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

October 8, 2010

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

10 CFR 50.73

Browns Ferry Nuclear Plant, Units 1, 2, and 3
Facility Operating License Nos. DPR-33, DPR-52, and DPR-68
NRC Docket Nos. 50-259, 50-260, and 50-296

Subject: **Licensee Event Report 50-259/2010-001-00**

The enclosed Licensee Event Report (LER) provides details related to Appendix R Safe Shutdown Instruction procedures where two required operator manual actions were incorrectly specified. This could lead to spurious overloading of the credited Emergency Diesel Generator during an Appendix R fire. The Tennessee Valley Authority is submitting this report in accordance with 10 CFR 50.73 (a)(2)(v)(A) as 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.

There are no commitments contained in this letter. Should you have any questions concerning this submittal, please contact James Emens, Site Licensing and Industry Affairs Manager at (256) 729-2636.

Respectfully,

K. J. Polson
Vice President

Enclosure: Licensee Event Report - Unit 1, 2, and 3 Appendix R Safe Shutdown Instruction Procedures Contain Incorrect Operator Manual Actions

cc (w/ Enclosure):
NRC Regional Administrator - Region II
NRC Senior Resident Inspector - Browns Ferry Nuclear Plant

IE22
NRR

Enclosure

Browns Ferry Nuclear Plant
Units 1, 2, and 3

Licensee Event Report - Units 1, 2, and 3 Appendix R Safe Shutdown Instruction Procedures Contain Incorrect Operator Manual Actions

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SEE ATTACHED

NRC FORM 366 (9-2007) U.S. NUCLEAR REGULATORY COMMISSION	APPROVED BY OMB NO. 3150-0104 EXPIRES 08/31/2010
LICENSEE EVENT REPORT (LER)	Estimated burden per response to comply with this mandatory collection request: 80 hours. Reported lessons learned are incorporated into the licensing process and fed back to industry. Send comments regarding burden estimate to the Records and FOIA/Privacy Service Branch (T-5 F52), U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, or by internet e-mail to infocollects@nrc.gov, and to the Desk Officer, Office of Information and Regulatory Affairs, NEOB-10202, (3150-0104), Office of Management and Budget, Washington, DC 20503. If a means used to impose an information collection does not display a currently valid OMB control number, the NRC may not conduct or sponsor, and a person is not required to respond to, the information collection.

1. FACILITY NAME	2. DOCKET NUMBER	3. PAGE
Browns Ferry Unit 1	05000259	1 of 6

4. TITLE: Unit 1, 2, and 3 Appendix R Safe Shutdown Instruction Procedures Contain Incorrect Operator Manual Actions

5. EVENT DATE			6. LER NUMBER			7. REPORT DATE			8. OTHER FACILITIES INVOLVED	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REV NO.	MONTH	DAY	YEAR	FACILITY NAME	DOCKET NUMBER
									Browns Ferry Unit 2	05000260
08	09	2010	2010	- 001 -	00	10	08	2010	Browns Ferry Unit 3	05000296

