



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

May 15, 2018

10 CFR 50.73

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

Browns Ferry Nuclear Plant, Units 1, 2, and 3
Renewed Facility Operating License No. DPR-33, DPR-52, and DPR-68
NRC Docket No. 50-259, 50-260, and 50-296

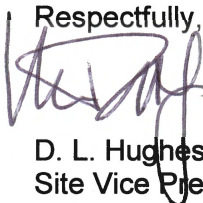
Subject: Licensee Event Report 50-259/2018-001-00

The enclosed Licensee Event Report provides details of an unanalyzed condition associated with the Browns Ferry Nuclear Plant Residual Heat Removal Service Water (RHRSW) System. During postulated fire events where RHR heat exchangers are credited for Fire Safe Shutdown, the RHRSW piping could experience water hammer damage which would cause a loss of function of credited heat exchangers. The Tennessee Valley Authority is submitting this report in accordance with 10 CFR 50.73(a)(2)(ii)(B), as any event or condition that results in the nuclear power plant being in an unanalyzed condition that significantly degraded plant safety.

There will be a supplement to this Licensee Event Report due to an ongoing risk evaluation.

There are no new regulatory commitments contained in this letter. Should you have any questions concerning this submittal, please contact J. L. Paul, Nuclear Site Licensing Manager, at (256) 729-2636.

Respectfully,

 W.K. PAULHARDT JR.
FOR D.L. HUGHES.

D. L. Hughes,
Site Vice President

Enclosure: Licensee Event Report 50-259/2018-001-00 – Fire Damage to Cables for Residual Heat Removal Service Water Valves Could Result in Water Hammer and Piping Damage

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cc (w/ Enclosure):

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

ENCLOSURE

**Browns Ferry Nuclear Plant
Units 1, 2, and 3**

Licensee Event Report 50-259/2018-001-00

**Fire Damage to Cables for Residual Heat Removal Service Water Valves Could Result in Water
Hammer and Piping Damage**

See Enclosed



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 Information Services Branch (T-2 F43), U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, or by e-mail to Infocollects.Resource@nrc.gov, and to the Desk Officer, Office of Information and Regulatory Affairs, NEOB-10202, (3150-0104), Office of Management and Budget, Washington, DC 20503. If a means used to impose an information collection does not display a currently valid OMB control number, the NRC may not conduct or sponsor, and a person is not required to respond to, the information collection.

1. Facility Name Browns Ferry Nuclear Plant (BFN), Unit 1	2. Docket Number 05000259	3. Page 1 OF 6
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4. Title
Fire Damage to Cables for Residual Heat Removal Service Water Valves Could Result in Water Hammer and Piping Damage

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
03	16	2018	2018	- 001	- 00	05	15	2018	BFN, Unit 2	05000260
									BFN, Unit 3	05000296

