TVA

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

May 11, 2015

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 50-327/2015-001-00, "Automatic Reactor Trip due to Negative Rate Trip as a Result of a Dropped Control Rod"**

The enclosed Licensee Event Report provides details concerning an automatic reactor trip following a dropped control rod. 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 and the Auxiliary Feedwater System. This condition had no impact on Unit 2. A supplement to this LER is planned for November 30, 2015 that includes the results of the associated root cause analysis.

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

Respectfully,

John T. Carlin
Site Vice President
Sequoyah Nuclear Plant

Enclosure: Licensee Event Report 50-327/2015-001

cc: NRC Regional Administrator – Region II
NRC Senior Resident Inspector – Sequoyah Nuclear Plant

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NRR

NNNRC FORM 366 U.S. NUCLEAR REGULATORY COMMISSION
(02-2014)

LICENSEE EVENT REPORT (LER)

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

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4. TITLE

Automatic Reactor Trip due to Negative Rate Trip as a Result of a Dropped Control Rod

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 05000
03	11	2015	2015	- 001	- 00	05	11	2015	FACILITY NAME	DOCKET NUMBER 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)(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	☐ 20.2203(a)(2)(ii)	☐ 50.36(c)(1)(ii)(A)	X **50.73(a)(2)(iv)(A)**	☐ 50.73(a)(2)(x)
99	☐ 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

LICENSEE CONTACT	TELEPHONE NUMBER *(Include Area Code)*
Jon Johnson, Program Manager	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
X YES *(If yes, complete 15. EXPECTED SUBMISSION DATE)* ☐ NO		11	30	2015

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

On March 11, 2015, at 0621 Eastern Daylight Time Sequoyah Nuclear Plant Unit 1 reactor automatically tripped due to a Negative Rate Trip as a result of Control Bank D Control Rod H-8 dropping into the core. Initial investigation revealed Control Rod H-8 dropped into the core approximately one second before the reactor trip. The dropped control rod caused a rapid decrease in power which was sensed by all four nuclear instrumentation system power range channels. Reactor trip logic is two out of four channels. No power changes or control rod motion were in progress prior to the reactor trip. All safety related equipment operated as designed, all control rods fully inserted as required, and auxiliary feedwater automatically initiated as expected. Unit 1 was stabilized in hot standby following the automatic reactor trip. The cause of the reactor trip was due to Control Rod H-8 failing to maintain its commanded position. Trouble shooting was performed on the electrical components associated with Control Rod H-8. All of the inspected components and testing were determined to be acceptable. Further troubleshooting is being conducted during the Unit 1 refueling outage and will be evaluated by Westinghouse and the licensee. A root cause evaluation is ongoing. As a result, the root cause and the associated corrective actions have not yet been established and will be provided in a revision to this LER. Unit 2 was unaffected by this event.

NARRATIVE

I. Plant Operating Conditions Before the Event

At the time of the event, Sequoyah Nuclear Plant (SQN) Unit 1 reactor was operating at approximately 99 percent rated thermal power (RTP). Unit 1 Turbine load was being decreased periodically in preparation for an upcoming planned outage. The condition described in this LER did not impact SQN Unit 2.

II. Description of Events

A. Event:

On March 11, 2015 at 06:21 Eastern Daylight Time (EDT), SQN Unit 1 reactor automatically tripped due to a Negative Rate Trip as a result of Control Bank D Control Rod H-8 [EIIS Code AA] dropping into the core. Investigation revealed Control Rod H-8 dropped into the core approximately one second before the reactor trip. Control Rod H-8 is located in the center of the core and is one of nine control rods in control bank D. The dropped control rod cause a rapid decrease in power which was sensed by all four nuclear instrumentation system (NIS) power range channels. The reactor trip logic is two out of four channels.

Trouble shooting was performed on the electrical components associated with Control Rod H-8. The components inspected and testing performed were determined to be acceptable.

All safety related equipment operated as designed, all control rods fully inserted as required, and auxiliary feedwater automatically initiated as expected. No complications were experienced during the reactor trip.

On March 11, 2015 at 0930 EDT, NRC was notified, in accordance with 10 CFR Part 50.72(b)(2)(iv)(B), due to a reactor protection system actuation and 10 CFR Part 50.72(b)(3)(iv)(A) due to a specified system actuation.

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
March 11, 2015 at 06:21 EDT	Integrated Computer System (ICS) indicates that Control Bank Control Rod H-8 starts dropping. All four NIS power range detectors indicate a drop in reactor power at the same time.

NARRATIVE

	A power range, neutron flux, negative rate trip is generated due to a rapid drop in reactor power. The remaining 52 control rods insert into the core.
	Operations perform immediate actions associated with procedure E-0, Reactor Trip or Safety Injection.
09:30	Control Rod H-8 stationary gripper fuses and blown fuse indicator are checked. No problems are identified.
1600	Continuity check performed on Control Rod H-8 coil and associated components with satisfactory results.