9. OPERATING MODE: 1

10. POWER LEVEL: 050

11. THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: *(Check all that apply)*

☐ 20.2201(b)	☐ 20.2203(a)(3)(i)	☐ 50.73(a)(2)(i)(C)	☐ 50.73(a)(2)(vii)
☐ 20.2201(d)	☐ 20.2203(a)(3)(ii)	☐ 50.73(a)(2)(ii)(A)	☐ 50.73(a)(2)(viii)(A)
☐ 20.2203(a)(1)	☐ 20.2203(a)(4)	☐ 50.73(a)(2)(ii)(B)	☐ 50.73(a)(2)(viii)(B)
☐ 20.2203(a)(2)(i)	☐ 50.36(c)(1)(i)(A)	☐ 50.73(a)(2)(iii)	☐ 50.73(a)(2)(ix)(A)
☐ 20.2203(a)(2)(ii)	☐ 50.36(c)(1)(ii)(A)	☐ 50.73(a)(2)(iv)(A)	☐ 50.73(a)(2)(x)
☐ 20.2203(a)(2)(iii)	☐ 50.36(c)(2)	☒ 50.73(a)(2)(v)(A)	☐ 73.71(a)(4)
☐ 20.2203(a)(2)(iv)	☐ 50.46(a)(3)(ii)	☐ 50.73(a)(2)(v)(B)	☐ 73.71(a)(5)
☐ 20.2203(a)(2)(v)	☐ 50.73(a)(2)(i)(A)	☐ 50.73(a)(2)(v)(C)	☐ OTHER Specify in Abstract below or in NRC Form 366A
☐ 20.2203(a)(2)(vi)	☐ 50.73(a)(2)(i)(B)	☐ 50.73(a)(2)(v)(D)	

12. LICENSEE CONTACT FOR THIS LER

NAME	TELEPHONE NUMBER (Include Area Code)
James Emens, Licensing Engineer	256-729-2636

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

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 August 9, 2010, during a review of design input calculations in support of the NFPA 805 transition from the 10CFR 50 Appendix R licensing basis, it was discovered that two 4kV breakers that should be tripped and/or tripped and isolated during certain Appendix R fires were not identified in the calculation summary. Specifically, for a postulated fire in Fire Zone 01-03, Breaker 14 on 4kV Shutdown Board B should be tripped. Also, for a postulated fire in Fire Zone 02-03, Breaker 19 on 4kV Shutdown Board A should be isolated and tripped in lieu of Breaker 10. Therefore, the Safe Shutdown Instructions (SSIs) for certain Appendix R fires do not contain two required operator manual actions (OMAs) to trip and/or trip and isolate these breakers. These OMAs are necessary to prevent spurious load additions on the credited Emergency Diesel Generator (EDG) during certain Appendix R fires. Without operator action to trip these breakers, spurious actuations could add loads on the credited EDG beyond what has been analyzed.

The cause of the omissions was that certain 4kV electrical loads identified in the appendices and attachments of the calculation were not transposed correctly to the main body of the calculation and subsequently were not identified in successor calculations and operating procedures.

NARRATIVE

I. PLANT CONDITION(S)

Browns Ferry Nuclear Plant (BFN) Units 1, 2, and 3 were operating in Mode 1 at approximately 50 percent thermal power.

II. DESCRIPTION OF EVENT

A. Event:

On August 9, 2010, during a review of design input calculations in support of the NFPA 805 transition from the 10CFR 50 Appendix R licensing basis, it was discovered that two 4kV breakers [BKR] that should be tripped and/or tripped and isolated during an Appendix R fire were not identified in the calculation summary. Specifically, for a postulated fire in Fire Zone 01-03, Breaker 14 on 4kV Shutdown Board B [EB] should be tripped. Also, for a postulated fire in Fire Zone 02-03, Breaker 19 on 4kV Shutdown Board A should be isolated and tripped in lieu of Breaker 10. Therefore, the Safe Shutdown Instructions (SSIs) for certain Appendix R fires do not contain two required operator manual actions (OMAs) to trip and/or trip and isolate these breakers. These OMAs are necessary to prevent spurious load additions on the credited Emergency Diesel Generator (EDG) [EK] during certain Appendix R fires. Without operator action to trip these breakers, spurious actuations could add loads on the credited EDG beyond what has been analyzed.

Operations personnel instituted compensatory actions by initiating interim operator actions to trip and/or trip and isolate the appropriate breakers during performance of the affected SSIs. Following permanent SSI procedure revisions, these compensatory actions were terminated.

The Tennessee Valley Authority (TVA) is submitting this Licensee Event Report (LER) in accordance with 10 CFR 50.73(a)(2)(v)(A), as 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.

