



Tennessee Valley Authority, Post Office Box 2000, Soddy Daisy, Tennessee 37384-2000

April 29, 2013

10 CFR 50.73

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

Sequoyah Nuclear Plant, Unit 1
Facility Operating License No. DPR-77
NRC Docket No. 50-327

Subject: Licensee Event Report 327/2013-002, "Loss of Auxiliary Control Room Instrumentation"

The enclosed Licensee Event Report provides details concerning the loss of auxiliary feedwater instrumentation in the auxiliary control room. This report is being submitted in accordance with 10 CFR 50.73(a)(2)(i)(B), as an event or condition that is prohibited by technical specifications.

There are no regulatory commitments contained in this letter. Should you have any questions concerning this submittal, please contact Mr. Michael McBrearty, Sequoyah Site Licensing Manager, at (423) 843-7170.

Respectfully,

A handwritten signature in blue ink, appearing to read "John T. Carlin". The signature is stylized and somewhat cursive.

John T. Carlin
Site Vice President
Sequoyah Nuclear Plant

Enclosure: Licensee Event Report
cc: Regional Administrator – Region II
NRC Senior Resident Inspector – Sequoyah Nuclear Plant

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 FOIA/Privacy Section (T-5 F53), U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, or by internet e-mail to infocollects.resources@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.

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4. TITLE:
Loss of Auxiliary Control Room Instrumentation

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
02	08	2013	2013	002	00	04	29	2013	FACILITY NAME	DOCKET NUMBER

9. OPERATING MODE 1	11. THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check all that apply)											
	<input type="checkbox"/> 20.2201(b)	<input type="checkbox"/> 20.2203(a)(3)(i)	<input type="checkbox"/> 50.73(a)(2)(i)(C)	<input type="checkbox"/> 50.73(a)(2)(vii)								
10. POWER LEVEL 100%	<input type="checkbox"/> 20.2201(d)	<input type="checkbox"/> 20.2203(a)(3)(ii)	<input type="checkbox"/> 50.73(a)(2)(ii)(A)	<input type="checkbox"/> 50.73(a)(2)(viii)(A)								
	<input type="checkbox"/> 20.2203(a)(1)	<input type="checkbox"/> 20.2203(a)(4)	<input type="checkbox"/> 50.73(a)(2)(ii)(B)	<input type="checkbox"/> 50.73(a)(2)(viii)(B)								
	<input type="checkbox"/> 20.2203(a)(2)(i)	<input type="checkbox"/> 50.36(c)(1)(i)(A)	<input type="checkbox"/> 50.73(a)(2)(iii)	<input type="checkbox"/> 50.73(a)(2)(ix)(A)								
	<input type="checkbox"/> 20.2203(a)(2)(ii)	<input type="checkbox"/> 50.36(c)(1)(ii)(A)	<input type="checkbox"/> 50.73(a)(2)(iv)(A)	<input type="checkbox"/> 50.73(a)(2)(x)								
	<input type="checkbox"/> 20.2203(a)(2)(iii)	<input type="checkbox"/> 50.36(c)(2)	<input type="checkbox"/> 50.73(a)(2)(v)(A)	<input type="checkbox"/> 73.71(a)(4)								
	<input type="checkbox"/> 20.2203(a)(2)(iv)	<input type="checkbox"/> 50.46(a)(3)(ii)	<input type="checkbox"/> 50.73(a)(2)(v)(B)	<input type="checkbox"/> 73.71(a)(5)								
<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)(C)	<input type="checkbox"/> OTHER									
<input type="checkbox"/> 20.2203(a)(2)(vi)	<input checked="" type="checkbox"/> 50.73(a)(2)(i)(B)	<input type="checkbox"/> 50.73(a)(2)(v)(D)	Specify in Abstract below or in NRC Form 366A									

12. LICENSEE CONTACT FOR THIS LER

FACILITY NAME Rusty Proffitt	TELEPHONE NUMBER (Include Area Code) (423) 843-6651
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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	003	FI	W351	Y	X	003	FM	M422	Y

14. SUPPLEMENTAL REPORT EXPECTED <input type="checkbox"/> YES (If yes, complete 15. EXPECTED SUBMISSION DATE) <input checked="" type="checkbox"/> NO	15. EXPECTED SUBMISSION DATE	MONTH	DAY	YEAR
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ABSTRACT (Limit to 1400 spaces, i.e., approximately 15 single-spaced typewritten lines)

