April 10, 2007

U.S. Nuclear Regulatory Commission 10 CFR 50.73
ATTN: Document Control Desk
Mail Stop OWFN, P1-35
Washington, D. C. 20555-0001

Dear Sir:

TENNESSEE VALLEY AUTHORITY - BROWNS FERRY NUCLEAR PLANT (BFN) - UNIT 2 - DOCKET 50-296 - FACILITY OPERATING LICENSE DPR - 63 - LICENSEE EVENT REPORT (LER) 50-296/2007-001-00

The enclosed report provides details of a Unit 3 automatic scram due to a low reactor water level caused by a loss of feedwater.

In accordance with 10 CFR 50.73 (a) (2) (iv) (A), TVA is reporting this event as a valid actuation of the reactor protection system and containment isolation valves in more than one system, actuation of emergency core cooling systems including high-pressure coolant injection system and reactor core cooling system. There are no commitments contained in this letter.

Sincerely,

Original signed by:

Brian O'Grady

cc: See page 2

Enclosure
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NRC FORM 366 (6-2004) U.S. NUCLEAR REGULATORY COMMISSION

LICENSEE EVENT REPORT (LER)

(See reverse for required number of digits/characters for each block)

1. FACILITY NAME	2. DOCKET NUMBER	3. PAGE
Browns Ferry Unit 3	05000296	1 OF 6

4. TITLE
Reactor Scram Due To Low Reactor Water Level Caused By Loss Of Feedwater.

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
									none	N/A
02	09	2007	2007-001-00			04	10	2007	FACILITY NAME none	DOCKET NUMBER N/A

9. OPERATING MODE: 1

10. POWER LEVEL: 100

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)	X	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

12. LICENSEE CONTACT FOR THIS LER

NAME	TELEPHONE NUMBER *(Include Area Code)*
Steve Austin, Licensing Engineer, Licensing and Industry Affairs	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	REPORTABLE TO EPIX

14. SUPPLEMENTAL REPORT EXPECTED	15. EXPECTED SUBMISSION DATE	MONTH	DAY	YEAR
☐ YES (*if yes, complete 15. EXPECTED SUBMISSION DATE*) ☒ NO				

ABSTRACT (*Limit to 1400 spaces, i.e., approximately 15 single-spaced type written lines*)

At 1208 Central Standard Time, on February 9, 2007, Unit 3 received an automatic reactor scram on low water level following the loss of condensate flow. Just prior to the scram, operations attempted to establish manual operation of the condensate and demineralizer system. Personnel were in the process of modifying the control logic for the condensate and demineralized water system backwash controller. With the primary controller in run mode and the secondary controller in the program mode, personnel were loading new software into the secondary controller. The personnel involved were experiencing difficulties loading the software onto the secondary controller, so they attempted to load software onto the primary controller. They placed primary controller, which was previously in the run mode, into the program mode. However, the secondary controller was not returned to the run mode. With neither controller the run mode the condensate and demineralizer water system demineralizers isolated. This resulted in a decrease in reactor water level and an automatic reactor scram. The root cause of this event was the individuals involved in the planning and implementation of the work order did not fully understand manual operation of the system. Additionally, there is inadequate guidance or limitations on the use of in-field decision making. TVA is revising the operating instructions for the condensate demineralizer system. TVA is revising trouble shooting guidance to eliminate trouble shooting under the direction of the system engineer or other individuals except under tightly controlled circumstances.

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

I. PLANT CONDITION(S)

Prior to the scram event, Units 2 and 3 were in operating in Mode 1 at 100 percent thermal power (approximately 3458 megawatts thermal). Unit 1 was shutdown and defueled. Units 1 and 2 were unaffected by the event.

II. DESCRIPTION OF EVENT

A. Event:

At 1208 Central Standard Time, on February 9, 2007, Unit 3 received an automatic reactor scram from 100 percent power on low water level following the loss of condensate flow. Just prior to the scram, operations attempted to establish manual operation of the condensate and demineralizer system [SF]. Personnel were in the process of modifying the control logic for the condensate and demineralized water system backwash controller. With the primary controller in run mode and the secondary controller in the program mode, personnel were loading new software into the secondary controller. The individuals experienced difficulties loading the software to the secondary controller, so they attempted to load software into the primary controller. They placed primary controller, which was previously in the run mode, the program mode. However, the secondary controller was not returned to the run mode. With neither controller the run mode the condensate and demineralizer water system demineralizers isolated. This resulted in a loss of condensate system [SD] booster pump suction, a decrease in feedwater flow, a decrease in reactor water level and a subsequent automatic reactor scram.

