperable from July 25, 2009, at 1127 hours to November 16, 2010 (Available)
- Loop II RHR Pumps 1B and 1D inoperable for approximately 1 minute on March 21, 2009, at 0406 hours (Unavailable)
- HPCI inoperable from September 1, 2009, at 1614 hours to September 3, 2009, at 0500 hours (Unavailable)
- Core Spray Loop II inoperable from September 1, 2009, at 1210 hours to September 2, 2009, at 1741 hours (Unavailable)
- RCIC System Inoperable approximately 29 hours from February 5, 2010, to February 7, 2010 (Unavailable)
- RCIC System Inoperable approximately 35 hours from August 30, 2010, to August 31, 2010 (Unavailable)
- HPCI inoperable from July 24, 2009, at 1130 hours to July 25, 2009, at 0125 hours (Unavailable)

Risk significant systems (or portions of risk significant systems) unavailable for extended periods of time reduces the margin of safety in the plant. A Probabilistic Risk Assessment (PRA) was performed which calculated the Incremental Core Damage probability Deficit (ICDPD) and Incremental Large Early Release Probability Deficit (ILERPD). The ICDPD was determined to be 2.75E-7 and the ILERPD was determined to be 2.66E-8.

Based on the PRA, this event posed minimal reduction to public health and safety; however, this PRA does not consider fire protection strategy impacts. TVA recognizes that failure of 1-FCV-74-66, when combined with the Appendix R impact, would yield a finding of greater safety significance.

## VI. CORRECTIVE ACTIONS - The corrective actions are being managed by TVA's Corrective Action Program.

### A. Immediate Corrective Actions

Key corrective actions from Problem Evaluation Reports (PERs) 271338 and 369800 are listed below.

The following immediate/interim actions have been provided:

- The initial 10 CFR Part 21 notification was made by LER 50-259/2010-003, Revision 1 on April 1, 2011.
- 1/2/3-FCV-74-52/66 assembly drawings were updated to correct historical issues.

NARRATIVE

- 1-FCV-74-66 was repaired to vendor specifications during the Unit 1 Refueling Outage (U1R8) in November 2010.
- The following valves were re-designed to install gussets with a structural weld to preclude the need for the correct thread engagement. Gussets with a structural weld have been installed on each of the valves.
  - 2-FCV-74-66, RHR System II LPCI Outboard Injection Valve
  - 3-FCV-74-66, RHR System II LPCI Outboard Injection Valve
  - 1-FCV-74-52, RHR System I LPCI Outboard Injection Valve
  - 2-FCV-74-52, RHR System I LPCI Outboard Injection Valve
  - 3-FCV-74-52, RHR System I LPCI Outboard Injection Valve
- To enhance operating margin, an interim action was implemented to ensure 1/2/3-FCV-74-52/66 are only stroked during modes 4 or 5 or when the pressure differential is verified to be less than 350 psid.
- 1-FCV-74-66 test procedure was revised to clarify acceptance criteria for performance of MOV Analysis Test System (MOVATS) testing.
- Issued guidance to Maintenance personnel, Planners, and Operations personnel that prior to any work involving the assembly/reassembly of safety-related valves, they are to verify that the applicable procedure has been revised to provide verification and inspection of critical dimensions when load bearing threaded connections involve the use of replacement parts.
- Operations personnel implemented the following restriction: When either FCV-74-52 or FCV-74-66 is not full open (i.e., not in its safety position), the RHR Loop shall be declared Inoperable for LPCI per TS LCO 3.5.1 or 3.5.2, as applicable.

Additional significant corrective actions are:

- Verified that other safety-related valves in the extent of condition do not have disc separation.
- Independent review of GL 89-10 program scope.

**B. Corrective Actions to Prevent Recurrence**

Corrective Actions to prevent recurrence include the following.

- 1-FCV-74-66 was repaired to vendor specifications during the U1R8 refueling outage.
- Gussets with structural welds were installed on 1/2/3-FCV-74-52/66 to preclude the need for the correct thread engagement.
- Procedures governing reassembly of safety-related valves were revised to provide verification and inspection of critical dimensions when load bearing threaded connections involve the use of replacement parts.
- Regulatory programs were reviewed to verify that applicable scoping criteria have been correctly applied to establish program scope.

NARRATIVE

- 1/2/3-FCV-74-52/66 were added into the GL 89-10 program.

## VII. ADDITIONAL INFORMATION

### A. Failed Components

The RHR Loop II LPCI flow control valve, 1-FCV-74-66, 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-74-66. 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 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 documents for this report are PER 271338 (mechanical failure mechanisms of the valve) and PER 369800 (broader issues associated with programs).

### 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) through (viii).

NARRATIVE

(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)

(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-74-66 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.

Initial TVA analysis of the risk impact of the 1-FCV-74-66 valve failure did not include the fire protection strategy impacts and produced an initial safety significance determination finding of low to moderate significance. However, it was recognized by TVA that failure of the valve, when combined with the Appendix R impact, would yield a finding of greater significance.

NARRATIVE

(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 for this basic component 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-74-52/66

(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.

Immediate corrective actions have been completed (See Section VI of this LER)

(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