

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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U.S. Nuclear Regulatory Commission Page 2 September 30, 2013

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.

Respectful S. BONO FOR K. POLSON 1/27 /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

See Enclosed

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Browns Ferry Nuclear Plant, Unit 1	05000259	2010	003	03	2 of 13	

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.

NRC FORM 366A		_	U.S. NUCLEA	RREGULATOR	Y COMMISSION	
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FACILITY NAME (1)	DOCKET (2)	YEAR	LER NUMBER (	6) REVISION	PAGE (3)	
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Browns Ferry Nuclear Plant, Unit 1	05000259	2010	003	03	3 of 13	
VARRATIVE Unit 1 Technical Specification (TS both RHR loops of LPCI to be ope discovery, the reactor was in Mod inoperable for the LPCI function of entered TS LCO 3.5.1 Condition A operable status within 7 days. W remained operable and TS LCO 3 Shutdown was satisfied. Within one hour of the determinat Cold Shutdown. Since TS LCO 3 Operations personnel exited TS L ECCS-Shutdown, became applica injection/spray subsystems to be (CS) [BM] subsystems operable a RHR Loop I was operable for LPC one LPCI subsystem may be con- decay heat removal if capable of TVA is submitting this report in ac non-compliance, and 10 CFR 50. the plant's Technical Specification 10 CFR 50.73(a)(2)(v), any event safety function or structures or sy and maintain it in a safe shutdown the consequences of an accident	erable in react e 3. Operatio of the Emerger A with a requir ith RHR Loop 3.4.7 for RHR ion of RHR Lo 5.1, ECCS - 6 .CO 3.5.1 Con able and requi operable. At t and one RHR Cl in accordan sidered opera being realigne cordance with 73(a)(2)(i)(B), ns. TVA is also or condition t stems that are n condition, (E	or Modes ns perso acy Core ed action I operabl Shutdow op II inol Dperating dition A. res two I hat time, subsyste ce with T ble during d and no 10 CFR any oper o submitt nat could e needed	s 1, 2, and 3 nnel declare Cooling Sys to restore R le, two RHR n Cooling Sy perability, Ur g, is not appl TS LCO 3.5 ow pressure Unit 1 had k m operable f S LCO 3.5.2 g alignment t otherwise i 21.2(c), rep ration or con ing this repo have prevei to: (A) shut	At the time of RHR Loop tem (ECCS) HR Loop II SDC subsys ystem - Hot hit 1 entered licable in Mo 5.2, ECCS poth Core Sp for ECCS. N 2, which state and operable. orting of defe dition prohib rt in accorda nted fulfillme down the re	e of o II o and to stems Mode 4, de 4, de 4, oray lote that es that on for ects and sited by ince with ent of the eactor	
<ul> <li>The past inoperability is based or analyzed time. The exact date at analyzed LPCI time is difficult to a LCOs 3.5.1 and 3.5.2 most likely pressure ECCS system inoperabilithat time, because the degraded due to mode change. Based on M reporting purposes, the discovery</li> <li>B. Inoperable Structures, Compore BFN Unit 1 RHR Loop II LPCI flow the valve operator in the open position. This is the outboard value closed to divert flow from the LPC There were six of these type valve Ferry Nuclear Plant, two per unit walworth Company 24-inch No. 5 butt welded, pressure-seal angle motor operator.</li> </ul>	which RHR L determine with occurred since lity due to ma condition was NUREG-1022 date will be re date will be re ments, or System w control valve sition. 1-FCV CI Loop II flow ve in the injec CI flow path wh es installed du with one in ea 5297PS, 600-I	oop II wo certainty March ntenance guidance etained a <u>ems tha</u> , 1-FCV 74-66 is path and tion path hen conta rring initia ch loop c b MSS S	build have fail . However, 13, 2009, back e and testing gnized, LCO e of event data is the event of t Contribute -74-66, failed a motor-ope d is normally , used to throw all construction of LPCI. The P-66 Rating	led to meet i violations of sed on othe g. Additional 3.0.4 was n ite reporting, date. <b>ed to the Ev</b> d to pass flow rated valve in the full-op ottle SDC flo ing is desire on of the Bro component cast carbon	ts TS r low ly, since ot met for <b>ent</b> w with (MOV) pen w and d. wms is a steel,	

