Tennessee Valley Authority, Sequoyah Nuclear Plant, P.O. Box 2000, Soddy Daisy, Tennessee 37384

July 1, 2020

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 50-327/2020-002-00, Safety Injection Signal with Reactor Trip Caused by a Failure with the Main Turbine Control System**

The enclosed licensee event report provides details concerning an automatic reactor trip due to a low steam line pressure safety injection signal. This event is being reported, in accordance with 10 CFR 50.73(a)(2)(iv)(A), as an event or condition that resulted in automatic actuation of any of the systems listed in paragraph (a)(2)(iv)(B) of 10 CFR 50.73: reactor protection system, general containment isolation signal, emergency core cooling systems, and auxiliary feedwater system.

There are no regulatory commitments contained in this letter. Should you have any questions concerning this submittal, please contact Mr. Jeffrey Sowa, Site Licensing Manager, at (423) 843-8129.

Respectfully,

Matthew Rasmussen
Site Vice President
Sequoyah Nuclear Plant

Enclosure: Licensee Event Report 50-327/2020-002-00
cc: NRC Regional Administrator – Region II
NRC Senior Resident Inspector – Sequoyah Nuclear Plant

NRC FORM 366 (04-2020)	U.S. NUCLEAR REGULATORY COMMISSION	APPROVED BY OMB: NO. 3150-0104 EXPIRES: 04/30/2020

LICENSEE EVENT REPORT (LER)

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

(See NUREG-1022, R.3 for instruction and guidance for completing this form http://www.nrc.gov/reading-rm/doc-collections/nuregs/staff/sr1022/r3/)

1. Facility Name	2. Docket Number	3. Page
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4. Title
Safety Injection Signal with Reactor Trip Caused by a Failure with the Main Turbine Control System

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	13	2020	2020	- 002	- 00	07	01	2020	NA	05000
									NA	05000

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)(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)
100	☐ 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)
	☐ 20.2203(a)(2)(v)	☐ 50.73(a)(2)(i)(A)	☐ 50.73(a)(2)(v)(D)	☐ 73.77(a)(2)(ii)
	☐ 20.2203(a)(2)(vi)	☐ 50.73(a)(2)(i)(B)	☐ 50.73(a)(2)(vii)	☐ 73.77(a)(2)(iii)
		☐ 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)
Scott Bowman	423-843-6910

13. Complete One Line for each Component Failure Described in this Report

Cause	System	Component	Manufacturer	Reportable To ICES	Cause	System	Component	Manufacturer	Reportable To ICES
X	TA	XM	SIEMENS	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 14 single-spaced typewritten lines)

On May 13, 2020, at 0208 eastern daylight time, Sequoyah Nuclear Plant, Unit 1, was at 100 percent power when an automatic reactor trip signal was received concurrent with a low steam line pressure safety injection signal. The low steam line pressure safety injection signal was actuated from the steam rate of decrease feature. Following the reactor trip, all plant safety systems responded as designed with the exception of the following components: Pressure Transmitter, 1-PT-1-23, was found to have a failed power supply and Glycol Inboard Isolation Valve, 1-FCV-61-122, failed to automatically isolate on the Phase A containment isolation signal (the corresponding outboard isolation valve actuated as designed such that the penetration was isolated, as required). A post-trip troubleshooting team determined that the cause of the event was the failure of a Speed Error Channel B card associated with the Analog Electro-Hydraulic (AEH) Control System for the Main Turbine. The card was replaced, calibrated, and successfully passed functional testing.

The failure of the Speed Error Amplifier B Card was caused by a failed chip on the card identified because of an over heated resistor. The corrective action to reduce similar events is to replace the Main Turbine AEH Control System.

NARRATIVE

I. Plant Operating Conditions Before the Event

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

II. Description of Event

A. Event Summary:

On May 13, 2020, beginning at approximately 0202 eastern daylight time (EDT), SQN Unit 1 main turbine [EIIS: TA] governor valves [EIIS: XCV] started oscillating causing steam generator (SG) [EIIS: SG] water levels, in SGs 1 and 3 to rise and megawatt electric (MWe) load to swing on three occasions (the actual MWe load swings varied each time). Shortly thereafter, at 0208, SQN Unit 1 experienced an automatic reactor trip. The reactor trip was preceded by a safety injection (SI) [EIIS: BQ] signal due to a low steam line pressure signal (signal actuated on rate of change). In addition to the SI and reactor trip, the following automatic responses occurred: Phase A Containment Isolation, Containment Ventilation Isolation, Main Control Room (MCR) Isolation, Auxiliary Building Isolation, Main Steam [EIIS: SB] Isolation Valve [EIIS: ISV] Closure, Main Feedwater Isolation and Auxiliary Feedwater (AFW) [EIIS: BA] start.

Following the reactor trip, all plant safety systems responded as designed with the exception of the following components: Pressure Transmitter [EIIS: PT], 1-PT-1-23, was found to have a failed power supply and Glycol Inboard Isolation Valve, 1-FCV-61-122, failed to automatically isolate on the Phase A containment isolation signal (the corresponding outboard isolation valve actuated as designed such that the penetration was isolated, as required).

