Tennessee Valley Authority, Post Office Box 2000, Soddy-Daisy, Tennessee 37379-2000

December 15, 2011

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 Nos. 50-327

Subject: **Licensee Event Report 327/2011-004, "Reactor Trip as a Result of a Loss of a Preferred Inverter - Revision 1"**

Reference: Letter from TVA to NRC, "Revised Submittal Schedule for Supplemental Report for License Event Report 327/2011-004, 'Reactor Trip as a Result of a Loss of a Preferred Inverter,'" dated November 2, 2011

The purpose of this letter is to provide the enclosed Revision 1 to licensee event report (LER) 327/2011-004. The schedule for submitting this revision was discussed in the referenced letter. The Revision 1 LER includes supplemental information concerning an automatic reactor trip and automatic engineered safety feature actuation of auxiliary feedwater following the loss of the Unit 1 Preferred Inverter. Changes to the previous report are indicated by revision bars in the right side margin of the page.

The original LER was submitted in accordance with 10 CFR 50.73(a)(2)(iv)(A), as a condition that resulted in automatic actuation of the reactor protection system and the auxiliary feedwater system.

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

Respectfully,

John T. Carlin
Site Vice President
Sequoyah Nuclear Plant

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

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NRR

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NRC FORM 366 (10-2010) U.S. NUCLEAR REGULATORY COMMISSION

APPROVED BY OMB: NO. 3150-0104 EXPIRES: 10/31/20

Estimated burden per response to comply with this mandatory collection request: 80 hours. Reported lessons learned are incorporated into the licensing process and fed back to industry. Send comments regarding burden estimate to the FOIA/Privacy Section (T-5 F52), U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, or by internet e-mail to infocollects.resources@nrc.gov, and to the Desk Officer, Office of Information and Regulatory Affairs, NEOB-10202, (3150-0104), Office of Management and Budget, Washington, DC 20503. If a means used to impose an information collection does not display a currently valid OMB control number, the NRC may not conduct or sponsor, and a person is not required to respond to, the information collection.

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 Unit 1	05000327	1 OF 6

4. TITLE:
Reactor Trip as a Result of a Loss of Preferred Inverter - Revision 1

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
07	20	2011	2011	- 004 -	01	12	15	2011	FACILITY NAME	DOCKET NUMBER

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)
- ☐ 20.2201(d)
- ☐ 20.2203(a)(1)
- ☐ 20.2203(a)(2)(i)
- ☐ 20.2203(a)(2)(ii)
- ☐ 20.2203(a)(2)(iii)
- ☐ 20.2203(a)(2)(iv)
- ☐ 20.2203(a)(2)(v)
- ☐ 20.2203(a)(2)(vi)
- ☐ 20.2203(a)(3)(i)
- ☐ 20.2203(a)(3)(ii)
- ☐ 20.2203(a)(4)
- ☐ 50.36(c)(1)(i)(A)
- ☐ 50.36(c)(1)(ii)(A)
- ☐ 50.36(c)(2)
- ☐ 50.46(a)(3)(ii)
- ☐ 50.73(a)(2)(i)(A)
- ☐ 50.73(a)(2)(i)(B)
- ☐ 50.73(a)(2)(i)(C)
- ☐ 50.73(a)(2)(ii)(A)
- ☐ 50.73(a)(2)(ii)(B)
- ☐ 50.73(a)(2)(iii)
- ☒ 50.73(a)(2)(iv)(A)
- ☐ 50.73(a)(2)(v)(A)
- ☐ 50.73(a)(2)(v)(B)
- ☐ 50.73(a)(2)(v)(C)
- ☐ 50.73(a)(2)(v)(D)
- ☐ 50.73(a)(2)(vii)
- ☐ 50.73(a)(2)(viii)(A)
- ☐ 50.73(a)(2)(viii)(B)
- ☐ 50.73(a)(2)(ix)(A)
- ☐ 50.73(a)(2)(x)
- ☐ 73.71(a)(4)
- ☐ 73.71(a)(5)
- ☐ OTHER
  Specify in Abstract below or in NRC Form 366A

12. LICENSEE CONTACT FOR THIS LER

FACILITY NAME	TELEPHONE NUMBER *(Include Area Code)*
SQN - Norm Thomas	(423) 843-7749

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
A	EE	INVT	I235	Y					

14. SUPPLEMENTAL REPORT EXPECTED

☐ YES *(If yes, complete 15. EXPECTED SUBMISSION DATE)* ☒ NO

15. EXPECTED SUBMISSION DATE

MONTH	DAY	YEAR

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

On July 20, 2011, at approximately 2129 Daylight Saving Time (DST), Sequoyah Nuclear Plant (SQN) Unit 1 automatically tripped as a result of loss of the Unit 1 Preferred Inverter. The loss of the preferred inverter caused the turbine governor valves to close, and the load rejection caused a Power Range Neutron Flux High Negative Rate Reactor Trip signal.