Isolation of the condensate system resulted in low reactor water level that tripped the reactor recirculation system [AD] pumps 3A and 3B and finally, the reactor water low level automatic scram. When the reactor water level reached level 2 (low low water level) the high pressure coolant injection (HPCI) [BJ] and reactor core isolation cooling (RCIC) [BN] systems auto initiated. By approximately 1210 hours HPCI and RCIC tripped on Hi water level and approximately 1 minute later, the feedwater pumps [SJ] tripped on hi water level. Operations briefly entered Emergency Operating Instruction (EOI) -1, Reactor Pressure Vessel Control, on low reactor water level and EOI-3, Secondary Containment Control, due to momentary Tip Room hi radiation.

During the event, all automatic functions resulting from the scram occurred as expected. All control rods inserted. As a result of the low water level, the Primary Containment Isolation System (PCIS) isolations [JM] Group 2 (Residual Heat Removal (RHR) System [BO] Shutdown Cooling), Group 3 Reactor Water Cleanup (RWCU) [CE] System, Group 6 (Ventilation), and Group 8 Traversing Incore Probe (TIP) [IG] were received along with the auto start of the Control Room Emergency Ventilation (CREV) [VI] System and the three Standby Gas Treatment (SGT) System trains.

At 1225 hours CST operations placed recirculation system pump 3B in service per 3-SR-3.4.9.3&4, Reactor Recirculation Start Limitations. By approximately 1233 hours, operations reset the reactor scram according to 3-AOI-100-1, Reactor Scram, and at 1235 hours CST operations reset the PCIS isolations and placed reactor feed pump 3C in service for level control. Reactor water level and normal heat rejection were being maintained by the feedwater and condensate system. At 1316 hours CST operations placed recirculation system pump 3A in service. SGT and CREV systems were secured by approximately 1646 hours CST.

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

This report is submitted 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 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 and any event or condition that results or should have resulted in ECCS discharging to the reactor coolant system).

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

None.

C. Dates and Approximate Times of Major Occurrences:

February 11, 2007, 1208 hours CST	Unit 2 received an automatic reactor scram.
February 11, 2007, 1235 hours CST	Operations reset the reactor scram.
February 11, 2007,1532 hours CST	TVA made a four hour non-emergency report per 10 CFR 50.72(b)(2)(iv)(A), 10 CFR 50.72(b)(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.

E. Method of Discovery

The event was immediately apparent to the operating crew through numerous indications and alarms in the Main Control Room.

F. Operator Actions

Operations personnel responded to the reactor scram according to applicable plant procedures. However, during the scram recovery the operations crew failed to properly implement steps in 3-SR-3.4.9.3&4, Reactor Recirculation Pump Start Limitations. The procedure requires that the difference between the recirculation loop coolant temperature and the reactor pressure vessel coolant temperature be less than or equal 50 degrees F prior to starting a recirculation pump. The 3B recirculation pump was started with a differential temperature of 72 degrees F.

G. Safety System Responses

The automatic reactor scram was uncomplicated. All automatic functions occurred as designed. All control rods inserted. The PCIS isolations Group 2, Group 6, and Group 8 TIP isolation were received along with the auto start of the CREV System and the three SGT System trains. Reactor water level was recovered by HPCI and RCIC system operation and subsequently, maintained by the feedwater system.

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

III. CAUSE OF THE EVENT

A. Immediate Cause

Reactor Scram

The immediate cause of the reactor scram was a loss of control of the condensate demineralizer valves. During the maintenance activity both the primary and secondary controllers were placed in the program mode. With neither controller in the run mode, all of the demineralizer outlet valves closed and the bypass valves opened. The flow path to the condensate booster pumps was restricted for a short period, thus; causing the lowering of the reactor water level.

Restart of Recirculation Pumps

The immediate cause for the premature restart of Recirculation pump 3B was inattention to detail. Pump 3B was started following an unsatisfactory performance of 3-SR-3.4.9.3&4 (the reactor dome temperature to recirculation system loop B temperature was greater than the maximum allowed.)

B. Root Cause

Reactor Scram

The root cause of this event was the individuals involved in the planning and implementation of the work order that provided instructions for establishing manual control of the condensate demineralized system did not fully understand manual operation of the system. The operating instructions for the condensate demineralizer system do not provide instructions for placing the system in manual operation, so the manual alignment was performed erroneously using a step-text work order. Since the work order was in error, those involved erroneously perceived that there was no risk involved in the manipulation of the controllers since they thought they had placed the system in the manual mode.

Additionally, there is inadequate guidance or limitations on the use of in-field decision making. Even though the system did not operate as expected, the individuals involved proceeded with trouble shooting activities.

Restart of Recirculation Pumps

The root cause for the failure to follow the guidance in 3-SR-3.4.9.3&4 was the operator misread the implementing step 7.10. Step 7.10 states: Verify the difference between the coolant temperature and the recirculation loop to be started and reactor pressure vessel coolant is ≤ 50 degrees F. (Whenever in Mode 2 and both recirculation pumps are not in operation, the difference may be ≤ 75 degrees F.)

