NRC Form 366 (9-83)

U.S. NUCLEAR REGULATORY COMMISSION
APPROVED OMB NO. 3150-0104
EXPIRES: 8/31/88

LICENSEE EVENT REPORT (LER)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
Sequoyah, Unit 1	0 5 0 0 0 3 2 7	1 OF 0 4

TITLE (4) An Inadequate Procedure Caused Inaccurate Primary To Secondary Leak Rates To Be Measured Resulting In A Potential Noncompliance With A Limiting Condition for Operation

EVENT DATE (5)			LER NUMBER (6)			REPORT DATE (7)			OTHER FACILITIES INVOLVED (8)	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	MONTH	DAY	YEAR	FACILITY NAMES	DOCKET NUMBER(S)
0 2	0 4	8 8	8 8	— 0 0 9 —	0 0	0 2	2 6	8 8	Sequoyah, Unit 2	0 5 0 0 0 3 2 8
										0 5 0 0 0

OPERATING MODE (9): 5

POWER LEVEL (10): 0 0 0

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more of the following) (11)

[ ] 20.402(b)	[ ] 20.405(c)	[ ] 50.73(a)(2)(iv)	[ ] 73.71(b)
[ ] 20.405(a)(1)(i)	[ ] 50.36(c)(1)	[ ] 50.73(a)(2)(v)	[ ] 73.71(c)
[ ] 20.405(a)(1)(ii)	[ ] 50.36(c)(2)	[ ] 50.73(a)(2)(vii)	[ ] OTHER (Specify in Abstract below and in Text, NRC Form 366A)
[ ] 20.405(a)(1)(iii)	[XX] 50.73(a)(2)(i)	[ ] 50.73(a)(2)(viii)(A)	
[ ] 20.405(a)(1)(iv)	[ ] 50.73(a)(2)(ii)	[ ] 50.73(a)(2)(viii)(B)	
[ ] 20.405(a)(1)(v)	[ ] 50.73(a)(2)(iii)	[ ] 50.73(a)(2)(x)	

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER: AREA CODE	
K. E. Meade, Plant Operations Review Staff	6 1 5	8 7 0 - 6 2 5 0

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13)

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS

SUPPLEMENTAL REPORT EXPECTED (14)

[ ] YES (If yes, complete EXPECTED SUBMISSION DATE) [XX] NO

EXPECTED SUBMISSION DATE (15): MONTH | DAY | YEAR

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

On February 4, 1988, with units 1 and 2 in mode 5 (0 percent power, 4 psig, 130 degrees F and 0 percent power, 320 psig, 189 degrees F, respectively), it was determined that Surveillance Instruction (SI)-137.2, "Reactor Coolant System Water Inventory," did not adequately address the technical specification (TS) requirement for calculating the primary to secondary leak rate in mode 1 when the plant was operating at less than 100 percent power or when the plant was in modes 2, 3, and 4. Limiting Condition for Operation (LCO) 3.4.6.2.c requires that the primary to secondary leak rate be limited to 1 gallon per minute (GPM) for all steam generators (EIIS Code SB) and 500 gallons per day for any one steam generator. Additional investigation revealed that the full power steam generator water volume incorporated into the SI-137.2 mathematical model was also incorrect. A modification which occurred before original startup changed the steam generator volume; however, the mathematical model was not revised. Because of the incorrect water volume, SI-137.2 underestimated the full power primary to secondary leak rate by approximately 28 percent.

The event was caused by an inadequate instruction for measuring the primary to secondary leak rate. The Sequoyah Nuclear Plant (SQN) TSs have specific limitations on the primary to secondary leak rate during plant operation in modes 1 through 4. SI-137.2 should have had provisions (i.e., correct steam generator water volumes) for measuring the subject leakage during all modes of TS applicability. The cause of the incorrect steam generator water volume has been attributed to an inadequate change control process. This prevented the Final Safety Analysis Report (FSAR) from being revised to the correct steam generator volume value. SI-137.2 has been revised to correct the subject deficiency. Also, the design change control and plant modification procedures have been revised to prevent recurrence of this type of event.

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DESCRIPTION OF EVENT

On February 4, 1988, with units 1 and 2 in mode 5 (0 percent power, 4 psig, 130 degrees F and 0 percent power, 320 psig, 189 degrees F, respectively), it was determined that Surveillance Instruction (SI)-137.2, "Reactor Coolant System Water Inventory," did not adequately address the Technical Specification (TS) requirement for calculating the primary to secondary leak rate in mode 1 when the plant was operating at less than 100 percent power or when the plant was in modes 2, 3 and 4. TS Limiting Condition for Operation (LCO) 3.4.6.2.c requires that the primary to secondary leak rate be limited to 1 gallon per minute (GPM) for all steam generators (EIIS Code SB) and 500 gallons per day for any one steam generator.