At the time of the reactor trip, plant personnel were installing test equipment inside the preferred inverter to monitor frequency. A ground was introduced on the preferred inverter output when the frequency check was performed, which caused the preferred inverter output to be lost. The direct cause of the event was that the acting Maintenance foreman incorrectly decided to perform missed steps in a preventative maintenance procedure. This resulted in the preferred inverter frequency check steps being performed out of sequence from the sequence in the test procedure. The root cause of this event was that Operations has not sustained written guidance for supervisory review of applicable work documents prior to returning equipment to service. Corrective actions will include establishing procedural guidance for Operations supervisory review of applicable work documents prior to returning equipment to service.

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NARRATIVE

I. PLANT CONDITION(S)

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

II. DESCRIPTION OF EVENT

A. Event:

On July 20, 2011, at approximately 2129 Daylight Saving Time (DST), SQN Unit 1 automatically tripped as a result of loss of the Unit 1 non-safety related Preferred Inverter [EIIS Code EE] and its associated Unit 1 non-safety related Preferred Power Board [EIIS Code EE]. The loss of the Preferred Inverter was caused by introduction of a ground when Electrical Maintenance personnel improperly performed a frequency check while the inverter was connected to the Unit 1 Preferred Power Board.

Just prior to the event, Operations had returned the Unit 1 Preferred Inverter to service following maintenance. The Unit 1 Preferred Power Board was then aligned to the inverter in accordance with system operating instructions. Electrical Maintenance personnel later discovered that the steps to perform a frequency check of the Preferred Inverter were not completed in the work order. The missed frequency check was not discussed with maintenance supervision. Electrical Maintenance notified Operations that the Preferred Inverter work order was not complete, and requested to perform the frequency check. The review of the work order was not adequate in that neither the involved Operations nor Electrical Maintenance personnel understood that the work order required the Preferred Inverter to be unloaded for the frequency check. In addition, the test instrument used to measure frequency was grounded, while the instrument normally used to perform this function is not grounded. When the test instrument was placed on the inverter output to measure frequency, a spark was noticed and a fuse opened in the inverter, which removed the output from the inverter and subsequently de-energized the Unit 1 Preferred Power Board. The transient voltage condition resulted in closure of the turbine governor valves [EIIS Code TA]. The loss of turbine load caused the rod control system to insert rods and reduce reactor power until a Power Range Neutron Flux High Negative Rate Reactor Trip signal was received.

During the transient, reactor coolant system (RCS) [EIIS Code AB] pressure increased to where the pressurizer power operated relief valves (PORVs) [EIIS Code AB] lifted. The PORVs subsequently reseated and RCS pressure recovered. Following the reactor trip, the auxiliary feedwater system [EIIS Code BA] automatically actuated as expected on loss of the main feedwater pumps. The main feedwater pumps were available for recovery using approved plant procedures following the reactor trip. The auxiliary feedwater and steam dump [EIIS Code SG]

systems operated as expected to remove decay heat and stabilize the RCS at the no-load value of 547 degrees Fahrenheit. Operations restored power to the Unit 1 Preferred Power Board by aligning the alternate power supply in accordance with Abnormal Operating Procedure (AOP)-P.09, "Loss of 120V AC Preferred Power."

The Tennessee Valley Authority originally submitted this report in accordance with 10 CFR 50.73(a)(2)(iv)(A), as a condition that resulted in automatic actuation of the reactor protection system and the auxiliary feedwater system.

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

None.

C. Dates and Approximate Times of Major Occurrences:

Date	Description
July 20, 2011 at approximately 20:49 DST	Operations returned the Unit 1 Preferred Inverter to service following maintenance. The Unit 1 Preferred Power Board is aligned to the Preferred Inverter in accordance with system operating instructions.
July 20, 2011 at approximately 21:00 DST	Electrical Maintenance personnel notifies Operations that the Preferred Inverter work order was not complete, and requests to perform the frequency check. The review of the work order was not adequate in that neither Operations nor Electrical Maintenance personnel determined that the work order required the Preferred Inverter to be unloaded for the frequency check.
July 20, 2011 at 21:28 DST	Performance of the frequency check at the Preferred Inverter results in de-energization of the Preferred Power Board.
July 20, 2011 at 21:28:28 DST	Loss of turbine load results in a Unit 1 automatic reactor trip from the Power Range Neutron Flux High Negative Rate Trip signal.
July 20, 2011 at 21:52:52 DST	Operations restores power to the Unit 1 Preferred Power Board in accordance with AOP-P.09, Loss of 120V AC Preferred Power.

