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July 6, 2009

U.S. Nuclear Regulatory Commission
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Washington, D.C. 20555-0001

10 CFR 50.73

TENNESSEE VALLEY AUTHORITY - SEQUOYAH NUCLEAR PLANT (SQN) UNIT 1 - DOCKET NO. 50-327 - FACILITY OPERATING LICENSE DPR-77 - LICENSEE EVENT REPORT (LER) 50-327/2009-005-00

The enclosed LER provides details concerning a manual reactor trip and engineered safety feature actuation of auxiliary feedwater. The manual trip was initiated because of a low and decreasing steam generator level when a feedwater regulating valve closed as a result of a ruptured actuator diaphragm. This report is being submitted in accordance with 10 CFR 50.73(a)(2)(iv)(A), as an event that resulted in an automatic actuation of the reactor protection system.

Sincerely,

for

Timothy P. Cleary

Enclosure
cc: See page 2

Enclosure
cc (Enclosure):

INPO Records Center
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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	2. DOCKET NUMBER	3. PAGE
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4. TITLE: Manual Reactor Trip Following a Loss of Flow Through Loop 1 Feedwater Regulating Valve

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
05	06	2009	2009	- 005 -	00	07	06	2009	FACILITY NAME	DOCKET NUMBER

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)	☒ 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)
Scott T Bowman	423-843-6910

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
X	SJ	FCV	F052	Y					

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 typewritten lines)*

On May 6, 2009, at 2256 Eastern daylight time with Unit 1 operating at 100 percent reactor power, the reactor was manually tripped because of imminent loss of steam generator level induced by the closure of the Loop 1 main feedwater regulating valve. During normal power operations, the control room received a steam flow/feedwater flow mismatch alarm. Operations placed the Loop 1 main feedwater regulating valve in manual and attempted to open the valve. Operations observed that there was still no flow to the Loop 1 steam generator. Operators took action to manually trip the reactor. The Loop 1 main feedwater regulating valve actuator diaphragm was found to be ruptured. The diaphragm failure is attributed to an improper clamping force of the diaphragm to the actuator stem. This insufficient clamping force was a result of insufficient torque applied to the fastening screw. The insufficient clamping force allowed stress concentrations at the diaphragm hole, which caused an initial tear in the diaphragm composite material and led to an instantaneous failure of the diaphragm.

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

I. PLANT CONDITION(S)

Unit 1 was in Mode 1 operating at approximately 100 percent power when the reactor trip was initiated.

II. DESCRIPTION OF EVENT

A. Event:

On May 6, 2009, at 2256 Eastern daylight time (EDT) with Unit 1 operating at 100 percent reactor power, the reactor was manually tripped because of imminent loss of steam generator (SG) [EIIS code AB] level induced by the closure of the Loop 1 main feedwater [EIIS code SJ] regulating valve. The control room received a steam flow/feedwater flow mismatch alarm. Operations placed the Loop 1 main feedwater regulating valve in manual and attempted to open the valve. Operations observed that there was still no flow to the Loop 1 SG. Operations initiated a manual trip. The unit was stabilized in Mode 3 and an event investigation was initiated.

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

None.

C. Dates and Approximate Times of Major Occurrences:

Date	Description
May 6, 2009 at 2256 EDT	Operations received various Loop 1 SG low feedwater and low level alarms. Loop 1 feedwater regulating valve was placed in manual in an attempt to control Loop 1 SG level.
May 6, 2009 at 2256 EDT	Operations initiated a manual reactor trip because of a loss of feedwater flow to Loop 1 SG and imminent Loop 1 SG low level.
May 6, 2009 at 2256 EDT	Unit 1 entered Mode 3.

D. Other Systems or Secondary Functions Affected:

No other systems or secondary functions were affected by this event.

E. Method of Discovery:

The main control room received a steam flow/feedwater flow mismatch alarm.

F. Operator Actions:

Operations placed the Loop 1 feedwater regulating valve in manual and attempted to open the valve with no result. Loop 1 SG level was approximately 25-26 percent and decreasing rapidly. Operations performed a manual reactor trip because of the imminent loss of Loop 1 SG level. After the reactor trip, the operating crew took actions necessary to stabilize the unit in a safe condition and maintained the unit in hot standby, Mode 3.