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

None

C. Dates and Approximate Times of Major Occurrences:

August 28, 2003	Calculation "Unit 1, 2, 3 Appendix R - Auxiliary Power System Alignments and Diesel Generator Loading" issued with Breaker 14 and Breaker 19 transposition errors.
August 9, 2010, at 0835 hours CDT	Engineering personnel notified Operations personnel that SSI procedures do not contain two required actions.
August 9, 2010, at 1616 hours CDT	Operations personnel completed an 8-hour Non-Emergency Notification System report to the NRC in accordance with 10 CFR 50.72(b)(3)(v)(A).
August 9, 2010, at 1654 hours CDT	Operations personnel initiated compensatory

interim operator actions.

D. Other Systems or Secondary Functions Affected

None

E. Method of Discovery

During a review of design input calculations in support of the NFPA 805 transition from the 10CFR 50 Appendix R licensing basis, it was discovered that two 4kV breakers that should be tripped and/or tripped and isolated during an Appendix R fire were not captured in the calculation summary, and therefore not appropriately incorporated into the SSIs.

F. Operator Actions

None

G. Safety System Responses

None

III. CAUSE OF THE EVENT

A. Immediate Cause

None

B. Root Cause

The cause of the SSI issues is that certain 4kV electrical loads identified in the appendices and attachments of the calculation were not carried through to the main body of the calculation and subsequently were not identified in successor calculations and instructions. This was determined to be a transposition error in the calculation due to human error. This transposition error was not found by the independent reviewer.

C. Contributing Factors

None

IV. ANALYSIS OF THE EVENT

The engineering calculation transposition error occurred in August 2003. As a result of this past design-related problem if certain Appendix R fires were to occur, under the guidance and alignments provided in the SSIs with the incorrect steps to trip and/or trip and isolate the two breakers, there existed the potential for an EDG to be overloaded and the ability to power the equipment necessary to achieve and maintain safe shutdown could be jeopardized.

For a postulated fire in Fire Zone 01-03, Breaker 14 on 4kV Shutdown Board B (0-BKR-211-000B/014, Normal Feeder to Transformers TS1E and TDE) is required to be tripped. There is normally no appreciable load on this breaker. However, if a specific fire induced failure of the control circuits to 480V [ED] transfer switches occurred, then loading of transformers TS1E and TDE could occur, and therefore loading on the credited EDG could increase to above analyzed limits.

For a postulated fire in Fire Zone 02-03, Breaker 19 on 4kV Shutdown Board A (2-BKR-074-0005, Residual Heat Removal (RHR) [BO] Pump 2A) should be isolated and tripped in lieu of Breaker 10 (0-BKR-023-0001, Residual Heat Removal Service Water (RHRSW) [BI] Pump A1). If the

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non-credited large load (i.e., 2000 hp RHR Pump 2A) is not isolated from the fire induced spurious start (i.e., Breaker 19 on 4kV Shutdown Board A isolated and tripped), then the postulated loading on the credited EDG could exceed the generator rating beyond what has been analyzed. Note that the incorrect tripping of the RHRSW Pump A1 breaker (Breaker 10) has no effect on the emergency core cooling system because it is not credited for an Appendix R fire.

If the operators do not take prompt action to remedy the EDG overload for certain Appendix R fires in Fire Zone 01-03 or 02-03, and the EDG fails, then the credited power source would not be available to power the credited RHR pumps. With rapid depressurization and no injection, the core could be uncovered for longer than what has been analyzed. Therefore, there is a potential for fuel damage as a result of this condition.

V. ASSESSMENT OF SAFETY CONSEQUENCES

The Low Pressure Coolant Injection (LPCI) mode of the RHR system is the minimum safe shutdown system to maintain reactor inventory. The RHR system in its LPCI mode is designed to provide a high capacity low pressure source of makeup water to the reactor vessel to assure adequate core cooling for a spectrum of conditions which can depressurize the reactor vessel. Following reactor depressurization, the RHR system may be manually operated to inject flow from the suppression pool to the reactor vessel through the recirculation line. In addition to providing makeup inventory, the LPCI mode is also used for decay heat removal. This function is accomplished by continuously pumping suppression pool water through the RHR heat exchanger to allow the RHRSW to remove the pool heat to the ultimate heat sink. This function is normally provided by the Suppression Pool Cooling mode of the RHR system. However, for minimum safe shutdown systems, the LPCI mode combined with the main steam relief valves relieving pressure will be adequate for decay heat removal.