D. Other Systems or Secondary Functions Affected:

Following the reactor trip, Operations noted that there were non-safety related indication lights not lit on panels in the main control room. These panels were

associated with the heater, drains, and vents system [EIIS Code SM] and the essential raw cooling water system [EIIS Code BI]. Also a pressure indicator for the gland seal steam system [EIIS Code TC] had failed downscale. Control fuses for these indications were found to be blown. The fuses were replaced and the indications were restored. Operations noticed that Steam Generator number 4 Atmospheric Relief Valve [EIIS Code SB] handswitch was indicating full open while the controller output was at zero. Operations closed the valve and initiated a work order to correct the condition.

E. Method of Discovery:

Control room alarms alerted operators to the start of the event.

F. Operator Actions:

Operations responded to the reactor trip by performing actions in accordance with Emergency Procedure E-0, "Reactor Trip or Safety Injection," and Emergency Subprocedure ES-0.1, "Reactor Trip Response." ES-0.1 is a subprocedure of procedure E-0. Operations performed AOP-P.09, "Loss of 120 V AC Preferred Power," to restore power to the Unit 1 Preferred Power Board. The operations crew responded to the event as expected.

G. Safety System Responses:

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

III. CAUSE OF THE EVENT

A. Immediate Cause:

The immediate cause of the reactor trip was the loss of the Unit 1 Preferred Inverter which resulted in a Power Range Neutron Flux High Negative Rate reactor trip.

B. Root Cause:

The direct cause of the event was that the acting Maintenance foreman incorrectly decided to perform missed steps in a preventative maintenance procedure. The root cause of this event was that Operations has not sustained written guidance for supervisory review of applicable work documents prior to returning equipment to service.

During two different periods in 2009 and 2010, Operations standing orders were used to establish an Operations supervisory work package review as a barrier to prevent inadequate work practices from impacting plant and equipment operation.

However, these standing orders were retired without sustaining the expectation in subsequent written guidance. As a result, in both cases, as well as this event, the requirement for Operations supervisory review of work packages prior to returning equipment to service was not maintained. Had this barrier been intact for this event, the operations reviewer of the work package would have identified missed procedure steps and not proceeded forward to re-align the Preferred Inverter and set equipment conditions for the event.

IV. ANALYSIS OF THE EVENT

Prior to the event, SQN Unit 1 was operating in Mode 1 at approximately 100 percent power. During the transient, RCS pressure increased to approximately 2344 psig, which is above the pressurizer PORV setpoint of 2335 psig. Both PORVs opened briefly, one for approximately 3 seconds, the other for approximately 7 seconds. The pressurizer safety valves did not actuate. PORV 1-PCV-334 reclosed close to the proper pressure of 2315 psig. The plant computer indicated that PORV 1-PCV-340A reclosed at approximately 2200 psig. The difference between PORV 1-PCV-340A reclosure pressure and expected reclosure pressure is attributed to normal setpoint drift, normal stroke time tolerance, computer scan rate, and the fact that the RCS Master Pressure Controller controls 1-PCV-68-340A with some time delay. Following the reactor trip, RCS pressure decreased due to the decreasing RCS temperature and the associated shrinking of coolant volume. The minimum RCS pressure was approximately 1997 psig, well above the pressure that would have initiated a safety injection signal (1870 psig).

Because of the turbine load reduction, RCS average temperature increased to approximately 581 degrees Fahrenheit (F), which is below the Technical Specification 3.2.5, "DNB Parameters," limit of 583 degrees F. Following the reactor trip, the loss of nuclear heat generation resulted in a rapid decrease in RCS temperature to approximately 545 degrees F. RCS temperature then increased to its no-load value of 547 degrees F as secondary side pressure recovered. The plant responded as expected for the conditions of the trip. No Technical Specification limits were exceeded and the Updated Final Safety Analysis Report analysis of the event remained bounding.

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:

Control room personnel responded to the reactor trip as prescribed by emergency procedures. The Unit 1 Preferred Power Board was re-energized from its alternate power supply in accordance with the abnormal operating procedure.

B. Corrective Actions to Prevent Recurrence:

The corrective actions are being managed through the SQN Corrective Action Program.

The corrective action to prevent recurrence will include establishing procedural guidance for Operations supervisory review of applicable work documents prior to returning equipment to service.

VII. ADDITIONAL INFORMATION

A. Failed Components:

The failed component was a model RU12.5K6631-HL-AB-MB Instrument AC Power Supply Inverter, made by International Computer Power.

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. Unplanned Scram with Complications:

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

VIII. COMMITMENTS

None.