S64 080304 800

March 4, 2008

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

10 CFR 50.73

Gentlemen:

TENNESSEE VALLEY AUTHORITY - SEQUOYAH NUCLEAR PLANT (SQN) UNIT 1 - DOCKET NO. 50-327 - FACILITY OPERATING LICENSE DPR-77 - LICENSEE EVENT REPORT (LER) 50-327/2008-001-00

The enclosed LER provides details concerning a manual reactor trip and engineered safety feature (ESF) actuation of auxiliary feedwater. The trip was initiated because of a low and decreasing steam generator level when a feedwater regulating valve closed as a result of maintenance personnel failing to follow procedures. This report is being submitted in accordance with 10 CFR 50.73(a)(2)(iv) as an event that resulted in the automatic actuation of engineered safety features, including the reactor protection system.

Sincerely,

Original signed by:

James D. Smith
Manager, Site Licensing and
Industry Affairs

Enclosure
cc: See page 2

Enclosure
cc (Enclosure):

INPO Records Center
Institute of Nuclear Power Operations
700 Galleria Parkway, SE, Suite 100
Atlanta, Georgia 30339-5957

Mr. Thomas H. Boyce, Senior Project Manager
U.S. Nuclear Regulatory Commission
Mail Stop 08G-9a
One White Flint North
11555 Rockville Pike
Rockville, Maryland 20852-2739

NRC Resident Inspector
Sequoyah Nuclear Plant
2600 Igou Ferry Road
Soddy-Daisy, Tennessee 37379

JDS:RRT:JWP:KTS
Enclosure
cc (Enclosure):

G. Arent, EQB 1B-WBN
P. M. Billingsley, OPS-4C-SQN (For Web posting)
W. R. Campbell, Jr., LP 3R-C
C. R. Church, POB 2B-SQN
J. E. Cipriani, OPS 4E-SQN
T. P. Cleary, OPS 4A-SQN
L. D. Clift, OPS 4G-SQN (Attn: R. N. McKaig)
K. R. Jones, OPS 4A-SQN
L. E. Nicholson, BR 4X-C
NSRB Support, BR 4X-C
M. R. Purcell, BR 4X-C
H. R. Rogers, LP 3R-C
S. A. Vance, ET 10A-K
E. J. Vigluicci, ET 10A-K
T. D. Wallace, STC 2A-SQN
WBN Site Licensing Files, ADM 1L-WBN
EDMS, WT CA-K

I:license/lers/2008/LER 1-2008-001.doc

NRC FORM 366 (9-2007) U.S. NUCLEAR REGULATORY COMMISSION	APPROVED BY OMB NO. 3150-0104 EXPIRES 08/31/2010
LICENSEE EVENT REPORT (LER) (See reverse for required number of digits/characters for each block)	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 Records and FOIA/Privacy Service Branch (T-5 F52), U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, or by internet e-mail to infocollects@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.

1. FACILITY NAME	2. DOCKET NUMBER	3. PAGE
Sequoyah Nuclear Plant Unit 1	05000327	1 of 5

4. TITLE:
Manual Reactor Trip Resulting From A Failure to Follow Procedures

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
01	16	2008	2008	- 001 -	00	03	04	2008	FACILITY NAME	DOCKET NUMBER

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)	☒ 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)
100	☐ 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

NAME	TELEPHONE NUMBER *(Include Area Code)*
Rusty Proffitt, Licensing Engineer	423-843-6651

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 January 16, 2008, at 1925 Eastern standard time, the Unit 1 reactor was manually tripped as a result of a low steam generator level on Loop 3. Maintenance personnel were performing a transmitter calibration on a Loop 3 steam generator pressure transmitter. During performance of the calibration, Maintenance personnel made an incorrect assumption that a place-keeping aid placed in the procedure was intended to be the procedure start point when in fact they should have started two pages previous to the place-keeping aid location. The two missed pages resulted in Operations not being notified to remove the channel being tested as a controlling channel. This resulted in performing testing on the pressure transmitter on a controlling channel, causing closure of the Loop 3 main feedwater regulating valve. Operations placed the feedwater regulating valve in manual to open the valve. However, before steam generator water level began to increase, the pre-established low steam generator level trigger-point was reached and the operating crew initiated a manual reactor trip. The cause was determined to be a failure to follow procedure because of personnel not performing proper place keeping during performance of a calibration procedure. Corrective actions include establishing and proceduralizing standard place-keeping requirements, strengthening and proceduralizing standard pre-job brief requirements, and training on these standard requirements.

LICENSEE EVENT REPORT (LER)

1. FACILITY NAME	2. DOCKET	6. LER NUMBER			3. PAGE
		YEAR	SEQUENTIAL NUMBER	REVISION	
Sequoyah Nuclear Plant (SQN) Unit 1	05000327	2008 --	001 --	00	2 OF 5

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

I. PLANT CONDITION(S)

Unit 1 was operating at 100 percent power when the reactor trip was initiated.

