April 12,2002

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

Gentlemen:

TENNESSEE VALLEY AUTHORITY - SEQUOYAH NUCLEAR PLANT (SQN) UNITS 1 AND 2 - DOCKET NOS. 50-327 AND 50-328- FACILITY OPERATING LICENSES DPR-77 AND 79 - LICENSEE EVENT REPORT (LER) 50-327/2002001

The enclosed report provides details concerning nonconservative low-low steam generator level setpoints. This event is being reported, in accordance with 10 CFR 50.73(a)(2)(i)(B), as an operation prohibited by technical specifications and 10 CFR 50.73(a)(2)(v)(A), as a condition that could have prevented the fulfillment of a safety function. This letter is being sent in accordance with NRC RIS 2001-05.

Sincerely,

Original signed by

Richard T. Purcell

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

NRC FORM 366 (7-2001)	U.S. NUCLEAR REGULATORY COMMISSION	APPROVED BY OMB NO. 3150-0104 EXPIRES 7-31-2004
LICENSEE EVENT REPORT (LER) (See reverse for required number of digits/characters for each block)		Estimated burden per response to comply with this mandatory information collection request: 50 hours. Reported lessons learned are incorporated into the licensing process and fed back to industry. Send comments regarding burden estimate to the Records Management Branch (T-6 E6), U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, or by internet e-mail to bjs1@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 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 (SQN) UNIT 1	05000327	1 OF 07

4. TITLE
Westinghouse Electrical Corporation Error Results in Nonconservative Steam Generator Level Setpoint

5. EVENT DATE			6. LER NUMBER			7. REPORT DATE			8. OTHER FACILITIES INVOLVED	
MO	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REV NO	MO	DAY	YEAR	FACILITY NAME	DOCKET NUMBER
									SQN 2	05000328
02	15	2002	2002	– 001 –	00	04	12	2002	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)(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)		50.73(a)(2)(iv)(A)		73.71(a)(4)
	20.2203(a)(2)(i)		50.36(c)(1)(ii)(A)	X	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)	X	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)
James W. Proffitt	(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 EXPECTED SUBMISSION DATE)		NO				

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

On February 15, 2002, Sequoyah performed an assessment of the narrow range steam generator level measurement instrument channels, based on a Nuclear Safety Advisory Letter from Westinghouse Electric Corporation. This assessment determined that the demonstrated accuracy calculation for low-low level trip setpoint narrow range span did not account for the measurement bias associated with the differential pressure (dP) created by the steam flow past the mid-deck plate in the moisture separator section of the steam generators. This dP phenomena could cause the steam generator narrow range level channels to read higher than actual water level at high steam flows. This could cause the low steam generator level trip setpoint to be nonconservative. The cause of the condition was the failure of Westinghouse personnel to account for the measurement bias, when establishing the low-low steam generator level setpoint calculation. TVA will modify the steam generator setpoints, as applicable, to account for the identified condition.

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

I. PLANT CONDITION(S)

Units 1 and 2 were in power operation at approximately 100 percent reactor power.

II. DESCRIPTION OF EVENT

A. Event:

On February 15, 2002, Sequoyah performed an assessment of the narrow range steam generator level measurement instrument channels (EIIS Code AB) based on a Nuclear Safety Advisory Letter from Westinghouse Electric Corporation. This assessment determined that the demonstrated accuracy calculation for low-low level trip setpoint narrow range span did not account for the measurement bias associated with the differential pressure (dP) created by the steam flow past the mid-deck plate in the moisture separator section of the steam generators. This dP phenomena could cause the steam generator narrow range level channels to read higher than actual water level at high steam flows. This could cause the low steam generator level trip setpoint to be nonconservative.

Sequoyah has an environmental allowance modifier feature that changes the low-low level steam generator setpoint. With this feature actuated the low-low setpoints are conservative and allow continued operation. As a conservative measure, the environmental allowance modifier will remain actuated until the steam generator setpoints are evaluated and modified, if applicable.

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

None.

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

C. Dates and Approximate Times of Major Occurrences:

February 9, 2002	Diablo Canyon issues an operating experience item resulting from the failure of the reactor to trip on a low-low steam generator level.
February 15, 2002	Westinghouse issued Nuclear Safety Advisory Letter (NASL) 02-3 concerning the steam generator water level setpoint analysis.
February 15, 2002	Sequoyah determined that the demonstrated accuracy calculation did not account for the measurement bias associated with the differential pressure created by the steam flow past the mid-deck plate in the moisture separator section of the steam generators.
February 15, 2002 at 1811 Eastern Standard Time (EST)	As a conservative measure, Operations personnel entered limiting condition for operation (LCO) 3.0.3 for potentially nonconservative steam generator low-low level setpoint on Unit 2. The environmental allowance modifier was determined to be actuated on Unit 1 since Unit 1 was venting at the time.
February 15, 2002 at 1814 EST.	The environmental allowance modifier was actuated on Unit 2 and Operations personnel exited LCO 3.0.3.
February 16, 2002	Plant procedures were revised to keep the environmental allowance modifier actuated, until the steam generator setpoints are evaluated and modified, if applicable.

