

February 27, 2008

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Mail Stop: OWFN, P1-35
Washington, D. C. 20555-0001

10 CFR 50.73

Dear Sir:

**TENNESSEE VALLEY AUTHORITY - BROWNS FERRY NUCLEAR PLANT
(BFN) - UNIT 3 - DOCKET 50-296 - FACILITY OPERATING LICENSE DPR - 68 -
LICENSEE EVENT REPORT (LER) 50-296/2007-005-00**

The enclosed report provides details of an automatic reactor scram due to a generator load reject. TVA is submitting this report in accordance with 10 CFR 50.73(a)(2)(iv)(A), as an event that resulted in an automatic actuation of the systems listed in paragraph (a)(2)(iv)(B) (i.e., Reactor Protection System including: reactor scram or reactor trip. There are no commitments contained in this letter.

Sincerely,
Original signed by:

S. M. Douglas, Interim
Site Vice President, BFN

cc: See page 2

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Enclosure

cc (Enclosure):

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Enclosure

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S:\lic\submit\LERs\296-2007-005 Reactor Scram.doc

NRC FORM 366 (9-2007)		U.S. NUCLEAR REGULATORY COMMISSION		APPROVED BY OMB NO. 3150-0104		EXPIRES 08/31/2010																																						
LICENSEE EVENT REPORT (LER) (See reverse for required number of digits/characters for each block)				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 Browns Ferry Unit 1				2. DOCKET NUMBER 05000296		3. PAGE 1 of 5																																						
4. TITLE: Automatic Reactor Scram Due To Main Generator Load Reject																																												
5. EVENT DATE			6. LER NUMBER		7. REPORT DATE		8. OTHER FACILITIES INVOLVED																																					
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9. OPERATING MODE 1			11. THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: <i>(Check all that apply)</i>																																									
10. POWER LEVEL 100			<table style="width:100%; font-size: small;"> <tr> <td>☐ 20.2201(b)</td> <td>☐ 20.2203(a)(3)(i)</td> <td>☐ 50.73(a)(2)(i)(C)</td> <td>☐ 50.73(a)(2)(vii)</td> </tr> <tr> <td>☐ 20.2201(d)</td> <td>☐ 20.2203(a)(3)(ii)</td> <td>☐ 50.73(a)(2)(ii)(A)</td> <td>☐ 50.73(a)(2)(viii)(A)</td> </tr> <tr> <td>☐ 20.2203(a)(1)</td> <td>☐ 20.2203(a)(4)</td> <td>☐ 50.73(a)(2)(ii)(B)</td> <td>☐ 50.73(a)(2)(viii)(B)</td> </tr> <tr> <td>☐ 20.2203(a)(2)(i)</td> <td>☐ 50.36(c)(1)(i)(A)</td> <td>☐ 50.73(a)(2)(iii)</td> <td>☐ 50.73(a)(2)(ix)(A)</td> </tr> <tr> <td>☐ 20.2203(a)(2)(ii)</td> <td>☐ 50.36(c)(1)(ii)(A)</td> <td>☒ 50.73(a)(2)(iv)(A)</td> <td>☐ 50.73(a)(2)(x)</td> </tr> <tr> <td>☐ 20.2203(a)(2)(iii)</td> <td>☐ 50.36(c)(2)</td> <td>☐ 50.73(a)(2)(v)(A)</td> <td>☐ 73.71(a)(4)</td> </tr> <tr> <td>☐ 20.2203(a)(2)(iv)</td> <td>☐ 50.46(a)(3)(ii)</td> <td>☐ 50.73(a)(2)(v)(B)</td> <td>☐ 73.71(a)(5)</td> </tr> <tr> <td>☐ 20.2203(a)(2)(v)</td> <td>☐ 50.73(a)(2)(i)(A)</td> <td>☐ 50.73(a)(2)(v)(C)</td> <td>☐ OTHER</td> </tr> <tr> <td>☐ 20.2203(a)(2)(vi)</td> <td>☐ 50.73(a)(2)(i)(B)</td> <td>☐ 50.73(a)(2)(v)(D)</td> <td style="font-size: x-small;">Specify in Abstract below or in NRC Form 366A</td> </tr> </table>						☐ 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	☐ 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
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12. LICENSEE CONTACT FOR THIS LER																																												
NAME Steve Austin, Licensing Engineer						TELEPHONE NUMBER (Include Area Code) 256-729-2070																																						
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																																				
14. SUPPLEMENTAL REPORT EXPECTED					15. EXPECTED SUBMISSION DATE																																							
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ABSTRACT <i>(Limit to 1400 spaces, i.e., approximately 15 single-spaced typewritten lines)</i>																																												
On December 31, 2007, at 2140 hours Central Standard Time (CST), Unit 3 reactor received an automatic scram signal following a main generator load reject. The reactor scram from the generator load reject was expected. All systems responded to the scram as expected. All control rods inserted. During the initial pressure transient, which peaked at 1141 psig, six of the main steam system relief valves opened. The reactor pressure was subsequently controlled with the main steam system bypass valves. The reactor water level was controlled by the Feedwater system, the normal heat removal path through the main condenser was maintained during the event. The reactor scram was reset December 31, 2007, by 2146 hours CST. TVA is submitting this report according to 10 CFR 50.73(a)(2)(iv)(A), as an event that resulted in a manual or automatic actuation of the systems listed in paragraph 10 CFR 50.73(a)(2)(iv)(B) (i.e., reactor protection system including reactor scram of trip, and general containment isolation signals affecting containment isolation valves in more than one system.)																																												