A post-trip troubleshooting team determined that the cause of the event was the failure of a Speed Error Channel B card [EIIS: XM] associated with the Analog Electro-Hydraulic (AEH) Control System for the Main Turbine. The card was replaced, calibrated, and successfully passed functional testing.

This event is reportable in accordance with 10 CFR 50.73(a)(2)(iv)(A), as an event or condition that resulted in automatic actuation of any of the systems listed in paragraph (a)(2)(iv)(B) of 10 CFR 50.73: reactor protection system [EIIS: JC], general containment isolation signal, emergency core cooling systems [EIIS: BQ/BP], and AFW system.

By design, following the SI signal, both Emergency Gas Treatment System (EGTS) [EIIS: 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 May 13, 2020, from 0208 until 0257, 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:

No inoperable structures, components, or systems contributed to this event.

C. Dates and approximate times of occurrences:

Date/Time (EDT)	Description
05/13/20, 0202	Main turbine governor valves started oscillating causing MWe swings.
02:02:47	Steam line pressure started going up.
02:02:48	Steam generator dump valves started opening and then started closing.
02:02:51	Steam line pressure started decreasing from peak.
02:07:11	Main turbine governor valve oscillations occurred a second time.
02:07:33	Steam generator dump valves started opening, again. Main turbine governor valve oscillations occurred a third and final time.
02:07:50	Safety Injection actuated. Both trains of containment ventilation isolation, containment isolation phase A, control room isolation, and auxiliary building isolation actuated. Both trains of the Unit 1 reactor trip solid state protection system actuated. The reactor trip was followed by a turbine trip.
0221	The SI was terminated.

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

The component that failed was a Siemens Speed Error Amplifier B Card associated with the Main Turbine (AEH) Control System.

E. Other systems or secondary functions affected:

Pressure Transmitter, 1-PT-1-23, was found to have a failed power supply and Glycol Inboard Isolation Valve, 1-FCV-61-122, failed to automatically isolate on the Phase A containment isolation signal (the corresponding outboard isolation valve actuated, as designed, such that the penetration was isolated, as required).

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

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

G. Failure mode, mechanism, and effect of each failed component:

The failure mechanism of the Speed Error Amplifier B Card was a subcomponent failure of a chip on the card. The failure mode was erroneous output associated with the failed chip that caused a resistor to overheat on the card.

H. Operator actions:

At approximately 0202 an unexpected transient occurred on the secondary side causing multiple alarms to annunciate. The operating crew observed that all twelve steam dumps cycled open and closed. Rod control indicated outward direction followed by inward direction. Steam Generator water levels in SGs 1 and 3 rose, then began to stabilize. After approximately four minutes, Control Bank D began stepping in, again, and it appeared that the unit was experiencing a run back. The unit supervisor recognized the runback was not stopping and ordered the operator at the controls (OATC) to trip the Unit 1 reactor. As the OATC reached to manually trip the reactor, it automatically tripped.

Following the trip, the operators entered the appropriate procedures and terminated the event. No human performance issues were identified.

I. Automatically and manually initiated safety system responses:

Following the SI actuation, the following automatic responses occurred: Reactor Trip, Phase A Containment Isolation, Containment Ventilation Isolation, Control Room Isolation, Auxiliary Building Isolation, Main Steam Isolation Valves Close, Main Feedwater Isolation and Auxiliary Feedwater start. All rods fully inserted as required.

III. Cause of the Event

A. Cause of each component or system failure or personnel error:

The failure of the Speed Error Amplifier B Card was caused by a failed chip on the card identified because of an over heated resistor.

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

There was no identified human performance related root cause.

IV. Analysis of the Event:

The plant safety system responses during and after the reactor trip were bounded by the responses described in the Updated Final Safety Analysis Report (UFSAR). The UFSAR Chapter 15 event that most closely matches the reactor trip is the accidental depressurization of the main steam system associated with an inadvertent opening of a single steam dump, relief, or safety valve. The features provided to protect against an accidental depressurization of the main steam system are the Safety Injection System, the overpower reactor trips (neutron flux and ΔT) and the reactor trip coincident with an SI signal, and redundant isolation of the main feedwater lines. Because the UFSAR transient bounds this event, there was no adverse affect associated with the health and safety of plant personnel or the general public.

V. Assessment of Safety Consequences

There were no actual safety consequences as a result of the reactor trip.

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

None.

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 did not occur when the reactor was shut down.

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

The reactor trip event was entered into the Tennessee Valley Authority Corrective Action Program (CAP) under CR 1607726.

A. Immediate Corrective Actions:

Troubleshooting determined that the cause of the event was the failure of a Speed Error Channel B card associated with the AEH Control System for the Main Turbine. The card was replaced, calibrated, and successfully passed functional testing.

1. FACILITY NAME	2. DOCKET NUMBER	3. LER NUMBER		
		YEAR	SEQUENTIAL NUMBER	REV NO.
Sequoyah Nuclear Plant Unit 1	05000-327	2020	- 002	- 00

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

The corrective action to reduce similar events is to replace the Main Turbine AEH Control System. The AEH control systems for both units are scheduled to be replaced during their respective Cycle 24 refueling outages.

VII. Previous Similar Events at the Same Site:

There were no previous similar events occurring at SQN within the last three years.

VIII. Additional Information

There is no additional information.

IX. Commitments:

There are no commitments.