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Browns Fe	rry Nuclear Plant, Unit 1	05000259	2010	003	03	4 of 13				
NARRATIVE										
	The disc skirt is threaded into the									
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	-		-		tly to the dis	С.				
C.	Dates and Approximate Times of	of Major Occu	rrences							
	1968-1969				ation of BFN	Unit 1				
		LPCI flow control valve 1-FCV-74-66.								
assembly surrounds the bottom section of the shoulder at the base. The shouldered stem to the disc through the skirt/disc connection, <b>C.</b> Dates and Approximate Times of Major O 1968-1969 Timefran LPCI flo 1983 Manufac 1-FCV-7 1997 The 2/3- "passive 2004 The 1-F "passive 2006 1-FCV-7 after an Before November 2008 1-FCV-7				ct not recogr	nized during					
		1-FCV-74-6	6 fit-up r	eassembly.						
	1997	The 2/3-FC	V-74-52/	66 valves w	ere classifie	d as				
		"passive" a	nd remov	ed from the	GL 89-10 s	cope.				
	2004	The 1-FCV-74-52/66 valves were classified as								
		"passive" and not included in the GL 89-10 scope.								
	2006	1-FCV-74-6	6 was re	furbished p	rior to Unit 1	restart				
		after an ext								
	Before November 2008	1-FCV-74-6	6 stem-t	o-disc sepa	ration occurr	ed based				
					no unseating					
	March 13, 2009, at 0553 hours	During the	Jnit 1 Cy	cle 7 refueli	ing outage, F	RHR				
		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	•		cle 8 refuel	• • •					
					d to place RI					
			l not be con	cordance wit firmed						
	Optobor 22 2010 at 1422 hours					SDC in				
	October 23, 2010, at 1433 hours	accordance			HR Loop I in					
	October 23, 2010, at 1505 hours	Unit 1 enter								
<b>P</b>				- 1.						
D.	Other Systems or Secondary FL	incuons Ane	cteu							
	None									
E.	Method of Discovery									
	The valve failure was discovered of									
	operating instruction 1-OI-74, "Res	sidual Heat R	emoval S	System," Se	ction 8.12.2,					

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.

NRC FORM 366A

(10-2010)

#### U.S. NUCLEAR REGULATORY COMMISSION

### LICENSEE EVENT REPORT (LER)

CC	NTINUATION	N SHEET				
FACILITY NAME (1)	DOCKET (2)	1	ER NUMBER (6	)	PAGE (3)	
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER		
Browns Ferry Nuclear Plant, Unit 1	05000259	2010	003	03	5 of 13	1

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.
- 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.