There are three levels of defense in depth related to Fire Protection.

1) Prevent fires from starting with administrative controls. Administrative controls are in place to control and track combustibles at BFN.
2) Identify and extinguish those fires that do start. At a minimum all fire areas have detection and the majority of areas have suppression systems. In addition, BFN has a full-time Fire Department that has an average response time of 10 minutes.
3) Ensure that a train of safe shutdown equipment is free of fire damage in the event of an Appendix R fire. The Appendix R Safe Shutdown Instructions have been walked down to verify their feasibility and reliability. Training is provided on a regular basis.

Once the fire is extinguished the possibility of fire induced overloading of the electrical boards ceases.

The SSIs are based on a minimum set of equipment to keep the reactor core covered and cooled; they do not list or credit all potentially available equipment. In both of the postulated fire events impacted by the discovered SSI issues, offsite power, other electrical boards, and other pumps other than the minimum specified in the SSIs may be available.

Additionally, the Condensate System [SD] may be available as an alternate water source to restore the reactor inventory which would provide additional time for recovery.

A risk evaluation was performed to determine the impact of failure to protect the EDG from potential overload conditions associated with a fire in fire zones 01-03 and 02-03. The fire frequency, which

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would result in an SSI entry, is approximately 1.6E-5/yr and 1.75E-5/yr, respectively. Since the calculation of the core damage frequency (CDF) is the multiplication of the ignition frequency by numbers less than one, the CDF cannot exceed the ignition frequency. Conservatively assuming that the core damage probability is 1.0 for simplification of this analysis, the delta-CDF would be no greater than 1.6E-5/yr and 1.75E-5/yr. Either taken separately or summed, these conditions do not result in a delta-CDF greater than 1E-4/year. Therefore, this condition is not considered to be risk significant.

Therefore, TVA concludes that there was no significant reduction in the protection of the public by this deficiency.

VI. CORRECTIVE ACTIONS

A. Immediate Corrective Actions

Interim operator action requirements were issued to trip and/or trip and isolate the appropriate breakers during applicable SSI entry conditions. These interim actions have been replaced by permanent incorporation into the appropriate SSI steps.

B. Corrective Actions to Prevent Recurrence

The human error associated with the calculation transposition error occurred during the Unit 1 re-start effort. The current Engineering Department at BFN is actively involved in the use of Human Performance Tools, such as self-checking and peer-checking and Technical Pre-Job Briefs. Technical Pre-Job Briefs are routinely performed for engineering tasks, and during these Pre-Job Briefs there is typically discussion of which Human Performance Tools should be used for the activity being performed. These Human Performance Tools are used on a daily basis and there are no adverse trends associated with calculation errors. The issue identified in this LER is a legacy issue. A review of the Corrective Action Program did not reveal any current trends related to calculation related human performance errors.

An additional review was made of the credited and non-credited 4kV loads and 480V loads associated with the calculation and no other deficiencies were found.

Therefore no additional corrective actions to prevent recurrence are necessary.

VII. ADDITIONAL INFORMATION

A. Failed Components

None

B. Previous LERs on Similar Events

There were no previous LERs on similar events identified.

C. Additional Information

Corrective action document for this report is Problem Evaluation Report PER 243955.

D. Safety System Functional Failure Consideration:

This event is a safety system functional failure in accordance with NEI 99-02.

E. Scram With Complications:

This event did not involve a scram.

NARRATIVE

VIII. COMMITMENTS

None