November 2, 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 1 - DOCKET 50-259 - FACILITY OPERATING LICENSE DPR - 33 - LICENSEE EVENT REPORT (LER) 50-259/2007-008-00

The enclosed report provides details of a manual reactor scram due to a electro hydraulic control system leak. TVA is reporting this in accordance with 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). There are no commitments contained in this letter.

Sincerely,

Original signed by:

Brian O'Grady

cc: See page 2

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NRC FORM 366 (6-2004)	U.S. NUCLEAR REGULATORY COMMISSION	APPROVED BY OMB NO. 3150-0104 EXPIRES 06/30/2007
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:: 50 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
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4. TITLE: Manual Reactor Scram due to an Electro Hydraulic Control System Leak

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 None	DOCKET NUMBER N/A
09	03	2007	2007-008-00			11	02	2007	FACILITY NAME None	DOCKET NUMBER N/A

9. OPERATING MODE	11. THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: *(Check all that apply)*							
1		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)
10. POWER LEVEL 072		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)*	X NO		N/A	N/A	N/A

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

On September 1, 2007, at approximately 1955 hours CDT, Unit 1 Operations was notified that there was a small EHC leak in the Unit 1 Moisture Separator (MS) room in the Turbine Building. Unit 1 was operating at 100 percent power. Video monitoring was established by 0657 hours CDT. Operations noted the leak rate was approximately 120 drops per minute. On September 3, 2007, at 0214 hours CDT Operations noted that the leak rate was increasing and manually scrammed the reactor from approximately 72 percent power.

This report is submitted in accordance with 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).

I. PLANT CONDITION(S)

Prior to the event, Unit 1 was operating at approximately 72 percent of rated thermal power (RTP) (2490 megawatts thermal). Units 2 and 3 were operating in Mode 1 at 100 percent RTP (3458 megawatts thermal). Units 2 and 3 were unaffected by the event.

II. DESCRIPTION OF EVENT

A. <u>Event:</u>

On September 3, 2007, at 0214 hours Central Daylight Time (CDT), Unit 1 was manually scrammed from approximately 72 percent power due to an un-isolatable electro hydraulic control [TG] (EHC) system leak. On September 1, 2007, at approximately 1955 hours CDT, Unit 1 Operations was notified that there was a small EHC leak in the Unit 1 Moisture Separator (MS) room in the Turbine Building [NM]. Unit 1 was operating at 100 percent power. On September 2, 2007, at 0100 hours, an entry into the MS room verified the leak was from the EHC system. The leak was initially identified by a surveillance camera in the moisture separator room Video monitoring was established by 0657 hours CDT to monitor the leak. Operations noted the leak rate was approximately 120 drops per minute.

A visual observation of the leak area determined that a wood isolator was missing from an EHC pipe support and the EHC line (thin-wall stainless steel tubing) was rubbing (fretting) against a steel support. On September 3, 2007, at approximately 0200 hours CDT, operations noted the EHC leak rate was increasing and initiated a reactor core flow runback. At 0214 hours CDT Operations noted that the leak rate had further increased and manually scrammed the reactor from approximately 72 percent power.

During the event, all automatic functions resulting from the scram occurred as expected. All of the control rods [AA] inserted. The reactor water level lowered to below level 3, 528 inches, hence; primary containment isolation system (PCIS) [JE] isolations Group 2 (residual heat removal (RHR) system [BO] shutdown cooling), Group 3 (reactor water cleanup (RWCU) system) [CE], Group 6 (ventilation), and Group 8 (traversing incore probe (TIP) [IG] system were received along with the autostart of the control room emergency ventilation (CREV) [VI] system and the three standby gas treatment (SGT) [BH] system trains. The reactor water level remained above level 2, 470 inches; accordingly, no emergency core cooling systems actuated. Reactor water level was recovered and maintained by the feedwater and condensate [SJ] system. Reactor pressure was controlled by the main steam bypass valves [JI].

The PCIS actuations were reset by 0224 hours CDT and SGT and CREV systems were secured by 0229 hours CDT.

Following the manual scram the Unit 2 Refuel Zone Exhaust Inboard Damper [VA] (1-FCO-064-0010) failed to fully close on PCIS isolation. The redundant damper did perform properly. TVA entered Technical Specification Limiting Condition for Operation (TS LCO) 3.6.4.2, Action A, which requires if one or more penetration flow paths with one secondary containment Isolation valves inoperable, within 8 hours, isolate the effected penetration flow path by use of at least one closed and de-activated automatic valve, closed manual valve or, blind flange. On September 3, 2007, at 2200 hours CDT, Outboard Refuel Zone Exhaust Damper (1-FCO-064-009) was verified closed and placed under an operations clearance for secondary containment under TS 3.6.4.2, Action A. BFN remained under the TS action

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

statement until the damper actuator was replaced and post maintenance testing was completed on September 12, 2007.