LICENSEE EVENT REPORT (LER)

FACILITY NAME (1)	DOCKET (2)	LER NUMBER (6)			PAGE (3)
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
Browns Ferry Nuclear Plant Unit 3	05000296	2007	-- 005	-- 00	2 of 5

NARRATIVE (If more space is required, use additional copies of NRC Form 366A) (17)

I. PLANT CONDITION(S)

Prior to the event Unit 3 was operating in Mode 1 at 100 percent RTP (3458 megawatts thermal). Units 1 and 2 were also at 100 percent power (3458 Megawatts thermal) and unaffected by the event.

II. DESCRIPTION OF EVENT**A. Event:**

On December 31, 2007, at 2140 hours Central Standard Time (CST), Unit 3 reactor received an automatic scram signal following a main generator [TB] load reject. The reactor scram from the generator load reject was expected.

All systems responded to the scram as expected. All control rods [AA] inserted. Because the reactor level lowered to level 3 (low level) primary containment isolation system (PCIS) [JM] isolations Group 2 (residual heat removal (RHR) [BO] system), Group 3, (reactor water cleanup (RWCU) [CE] system), Group 6 (ventilation), and Group 8 (traversing incore probe (TIP) [IG] system) signals were received. The low water level also initiated the standby gas treatment (SGT) [BN] system and the control room emergency ventilation (CREV) [VI] system. The reactor water level remained above level 2 (low-low Level); accordingly, no emergency core cooling systems were actuated.

The reactor scram and PCIS actuations were reset December 31, 2007, by 2146 hours CST. SGT and CREV systems were secured by 2153 hours CST.

During the initial pressure transient, which peaked at 1141 psig, six (6) of the main steam system [SB] relief valves [RV] opened. The reactor pressure was subsequently controlled with the main steam system bypass valves [PCV]. The reactor water level was controlled by the feedwater [SJ] system; the normal heat removal path through the main condenser [SG] was maintained during the event.

TVA is submitting this report according to 10 CFR 50.73(a)(2)(iv)(A), as an event that resulted in a manual or automatic actuation of the systems listed in paragraph 10 CFR 50.73(a)(2)(iv)(B) (i.e., reactor protection system including reactor scram or trip, and general containment isolation signals affecting containment isolation valves in more than one system.)

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

None.

C. Dates and Approximate Times of Major Occurrences:

December 31, 2007	2140 hours CST	Unit 3 received and automatic scram.
January 1, 2008	0058 hours CST	TVA completed an four hour non-emergency report per 10 CFR 50.72(2)(iv)(B) and an eight hour non-emergency report per 10 CFR 50.72(b)(3)(iv)(A).

D. Other Systems or Secondary Functions Affected

None.

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NARRATIVE (If more space is required, use additional copies of NRC Form 366A) (17)

E. Method of Discovery

The turbine trip and reactor scram were immediately apparent to the control room staff through numerous alarms and indications.

F. Operator Actions

Operations personnel responded to the event according to applicable plant procedures. Operations momentarily entered Emergency Operating Instruction, 3-EOI-001, Reactor Pressure Control and Abnormal Operating Instruction, 1-AOI-100-1, Reactor Scram, as required.

