April 25, 2005

U.S. Nuclear Regulatory Commission 10 CFR 50.73
ATTN: Document Control Desk
Washington, D.C. 20555-0001

Gentlemen:

TENNESSEE VALLEY AUTHORITY - SEQUOYAH NUCLEAR PLANT (SQN) UNIT 2 - DOCKET NO. 50-328 - FACILITY OPERATING LICENSE DPR-79 - LICENSEE EVENT REPORT (LER) 50-328/2005-001-00

The enclosed LER provides details concerning the Unit 2 reactor trip and engineered safety feature (ESF) actuation of auxiliary feedwater. The automatic reactor trip occurred as a result of low-low steam generator level when the main feedwater regulation valves closed upon inadvertent loss of control power. This event is being 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 ESF actuation.

Sincerely,

Original signed by

P. L. Pace
Manager, Site Licensing and
Industry Affairs

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NRC FORM 366 (6-2004) U.S. NUCLEAR REGULATORY COMMISSION

APPROVED BY OMB: NO. 3150-0104 EXPIRES: 06/30/2007

LICENSEE EVENT REPORT (LER)

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

1. FACILITY NAME	2. DOCKET NUMBER	3. PAGE
Sequoyah Nuclear Plant (SQN) Unit 2	05000328	1 OF 6

4. TITLE
Unit 2 Reactor Trip Following Closure of Main Feedwater Upon Inadvertent Opening of Control Breakers

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
02	23	2005	2005	- 001 -	00	04	25	2005	FACILITY NAME	DOCKET NUMBER 05000

9. OPERATING MODE: 1

10. POWER LEVEL: 100

11. THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR§: *(Check all that apply)*

☐ 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)
☐ 20.2203(a)(2)(ii)	☐ 50.36(c)(1)(ii)(A)	☒ 50.73(a)(2)(iv)(A)	☐ 50.73(a)(2)(x)
☐ 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

FACILITY NAME	TELEPHONE NUMBER (Include Area Code)
Z. T. Kitts, Licensing Engineer	423-843-7018

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 23, 2005, at approximately 2030 Eastern standard time (EST), Maintenance personnel were performing pre-refueling outage maintenance activities on a 125-Volt (V) direct current (dc) vital battery board molded case circuit breaker. After removing the breaker from the panel, Maintenance personnel began installation of a breaker board panel cover to its normal position. While aligning the cover, it shifted downwards, opening two adjacent breakers. One of the adjacent breakers controlled a fuse column providing power to all four main feedwater regulation valves, which subsequently closed on loss of control power. A reactor trip was initiated by the reactor protection system on low-low steam generator level. The control room operators responded to the event in accordance with plant procedures. They diagnosed the plant condition, took actions to stabilize the unit, and maintained the unit in Hot Standby, Mode 3. The root cause of the inadvertent closure of the trip sensitive breakers was determined to be insensitivity to trip risks. The maintenance procedure has been revised to provide positive position control of panels when working on molded case breakers. Moreover, SQN has implemented a procedure that provides additional processes for review and control of site work activities concerning potential nuclear safety and generation risk above and beyond that specified in the TVA nuclear online and outage work management procedures.

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17. NARRATIVE *(If more space is required, use additional copies of NRC Form 366A)*

I. PLANT CONDITION(S)

Unit 2 was in Power Operation (Mode 1) at approximately 100% thermal power.

II. DESCRIPTION OF EVENT

A. Event:

On February 23, 2005, at approximately 2030 Eastern standard time (EST), Maintenance personnel were performing pre-refueling outage maintenance activities on a 125-Volt (V) direct current (dc) vital battery board [EIIS Code BYBD] molded case circuit breaker [EIIS Code BKR]. After removing the breaker from the panel, Maintenance personnel began installation of a battery board panel cover to its normal position. While aligning the cover to insert the holding bolts, the cover shifted downward, opening two adjacent breakers. One of the adjacent breakers controlled a fuse [EIIS Code FU] column providing power to all four main feedwater regulation valves [EIIS Code FCV], which subsequently closed on loss of control power. A reactor trip was initiated by the reactor protection system [EIIS Code JC] on low-low steam generator (SG) [EIIS Code SG] level. As designed, auxiliary feedwater (AFW) system [EIIS Code BA] was automatically initiated on SG low-low level.

