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

April 11, 2016

10 CFR 50.73

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

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

Subject: **Licensee Event Report 50-327/2016-001-00,** "Automatic Safety Injection due to Low Steam Line Pressure on Loop 2 Main Steam"

The enclosed Licensee Event Report (LER) provides details concerning an automatic safety injection due to low steam line pressure on loop 2 of the main steam line. This report is being 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 Reactor Protection System, Auxiliary Feedwater System, and Emergency Core Cooling System.

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

Respectfully,

Christopher J. Schwarz
Site Vice President
Sequoyah Nuclear Plant

Enclosure: Licensee Event Report 50-327/2016-001-00
cc: NRC Regional Administrator – Region II
NRC Senior Resident Inspector – Sequoyah Nuclear Plant

IE22
NRR

NRC FORM 366 U.S. NUCLEAR REGULATORY COMMISSION
(11-2015)

LICENSEE EVENT REPORT (LER)

(See Page 2 for required number of digits/characters for each block)

APPROVED BY OMB: NO. 3150-0104 EXPIRES: 10/31/2018

1. FACILITY NAME	2. DOCKET NUMBER	3. PAGE
Sequoyah Nuclear Plant Unit 1	05000327	1 OF 7

4. TITLE

Automatic Safety Injection due to Low Steam Line Pressure on Loop 2 Main Steam

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: NA	DOCKET NUMBER
02	09	2016	2016 -	001 -	00	04	11	2016	FACILITY NAME: NA	DOCKET NUMBER

9. OPERATING MODE	11. THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: *(Check all that apply)*			
3	☐ 20.2201(b)	☐ 20.2203(a)(3)(i)	☐ 50.73(a)(2)(ii)(A)	☐ 50.73(a)(2)(viii)(A)
	☐ 20.2201(d)	☐ 20.2203(a)(3)(ii)	☐ 50.73(a)(2)(ii)(B)	☐ 50.73(a)(2)(viii)(B)
	☐ 20.2203(a)(1)	☐ 20.2203(a)(4)	☐ 50.73(a)(2)(iii)	☐ 50.73(a)(2)(ix)(A)
	☐ 20.2203(a)(2)(i)	☐ 50.36(c)(1)(i)(A)	☒ **50.73(a)(2)(iv)(A)**	☐ 50.73(a)(2)(x)
10. POWER LEVEL	☐ 20.2203(a)(2)(ii)	☐ 50.36(c)(1)(ii)(A)	☐ 50.73(a)(2)(v)(A)	☐ 73.71(a)(4)
	☐ 20.2203(a)(2)(iii)	☐ 50.36(c)(2)	☐ 50.73(a)(2)(v)(B)	☐ 73.71(a)(5)
	☐ 20.2203(a)(2)(iv)	☐ 50.46(a)(3)(ii)	☐ 50.73(a)(2)(v)(C)	☐ 73.77(a)(1)
00	☐ 20.2203(a)(2)(v)	☐ 50.73(a)(2)(i)(A)	☐ 50.73(a)(2)(v)(D)	☐ 73.77(a)(2)(i)
	☐ 20.2203(a)(2)(vi)	☐ 50.73(a)(2)(i)(B)	☐ 50.73(a)(2)(vii)	☐ 73.77(a)(2)(ii)
		☐ 50.73(a)(2)(i)(C)	☐ OTHER Specify in Abstract below or in NRC Form 366A	

12. LICENSEE CONTACT FOR THIS LER

LICENSEE CONTACT	TELEPHONE NUMBER *(Include Area Code)*
Jon Johnson	423-843-8129

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 typewritten lines)*

On February 9, 2016, at 1415 Eastern Standard Time, Sequoyah Nuclear Plant (SQN) Unit 1 experienced a Low Steam Line Pressure Safety Injection (SI) signal. At the time of the event, Operations was warming the main steam (MS) line downstream of the main steam isolations valves (MSIV) via the Loop 2 MSIV bypass valve. During this controlled evolution, a sudden drop in MS line pressure occurred that actuated the two-out-of-three logic as sensed on all three Loop 2 steam line pressure bi-stables. All safety systems responded as designed and the SI was terminated per procedure.

Prior to the event, Unit 1 was recovering from a forced outage involving a Main Generator Blower repair that began on December 26, 2016. Unit 1 was maintained in Mode 3 during this time. The most probable cause of the SI actuation was determined to be water accumulation in MS Loop 2 upstream of the MSIV as a result of steam condensation not being drained during the extended Mode 3 forced outage.

