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

April 25, 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 50-327/2019-003-01, Automatic Reactor Trip due to Negative Rate Trip as a Result of a Dropped Control Rod**

Reference: TVA letter submitted to NRC dated October 23, 2019, "Licensee Event Report 50-327/2019-003-00, Automatic Reactor Trip due to Negative Rate Trip as a Result of a Dropped Control Rod."

The enclosed licensee event report has been revised with supplemental information concerning an automatic reactor trip following a dropped control rod and an inoperable steam generator pressure transmitter affecting the Engineered Safety Features Actuation System. This revised report reflects the results of the causal analyses and associated corrective actions. These events were previously reported, 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 and the auxiliary feedwater system; 10 CFR 50.73(a)(2)(i)(B), as an event that resulted in a condition prohibited by Technical Specifications; and in accordance with 10 CFR 50.73(a)(2)(v)(D), as an event that resulted in a condition which could have prevented the fulfillment of a safety function necessary to mitigate the consequences of an accident. Changes to the reference report are indicated by revision bars on the right side margin of the page.

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 Plan

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 |
|---|---|---|
| Sequoyah Nuclear Plant Unit 1 | 05000327 | 1 OF 7 |

**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: NA | Docket Number: 05000 |
| 08 | 27 | 2019 | 2019 | - 003 | - 01 | 04 | 25 | 2020 | Facility Name: NA | 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)(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) |
| 100 | ☐ 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 | AA | 75 | W120 | Y | D | SB | PT | F180 | 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 August 27, 2019, at 0109 eastern daylight time, Sequoyah Nuclear Plant, Unit 1, automatically tripped as a result of Control Bank D Control Rod H-8 dropping into the core. All safety related equipment operated as designed, all control rods fully inserted as required, and auxiliary feedwater automatically initiated as expected. Additionally, during a post trip review, it was identified that the Steam Generator #3 pressure transmitter, 1-PT-1-23, demonstrated sluggish behavior. This pressure transmitter had previously been identified as inoperable during a Unit 1 reactor trip occurring April 14, 2019. Maintenance was performed following the April event and the transmitter was declared operable. After further review, it was determined that the transmitter was inoperable since November of 2015.

For Control Rod H-8, the root cause was determined to be that excessive wear to the H-8 stationary gripper latch mechanism resulted in the inability to maintain the control rod in the fully, or nearly fully, withdrawn position for an extended period of time. The corrective action is to submit a license amendment request and develop an engineering design change to permanently remove Control Rod H-8. For 1-PT-1-23, the cause was determined to be the procedure associated with post maintenance testing did not identify that maintenance can affect the response time of the transmitters. The corrective action is to revise the procedure to note response time aspects should be considered when developing post maintenance testing for these transmitters.

| 1. FACILITY NAME | 2. DOCKET NUMBER | 3. LER NUMBER | | |
|---|---|---|---|---|
| | | YEAR | SEQUENTIAL NUMBER | REV NO. |
| Sequoyah Nuclear Plant Unit 1 | 05000-327 | 2019 | - 003 | - 01 |

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:

Prior to beginning the coast-down for the Unit 1 refueling outage, scheduled to begin October 12, 2019, Control Bank D was stepped out of the core from 220 steps to 228 steps. The rods were in this position for approximately 1.5 hours. At 0109 eastern daylight time (EDT) on August 27, 2019, SQN Unit 1 automatically tripped due to a negative rate trip as a result of Control Bank D, Control Rod, H-8 [EIIS: AA] dropping into the core. Investigation revealed Control Rod H-8 dropped into the core approximately 1.3 seconds before all other control rods.

Post-trip troubleshooting was performed and did not identify any immediate concerns with the Control Rod Drive Mechanism (CRDM) System. With no cause identified, a monitoring plan was developed to enable site personnel to identify a cause in the event the condition was repeated prior to the Unit 1 refueling outage. Additionally, rod control current-order testing was performed to ensure the rod control system functioned as designed. Results from the current-order testing identified acceptable results from 0 to approximately 20 steps and 220 to 231 steps. A temporary modification was implemented to energize both the stationary gripper and the movable gripper to ensure Control Rod H-8 does not drop. The modification would not prevent Control Rod H-8 from dropping into the core upon opening of the reactor trip breakers.

Following the reactor trip, all safety related equipment operated as designed. No complications were experienced.

The Unit 1 reactor trip is reportable 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 [EIIS: JC] and the Auxiliary Feedwater (AFW) [EIIS: BA] System.