9. Operating Mode	11. This Report is Submitted Pursuant to the Requirements of 10 CFR §: (Check all that apply)			
1	<input type="checkbox"/> 20.2201(b)	<input type="checkbox"/> 20.2203(a)(3)(i)	<input type="checkbox"/> 50.73(a)(2)(ii)(A)	<input type="checkbox"/> 50.73(a)(2)(viii)(A)
	<input type="checkbox"/> 20.2201(d)	<input type="checkbox"/> 20.2203(a)(3)(ii)	<input checked="" type="checkbox"/> 50.73(a)(2)(ii)(B)	<input type="checkbox"/> 50.73(a)(2)(viii)(B)
	<input type="checkbox"/> 20.2203(a)(1)	<input type="checkbox"/> 20.2203(a)(4)	<input type="checkbox"/> 50.73(a)(2)(iii)	<input type="checkbox"/> 50.73(a)(2)(ix)(A)
	<input type="checkbox"/> 20.2203(a)(2)(i)	<input type="checkbox"/> 50.36(c)(1)(i)(A)	<input type="checkbox"/> 50.73(a)(2)(iv)(A)	<input type="checkbox"/> 50.73(a)(2)(x)
10. Power Level	<input type="checkbox"/> 20.2203(a)(2)(ii)	<input type="checkbox"/> 50.36(c)(1)(ii)(A)	<input type="checkbox"/> 50.73(a)(2)(v)(A)	<input type="checkbox"/> 73.71(a)(4)
100	<input type="checkbox"/> 20.2203(a)(2)(iii)	<input type="checkbox"/> 50.36(c)(2)	<input type="checkbox"/> 50.73(a)(2)(v)(B)	<input type="checkbox"/> 73.71(a)(5)
	<input type="checkbox"/> 20.2203(a)(2)(iv)	<input type="checkbox"/> 50.46(a)(3)(ii)	<input type="checkbox"/> 50.73(a)(2)(v)(C)	<input type="checkbox"/> 73.77(a)(1)
	<input type="checkbox"/> 20.2203(a)(2)(v)	<input type="checkbox"/> 50.73(a)(2)(i)(A)	<input type="checkbox"/> 50.73(a)(2)(v)(D)	<input type="checkbox"/> 73.77(a)(2)(i)
	<input type="checkbox"/> 20.2203(a)(2)(vi)	<input type="checkbox"/> 50.73(a)(2)(i)(B)	<input type="checkbox"/> 50.73(a)(2)(vii)	<input type="checkbox"/> 73.77(a)(2)(ii)
	<input type="checkbox"/> 50.73(a)(2)(i)(C)	<input type="checkbox"/> OTHER	Specify in Abstract below or in NRC Form 366A	

12. Licensee Contact for this LER

Licensee Contact Baruch Calkin, Licensing Engineer	Telephone Number (Include Area Code) (256) 614-6713
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13. Complete One Line for each Component Failure Described in this Report

Cause	System	Component	Manufacturer	Reportable to ICES	Cause	System	Component	Manufacturer	Reportable to ICES
N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A

14. Supplemental Report Expected <input checked="" type="checkbox"/> Yes (If yes, complete 15. Expected Submission Date) <input type="checkbox"/> No	15. Expected Submission Date	Month 07	Day 16	Year 2018
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Abstract (Limit to 1400 spaces, i.e., approximately 14 single-spaced typewritten lines)

On March 16, 2018, at 1604 Central Daylight Time, BFN Engineering discovered an unanalyzed condition affecting the Residual Heat Removal (RHR) heat exchangers (HX) in a postulated fire event. It was discovered that the RHR Service Water (RHRSW) HX piping associated with the credited HXs in the National Fire Protection Association 805 Nuclear Safety Capability Analysis could experience water hammer. A legacy condition existed where credited RHR HXs were subject to voiding due to single spurious opening of the RHRSW outlet valve and subsequent manual or spurious start of a pump. Spurious opening of the valves was recognized and addressed for flow diversion, but the potential effects of water hammer were not addressed.

Operations personnel immediately established compensatory fire watch measures, and provided an eight-hour report to the NRC under Event Notification 53267.

The cause of this condition was a legacy personnel error resulting in inadequate analysis of the potential for flow diversion to lead to pipe voiding and water hammer. Corrective actions for this condition include implementing design changes to install vacuum breakers on the RHRSW side of the RHR HX.



**LICENSEE EVENT REPORT (LER)
CONTINUATION SHEET**

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 Information Services Branch (T-2 F43), U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, or by e-mail to Infocollects.Resource@nrc.gov, and to the Desk Officer, Office of Information and Regulatory Affairs, NEOB-10202, (3150-0104), Office of Management and Budget, Washington, DC 20503. If a means used to impose an information collection does not display a currently valid OMB control number, the NRC may not conduct or sponsor, and a person is not required to respond to, the information collection.