D. Other Systems or Secondary Functions Affected:

None.

E. Method of Discovery:

Engineering review of the Westinghouse Nuclear Safety Advisory Letter on steam generator level concluded that insufficient margin existed in the present setpoint analysis to overcome the error in the Westinghouse calculation.

F. Operator Actions:

Operations personnel entered LCO 3.0.3 for Unit 2 and then actuated environmental allowance modifier.

G. Safety System Responses:

Not applicable - no safety system response was required.

III. CAUSE OF THE EVENT

A. Immediate Cause:

The immediate cause of this condition was the error in the Westinghouse low-low level setpoint calculation potentially resulting in the Sequoyah setpoint being nonconservative.

B. Root Cause:

The root cause of the condition was the failure of Westinghouse personnel to account for the measurement bias associated with the differential pressure created by the steam flow past the mid-deck plate in the moisture separator section of the steam generators, when establishing the low-low steam generator level setpoint calculation.

C. Contributing Factor:

None.

IV. ANALYSIS OF THE EVENT

The Sequoyah FSAR accident and transient analyses credit reactor trip on low SG level for the loss of normal feedwater (LONF), loss of off-site power (LOOP), and feedwater line break transients. Since the dP phenomena does not exist for the feedwater line break transient, this analysis is not affected. In the unlikely event of a loss of main feedwater or a loss of offsite power, the potential exists that the required reactor trip on low steam generator level may be delayed or not be received. In this scenario, alternate reactor trip signals actuating on over-temperature differential temperature, high pressurizer pressure or high pressurizer level would most likely be generated prior to the loss of nucleate boiling and actuation of the pressurizer safety relief valves. However, these reactor trip signals would arrive at a different point in the transient than currently analyzed and the net effect of the delay on the UFSAR Chapter 15 accident analysis results is unknown. Thus, a condition exists in which a credited safety function could have failed to operate.

V. ASSESSMENT OF SAFETY CONSEQUENCES

The function of the reactor protection circuits associated with low-low steam generator water level and low feedwater flow is to preserve the steam generator heat sink for removal of long term residual heat. Reactor trips on RCS temperature and pressurizer pressure will trip the unit before there is any damage to the core or the reactor coolant system. For the LONF or LOOP event, the consequences are bounded by a small break loss of coolant accident (LOCA) and therefore, the effect of the mid-deck pressure loss on this event does not represent a substantial safety hazard.

Concerning component integrity, each steam generator is analyzed for one occurrence of steam generator dryout.

For the ATWS Mitigation System Actuation Circuitry (AMSAC) system, an increase in steam generator level uncertainty could impact the AMSAC actuation setpoint and consequently, the operation of AMSAC.

The AMSAC actuation setpoint is not directly assumed in any safety analysis. AMSAC is set to operate upon reaching a low level setting of 8 percent of SG Narrow Range. The additional error added to the loop error results in a total instrument channel uncertainty of 7.6 percent of span. Even for the worst case, the AMSAC would still actuate before reaching the safety limit of 0 percent of span. The effect of this additional bias would be a slight delay in AMSAC actuation.

Steam line breaks (SLBs) outside containment producing mass and energy releases assume a reactor trip on a low-low steam generator water level as a convenience trip if no other trip signals are received. The effect of any decrease in the water level reactor trip setpoint (if greater than 0 percent of the narrow range span, and not bounded by the increase in the calculated uncertainty) is bounded by the very conservative assumptions in the analysis related to main feedwater flow and steam generator tube uncovery. The calculated SLB mass and energy releases outside containment remain conservative.

Additionally, Engineering personnel reviewed previous plant trips on low-low steam generator level, and no observable bias was found to exist.

VI. CORRECTIVE ACTIONS

A. <u>Immediate Corrective Actions</u>:

As a conservative measure, the environmental allowance modifiers were actuated. Plant procedures were revised to keep the environmental allowance modifier actuated, until the steam generator setpoints are modified, if applicable.

B. <u>Corrective Actions to Prevent Recurrence</u>:

TVA will modify the steam generator setpoints, if applicable, to account for the identified condition.

LICENSEE EVENT REPORT (LER)

FACILITY NAME (1)	DOCKET	LER NUMBER (6)			PAGE (3)
		YEAR	SEQUENTIAL NUMBER	REVISION	
Sequoyah Nuclear Plant (SQN) Unit 1	05000327	2002 --	001 --	00	7 OF 7

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 previous events.

C. Additional Information:

None

D. Safety System Functional Failure:

This condition is considered to be a safety system functional failure in accordance with NEI 99-02, because of the low-low steam generator setpoints being nonconservative.

VIII. COMMITMENTS

None.