Additionally, during a post trip review, it was identified that Steam Generator (SG) [EIIS: SB] #3 pressure transmitter [EIIS: PT], 1-PT-1-23, demonstrated sluggish behavior. The sluggish response could have challenged the Engineered Safety Features Actuation System (ESFAS) [EIIS: JE] input function associated with SG pressure. 1-PT-1-23 is one of three pressure transmitters used to monitor SG #3 pressure. These three transmitters are used, with their instrument loops, to provide two-out-of-three logic (per SG) required by technical specifications for ESFAS. The affected portion of ESFAS included actuation signals for Safety Injection (SI) [EIIS: BQ] and Steam Line Isolation (SLI).

Maintenance was performed to backfill the sensing line and the transmitter tubing was removed and checked to verify that there were no blockages. 1-PT-1-23 was declared operable on August 28 at 0142 EDT.

1-PT-1-23 had previously been identified as inoperable during a Unit 1 reactor trip occurring April 15, 2019. Maintenance was performed following the April event and 1-PT-1-23 was declared operable. A past operability evaluation determined that the maintenance performed in April did not correct the condition and 1-PT-1-23 was actually inoperable since November of 2015.

With 1-PT-1-23 inoperable, Technical Specification (TS) Limiting Condition for Operation (LCO) 3.3.2.D for ESFAS Instrumentation required the channel be placed in trip within 72 hours, or be in Mode 3 within 78 hours. Failing to complete the Required Actions led to a condition prohibited by TS, and is reportable under 10 CFR 50.73(a)(2)(i)(B). Additionally, a past operability evaluation determined other steam line pressure channels in addition to 1-PT-1-23 were removed from service for testing since 2015. With 1-PT-1-23 and an additional instrument for that channel inoperable, ESFAS instrumentation would not have provided sufficient logic for actuation. This condition could have prevented the fulfillment of a safety function necessary to mitigate the consequences of an accident, which is reportable under 10 CFR 50.73(a)(2)(v)(D).

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 |
|---|---|
| November 2015 | Trace analysis of plant computer data indicated that 1-PT-1-23 pressure response was lagging. A condition report (CR) was initiated, but 1-PT-1-23 was not repaired. 1-PT-1-23 was not recognized as inoperable. |
| April 15, 2019, 1229 | Following a Unit 1 trip associated with the loss of an operating Main Feedwater Pump, a post trip review identified that 1-PT-1-23 pressure response was lagging. TS LCO 3.3.2.D was entered. |
| April 17, 2019, 0332 | Maintenance was completed with the sensing line blown down. 1-PT-1-23 was declared operable and TS LCO 3.3.2.D was exited. However, 1-PT-1-23 was still inoperable. |

| Date/Time (EDT) | Description |
|---|---|
| August 26, 2019 2332 | Control Bank D stepped out to 228 steps. |
| August 27, 2019, 0109 | The Unit 1 reactor tripped on power range high neutron flux rate due to Control Rod H-8 dropping into the core. Unit 1 entered Mode 3. Main control room (MCR) operators took appropriate actions in accordance with applicable procedures. |
| August 27, 2019, 0424 | A post trip review identified that 1-PT-1-23 pressure response was lagging. TS LCO 3.3.2.D was entered. |
| August 28, 2019, 0142 | Maintenance was completed and 1-PT-1-23 was declared operable. TS LCO 3.3.2.D was exited. |
| November 2015 – August 28, 2019 | Various maintenance and testing activities were performed (cumulative time 29 hours and 43 minutes) with both 1-PT-1-23 and additional channels removed from service. |

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

For the reactor trip, the component that failed is the magnetic control rod drive mechanism, model number, L-106A, manufactured by Westinghouse Electric Corporation.

The failed pressure transmitter is a safety-related, 10-50 milliAmp Foxboro Model E11GM. The component identification at SQN is SQN-1-PT-001-0023-F.

E. Other systems or secondary functions affected:

1-PT-1-23 provides input to the ESFAS.

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

MCR alarms and annunciators provided indication to the operators during the reactor trip.

1-PT-1-23 was discovered to be inoperable during a post trip review.

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

Flow induced vibration caused excessive wear to the H-8 stationary gripper latch mechanism, resulting in the inability to maintain the rod in the fully withdrawn or nearly fully withdrawn position for an extended period of time.

1-PT-1-23 failed because the procedure associated with post maintenance testing did not identify that maintenance can affect the response time of the transmitter.