On February 26, 2013, it was determined the Sequoyah Unit 1 auxiliary feedwater (AFW) flow indicators in the auxiliary control room (ACR) had been inoperable from February 8, 2013, until February 16, 2013. The AFW flow indicators in the ACR normally indicate zero. TVA was unable to identify evidence of when exactly they failed. Operators noticed that an Essential Raw Cooling Water indicator failed on February 8, 2013. During troubleshooting on February 15, 2013, it was discovered that two fuses in the circuit cleared. These fuses supply power to instrument loops including power to the AFW flow indicators. When these fuses are blown, power is lost to the flow modifiers that provide indication. The AFW flow indicators are fed from the same power supply that feeds two other instrument loops. A flow modifier that drives a plant computer data point failed resulting in the blown fuses. The fuses and flow modifier were replaced and the indicators were returned to operable status. Based on the review of the data, it was determined the AFW flow indicators had been out of service longer than allowed by Technical Specification 3.3.3.5.

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NARRATIVE

I. Plant Operating Conditions Before the Event

At the time of the event, Sequoyah Nuclear Plant (SQN) Unit 1 was operating at approximately 100 percent rated thermal power.

II. Description of Events

A. Event:

On February 26, 2013, it was determined the Sequoyah Unit 1 auxiliary feedwater (AFW) (EIS Code BA) flow indicators in the auxiliary control room (ACR) had been inoperable from February 8, 2013, until February 16, 2013.

The AFW flow indicators in the ACR normally indicate zero; TVA was unable to identify evidence of when exactly they failed. On February 8, 2013, operators noticed that an Essential Raw Cooling Water (ERCW) (EIS Code BI) indicator had failed. A troubleshooting plan was performed on February 15, 2013. During troubleshooting it was discovered that two fuses in the circuit were cleared. These fuses supply power to the instrument loops including power to the AFW flow indicators. When these fuses cleared, power is lost to the associated AFW and ERCW flow modifiers that provide the flow indication. LCO 3.3.3.5 was entered at 1400 on February 15, 2013, for the two AFW ACR indicators. LCO 3.3.3.5 was exited at 2344 on February 16, 2013, after replacing the blown fuses. The AFW flow indicators are fed from the same power supply that feeds two other instrument loops. Based on the review of the data, it was determined the AFW flow indicators had been out of service longer than allowed by Technical Specification 3.3.3.5.

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

There were no inoperable structures, components or systems that contributed to this event.

C. Dates and approximate times of occurrences:

Dates and Times	Description
February 8, 2013	Operations noticed that an ERCW flow indicator in the auxiliary control room had failed.
February 15	Troubleshooting determined two fuses in the circuit were blown.

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February 15 at 1400	LCO 3.3.3.5 was entered for the two AFW ACR indicators.
February 16 at 2344	The fuses and flow modifier were replaced and LCO 3.3.3.5 was exited.
February 26	It was determined the Sequoyah Unit 1 AFW flow indications in the ACR were inoperable from February 8, 2013, to February 16, 2013.

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

The failed components were the auxiliary feedwater inlet flow indicators, 1-FI-3-147C and 1-FI-3-163C, for Steam Generator No. 3 and Steam Generator No. 1 respectively (EIS Code AB) and ERCW flow indicator, 1-FI-67-62C. The flow modifier, 1-FM-3-147C, is a current to current modifier that drives a plant computer data point for AFW flow to Steam Generator No. 3 and is in the same instrument loop as the other AFW flow indicator and the ERCW flow indicator.

The AFW inlet flow indicators and the ERCW flow indicator are Westinghouse VX252 indicators.

The flow modifier, 1-FM-3-147C, is a Moore Industries SCT module.

E. Other systems or secondary functions affected:

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

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

Discovery of the failed ERCW indicator (1-FI-67-62C) was made by Operations personnel during performance of operator rounds. During troubleshooting of the ERCW flow indicator failure it was determined that two fuses in the circuit were blown. These fuses also supply power to the instrument loops for the AFW flow indicators.

G. The failure mode, mechanism, and effect of each failed component, if known:

TVA determined the failure of the fuses resulted in the failure of the ERCW and AFW flow indicators. The failure of the fuses was the result of failure of a flow

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modifier that drives a plant computer data point powered from the same instrument loop.

H. Operator actions:

Operations personnel responded to the failure of the ERCW flow indicator by initiating a Service Request to troubleshoot and correct the identified failure. TVA determined the failed fuses resulted in the loss of indication for AFW flow to Steam Generator No. 1 and Steam Generator No. 3, and Operations personnel subsequently entered LCO 3.3.3.5.