1. FACILITY NAME	2. DOCKET NUMBER	3. LER NUMBER		
		YEAR	SEQUENTIAL NUMBER	REV NO.
Browns Ferry Nuclear Plant, Unit 1	05000259	2018	- 001	- 00

NARRATIVE

I. Plant Operating Conditions Before the Event
 At the time of discovery, Browns Ferry Nuclear Plant (BFN), Units 1 and 2, were in Mode 1 at approximately 100 percent rated thermal power, and BFN, Unit 3, was in Mode 5 for a refueling outage at zero percent power.

II. Description of Event

A. Event Summary
 On February 19, 2016, at 1235 Central Standard Time (CST), BFN personnel discovered a failure to consider Residual Heat Removal (RHR) [BO] Service Water (SW) [KE] water hammer in Fire Safe Shutdown (FSS) analyses. This condition affected the RHR heat exchangers (HX) [HX] in a postulated fire event. It was discovered that the RHRSW piping associated with the HXs in the National Fire Protection Association (NFPA) 805 Nuclear Safety Capability Analysis could experience water hammer. A legacy condition existed where RHR HXs were subject to voiding and water hammer due to a single spurious opening of the RHRSW outlet valve [V] and subsequent manual or spurious start of a pump [P]. Spurious opening of the valves was recognized and addressed for flow diversion, but the potential effects of water hammer were not addressed.

The Functional Evaluation (FE) for this condition (which serves as an input to reportability determination) contained a non-conservative evaluation error leading to the conclusion that heat exchangers deterministically credited for Decay Heat Removal (DHR) in the Appendix R and NFPA 805 analyses would not be impacted. As a result, it had originally been determined that the condition was not reportable. In March 2018, the evaluation error was discovered as the result of a periodic review of degraded and non-conforming conditions.

On March 16, 2018, at 1604 Central Daylight Time (CDT), BFN Engineering determined that an unanalyzed condition which significantly affected plant safety existed. The previously identified water hammer condition could potentially result in damage to the RHRSW piping. This could render the credited HX for DHR non-functional, and it was determined that this condition was reportable.

The Tennessee Valley Authority (TVA) is submitting this report in accordance with 10 CFR 50.73(a)(2)(ii)(B), as any event or condition that results in the nuclear power plant being in an unanalyzed condition that significantly degraded plant safety.



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B. Status of structures, components, or systems that were inoperable at the start of the event and that contributed to the event

There were no structures, components, or systems that were inoperable at the time of discovery, which contributed to this condition.

C. Dates and approximate times of occurrences

Date	Time	Event
February 19, 2016 CST	1235	Failure to consider RHRSW water hammer in FSS analyses discovered during the NFWA 805 transition.
March 16, 2018 CDT	1604	A revised FE by BFN engineering determined that an unanalyzed condition existed which affected the RHRSW piping, after a non-conservative evaluation error was identified in the original FE for this condition.
March 16, 2018 CDT	2104	Event Notification 53267 made to the NRC.

D. Manufacturer and model number of each component that failed during the event

This condition did not involve any failed components.

E. Other systems or secondary functions affected

There were no other systems or secondary functions affected by this condition.

F. Method of discovery of each component or system failure or procedural error

There were no component failures, system failures, or procedural errors associated with this condition.

G. Failure mode, mechanism, and effect of each failed component

This condition did not involve any failed components.

H. Operator actions

Operations personnel established compensatory fire watch measures as an immediate action.

I. Automatically and manually initiated safety system responses

There were no safety system responses initiated as a result of this condition.



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III. Cause of the Event
 The cause of this condition was a legacy personnel error resulting in inadequate analysis of the potential for flow diversion to lead to pipe voiding and water hammer.

A. Cause of each component or system failure or personnel error
 This deficiency originated in the Appendix R Safe Shutdown analysis. Credited RHR HXs were subject to voiding due to single spurious opening of the RHRSW outlet valve and subsequent manual or spurious start of a pump. Spurious opening of the valves was recognized and addressed for flow diversion, but the potential effects of water hammer were not addressed.

B. Cause(s) and circumstances for each human performance related root cause
 This is a legacy issue dating back to the 1980s when the potential effects of water hammer were not addressed in the initial Appendix R Safe Shutdown Analysis for BFN.