LICENSEE EVENT REPORT (LER)           CONTINUATION SHEET           CONTINUATION SHEET           FACILITY NAME (1)         DOCKET (2)         LER NUMBER (8)         PAGE (3)           Browns Ferry Nuclear Plant, Unit 1         DOCKET (2)         LER NUMBER (1)         PAGE (3)           NARRATIVE           FOV-74-68 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-Ol-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 neadreed the Safe Shutdown Instructions (SISIs) 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	LICENSEE EVENT REPORT (LER)           CONTINUATION SHEET           PAGE (3)           VEAR         SOURTINUATION SHEET           Browns Ferry Nuclear Plant, Unit 1         OCKET (2)         LER NUMBER (6)         PAGE (3)           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 (ff 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 1/2/3-0I-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 had the valve remained closed during a fire event, the plant operators would have had to exit from the Safe Shuddown 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 (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 reac	(10-2010)							
CONTINUATION SHEET           FACILITY NAME (1)         DOCKET (2)         LER NUMBER (8)         PAGE (3)           Browns Ferry Nuclear Plant, Unit 1         DOCKET (2)         LER NUMBER (8)         PAGE (3)           NARRATIVE         CONTINUATION Structure Revision           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 (f 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-O/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 (SISI) 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	CONTINUATION SHEET           FACILITY NAME (1)         DOCKET (2)         LER NUMBER (6)         PAGE (3)           Browns Ferry Nuclear Plant, Unit 1         DOCKET (2)         VERM SEQUENTIAL REVEICE           Browns Ferry Nuclear Plant, Unit 1         DOCKET (2)         CONTINUATION SHEET           NUMBER (1)         CONTINUATION SHEET         PAGE (3)           REVENTING TO CONTINUATION SHEET           TERN Vaclear Plant, Unit 1         DOCKET (2)         CONTINUATION SHEET           Browns Ferry Nuclear Plant, Unit 1         DOCKET (2)         CONTINUATION SHEET           NUMBER (2)         CONTINUATION SHEET           NUMER STATE           DOCKET (2)         CONTINUATION SHEET           REVAIL SECTION           NUMER (2)         Control Colspan="2">Control Colspan="2"           STATE ST		LICENSE	E EVENT R	EPORT	(LER)			
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Loop injection lines filled with water. During SDC operations, the valve may be throttled by operator action (as cooling needs require). 1-FCV-74-86 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-74-86 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, 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 staregy for the threaded connectio	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 reductant 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-DI-74. Following preliminary investigations, the failed valve was reworked and tested, and RHR Loop II was returned to service. The TVA investigation disc veceeding the strength of the threaded connection, allowing the valve skint and opull-out of the valve disc. This resulted from pressure	NARRATI	/E		•				
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	valve stem in the closed direction lodged the disc into the seat. The required unseating		FCV-74-66 and FCV-74-52 are left op Loop injection lines filled with water. In operator action (as cooling needs required open on a LPCI signal (if closed) and valve can be throttled. The valve rece- operator action (RHR System Operation 1-FCV-74-66 is in a portion of the Divi- necessary for vessel injection and has closed or blocked. When the valve di- condition identified on October 23, 20 fires in the Division I RHR areas unus event, the plant operators would have (SSIs) procedures and either re-energy mplement alternate plans for establis Under all other (non-Appendix R) con available to provide the LPCI function On October 23, 2010, BFN Unit 1, 1B core in support a refueling outage. At the pump was secured. Subsequent become separated from the stem and flow to the reactor. Operations person using RHR Loop I in accordance with failed valve was reworked and tested, The TVA investigation determined the manufacturer's defect, undersized dis mechanism was opening thrust exceet the disc skirt and disc. The over thrus pressure on the top of the valve disc to allowing the valve skirt and stem to pu- entrapment between the inboard and resulting in failure of the valve disc to given an open signal. If the threaded withstand system back pressure. The disc could not be removed from to the operator ling the seats freed the dis- stroked one time after securing from a	During SDC o uire). 1-FCV- remain interlo eives no autor ing Instruction ision II RHR s s the ability to sc separation 10, it would have able. Had the stable. Had the stable. Had the stable. Had the stable. Had the stable. Had the stable of the hing flow from ditions the red needed for a RHR pump w fter 110 secon investigation of lass skirt and nuel secured 1-OI-74. Foll and RHR Lo sc skirt threads ading the stren st condition was exceeded the ull-out of the v outboard inje- lift off of the v connection m he body by con of hydraulic j sc. It should b attempting shu	perations 74-66 and cked open atic clos s 1/2/3-O ystem that prevent fill eventual ave rende on the Sants in the other low lundant low dequate of ras starte ds of obs liscovere lodged in RHR Loo owing pre- op II was f the stem s at skirt/or action valv alve disc ction valv alve seat et design	the valve m d FCV-74-52 en for five min sure signal an ol-74). at is part of a flow from real ly led to the l ered the App mained close afe Shutdow fire affected w pressure s core cooling. d to provide serving no flo d the 1-FCV nto the valve p II SDC and eliminary inve returned to s n-to-disc sep disc connecti e threaded c axial directio of the thread for the thread and the second at a specificatio al means (e. heating the hat the valve poling and at	hay be the will auto nutes before a flow pate aching the blocked fore endix R set ed during n Instruct areas or ources. would ha SDC of to ources. would ha SDC of to ources. would ha SDC of to ources. areas or ources. would ha SDC of to ources. areas or ources. would ha set of to ources. areas or ources. areas ources. areas or ources. areas or ources. areas ources. areas ource	rottled by protection, protection, processure e testing ator was ee	