I. Automatically and manually initiated safety system responses:

There were no automatic or manual initiated safety system responses.

III. Cause of the event

A. The cause of each component or system failure or personnel error, if known:

The cause of the flow indicator failure was the result of two blown fuses. The cause of the blown fuses was the result of the failure of flow modifier module 1-FM-3-147C.

B. The cause(s) and circumstances for each human performance related root cause:

There were no human performance related issues for the identified condition.

IV. Analysis of the event:

The AFW system is safety related. The AFW system safety function involves supplying feedwater to the steam generators to remove reactor coolant system (RCS) (EISS Code AB) decay heat and other stored energy. The AFW system is required to provide feedwater when the non safety grade main feedwater (MFW) system (EISS Code SJ) is unavailable and the RCS is above the maximum pressure/temperature for initiation of residual heat removal (RHR) (EISS Code BP) system operation. There are three AFW trains, two motor driven pumps and one turbine driven pump.

As described in the updated Final Safety Analysis Report Sections 10.4.7.2, 15.2 and 15.4, the design basis events (DBEs) that impose safety related performance requirements on the AFW system include the loss of main feedwater (MFW) transient, main feedwater line break (MFLB), main steam line break (MSLB), loss of all alternating

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current power (station blackout) and a small break loss of coolant accident (SB LOCA). The AFW system performs safety functions to both mitigate and recover from these transients. These transients are divided into two broad classes which include 1) primary system cooldown events which limit the maximum amount of AFW flow, and 2) primary system heatup events that establish minimum AFW flow requirements. The MSLB is the limiting RCS cooldown event and establishes the maximum AFW flow limit (2250 gallons per minute (gpm) AFW flow delivered to a faulted loop until isolated). The loss of MFW and MFLB transients are the limiting RCS heatup events. The loss of MFW transient establishes a minimum AFW flow requirement of 410 gpm to at least two steam generators one minute following the initiation of a low-low steam generator water level reactor trip signal. The MFLB transient establishes a minimum AFW flow requirement of 1070 gpm to 3 steam generators within 10 minutes of a low-low steam generator water level reactor trip under the worst case single failure assumption. In addition to the DBE mitigation, the AFW system supports recovery from the DBEs listed above by providing sufficient flow to remove decay heat, stored energy and reactor coolant pump heat until such time as the RHR system can be placed in service. The requirements for performing this recovery function are identical to the minimum AFW requirements for the loss of MFW transient mitigation. The system is designed to start automatically in the event of a loss of off-site electrical power, a safety injection signal, low-low steam generator water level, a trip of one or both main feed-water pumps, any of which will result in, may be coincident with, or may be caused by a reactor trip, or an anticipated transient without scram mitigating systems actuating circuitry initiation.

The function of the ACR inlet flow indicators is to provide flow indication while Operations is using AFW to provide flow to the steam generators during a DBE, when the MCR is abandoned. The two AFW circuit loops that provide flow indication for AFW do not provide a control function of the AFW system. The electric pumps start on a two-out-of-three low-low-level signal in any steam generator; and the turbine pump starts on a two-out-of-three low-low-level signal in any two steam generators.

V. Assessment of Safety Consequences

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

Safety-related systems that were needed to shut down the reactor, maintain safe shutdown conditions, remove residual heat or mitigate the consequences of an accident remained available throughout the event.

- 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

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material, or mitigate the consequences of an accident:

This event did not occur when the reactor was shut down. Safety-related systems that were needed to shut down the reactor, maintain safe shutdown conditions, remove residual heat or mitigate the consequences of an accident remained available throughout the event.

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

There was no failure that rendered a train of a safety system inoperable during this event.

VI. Corrective Actions

Corrective Actions are being managed by TVA's corrective action program under PER 683145 and 688013.

- A. Immediate Corrective Actions:

Following troubleshooting of the ERCW flow indicator failure it was determined that two fuses in the circuit were blown and the 1-FM-3-147C flow modifier had failed. These fuses supply power to the instrument loops including power to the AFW flow indicators. The fuses and flow modifier were replaced.

- B. Corrective Actions to prevent recurrence or to reduce probability of similar events occurring in the future:

To reduce probability of similar events occurring in the future a corrective action established an alarm on the plant computer system should the AFW flow indicator instrument loop lose power.

VII. Additional Information

- A. Previous similar events at the same plant:

There were no previous similar events for this failure.

- B. Additional Information:

None.

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D. Safety System Functional Failure Consideration:

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

E. Scrams with Complications Consideration:

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

VIII. Commitments:

None