The operator missed the clarification that ≤ 75 degrees F was only valid in Mode 2 with both pumps not in operation. The operator keyed in on the value, 75 degrees F.

C. Contributing Factors

None.

IV. ANALYSIS OF THE EVENT

The reactor scram was uncomplicated. The temporary lowering of the reactor water level is an expected response to loss of feedwater flow at 100 percent thermal power.

Equipment response following the reactor scram was in accordance with plant design for a loss of feedwater. The short term lowering of the reactor water level was recovered by HPCI and RCIC operation. Following the initial transient, reactor water level was controlled by the feedwater system. The operation of other systems post scram (e.g., containment isolation, start-up of SGT and CREV systems, isolation of normal reactor building ventilation, RWCU isolation, TIP isolation, etc) also occurred according to the plant design. The main condenser continued to function as the heat sink following the reactor scram. Except for the restart of the 3B recirculation pump, the operator actions in response to the event were appropriate.

V. ASSESSMENT OF SAFETY CONSEQUENCES.

UFSAR Section 14.5.5.3 addresses the complete loss of feedwater transient. Loss of feedwater flow is the most severe from high power conditions. Transient analysis is performed at 102 percent thermal power and 100 percent rated core flow. Conservative heat values are used to maximize the heat addition to the vessel, Main Steam Relief Valve challenges, and inventory loss. Water Level is considered to be normal. The three feed pumps are assumed to coast down in one second. RCIC is assumed to initiate at level 2 (low low water level). HPCI is not assumed to start and no MSIV closure takes place. The analysis shows that no safety limits are exceeded for the loss of feedwater transient.

The transient described in this LER is bounded by the analysis presented in UFSAR Section 14.5.5.3. First, when the reactor water level reached Level 3 (low water level) the reactor scram occurred as designed. The water level continued to fall, and, at the Level 2 setpoint RCIC and HPCI initiated, returning water level to normal. The feedwater system was used to maintain reactor water level and the main condenser continued to serve as a heat sink.

Although recirculation pump 3B was placed in service following an unsatisfactory performance of 3-SR-3.4.9.3&4, the Technical Specifications Surveillance Requirement SR 3.4.9.4 states that the temperature difference of 75 degrees F or lower is permissible only when the plant is in Mode 2 (Startup) and both recirculation pumps are not in operation. Otherwise, the difference of 50 degrees F is applicable. GE SIL No. 517, Supplement 1, Analysis Basis for Idle Recirculation Loop Startup, states that the temperature of an idle recirculation loop to be started must be within 50 degrees of the vessel dome saturation temperature for restart of the first loop, but the basis for this value upon an initial idle loop temperature of 100 degrees F, implying Mode 4 (Cold Shutdown). At the time of the event BFN had just transitioned from Mode 1 (Run) to Mode 3 and Mode 2 after being in operation at normal pressure and temperature conditions for an extended length of time. Being in Mode 2 with an initial loop temperature of 100 degrees F (an outage startup condition) is a more adverse condition with regards to the reactor pressure vessel thermal stress. Therefore, the thermal stresses generated in going from Mode 1 to Mode 3 are bounded by thermal stresses in Mode 2. The actual temperature difference between the reactor coolant temperatures in the in the recirculation loop to be started and the reactor pressure vessel coolant temperature was less than 75 degrees F. Therefore, the reactor coolant pressure boundary was not adversely affected by this event. TVA concludes that the health and safety of the public was not impacted by this event.

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VI. CORRECTIVE ACTIONS

A. Immediate Corrective Actions

Reactor Scram

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

Restart of Recirculation Pumps

Operations entered TS LCO 3.4.9.A1 and A2 which requires that an engineering evaluation be performed within 72 hours to determine if the reactor coolant system (RCS) is acceptable for continued operation. The evaluation found the RCS acceptable for continued operation.

B. Corrective Actions to Prevent Recurrence[1]

- TVA is revising the operating instructions for the condensate demineralizer system to include manual operation.
- TVA is revising trouble shooting guidance to eliminate trouble shooting under the direction of the system engineer or other individuals except under tightly controlled circumstances.

VII. ADDITIONAL INFORMATION

A. Failed Components

None.

B. Previous LERs on Similar Events

None.

C. Additional Information

The corrective action document for the reactor scram is PER 119490. The recirculation pump start issue is discussed in corrective action document PER 119489.

D. Safety System Functional Failure Consideration:

No safety functions were compromised as a result of this event. Therefore, this event is not considered a safety system functional failure in accordance with NEI 99-02 in that functional capability of the overall system was maintained.

E. Loss of Normal Heat Removal Consideration:

The main condenser and feedwater system provided a normal heat removal path following the reactor scram. Accordingly, this event did not result in a scram with a loss of normal heat removal as defined in NEI 99-02.

VIII. COMMITMENTS

None.

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