Troubleshooting was initiated and Unit 1 was placed in Mode 4 to facilitate draining of Loop 2 and 3 MS lines. Corrective actions include actions to address operating for extended periods of time in Mode 4 and above with MSIVs closed.

LICENSEE EVENT REPORT (LER) CONTINUATION SHEET

1. FACILITY NAME	2. DOCKET NUMBER	3. LER NUMBER YEAR	SEQUENTIAL NUMBER	REV NO.
Sequoyah Nuclear Plant Unit 1	05000327	2016	- 001 -	00

NARRATIVE

I. Plant Operating Conditions Before the Event

At the time of the event, Sequoyah Nuclear Plant (SQN) Unit 1 reactor was in Mode 3 due to a forced outage involving the Main Generator. No inoperable systems, structures, or components contributed to this event.

II. Description of Events

A. Event:

On February 9, 2016 at 1415 Eastern Standard Time (EST), SQN Unit 1 experienced an automatic Low Steam Line Pressure Safety Injection (SI) [EIIS Code BQ] signal. The SI resulted in actuation of both trains of Emergency Core Cooling Systems (ECCS) [EIIS Codes BQ/BP] and injection into the Reactor Coolant System (RCS) [EIIS Code AB].

Just prior to the event, Operations was preparing to warm the main steam (MS) [EIIS code SB] lines downstream of the main steam isolation valves (MSIV) [EIIS Code ISV] in accordance with System Operating Instructions. The System Operating Instruction allows for MS line warming using up to three MSIV bypass valves based on the RCS temperature. SQN Unit 1 was in Mode 3 with two reactor coolant pumps (RCPs) [EIIS Code P] in service and RCS temperature and pressure at 524 degrees Fahrenheit (F) and 2235 pound per square inch gauge (psig), respectively. The Loop 2 MSIV bypass valve was selected to conduct the MS line warming because it had a running RCP, and the steam pressure trends noted during MSIV bypass valve operability testing provided confidence that RCS temperature and steam generator (SG) [EIIS Code SG] pressure would be adequately controlled. Operations briefed a contingency to close the Loop 2 MSIV bypass valve if plant conditions were not as expected.

At approximately 1413 EST, the Loop 2 MSIV bypass valve was opened. The pressure drop in MS Loop 2 was approximately 10 to 12 psig per minute. At 1415 EST, a sudden pressure drop in MS Loop 2 occurred at approximately 50 psig per second and the automatic SI signal initiated. All three MS Loop 2 pressure bi-stables that feed the protective sets actuated up within 20 milliseconds of each other making up the two-out-of-three logic. All safety systems responded as designed and the SI was terminated per procedure.

During the warming of MS Loop 2, the water was transferred to piping downstream of the MSIV via the 2-inch MSIV bypass line. Once the vertical sections of piping were void of water, the horizontal sections began emptying and opened a flow path for steam to move into the vertical sections of piping resulting in a steam pressure drop in the Loop 2 SG. Ultrasonic water level measurements identified the MS Loop 2 36-inch horizontal line contained about 18 inches of

1. FACILITY NAME	2. DOCKET	6. LER NUMBER		
		YEAR	SEQUENTIAL NUMBER	REV NO.
Sequoyah Nuclear Plant Unit 1	05000327	2016	- 001 -	00

NARRATIVE

water. The 36-inch MS Loop 3 horizontal line was determined to have approximately 14 inches of water.

Plans were developed and initiated to drain the water from the affected MS lines. Unit 1 was placed in Mode 4 to facilitate draining. All work was completed as planned.

The event is reportable in accordance with 10 CFR 50.73(a)(2)(iv)(A), as an event that resulted in a manual or automatic actuation of the Reactor Protection System (RPS), the Auxiliary Feedwater System (AFW) [EIIS Code BA] , and the ECCS.

By design, following the SI signal, both Emergency Gas Treatment System (EGTS) [EIIS Code BH] air cleanup subsystem trains aligned to support Unit 1. This action caused both trains of the EGTS air cleanup subsystem to be inoperable for Unit 2 on February 9, 2016, from 1416 EST until 1602 EST, when the EGTS was restored per procedure.

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:

On February 9, 2016, at 1412 EST, operators commenced warming of the Unit 1 MS header using the loop 2 MSIV bypass valve. At 1415 EST, Unit 1 experienced a low steam line pressure safety injection signal.