NRC FORM 366A					ORY COMMISSION
NRC FORM 300A (10-2010)			U.S. NUCLEA	R REGULAT	
LICENS	SEE EVENT R	EPORT	(LER)		
	CONTINUATION			<u> </u>	
FACILITY NAME (1)	DOCKET (2)		LER NUMBER (	6) REVISION	PAGE (3)
	1	YEAR	NUMBER	NUMBER	
Browns Ferry Nuclear Plant, Unit 1	05000259	2010	003	03	7 of 13
NARRATIVE					
The disc skirt was part of an original (1968-69 timeframe). The safety-rel Walworth Company which has since the vendor indicated that no historic available. TVA determined that the denoted on the vendor drawing. A r valve skirt was part of the original as substitution. Thus, the defective un- each of the 1/2/3-FCV-74-52/66 valv To address the extent of condition, <sup>–</sup> functions which could block flow and Valves 2-FCV-74-66, 3-FCV-74-66,	ated 24-inch gl been acquired inspection doc disc skirt thread eview of the va sembly and ha dersized disc sl ves. TVA evaluated the disc-skirt/o	obe valve by Cran umentation ded conn lve maint is not bee kirt thread all globe disc sepa	e was manu e Nuclear, li on of the val ection dime cenance hist en replaced ds may have valves with ration could	factured b nc. Discus ve interna nsions we ory indicat with other been pre safety rela go undete	y the ssions with ls is re not as tes the parts by sent in ated acted.
thread engagement. This condition need for correct thread engagement As a result of TVA investigation, it w	was addressed	l by insta	lling gussets	s to preclu	de the
missed opportunities to identify the	manufacturing	defect pri	or to valve f	ailure.	
<ul> <li>Flow-induced vibration problems replaced the existing valve disc Since the skirt was not replaced would have been limited to the in threads would have again gone access to thread measurements reasonable to measure the skirt appropriate match.</li> </ul>	with a new V-no during this effo nternal threads undetected. Gi of the new disc	otch disc int, any m of the ne iven that c and exis	with linear f easurement w disc, and the 1983 de sting skirt, it	low charac ts taken at the under sign chan would hav	cteristics. the time sized skirt ge had ve been
<ul> <li>In the mid 1990s, when TVA wa of Browns Ferry Units 2 and 3, a compliance with Generic Letter Surveillance). This program est requirements and to verify that M testing. TVA initially included th numerous other safety-related v established at Browns Ferry. In the requirements of the GL 89-1 valve requirements as they appl identifying those valves required events with the purpose of remo function from the program. TVA Units 2 and 3, and May 5, 2004, GL 89-10 program. Misapplied 1-FCV-74-66 resulted in inappro program. This contributed to the 1-FCV-74-66 valve been include separation may have been more</li> </ul>	In area that req (GL) 89-10 (Mo ablished the red AOVs continue e FCV-74-52 an alves, in the GL 1994, TVA con 0 program to pl ied to Units 2 a 1 to have an act ving those dete submitted lette for Unit 1, whic criteria for dete priate classifica untimely identi ed in the GL 89-	uired sign tor Opera quirement to function nd FCV-7 . 89-10 p tracted a rovide an nd 3. The ive safety ermined ne ers to NR ch exclud rmination ation and fication o	nificant cons ated Valve T ts to ensure on as design '4-66 valves rogram whe in engineerin assessmer e assessmer e assessmer of to have a C on Januar ed these va of active/pa not included f the valve f	sideration esting and MOVs m ed throug , along wir n it was fin ng firm fan t of the G r design b n active s ry 6, 1997 lves from assive fund d in the G ailure. Ha	was d eet design h periodic th rst niliar with L 89-10 cused on asis afety , for the ction of _ 89-10 ad the
NBC EORM 366A (10-2010)					,,, <b>,</b> ,,

NRC FORM 366A (10-2010)			U.S. NUCLEAF	REGULATOR	Y COMMISSION					
	NSEE EVENT R CONTINUATION		(LER)							
FACILITY NAME (1) DOCKET (2) LER NUMBER (6) PAGE (3)										
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER						
Browns Ferry Nuclear Plant, Unit 1	05000259	2010	003	03	8 of 13	1				
NARRATIVE		<b>.</b>								
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.