H. Operator actions:

MCR operators responded to the reactor trip, as required. They promptly identified the condition and took appropriate actions.

I. Automatically and manually initiated safety system responses:

The reactor protection system, including main feedwater isolation and AFW start, responded to the trip, as designed. All rods fully inserted as required.

III. Cause of the Event

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

For Control Rod H-8, the root cause was determined to be that excessive wear to the H-8 stationary gripper latch mechanism resulted in the inability to maintain the control rod in the fully, or nearly fully, withdrawn position for an extended period of time.

For 1-PT-1-23, the cause was determined to be the procedure associated with post maintenance testing did not identify that maintenance can affect the response time of the transmitters.

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

There were no human performance related causes for the reactor trip and the inoperability of 1-PT-1-23.

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 Rod Cluster Control Assembly Misalignment. The degraded response of 1-PT-1-23 did not cause abnormal degradation of, or stress upon, the principle safety barriers (fuel cladding, reactor coolant system, or containment). Therefore, this event did not adversely affect 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 probabilistic risk assessment (PRA) evaluation was completed for the inoperable 1-PT-1-23.

The evaluation considered 1-PT-1-23 inoperable in conjunction with not maintaining a two-out-of-three logic for ESFAS instrumentation (SI signal on low SG pressure, SLI signal on low SG pressure, SLI signal on a High Negative Rate) for a duration covering 29 hours and 43 minutes (28 hours and 10 minutes previously reported in LER 1-2019-002 plus 93 minutes since previous LER), and the safety significance was determined to be low.

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:

1-PT-1-23 was declared inoperable and returned to service in approximately 22 hours.

## VI. Corrective Actions

The reactor trip event was entered into the Tennessee Valley Authority Corrective Action Program (CAP) under CR 1544222. The inoperability of 1-PT-1-23 was entered into the CAP under CR 1544227 and CR 1561136.

A. Immediate Corrective Actions:

For Control Rod H-8, troubleshooting was performed and did not identify any immediate concerns with the Control Rod Drive Mechanism (CRDM) System. With no cause identified, a monitoring plan was developed to enable site personnel to identify a cause in the event the condition was repeated prior to the Unit 1 refueling outage. Additionally, rod control current-order testing was performed to ensure the rod control system functioned as designed. Results from the current-order testing identified acceptable results from 0 to approximately 20 steps and 220 to 231 steps. However, unexpected anomalies were identified, during the test, which will be further investigated. Westinghouse was consulted on the issue and concluded that the current-order issue is a potential failure mechanism in conjunction with another potential failure mechanism (during the refueling outage, additional current order traces were analyzed and found to be within acceptable limits). Actions were put into place to not exceed 220 steps to ensure reliability prior to beginning the refueling outage to preclude another event. A temporary modification was implemented to energize

both the stationary gripper and movable gripper to ensure Control Rod H-8 does not drop. The modification would not prevent Control Rod H-8 from dropping into the core upon opening of the reactor trip breakers.

For 1-PT-1-23, work orders were completed to backfill and blow out the sensing lines.

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

For the reactor trip, the corrective action to prevent recurrence is to submit a license amendment request and develop an engineering design change to permanently remove Control Rod H-8.

For 1-PT-1-23, the corrective action is to revise the procedure to note response time aspects should be considered when developing post maintenance testing for these transmitters (the procedure revision was completed on December 20, 2019).

## VII. Previous Similar Events at the Same Site:

A review of previous reportable events related to an automatic reactor trip due to a negative rate trip identified an automatic reactor trip as a result of Control Rod, H-8, falling into the Unit 1 core reported in LER 1-2015-001. The root cause of the event was failure of a maintenance procedure to provide inspection guidance and acceptance criteria on CRDM vertical panel connections. The corrective action to prevent recurrence was revising the maintenance procedure and periodic preventive maintenance of CRDM connections.

Additionally, a review of previous reportable events for the past three years identified that LER 1-2019-002 involved a condition prohibited by technical specifications and an event that could have prevented the fulfillment of a safety function associated with inoperable pressure transmitter, 1-PT-1-23. The maintenance performed on the transmitter, to restore operability, was unsuccessful and not identified until another reactor trip approximately four months later. The corrective action was to create preventative maintenance instructions to clear sensing lines for Main Steam transmitters.

## VIII. Additional Information

There is no additional information.

## IX. Commitments:

There are no commitments.