Dates and Times	Description
December 26, 2015 at 0000 EST	Unit 1 commenced power reduction for a scheduled forced outage to inspect Unit 1 Main Generator
1014 EST	Unit 1 reactor was manually tripped and enterd Mode 3
February 9, 2016 at 1412 EST	Unit 1 commenced warming of the MS header using Steam Generator Loop No. 2 MSIV Bypass Valve
1415 EST	Unit 1 experienced a Low Steam Line Pressure Safety Injection Signal. Unit 1 enters Emergency Procedure E-0, Reactor Trip or Safety Injection.
1427 EST	Unit 1 operators terminate SI per procedure.

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

There were no components that failed during the event.

1. FACILITY NAME	2. DOCKET	6. LER NUMBER		
		YEAR	SEQUENTIAL NUMBER	REV NO.
Sequoyah Nuclear Plant Unit 1	05000327	2016	- 001 -	00

NARRATIVE

E. Other systems or secondary functions affected:

Because the centrifugal charging pumps (CCP) [EIIS Code BQ] injected into the RCS loops during the event, the Pressurizer (PZR) [EIIS Code PZR] level increased and then decreased, which is indicative of a PZR insurge followed by a rapid heatup of PZR liquid and an outsurge. The PZR water temperature decreased from 662 degrees F to 544 degrees F and then increased back to 663 degrees F. SQN's Technical Requirements Manual (TRM) limits changes to the PZR temperature to a maximum heatup of 100 degrees F and a maximum cool down of 200 degrees F in any one hour period. As a result, an evaluation was conducted to analyze thermal fatigue induced damage to the PZR Lower Head and Surge Line Nozzle.

The PZR insurge and outsurge did cause a small increase in fatigue usage of the PZR Lower Head and Surge Line Nozzle. The increase has been demonstrated to be well within available fatigue margins.

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

Control Room alarms annunciated alerting operators to the start of the event.

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

No failure occurred.

H. Operator actions:

The operators entered Emergency Procedure E-0, Reactor Trip or Safety Injections and transitioned to ES-1.1, SI Termination. No human performance issues were identified.

I. Automatically and manually initiated safety system responses:

Following the automatic SI actuation, all plant systems responded as designed. The following automatic responses occurred: Phase A Containment Isolation, Containment Ventilation Isolation, Control Room Isolation and Auxiliary Building Isolation. All four emergency diesel generators [EIIS Code EK] started and operated in standby mode without tying to the engineered safety feature boards [EIIS Code EB]. All control rods were previously inserted as part of the controlled shutdown and the reactor trip breakers were open.

III. Cause of the event

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

The most probable cause of the automatic SI actuation was determined to be water accumulation in MS Loop 2 as a result of steam condensation not being drained during the extended Mode 3 forced outage.

1. FACILITY NAME	2. DOCKET	6. LER NUMBER		
		YEAR	SEQUENTIAL NUMBER	REV NO.
Sequoyah Nuclear Plant Unit 1	05000327	2016	- 001 -	00

NARRATIVE

SQN personnel did not anticipate the condition resulting from an extended period of time in Mode 3. Specifically, a process to evaluate the accumulation of water upstream of the MSIVs was not considered. An effective use of operating experience, prior to entering the period of extended Mode 3 operation was not used.

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

No human performance issues were identified.

The cause analysis is documented in Condition Report 1135308.

IV. Analysis of the event:

Prior to the automatic SI event, SQN Unit 1 was in Mode 3 for the previous 45 days. The Updated Final Safety Analysis Report (UFSAR) event most similar to this event is the Spurious SI at Power Event from 102 percent of 3423 megawatts (MW) thermal RTP (Rated Thermal Power). In the UFSAR analysis, the reactor trips with the SI, or it can be tripped by the RPS. It is assumed that the reactor trip occurs due to low pressurizer pressure. During the event described in this LER, the PZR pressure was maintained well over the SI setpoint and remained within the UFSAR analysis.