G. Safety System Responses:

The plant responded to the reactor trip as designed.

III. CAUSE OF THE EVENT

A. Immediate Cause:

The immediate cause was the failure of the Loop 1 feedwater regulating valve air operated diaphragm.

B. Root Cause:

The diaphragm failure is attributed to an improper clamping force of the diaphragm to the actuator stem. This insufficient clamping force was a result of insufficient torque applied to the fastening screw. The insufficient clamping force allowed stress concentrations at the diaphragm hole, which caused an initial tear in the diaphragm composite material and led to an instantaneous failure of the diaphragm.

The root cause of the equipment failure and subsequent reactor trip was determined to be that the governing vendor manual control procedure does not consider applicability to critical components. The vendor manual for the valves originally did not specify a torque value for diaphragm installation. The vendor updated the vendor manual to specify a torque value, but the TVA vendor manual program did not require checking for updates to non-safety related vendor manuals.

C. Contributing Factor:

Contributing factors include the absence of a torque specification in the vendor manual and maintenance instructions, which can lead to various interpretations of the instructions.

IV. ANALYSIS OF THE EVENT

The plant responded to the manual reactor trip as designed. The reactor coolant system (RCS) average temperature was at approximately 578 degrees Fahrenheit (°F) prior to the plant transient. Following the reactor trip, the loss of nuclear heat generation resulted in a decrease in RCS average temperature to approximately 535°F. As heat removal in the SG decreased, as a result of increased steam pressure, the decrease in RCS temperature slowed. The introduction of cold auxiliary feedwater (AFW) resulted in a slower, but continued reduction in RCS temperature until AFW flow was reduced after the reactor trip. RCS temperature then started to increase. RCS temperature remained within technical specification (TS) limits and was bounded by the Updated Final Safety Analysis Report (UFSAR) analysis.

The plant responded as expected for the conditions of the trip. No TS limits were exceeded and the UFSAR analysis of this event remained bounding.

V. ASSESSMENT OF SAFETY CONSEQUENCES

Based on the above "Analysis of The Event," this event did not adversely affect the health and safety of plant personnel or the general public.

VI. CORRECTIVE ACTIONS

A. Immediate Corrective Actions:

The valve was disassembled, the diaphragm replaced and returned to service.

B. Corrective Actions to Prevent Recurrence:

The vendor manual was revised to include the torque requirement for clamping the diaphragm to the actuator stem. Other corrective actions include revising the maintenance procedures to clearly define the torque requirements, revising the procedure governing the vendor manual to review vendor manual classifications, and replacing the diaphragms in the feedwater regulating and bypass valves on both Units.

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

VII. ADDITIONAL INFORMATION

A. Failed Components:

The failed component was a diaphragm in a Fisher Controls International Inc. Size 80 Type 667 actuator in the Loop 1 main feedwater regulating valve. The diaphragm failure was determined to be improper clamping force of the diaphragm plates that support and attach the diaphragm to the actuator stem. This insufficient clamping force was a result of insufficient torque applied to the fastening cap screw. The insufficient clamping force allowed stress concentrations at the diaphragm hole to be approximately three times nominal values, which caused an initial tear in the diaphragm composite material and led to the instantaneous failure of the diaphragm.

All equipment responded as required except the failed main feedwater regulating valve had a “not closed” indication in the main control room following a feedwater isolation signal.

B. Previous LERs on Similar Events:

A review of previous reportable events for the past 10 years indicated that there have been reactor trips regarding main feedwater regulating valves; however, there are no previous events that were a result of a diaphragm failure in a feedwater regulating valve.

C. Additional Information:

None.

D. Safety System Functional Failure:

This event did not result in a safety system functional failure in accordance with 10 CFR 50.73(a)(2)(v).

E. Unplanned Scram with Complications:

This condition did not result in an unplanned scram with complications.

VIII. COMMITMENTS

None.