LICENSEE EVENT REPORT (LER) CONTENUATION SHEET           FACILITY NAME (f)         DOCKET (2)         LER NUMBER (6)         PAGE (3)           BROWNE For TWAR         EXPLORE TO TWAR         PAGE (3)           BROWNE Ferry Nuclear Plant, Unit 1         00000259         2010         -003         -03         9 of 13           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 Norember 2008, to November 16, 2010 (Available)           LOO 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 Representation 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 1474 hours (Unavailable)           - Core Spray Loop II Inoperable from September 1, 2009, at 1210 hours to September 2, 2009, at 1741 hours (Unavailable)           - Core Spray Loop II Inoperable approximately 35 hours from August 30, 2010, to August 31, 2010 (Unavailable)           - RCIC System Inoperable approximately 35 hours from August 30, 2010, to A	NRC FORM	366A	- <u></u> . – <u>_</u> .	· ···			U.S. NUCLEA	R REGULATOR	RY COMMISSION	<u> </u>
FAGILITY NAME (1)         DOCKET (2)         LER NUMBER (6)         PAGE (3)           Browns Ferry Nuclear Plant, Unit 1         05000259         2010         -003         -03         9 of 13           NARRATIVE         An evaluation was performed for the past concurrent inoperabilities of safety systems for the period of the time 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 1141 hours (Available)         •         Loop I RHR Pumps 1B and 1D inoperable from July 25, 2009, at 1127 hours to November 16, 2010 (Available)         •         Loop I RHR Pumps 1B and 1D inoperable from approximately 1 minute on March 21, 2009, at 0406 hours (Unavailable)         •         Loop II RHR Pumps 1B and 1D inoperable for approximately 1 minute on March 21, 2009, at 1741 hours (Unavailable)         •         Loop II RHR Pumps 1B and 1D 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)         •         HCI inoperable from July 24, 2009, at 1130 hours to July 25, 2009, at 0125 hours (Unavailable)         •         HCI inoperable from July 24, 2009, at 1130 hours to July 25, 2009, at 0125 hours (Unavailable)         • <t< th=""><th>(10-2010)</th><th></th><th></th><th>LICENSE</th><th>E EVENT R</th><th>EPORT</th><th>(LER)</th><th></th><th></th><th></th></t<>	(10-2010)			LICENSE	E EVENT R	EPORT	(LER)			
VEAR         SECURPTLI         REVISION           Browns Ferry Nuclear Plant, Unit 1         05000259         2010         -003         -03         9 of 13           NARRATVE         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: <ul> <li>LPCI Loop I (RHR Pump 1C) inoperable from November 2008, to November 16, 2010 (Available)</li> <li>Loop I RHR Pumps 1A and 1C inoperable from March 20, 2009, at 2055 hours to March 22, 2009, at 1414 hours (Available)</li> <li>Unit 1/2 A Emergency Dissel Generator inoperable from July 25, 2009, at 1127 hours to November 16, 2010 (Available)</li> <li>Loop II RHR Pumps 1B and 1D inoperable for approximately 1 minute on March 21, 2009, at 0406 hours (Unavailable)</li> <li>HPCI inoperable from September 1, 2009, at 1614 hours to September 3, 2009, at 0500 hours (Unavailable)</li> <li>Core Spray Loop II inoperable from September 1, 2009, at 1210 hours to September 2, 2009, at 1741 hours (Unavailable)</li> <li>RCIC System Inoperable approximately 25 hours from February 5, 2010, to February 7, 2010 (Unavailable)</li> <li>RCIC System Inoperable approximately 35 hours from August 30, 2010, to August 31, 2010 (Unavailable)</li> <li>RCIC System Inoperable approximately 35 hours to July 25, 2009, at 0125 hours (Unavailable)</li> <li>RCIC System (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) was determined to be 2</li></ul>				CO						
Browns Ferry Nuclear Plant, Unit 1         05000259         101         NUMBER         NUMB		FA	CILITY NAME (1)		DOCKET (2)		· · · · · · · · · · · · · · · · · · ·		PAGE (3)	
<ul> <li>NARRATIVE</li> <li>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:         <ul> <li>LPCI Loop I (RHR Pump 1C) inoperable from November 2008, to November 16, 2010 (Available)</li> <li>Loop I RHR Pumps 1A and 1C inoperable from March 20, 2009, at 2055 hours to March 22, 2009, at 1414 hours (Available)</li> <li>Unit 1/2 A Emergency Diesel Generator inoperable from July 25, 2009, at 1127 hours to November 16, 2010 (Available)</li> <li>Loop II RHR Pumps 1B and 1D inoperable for approximately 1 minute on March 21, 2009, at 0406 hours (Unavailable)</li> <li>Loop II II RHR Pumps 1B and 1D inoperable for approximately 1 minute on March 21, 2009, at 0406 hours (Unavailable)</li> <li>Core Spray Loop II inoperable from September 1, 2009, at 1210 hours to September 2, 2009, at 1741 hours (Unavailable)</li> <li>Core Spray Loop II inoperable from September 1, 2009, at 1210 hours to September 2, 2009, at 1741 hours (Unavailable)</li> <li>RCIC System Inoperable approximately 29 hours from August 30, 2010, to February 7, 2010 (Unavailable)</li> <li>RCIC System Inoperable approximately 35 hours from August 30, 2010, to August 31, 2010 (Unavailable)</li> <li>HPCI inoperable from July 24, 2009, at 1130 hours to July 25, 2009, at 0125 hours (Unavailable)</li> <li>HPCI inoperable from July 24, 2009, at 1130 hours to July 25, 2009, at 0125 hours (Unavailable)</li> <li>HPCI inoperable from July 24, 2009, at 1130 hours to July 25, 2009, at 0125 hours (Unavailable)</li> <li>HPCI inoperable from July 24, 2009, at 1130 hours to July 25, 2009, at 0125 hours (Unavailable)</li> <li>HPCI inoperable from protions of risk significant systems) unavailable for extended periods of time reduces the margin of safety in the</li></ul></li></ul>										