There was no transient for reactor power. A reactor trip signal was generated, but the reactor trip breakers were already open. Prior to the event, RCS pressure was approximately 2235 psig. Following SI, RCS pressure increased to around 2250 psig, decreased to around 2220 psig, then increased to around 2278 psig. The RCS temperature averaged about 527 degrees F prior to the event. The lowering steam pressure, the influx of cold AFW generated by the SI, and the CCP injection tank (CCPIT) injection into the RCS reduced the RCS temperature by 10-12 degrees in approximately 5 minutes. PZR level was on program at approximately 25 percent at the time of the SI. As the RCS temperature experienced no immediate reduction following the SI, the PZR level increased due to the influx of water via the CCPIT and the letdown isolation. The PZR level had reached 50 percent at the time the SI was terminated and the CCPIT flow isolated. The PZR level continued to rise until excess letdown was established. The peak PZR level was 65 percent. In the UFSAR analysis, the PZR would approach 100 percent level. Two RCPs were in service at all times during the transient and forced flow was maintained with no anomalies noted. Containment pressure and temperature were unaffected by this transient. Radiation levels following the unit trip were within expected post trip values. The non-response of these indicators is consistent with a leak-tight RCS and no transient induced fuel failures.

The RCS bulk temperature change was small in scope. There was no challenge to the Technical Specification (TS) 3.4.3, "RCS Pressure and Temperature (P/T) Limits." The actual PZR heatup during the SI event was 119 degrees F in approximately 55 minutes, which violated Technical Requirement 8.4.2. The pressure remained within TS limits and bounded by the UFSAR analysis. While this transient caused a small increase in the total fatigue usage of the lower pressurizer and surge line, these components remain well within design basis fatigue limits.

NRC FORM 366A (11-2015)	LICENSEE EVENT REPORT (LER) CONTINUATION SHEET		U.S. NUCLEAR REGULATORY COMMISSION	
1. FACILITY NAME	2. DOCKET	6. LER NUMBER		
		YEAR	SEQUENTIAL NUMBER	REV NO.
Sequoyah Nuclear Plant Unit 1	05000327	2016	- 001 -	00

NARRATIVE

With Unit 1 initially in Mode 3, the A and B motor driven AFW pumps were already in service at the time of the event. The turbine driven AFW pump did start. The AFW flow on all four loops increased rapidly to the maximum flow rate at the time of the SI signal, consistent with all of the AFW level control valves fully opening as designed. The Operators decreased the AFW flow rate to a low value within 4 minutes of the SI signal. Main feedwater [EIIS Code SJ] was already isolated at the time of the event. The MSIVs were closed prior to and during the event. The relatively small RCS temperature change had little effect on the SG water level.

The plant responded as expected for the conditions of the event. Plant response remained within TS limits and was bounded by the UFSAR analysis during this event.

V. Assessment of Safety Consequences

There were no safety consequences as a result of the event. All safety systems functioned as designed. No TS limits were exceeded and the event was bounded by the UFSAR.

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

No systems or components failed as a result of the SI actuation.

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 material, or mitigate the consequences of an accident:

The event occurred with Unit 1 in Mode 3. With Unit 1 reactor shutdown, the systems and components needed to maintain safe shutdown conditions and remove residual heat, control the release of radioactive material or mitigate the consequences of an accident were not affected.

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.

VI. Corrective Actions

Corrective Actions are being managed by TVA's corrective action program under Condition Report 1135308.

A. Immediate Corrective Actions:

Operations implemented E-0 and recovery actions.

NRC FORM 366A
(11-2015)

U.S. NUCLEAR REGULATORY COMMISSION

LICENSEE EVENT REPORT (LER)
CONTINUATION SHEET

1. FACILITY NAME	2. DOCKET	6. LER NUMBER		
		YEAR	SEQUENTIAL NUMBER	REV NO.
Sequoyah Nuclear Plant Unit 1	05000327	2016	- 001 -	00

NARRATIVE

B. Corrective Actions to Prevent Recurrence or to reduce probability of similar events occurring in the future:

Evaluate procedural changes to include precautions and limitations, and to address operating for extended periods of time in Mode 4 and above with MSIVs closed.

Evaluate and lower MS line pressure to allow MSIVs to leak by to clear lines.

Develop a periodicity of monitoring MS lines for condensation built up both by temperature and ultrasonic measurement.

Evaluate and develop a schedule to drain paths by periodically opening the Atmospheric Relief Valves and/or MSIV bypass valves to clear MS lines.

VII. Additional Information

A. Previous similar events at the same plant:

There were no other similar events at SQN.

B. Additional Information:

None.

C. Safety System Functional Failure Consideration:

The event did not result in a safety system functional failure.

D. Scrams with Complications Consideration:

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

VIII. Commitments:

None.