B. Inoperable Structures, Components, or Systems that Contributed to the Event:

None.

C. Dates and Approximate Times of Major Occurrences:

February 23, 2005 at ~2030 EST	Maintenance personnel began physical breaker work activities in the dc vital battery board panel.
February 23, 2005 at ~2058 EST	Maintenance personnel began restoration of the dc vital battery board panel.
February 23, 2005 at ~2106 EST	Direct current vital battery board panel cover shifts downward opening two adjacent breakers, resulting in closure of the main feedwater regulation valves; subsequently, the unit received a low-low generator level signal initiating a reactor trip.

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17. NARRATIVE *(If more space is required, use additional copies of NRC Form 366A)*

D. Other Systems or Secondary Functions Affected:

None.

E. Method of Discovery:

The indication of an event began with an annunciation for high feedwater heater pressure [EIIS Code PI] on the main control panel. Shortly thereafter, multiple alarms were received, including low SG level in the main control room. Subsequently, a reactor trip was annunciated on the main control panel [EIIS Code CBD].

F. Operator Actions:

Control room operators responded to the event in accordance with plant procedures. The first alarm received was for high feedwater heater pressure, which operators responded to by placing the main feed pump master controller [EIIS Code SCO] in manual. Subsequently, multiple alarms were received including low SG level. Operators found the main feedwater regulation valves closed and non-responsive when placing them in manual control in response to the low SG alarm. Operators determined a manual reactor trip was necessary; however, an automatic reactor trip occurred as a result of low-low SG level before the manual reactor trip was carried out. Following the automatic reactor trip, operators took actions necessary to stabilize the unit, and maintained the unit in Hot Standby, Mode 3.

G. Safety System Responses:

The plant responded to the reactor trip as designed, including automatic start of all trains of the AFW system. Operators reduced the flow of AFW following the reactor trip to mitigate a decrease in reactor coolant system (RCS) [EIIS Code AB] average temperature. Operators noted the Rod Position Indication (RPI) System [EIIS Code BWR] indicated Shutdown Bank D Rod Cluster Control Assembly (RCCA) approximately 50 steps withdrawn during the reactor trip and later indicated zero steps. However, the RPI also indicated the RCCA fully inserted by the rod bottom indication light. A review of the integrated computer system [EIIS Code ID] verified the RCCA did not stop at approximately 50 steps withdrawn as indicated by the RPI system. This abnormality has been documented in the past as a result of static electricity or the meter [EIIS Code MTR] pointer rubbing on the meter scale of the RPI.

III. CAUSE OF THE EVENT

A. Immediate Cause:

The immediate cause of the unit trip was the initiation of an automatic reactor trip signal, on low-low level in the Loop 2 SG, when the main feedwater pumps were unable to provide feedwater flow to maintain SG level as a result of the closed feedwater regulation valves.

B. Root Cause:

The root cause of the inadvertent closure of the trip sensitive breakers was determined to be insensitivity to trip risks.

C. Contributing Factor:

Contributing to the event was a lack of risk sensitivity controls for determining at what plant operational mode this maintenance activity should be performed. Site barriers, including work week senior reactor operator review, critical evolutions committee review meeting and the maintenance foreman provided pre-job brief failed to understand the scope of the risk. The first of the two pre-job briefs had not addressed the risk to generation; however, once recognized in the second briefing, no escalation to management was taken nor was the activity stopped or mitigative action taken to reduce the risk. Also, placards warning of unit trip hazards were not in place to enhance sensitivity of the work environment and the work instruction did not contain cautions warning of unit trip hazards.