This report is submitted in accordance with 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:

September 3, 2007 at 0214 hours CDT	Unit 1 reactor was manually scrammed.
September 3, 2007 at 0405 hours CDT	TVA made a four hour non-emergency report per 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 leak was identified by a surveillance camera in the moisture separator room. A visual walkdown confirmed the fluid was from the EHC system. The reactor manually scrammed by the control room staff.

F. Operator Actions

Operations personnel responded to the event according to applicable plant procedures. Operations momentarily entered Emergency Operating Instruction, 1-EOI-1, Reactor Pressure Control, and Abnormal Operating Instruction, 1-AOI-100-1, Reactor Scram. The operator actions taken in response to the manual reactor scram were appropriate. These included the verification that the reactor was shutdown, the expected system isolations had occurred, and restoration of the affected systems.

G. Safety System Responses

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 the CREV system and the three SGT system trains. Reactor level was automatically restored with reactor feedwater; as such, no emergency core cooling systems actuated, and no relief valves opened.

Additionally, Operations personnel responded to the failure of damper 2-FCO-064-0010 to fully isolate by verifying 2-FCO-064-0009 was fully closed and entering the appropriate Limiting Condition for Operation Action Statement.

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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

The immediate cause of the manual scram is a through wall leak due to the fretting of the EHC tubing against a steel support member.

Operations entered TS LCO 3.6.4.2, Action A, when secondary containment damper 1-FCO-064-0010 failed to completely close upon PCIS isolation signal.

B. Root Cause

No protective isolation block was installed between the EHC tubing and the steel support allowing the tubing to fret against a support.

A failed American Solenoid Company (ASCO) solenoid valve [FSV] on 1-FCO-064-0010 resulted in the secondary containment damper failing to close.

C. Contributing Factors

None.

IV. ANALYSIS OF THE EVENT

The location of the protective/isolation blocks is controlled by notes for field routed tubing. A post scram walkdown of the area underneath the failed tube did not identify any evidence that a protective block had been previously installed. In this case, a walkdown by an engineer would have identified the tubing resting on the steel support which would have not looked out of the ordinary. The EHC System would have been in operation for the flow-induced vibration effects on the tubing to become apparent. Because the configuration did not appear out of the ordinary, the wood isolation block was not noted as missing during the pre-startup walkdowns.

V. ASSESSMENT OF SAFETY CONSEQUENCES

The safety consequences of this event were not significant. The reactor scram was not complicated. All safety systems operated as required. PCIS groups 2, 3, 4, 6, and 8 isolations were as expected. The reactor water level lowered to level 3, but remained above level 2; therefore, ECCS systems did not actuate. Reactor water level was recovered and maintained by the reactor feed pumps. Manual reactor scram from 100% power is a transient for which BFN is analyzed. TVA scrammed Unit 1 from approximately 72 percent power which is less severe than a scram from full power. 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.

TVA replaced the fretted EHC tubing and installed a wood isolation block between the tubing and the steel support.

The ASCO solenoid valve on 1-FCO-064-0010 was replaced. Following post maintenance testing operations released the secondary containment clearance, exiting TS 3.6.4.2, Action A.

B. **Corrective Actions to Prevent Recurrence**

Prior to restart, TVA performed a walkdown of the remaining EHC lines and confirmed proper protective isolation of the EHC tubing from nearby support steel.

VII. ADDITIONAL INFORMATION

A. **Failed Components**

None.

B. **Previous LERs on Similar Events**

Unit 1 License Event Report 259/2007-002 provides details of a manual scram of the Unit 1 Reactor due to an un-isolatable EHC leak. Although both LERs discuss manual shutdown of Unit 1 because of an EHC leak, the root cause of 259/2007-002 was over tightening of a compression fitting. As such, the corrective actions taken in 259/2007-002 would not preclude the event discussed in this LER.

C. **Additional Information**

Corrective action document for this event is PER 129791.

D. **Safety System Functional Failure Consideration:**

This event is not considered a safety system functional failure according to NEI 99-02.

E. **Scram With Complications Consideration:**

This event did not result in a complicated scram according to NEI 99-02.

VIII. COMMITMENTS

None.