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<ul> <li>period of the time period that Loop II LPCI was inoperable. The following conditions were considered:</li> <li>LPCI Loop I (RHR Pump 1C) inoperable from November 2008, to November 16, 2010 (Available)</li> <li>Loop I RHR Pumps 1A and 1C inoperable from March 20, 2009, at 2055 hours to March 22, 2009, at 1414 hours (Available)</li> <li>Unit 1/2 A Emergency Diesel Generator inoperable from July 25, 2009, at 1127 hours to November 16, 2010 (Available)</li> <li>Loop II RHR Pumps 1B and 1D inoperable for approximately 1 minute on March 21, 2009, at 0406 hours (Unavailable)</li> <li>HPCI inoperable from September 1, 2009, at 1614 hours to September 3, 2009, at 0500 hours (Unavailable)</li> <li>Core Spray Loop II inoperable from September 1, 2009, at 1210 hours to September 2, 2009, at 1741 hours (Unavailable)</li> <li>RCIC System Inoperable approximately 29 hours from February 5, 2010, to February 7, 2010 (Unavailable)</li> <li>RCIC System Inoperable approximately 35 hours from August 30, 2010, to August 31, 2010 (Unavailable)</li> <li>HPCI inoperable from July 24, 2009, at 1130 hours to July 25, 2009, at 0125 hours (Unavailable)</li> <li>HPCI inoperable from July 24, 2009, at 1130 hours to July 25, 2009, at 0125 hours (Unavailable)</li> <li>HPC inoperable from July 24, 2009, at 1130 hours to July 25, 2009, at 0125 hours (Unavailable)</li> <li>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 ICOPD was determined to be 2.75E-7 and the ILERPD was determined to be 2.66E-8.</li> <li>Based on the PRA, this event posed minimal reduction to public health and safety; however, this PRA does not consider fire protections strategy impact. TVA recognizes that failure of 1.FCV-74-66, when combined with the Appendix R impact, woul</li></ul>	NARRATIVE								•	
<ul> <li>Key corrective actions from Problem Evaluation Reports (PERs) 271338 and 369800 are listed below.</li> <li>The following immediate/interim actions have been provided:</li> <li>The initial 10 CFR Part 21 notification was made by LER 50-259/2010-003, Revision 1 on April 1, 2011.</li> </ul>	NARRATIVE An pe co Ri: pe (P (IC wa Ba thi 1-  sa VI. CC	n evalua eriod of onsidered • Ll • Lo • Lo • Lo • Lo • Lo • C • C • C • C • C • C • C • C • C • C	ation was perfor the time period ed: PCI Loop I (RHF ovember 16, 20 pop I RHR Pum larch 22, 2009, a nit 1/2 A Emerge 127 hours to No pop II RHR Pum larch 21, 2009, a PCI inoperable 500 hours (Unav ore Spray Loop eptember 2, 200 CIC System Ino ebruary 7, 2010 CIC System Ino ugust 31, 2010 PCI inoperable Jnavailable) ificant systems of time reduces to as performed wh and Incrementa rmined to be 2.7 of the PRA, this e does not conside 4-66, when com gnificance. CTIVE ACTIONS	that Loop II L R Pump 1C) in 10 (Available) os 1A and 1C at 1414 hours ency Diesel C vember 16, 2 ups 1B and 1E at 0406 hours from Septemi vailable) II inoperable op, at 1741 ho perable appro (Unavailable) from July 24, (or portions of he margin of hich calculate I Large Early 75E-7 and the event posed no ler fire protect bined with the	ast concurrer PCI was inop noperable from inoperable from inop	I inopera erable. m Novem om Marco berable fi e) or appro: ) it 1614 he ber 1, 20 able) hours fro hours fro o hours fro 0 hours tro chours fro 0 hours tro chours tro blant. A chability D s determining tion to pu impacts.	abilities of sa The following aber 2008, to th 20, 2009, rom July 25, ximately 1 m ours to Sept 09, at 1210 m February m August 30 o July 25, 20 ms) unavaila Probabilistic e Damage p eficit (ILERF ned to be 2. ablic health a TVA recogr would yield a	afety system g conditions at 2055 hou 2009, at inute on ember 3, 20 hours to 5, 2010, to 0, 2010, to 0, 2010, to 009, at 0125 ble for exte Risk Asses robability De DD. The IC 66E-8. and safety; I hizes that fa a finding of s	as for the swere urs to 009, at 5 hours ended ssment eficit 2DPD however, illure of greater	
<ul> <li>listed below.</li> <li>The following immediate/interim actions have been provided:</li> <li>The initial 10 CFR Part 21 notification was made by LER 50-259/2010-003, Revision 1 on April 1, 2011.</li> </ul>	А.	. <u>Imm</u> e	ediate Correctiv	ve Actions						
<ul> <li>The initial 10 CFR Part 21 notification was made by LER 50-259/2010-003, Revision 1 on April 1, 2011.</li> </ul>				s from Proble	em Evaluation	Reports	(PERs) 271	338 and 36	9800 are	
Revision 1 on April 1, 2011.		The f	ollowing immed	iate/interim a	ctions have b	een provi	ded:			
					fication was n	nade by I	LER 50-259/	/2010-003,		
<ul> <li>1/2/3-FCV-74-52/66 assembly drawings were updated to correct historical issues.</li> </ul>		• 1.	/2/3-FCV-74-52/	66 assembly	drawings we	re update	ed to correct	historical is	sues.	

