TVA

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

February 26, 2003

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

Gentlemen:

TENNESSEE VALLEY AUTHORITY - SEQUOYAH NUCLEAR PLANT UNIT 2 - DOCKET NO. 50-328 - FACILITY OPERATING LICENSE DPR-79 - LICENSEE EVENT REPORT (LER) 50-328/2003-001-00

The enclosed report provides details concerning a reactor trip signal. 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.

This letter is being sent in accordance with NRC RIS 2001-05.

Sincerely,

R.T. Purcell

Richard T. Purcell

Enclosure
cc (Enclosure):
INPO Records Center
Institute of Nuclear Power Operations
700 Galleria Parkway
Atlanta, Georgia 30339-5957

Mr. Raj K. Anand, Senior Project Manager
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NRC FORM 366 (7-2001) U.S. NUCLEAR REGULATORY COMMISSION

LICENSEE EVENT REPORT (LER)

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

APPROVED BY OMB NO. 3150-0104 EXPIRES 7-31-2004

1. FACILITY NAME	2. DOCKET NUMBER	3. PAGE
Sequoyah Nuclear Plant (SQN) UNIT 2	05000328	1 OF 5

4. TITLE
Reactor Trip Signal as a Result of a Low-Low Steam Generator Level

5. EVENT DATE MO	DAY	YEAR	6. LER NUMBER YEAR	SEQUENTIAL NUMBER	REV NO	7. REPORT DATE MO	DAY	YEAR	8. OTHER FACILITIES INVOLVED FACILITY NAME	DOCKET NUMBER
										05000
01	01	2003	2003	– 001 –	00	02	26	2003	FACILITY NAME	DOCKET NUMBER 05000

9. OPERATING MODE	5
10. POWER LEVEL	000

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

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

12. LICENSEE CONTACT FOR THIS LER

NAME	TELEPHONE NUMBER (Include Area Code)
Jan Bajraszewski	(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

14. SUPPLEMENTAL REPORT EXPECTED			15. EXPECTED SUBMISSION DATE	MONTH	DAY	YEAR
YES (If yes, complete EXPECTED SUBMISSION DATE)	X	NO				

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

On January 1, 2003, at 0626 Eastern Standard Time, a steam generator Loop No. 2 low-low level initiated a reactor trip signal. The reactor trip signal did not initiate any pump or valve operation because the unit was already in Mode 5, cold shutdown, and related equipment had been secured or was already in the required position. Before the event, it was determined that the secondary side of the steam generators needed to be drained to support chemistry. In accordance with plant procedures, instrument maintenance technicians were dispatched to bypass the steam generator level reactor trip set points of the reactor protection system. However, the technicians inadvertently bypassed the steam generator low level alarms rather than the low-low level trips. As a result, a trip signal was generated upon draining the steam generator. The cause of the event was an inadequate procedure; the information contained in the procedure was too generic. Contributing to the event, supervisors incorrectly selected which portions of the procedure was to be performed and did not verify the information. Subsequent to the event, the correct steam generator level trip set points were bypassed and work was completed without incident. The involved individuals were counseled and the event was documented in the corrective action program where corrective actions were developed to provide more detailed procedural information.

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Sequoyah Nuclear Plant (SQN) Unit 2	05000328	2003 --	001 --	00	2 OF 5

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

I. PLANT CONDITION(S)

Unit 2 was in Mode 5, cold shutdown.

II. DESCRIPTION OF EVENT

A. Event:

On January 1, 2003, at 0626 Eastern Standard Time, a steam generator (SG) [EIIS Code SB] Loop No. 2 low-low level initiated a reactor trip signal. The reactor trip signal did not initiate any pump or valve operation because the unit was already in Mode 5, cold shutdown, and related equipment had been secured or was already in the required position. Before the event, it was determined that the secondary side of the SGs needed to be drained to support chemistry. In accordance with plant procedures, instrument maintenance technicians were dispatched to bypass the steam generator level reactor trip set points of the reactor protection system [EIIS Code SJ]. However, the technicians inadvertently bypassed the SG low level alarms rather than the low-low level trips. As a result, a trip signal was generated upon draining the SG.

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

None.

C. Dates and Approximate Times of Major Occurrences:

December 31, 2002, at 2200 EST	Operations requests maintenance personnel to ensure the high-high and low-low level functions are bypassed for each of the four SGs.
December 31, 2002, at 2245 EST	Maintenance personnel notify Operations that SG level trips are in bypass.
January 1, 2003, at 0513 EST	Operations begins to drain the Loop 2 SG.
January 1, 2003, at 0626 EST	A reactor trip signal was annunciated in the main control room.

D. Other Systems or Secondary Functions Affected:

None.

E. Method of Discovery:

The reactor trip signal annunciated on the main control room panels.

F. Operator Actions:

No operator action was required for the trip signal, control room operators promptly diagnosed the condition as being related to draining of the SG.

G. Safety System Responses:

The reactor trip signal did not initiate any pump or valve operation because the unit was already in Mode 5, cold shutdown, and related equipment had been secured or was already in the required position.

III. CAUSE OF THE EVENT

A. Immediate Cause:

The immediate cause of the event was the draining of the Loop 2 SG. Once the water level passed SG level set points, the reactor trip signal was generated.

B. Root Cause:

The root cause of the event was inadequate procedure; the information contained in the procedure was too generic. The procedure did not contain noun descriptions of the set point functions and the layout of the procedure suggested that SG trip set points were grouped together for each SG where actually the low-low level SG trip was separate from the other trip set points. This led to misinterpretation by the performers.

C. Contributing Factor:

Contributing to the event was that the supervisors incorrectly selected portions of the procedure for performance and did not verify and validate the selection. Additionally, inattention to detail and unawareness led the performers to miss opportunities to identify the error during procedure performance.

IV. ANALYSIS OF THE EVENT

The reactor protection system automatically initiates a reactor trip when any monitored variable or combination of variables exceeds a predetermined setpoint value. Set points

are provided to envelope safe operating conditions with adequate margin for uncertainties to ensure that fuel design limits are not exceeded. The basic function of the reactor protection

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

system, associated with the steam generator, is the low-low level setpoint. This function is associated with the water level necessary to preserve the steam generator heat sink for removal of long term residual heat from the reactor coolant system, including the fuel rods. Should a complete loss of feedwater occur, the reactor would be tripped on low-low steam generator narrow range water level and the auxiliary feedwater pumps would start. This reactor trip acts before the steam generators are dry to reduce the required capacity and starting time requirements of auxiliary feedwater pumps and to minimize thermal transients on the reactor coolant system and steam generators. The reactor protection system and the low-low steam generator level trip functioned as described in the Final Safety Analysis Report.

V. ASSESSMENT OF SAFETY CONSEQUENCES

At the time of the event, the unit was in Mode 5 (cold shutdown) operation with the auxiliary feedwater pumps secured. Additionally, the low-low steam generator trip variable of the reactor protection system was not required for that mode of operation. Therefore, based on plant conditions and 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:

The correct SG level set points were bypassed and work was completed without incident.

B. Corrective Actions to Prevent Recurrence:

The event was documented in the corrective action program and an action is in place for development of a procedure revision to add more detailed information. A review was performed of other similar instrument maintenance procedures for layout and description content problems. The review did not identify any other procedures with layout or description content problems.

The involved individuals were counseled on the use of error reduction tools and the requirements for following procedures. Briefings were conducted with appropriate instrument maintenance personnel to communicate event lessons learned.

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

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 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).

VIII. COMMITMENTS

None.