G. Safety System Responses

The reactor protection [JC] system logic responded to the generator trip per plant design by initiating the reactor scram. All control rods inserted. The PCIS Group 2 (RHR system shutdown cooling), Group 3 (RWCU system), Group 6 (ventilation), and Group 8 (TIP) isolations were received as expected due to the lowering of the reactor water level, along with the auto start of CREV system and three SGT system trains. The reactor water level was immediately restored with reactor feedwater, and no ECCS actuations were required.

III. CAUSE OF THE EVENT**A. Immediate Cause**

The immediate cause for the reactor scram was the turbine load reject signal.

B. Root Cause

The generator trip was the result of spurious operation of the Unit 3 generator breaker phase discordance Relay (20-7 relay) which resulted in a generator breaker trip signal. Although TVA could not identify the cause for the spurious actuation of the 20-7 relay, TVA postulates that either an electrical or mechanical agitation of the 20-7 relay resulted in the generation of the trip signal.

C. Contributing Factors

None.

IV. ANALYSIS OF THE EVENT

The reactor scram was initiated by control valve fast closure signals. This was the result of a power load unbalance (PLU) signal. The PLU signal is generated whenever normalized turbine power as indicated by moisture separator 3C1 inlet pressure is 40 percent greater than the normalized generator current within rate limitations. The PLU signal was the result of a loss of generator load when the generator output breaker (PCB-234) tripped open. PCB-234 tripped because the 20-7 relay spuriously actuated.

The cause for the generator load reject was the opening of PCB-234. This was the only breaker in the generator output path. The apparent cause of PCB-234 opening was a spurious actuation of the 20-7 relay. The appropriate alarm was actuated in the control room for the 20-7 relay during the turbine load reject and reactor scram.

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NARRATIVE (If more space is required, use additional copies of NRC Form 366A) (17)

TVA has determined that no phase discordance occurred. All three phases of the generator output breaker opened simultaneously. Post event testing of the contacts that initiate the 20-7 relay found them working properly. Other relays capable of tripping PCB-234 were inspected for abnormalities that could have resulted in the trip and none were identified.

Just prior to the scram, an Assistant Unit Operator (AUO) had been dispatched to the local panel that housed the generator breaker control equipment to clear a Generator Breaker Air Pressure Abnormal alarm. The steps taken by the individual to clear the alarm were reviewed and no actions were identified that would trip PCB-234.

V. ASSESSMENT OF SAFETY CONSEQUENCES

The safety consequences of this event were not significant. Reactor scram from a generator load reject at 100 percent power is a transient for which BFN is analyzed. The reactor scram was not complicated. All safety systems operated as required. PCIS Groups 2, 3, 6, and 8 isolations were as required. The reactor water level dropped to below level 3, but remained above level 2, therefore ECCS did not actuate. The reactor water level was recovered immediately by the feedwater system pumps. The reactor pressure reached 1141 psig, and following relief valve operation, reactor pressure was then controlled by the main steam system bypass valves. The short term actuation by 6 relief valves following a generator load reject from 100% power was as expected.

Operator actions taken in response to the reactor scram were appropriate. These included the verification that the reactor was shutdown, the expected actuations and isolations had occurred, and the restoration of these systems to normal post scram alignment. Therefore, TVA concludes that the health and safety of the public was not affected by this event.

VI. CORRECTIVE ACTIONS**A. Immediate Corrective Actions**

Operations personnel placed the reactor in a stable condition according to plant procedures.

B. Corrective Actions to Prevent Recurrence⁽¹⁾

The corrective actions address both postulated causes. To address postulated mechanical agitation, the 20-7 relay will be labeled as a trip sensitive component. Although post scram testing could not identify any electrical faults with the 20-7 relay to address any postulated electrical agitation, the relay was replaced.

VII. ADDITIONAL INFORMATION**A. Failed Components**

None.

B. Previous LERs on Similar Events

None.

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NARRATIVE (If more space is required, use additional copies of NRC Form 366A) (17)

(1) TVA does not consider the corrective actions regulatory requirements. The completion of the actions will be tracked in TVA's Corrective Action Program.

C. Additional Information

Corrective action document for this report is PER 135878.

D. Safety System Functional Failure Consideration:

This event did not involve a safety system functional failure according to NEI 99-02.

E. Scram With Complications Consideration:

This event was not a complicated scram according to NEI 99-02.

VIII. COMMITMENTS

None.