NRC FORM 366A	· · · · · · · · · · · · · · · · · · ·			U.S. NUCLEAR	REGULATO	RY COMMISSION	
(10-2010)	LICENSE	E EVENT R	EPORT	(LER)			
	CO	NTINUATION	<b>SHEET</b>		<u> </u>		
	FACILITY NAME (1)	DOCKET (2)	YEAR	ER NUMBER (6) SEQUENTIAL	REVISION	PAGE (3)	{
Browns Ferry I	Nuclear Plant, Unit 1	05000259	2010	NUMBER 003	NUMBER	10 of 13	_ ,
NARRATIVE			2010				
•	1-FCV-74-66 was repaired to v Outage (U1R8) in November 2	•	cations d	uring the Uni	it 1 Refueli	ing	
•	The following valves were re-d preclude the need for the corre have been installed on each of	ect thread eng					
	<ul> <li>2-FCV-74-66, RHR System</li> <li>3-FCV-74-66, RHR System</li> <li>1-FCV-74-52, RHR System</li> <li>2-FCV-74-52, RHR System</li> <li>3-FCV-74-52, RHR System</li> </ul>	n II LPCI Outb n I LPCI Outbo n I LPCI Outbo	oard Inje bard Injec bard Injec	ction Valve ction Valve ction Valve			
•	To enhance operating margin, 1/2/3-FCV-74-52/66 are only s differential is verified to be less	troked during	modes 4				
•	1-FCV-74-66 test procedure w performance of MOV Analysis		-	•	teria for		
•	Issued guidance to Maintenand prior to any work involving the are to verify that the applicable and inspection of critical dimer involve the use of replacement	assembly/rea procedure hasions when lo	ssembly as been r	of safety-relative of safety-relative of safety-relative of the safety o	ated valves ovide verifi	s, they ication	
•	Operations personnel impleme or FCV-74-66 is not full open ( declared Inoperable for LPCI p	i.e., not in its	safety po	sition), the R	HR Loop :		
Ac	dditional significant corrective ac	tions are:					_ ,
•	Verified that other safety-relate separation.	ed valves in th	e extent	of condition (	do not hav	ve disc	
٠	Independent review of GL 89-	10 program so	ope.				
В. <u>С</u>	orrective Actions to Prevent R	<u>ecurrence</u>					
Co	orrective Actions to prevent recu	rrence include	e the follo	wing.			
•	1-FCV-74-66 was repaired to volutage.	vendor specifi	cations d	uring the U1	R8 refuelir	ng	
•	Gussets with structural welds need for the correct thread eng		on 1/2/3	-FCV-74-52/	66 to prec	lude the	
•	Procedures governing reasser verification and inspection of c connections involve the use of	ritical dimensi	ions whei				I
•	Regulatory programs were rev been correctly applied to estab			plicable scop	oing criteria	a have	