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

The root cause for this event is still under investigation based on data obtained during the current refueling outage. SQN plans to re-analyze all the data following the outage which could include additional lab analyses. When the final investigation is completed a supplement to this LER will be provided.

E. Other systems or secondary functions affected:

There were no other systems or functions affected by this event.

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

Reactor and turbine trip alarms annunciated alerting operators to the start of the event.

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

The root cause for this event is still under investigation based on data obtained during the current refueling outage. SQN plans to re-analyze all the data following the outage which could include additional lab analyses. When the final investigation is completed a supplement to this LER will be provided.

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H. Operator actions:

The operators entered Emergency Procedure E-0, Reactor Trip or Safety Injection and then transitioned from E-0 to Emergency Subprocedure ES-0.1, Reactor Trip Response. There were no identified complications or human performance issues associated with the trip response. Problem Evaluation Report (PER) 997605 was initiated to document that Unit 1 had tripped and start the investigation for the cause of the trip.

I. Automatically and manually initiated safety system responses:

Following the reactor trip, all plant safety systems responded as designed. All rods fully inserted as required. Auxiliary feedwater (AFW) automatically initiated from the feedwater isolation signal as expected.

III. Cause of the event

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

The root cause for this event is still under investigation based on data obtained during the current refueling outage. SQN plans to re-analyze all the data following the outage which could include additional lab analyses. When the final investigation is completed a supplement to this LER will be provided.

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

The root cause for this event is still under investigation based on data obtained during the current refueling outage. SQN plans to re-analyze all the data following the outage which could include additional lab analyses. When the final investigation is completed a supplement to this LER will be provided.

IV. Analysis of the event:

Prior to the event, SQN Unit 1 was operating in MODE 1 at approximately 99 percent RTP with the Reactor Coolant System (RCS) [EIIS Code AB] pressure and temperature near the nominal value of approximately 2235 pounds per square inch gauge (psig) and approximately 578 degrees Fahrenheit (F). Both the motor driven and the turbine driven auxiliary feedwater (AFW) [EIIS Code BA] pumps and steam dump valves (SDV) and the atmospheric relief valves (ARV) were available.

Following the reactor trip, RCS pressure rapidly decreased due to the decreasing RCS average temperature and the associated shrinking of coolant volume. The minimum RCS pressure was approximately 2019 psig, well above the pressure that would have initiated a safety injection signal (1870 psig). Pressurizer [EIIS Code AB] pressure recovered gradually before dropping back to normal operating pressure.

NARRATIVE

The DNB limit for RCS average temperature of less than or equal to 583 degrees F was not exceeded. The loss of nuclear heat generation resulted in a decrease in RCS temperature to approximately 538 degrees F.

The reactor coolant pumps (RCP) [EIIS Code AB] were in service at all times during the transient and forced flow was maintained with no anomalies noted.

Prior to the trip, PZR level was being maintained in the normal program band of approximately 60 percent. Following the trip, pressurizer level followed the RCS temperature response; increasing and decreasing in magnitude, slope and duration with the RCS temperature. The minimum PZR level following the trip was approximately 22 percent. Over a 15 minute period, pressurizer level stabilized within its program value.

The main feedwater flow rate was at nominal full power value prior to the reactor trip. When RCS average temperature dropped below 550 degrees F, main feedwater was isolated [EIIS code SJ]. The AFW system was initiated following the reactor trip on steam generators (SG) [EIIS Code AB] low-low level. AFW flow was reduced at 6 minutes after the trip to less than approximately 215 gallons per minute (gpm) to mitigate the decrease in RCS average temperature and also due to recovering SG levels.

The plant responded as expected for the conditions of the trip.

V. Assessment of Safety Consequences

There were no safety consequences as a result of the event. All safety systems functioned as designed and no complications were experienced. No Technical Specification limits were exceeded and the Updated Final Safety Analysis Report (UFSAR) analyses of the event remained bounding.

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:

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.

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NARRATIVE

VI. Corrective Actions

Corrective Actions are being managed by TVA's corrective action program under PER 997605.

A. Immediate Corrective Actions:

- Reactor trip recovery completed.
- Initial troubleshooting of fuses and circuit cards associated with Control Rod H-8 performed with no issues identified.
- Westinghouse troubleshooting guide steps performed with no issues identified (WCAP-15360-P, Westinghouse Rod Control Corrective Maintenance Guide).
- Control rods were successfully exercised with no issues.
- Installed testing equipment to monitor Control Rod H-8 during operation with no issues identified going into the planned refueling outage.

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

The root cause for this event is still under investigation based on data obtained during the current refueling outage. SQN plans to re-analyze all the data following the outage which could include additional lab analyses. When the final investigation is completed a supplement to this LER will be provided.

VII. Additional Information

A. Previous similar events at the same plant:

A review of previous reportable events for the past three years did not identify any previous similar events.

B. Additional Information:

None.

C. Safety System Functional Failure Consideration:

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

NARRATIVE

D. Scrams with Complications Consideration:

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

VIII. Commitments:

None.