IV. Analysis of the Event
 The ultimate safety objective of the RHRSW system is heat removal from the primary water of the RHR systems, using cooling water from the ultimate heat sink [BS] delivered by service water pumps located within the intake pumping station. The RHRSW is a twelve-pump, four-header system with four pairs of pumps normally assigned to the RHR System. Each of the pairs feeds one independent RHR service water header which, in turn, feeds one RHR heat exchanger in each unit. Each RHR service water header is physically, mechanically, and electrically independent of the alternate headers performing the same function.

In a fire event, one RHR HX and one RHRSW pump per unit, in conjunction with RHR Suppression Pool Cooling [DA] or Alternate Shutdown Cooling, are credited for DHR. The containment cooling analyses assume that DHR with RHRSW is established within two hours of reactor SCRAM. The current licensing basis for NFPA 805 requires the RHRSW system function to remove decay heat be free of fire damage) or be evaluated for risk using a risk informed, performance-based approach if deterministic requirements are not met.

Fire damage to the cables for the RHRSW outlet motor operated valves could cause the valves to spuriously open and drain the RHRSW piping. Subsequent starting of the RHRSW pumps on the affected header could cause water hammer loads and damage the piping. This could result in a loss of safety function for a required RHR HX during a fire event. Therefore, this condition constitutes an unanalyzed condition that significantly degraded plant safety and is reportable in accordance with 10 CFR 50.73(a)(2)(ii)(B).



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NARRATIVE

V. Assessment of Safety Consequences

A water hammer analysis is being performed to more accurately characterize the potential for system impacts including HX functionality, pipe rupture, and valve impacts in the scenarios of concern. A risk evaluation is being performed to determine the change in plant fire risk due to this condition. The results of the water hammer analysis and of the risk evaluation will be included in a planned supplement to this report.

A. Availability of systems or components that could have performed the same function as the components and systems that failed during the event

No SSCs failed as a result of this condition. The availability of other systems and components is determined by the Fire Probabilistic Risk Assessment model for each fire scenario. These results will be provided in the supplement.

B. For events that occurred when the reactor was shut down, availability of systems or components needed to shutdown the reactor and maintain safe shutdown conditions, remove residual heat, control the release of radioactive material, or mitigate the consequences of an accident

During the period when this condition existed there were multiple time periods when one or more BFN reactors were shut down. For Non-Power Operations (NPO), the Fire Areas of concern where spurious operation for RHRSW outlet valves can spuriously open are already considered pinch point locations in the NFPA 805 NPO analysis for the affected Unit's decay heat removal. These are already included in the BFN NPO procedure for compensatory measures, with the exception of HX 2D for Fire Areas 03-03 and 23 where there are alternate paths available for Unit 2 decay heat removal.

C. For failure that rendered a train of a safety system inoperable, an estimate of the elapsed time from the discovery of the failure until the train was returned to service

This condition did not result in the Technical Specification inoperability of any safety system. However, this condition rendered HXs credited for some fire scenarios non-functional for the purposes of those scenarios.

VI. Corrective Actions

This condition was entered into the TVA Corrective Action Program (CAP) and is being tracked under Condition Reports 1139620 and 1397087.

A. Immediate Corrective Actions

The immediate corrective action upon discovery of the unanalyzed condition was to establish compensatory fire watch measures.



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NARRATIVE

B. Corrective Actions to Prevent Recurrence or to reduce probability of similar events occurring in the future
 Design Changes will be issued and implemented to install vacuum breakers [VACB] on the RHRSW side of the RHR HXs for the affected units.

VII. Previous Similar Events at the Same Site
 A review of the BFN CAP and Licensee Event Reports for Units 1, 2, and 3 found no instances within the past five years of degraded or unanalyzed conditions on the RHR HXs which resulted from insufficiently justified assumptions.

VIII. Additional Information
 There is no additional information.

IX. Commitments
 There are no new commitments.