II. DESCRIPTION OF EVENT

A. Event:

On January 16, 2008, at 1925 Eastern standard time (EST), the Unit 1 reactor was manually tripped as a result of a low steam generator (EIIS code AB) level on Loop 3. Maintenance personnel were performing a transmitter calibration on a Loop 3 steam generator pressure transmitter. During performance of the calibration, Maintenance personnel made an incorrect assumption that a place-keeping aid placed in the procedure was intended to be the procedure start point when in fact they should have started two pages previous to the place-keeping aid location. The two missed pages resulted in Operations not being notified to remove the channel being tested as a controlling channel. This resulted in performing testing on the pressure transmitter on a controlling channel, causing closure of the Loop 3 main feedwater regulating valve (EIIS code SJ). Operations placed the feedwater regulating valve in manual to open the valve. However, before steam generator water level began to increase, the pre-established low steam generator level trigger-point was reached and the operating crew initiated a manual reactor trip.

Following the reactor trip, the plant systems responded as designed. The rods fully inserted as required. Immediate operator actions were performed and auxiliary feedwater (AFW) flows were adjusted to control the reactor coolant system (RCS) cooldown. The unit entered Mode 3 and an event investigation was initiated.

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

None.

C. Dates and Approximate Times of Major Occurrences:

January 16, 2008 at 1649 EST	Operations authorized performance of a calibration on steam generator Loop 3 pressure transmitter.
January 16, 2008 at 1734 EST	Operations enter LCO 3.3.2 action 17 and LCO 3.3.3.7 action 1 for performance of the calibration on steam generator Loop 3 pressure transmitter.
January 16, 2008 at 1925 EST	Operations initiated a manual reactor trip as a result of a low steam generator level on Loop 3.

D. Other Systems or Secondary Functions Affected:

No other systems or secondary functions were affected by this event.

E. Method of Discovery:

The steam generator low-level alarm was annunciated in the Main Control Room.

F. Operator Actions:

Operations placed the feedwater regulating valve in manual to open the valve. However, before steam generator water level began to increase, the pre-established low steam generator level trigger-point was reached and the operating crew initiated a manual reactor trip. After the reactor trip, the operating crew took actions necessary to stabilize the unit in a safe condition and maintained the unit in hot standby, Mode 3.

G. Safety System Responses:

The plant responded to the reactor trip as designed.

III. CAUSE OF THE EVENT

A. Immediate Cause:

The immediate cause of the event was closure of the Loop 3 main feedwater regulating valve as a result of an error during performance of a calibration procedure. The closure of the feedwater regulating valve resulted in a low steam generator level.

B. Root Cause:

The root cause was determined to be a failure to follow procedure because of personnel not performing proper place keeping during performance of a calibration procedure.

C. Contributing Factor:

Contributing to the event was an inadequate pre-job brief by Maintenance and Operations personnel and Operations allowed Maintenance to perform work on a controlling channel.

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

IV. ANALYSIS OF THE EVENT

The plant systems responded to the reactor trip as designed. The reactor coolant system (RCS) average temperature was near 578.2 degrees F prior to the loss of main feedwater to Loop 3 steam generator. When feedwater was lost, RCS average temperature made a slight increase before the reactor trip. Following the reactor trip, the loss of nuclear heat generation resulted in a rapid decrease in RCS average temperature to 548 degrees F. As heat removal in the steam generators decreased as a result of increased steam pressure, the decrease in RCS temperature slowed. The introduction of cold auxiliary feedwater (AFW) resulted in a slower, but continued reduction in RCS temperature until AFW flow was reduced about 5 minutes after the reactor trip. RCS temperature then started to increase. The minimal RCS temperature was 543 degrees F. RCS temperature remained within Technical Specification limits and bounded by the Safety Analysis Report (SAR) analysis.

The plant responded as expected for the conditions of the trip. No Technical Specification limits were exceeded and the SAR analysis of this 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 as prescribed by emergency procedures. They diagnosed the plant condition and took action necessary to stabilize the unit in a safe condition.

B. Corrective Actions to Prevent Recurrence:

Corrective actions include establishing and proceduralizing standard place-keeping requirements, strengthening and proceduralizing standard pre-job brief requirements, and training on these standard requirements.

LICENSEE EVENT REPORT (LER)

1. FACILITY NAME	2. DOCKET	6. LER NUMBER			3. PAGE
		YEAR	SEQUENTIAL NUMBER	REVISION	
Sequoyah Nuclear Plant (SQN) Unit 1	05000327	2008 --	001 --	00	5 OF 5

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

VII. ADDITIONAL INFORMATION

A. Failed Components:

None.

B. Previous LERs on Similar Events:

A review of previous reportable events 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 condition did not result in an unplanned scram with complications.

VIII. COMMITMENTS

None.