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NRC FORM 366A

(10-2010)

#### U.S. NUCLEAR REGULATORY COMMISSION

#### LICENSEE EVENT REPORT (LER)

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FACILITY NAME (1)	DOCKET (2)	1	ER NUMBER (6	)	PAGE (3)	
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER		
Browns Ferry Nuclear Plant, Unit 1	05000259	2010	003	03	11 of 13	I

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).

NRC	FORM	366	Ā

(10-2010)

U.S. NUCLEAR REGULATORY COMMISSION

LICENSEE EVENT REPORT (LER) CONTINUATION SHEET										
FAC	FACILITY NAME (1)		LER NUMBER (6)			PAGE (3)				
		DOCKET (2)	YEAR	SEQUENTIAL	REVISION					
Browns Ferry Nucle	ear Plant, Unit 1	05000259	2010	003	NUMBER 03	12 of 13				
NARRATIVE										
(i) Nan	ne and address of the individ	dual or individ	uals infor	ming the Co	mmission.					
Ten Brov Pos	Polson, Vice President nessee Valley Authority wns Ferry Nuclear Plant t Office Box 2000 atur, Alabama 35609-2000									
facil	<li>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.</li>									
Fac	Facility: Browns Ferry Nuclear Plant									
cart	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									
	<ul> <li>Identification of the firm constructing the facility or supplying the basic component which fails to comply or contains a defect.</li> </ul>									
Stea 2) a No.	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)									
	) 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.									
due	<u>Nature of the defect</u> : 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.									
Unit by ti Disc valv dime mair	disc skirt was part of an orig 1 (installed in 1968-69 time he Walworth Company, whic cussions with the vendor ind e internals is available. TV/ ensions were not as denoted ntenance history indicates the not been replaced with othe	frame). The t ch has since t icated that no A determined d on the vend ne valve skirt	24-inch g been acqu historic i that the c or drawin was part	lobe valve w uired by Crainspection do disc skirt three g. A review of the origina	vas manufac ne Nuclear, ocumentatic eaded conne of the valve	etured Inc. In of the Action				
valv	ety hazard which could be c e stem-to-disc separation, h one RHR system loop wou	ad this condit	ion existe	ed during an	accident co					
the t dete TVA	al TVA analysis of the risk in fire protection strategy impa ermination finding of low to n that failure of the valve, wh iding of greater significance.	cts and produ noderate sign en combined	iced an ir ificance.	nitial safety s However, it	ignificance was recogn	ized by				

NRC FORM 366A	M 366A U.S. NUCLEAR REGULATORY COMMISSION										
(10-2010)											
LICENSEE EVENT REPORT (LER)											
CONTINUATION SHEET											
FACILITY NAME (1)	DOCKET (2)	LER NUMBER (6)			PAGE (3)						
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER							
Browns Ferry Nuclear Plant, Unit 1	05000259	2010	003	03	13 of 13						
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 h	Immediate corrective actions have been completed (See Section VI of this LER)										
	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											

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