IV. ANALYSIS OF THE EVENT

This event is most similar to and bounded by the analyzed event of Loss of Normal Feedwater in the Final Safety Analysis Report, with all plant safety systems operating as designed during and following the reactor trip. Prior to the event, the plant was in Mode 1 with the following conditions: RCS pressure was approximately 2235 pound per square inch (psig) with an average temperature near program value of 578.2 degrees Fahrenheit (°F); pressurizer [EIIS Code PZR] level was on program at approximately 60 percent (%); secondary side steam pressure was approximately 845 psig with normal main feedwater supply and nominal full power steam flow; and SG levels were at 44% narrow range (NR). Following the automatic reactor trip, the RCS pressure increased to a maximum of 2285 psig upon loss of normal heat removal, subsequently reducing to a minimum of 2070 psig as a result of RCS temperature decrease and coolant volume shrinkage. The loss of nuclear heat generation and the introduction of cold AFW resulted in a rapid decrease in RCS temperature to below 540°F for approximately 4 minutes, and reached a minimum of 537°F. Operators took action to reduce RCS

cooldown, and before procedural steps to emergency borate were carried out, RCS temperature had increased above 540°F no longer requiring emergency boration. RCS temperature was restored to 550°F approximately 30 minutes after the reactor trip. Pressurizer level decreased to 22% and stabilized near 30% following the event. Steam pressure increased to approximately 1060 psig when the turbine stop valves closed. The atmospheric relief valves (ARVs) [EIIS Code RV] on SG Nos. 1, 2, & 4 opened as designed. The ARV on SG No. 3 did not open during the transient and was investigated and repaired under a work document. Steam generators' safety relief valves did not open and secondary side pressure limits were not exceeded. Additionally, steam dumps to the condenser [EIIS Code COND] operated as expected and remained available. AFW system actuated as designed on the SG low-low level signal, recovering the SG water levels following the reactor trip.

V. ASSESSMENT OF SAFETY CONSEQUENCES

Based on the above "Analysis of The Event," this event did not adversely affect the health and safety of plant personnel or the general public.

VI. CORRECTIVE ACTIONS

A. Immediate Corrective Actions:

Immediate action was taken to require the use of guide rods (i.e., positive position control) for work on similar breakers. Additionally, the maintenance procedure in use prior to the event has been revised to include the use of guide rods for all 120-V alternate current and 125-V dc boards. Pre-outage work was stopped until a procedure was implemented to provide an additional review of nuclear safety and generation risks against site work. Plant management reviewed pre-outage work for release to work until the additional procedure was implemented, following the plant event. Warning placards have been installed on the battery board room doors to enhance the level of sensitivity of the work environment. Additionally, the craftsmen involved in the maintenance activity have been restricted from generation risk maintenance activities until they demonstrated the correct work standards.

B. Corrective Actions to Prevent Recurrence:

SQN has implemented a procedure that provides additional barriers for review and control of site work activities to minimize and manage potential nuclear safety and generation risk above and beyond that specified in the TVA nuclear online and outage work management procedures. The procedure includes the following attributes: specific questioning of the work activity relative to it causing a plant transient; questions concerning reactivity management, containment integrity, and

electrical systems; contingency requirement if the proximity of other equipment poses a risk; and a graduated level of reviews up to a review by senior management.

The maintenance procedure for work on molded case breakers has been revised to include positive position control of all panel covers.

VII. ADDITIONAL INFORMATION

A. Failed Components:

None

B. Previous LERs on Similar Events:

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

C. Additional Information:

None

D. Safety System Functional Failure:

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

E. Loss of Normal Heat Removal Consideration

This event did result in a loss of normal heat removal because the closure of the main feedwater regulation valves precipitated the reactor trip.

VIII. COMMITMENTS

None.