SI-137.2, which is performed every 72 hours when the plant is in modes 1 through 4, calculates the primary to secondary leak rate for any one steam generator based on the activity level of a known isotope in the reactor coolant system (RCS) (EIIS Code AB) and the particular steam generator. The test is initiated by obtaining a liquid sample from the RCS and steam generator (blowdown), and determining the initial activity level of a specific isotope (usually Na-24). Once a sample is taken, steam generator blowdown is isolated, and RCS conditions are maintained essentially constant. At the end of the test (usually 3 hours), a second steam generator blowdown sample (EIIS Code WF) is taken and analyzed for isotopic activity. Based on the increase in steam generator isotopic activity, the known RCS isotopic activity and the steam generator liquid volume, the primary to secondary leak rate can be calculated.

During a routine audit of Sequoyah Nuclear Plant (SQN) procedures, an NRC inspector discovered that the steam generator water volume used by SI-137.2 was applicable only when SQN units 1 or 2 were in mode 1 at 100 percent power. As a result, performance of SI-137.2, when either SQN unit was in mode 1 at less than 100 percent power or in modes 2 through 4, incorrectly calculated the primary to secondary leak rate due to different steam generator volumes.

Additional investigation revealed that the full power steam generator water volume (from the Final Safety Analysis Report (FSAR) Section 5.2.7) incorporated into the SI-137.2 mathematical model was also incorrect. A modification which occurred before original startup changed the steam generator volume; however, the mathematical model was not revised. Because of the incorrect water volume, SI-137.2 underestimated the full power primary to secondary leak rate by approximately 28 percent.

CAUSE OF EVENT

The event was caused by an inadequate procedure for measuring the primary to secondary leak rate. The SQN TSs have specific limitations on the primary to secondary leak rate during plant operation in modes 1 through 4. SI-137.2 should have had provisions (i.e., correct steam generator water volumes) for measuring the subject leakage during all modes of TS applicability. The cause of the incorrect steam generator water volume has been attributed to an inadequate change control process. This prevented the FSAR from being revised to the correct steam generator volume value.

ANALYSIS OF EVENT

This event is reportable in accordance with 10 CFR 50.73, paragraph a.2.i, as an operation prohibited by the SQN TSs.

The intent of maintaining the primary to secondary leak rate within limits is to ensure that radiological consequences of the postulated accidents described in Chapter 15 of the SQN FSAR remain within the limitations of 10 CFR 100. That is, the offsite doses resulting from a postulated accident have been calculated based on the maximum allowed TS leak rate, and have been shown to be acceptable.

The effect of a primary to secondary leak rate in excess of what is allowed by the SQN TSs would affect the radiological consequences of all FSAR Chapter 15 events that involved secondary steam releases to the atmosphere. However, a review of plant data corresponding to full power operating conditions showed no cases where the FSAR Chapter 15 limits had been exceeded. Full power operation is the most likely condition to create the design basis event. TVA believes that the available full power data represent a reasonable basis to conclude that the radiological consequences discussed in FSAR Chapter 15.5.3 would remain bounding.

CORRECTIVE ACTIONS

As immediate corrective action, TVA has issued Instruction Change Form (ICF)-88-210, which was approved on January 30, 1988, to permanently change SI-137.2. This ICF changed SI-137.2 to include a requirement to correct for RCS temperature differences between the start and end points of the test, and a figure providing steam generator water volumes (at 33 percent narrow range level) during plant operation in modes 3, 4, and 5. In addition, TVA is currently developing a figure that will provide steam generator water volumes during mode 2 operation and mode 1 operation below 100 percent power. This information will be incorporated into SI-137.2 before startup (mode 2) of unit 2

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TEXT (If more space is required, use additional NRC Form 366A's) (17)

To correct the inaccurate reference to steam generator water volume contained in FSAR Section 5.2.7, TVA will revise the SQN FSAR. This FSAR change will be included with the next scheduled revision of the FSAR as required by 10 CFR 50.71 (e).

In October 1987, TVA issued Nuclear Engineering Procedures (NEP) 6.3, "Operating Plant Modifications;" NEP 6.4, "Plant Modifications Packages;" NEP 6.5, "Plant Modification Studies;" and NEP 6.7, "Document Update Process - Modifications." These procedures consolidated the design change control and plant modification processes. Specifically, Section 3.5 and Appendix K of NEP 6.4 provide the requirements necessary to ensure that the effect of a design change on plant operating procedures has been reviewed before the change has been released for implementation. This will prevent recurrence of this type of event.

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TENNESSEE VALLEY AUTHORITY
Sequoyah Nuclear Plant
Post Office Box 2000
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February 26, 1988

U. S. Nuclear Regulatory Commission
Document Control Desk
Washington, DC 20555

Gentlemen:

TENNESSEE VALLEY AUTHORITY - SEQUOYAH NUCLEAR PLANT UNIT 1 - DOCKET NO. 50-327 - FACILITY OPERATING LICENSE DPR-77 - REPORTABLE OCCURRENCE REPORT SQRO-50-327/88009

The enclosed licensee event report provides details concerning an inadequate procedure causing inaccurate primary to secondary leak rates to be measured resulting in a potential noncompliance with a Limiting Condition for Operation. This event is reported in accordance with 10 CFR 50.73, paragraph a.2.i.

Very truly yours,

TENNESSEE VALLEY AUTHORITY

S. J. Smith

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Plant Manager

Enclosure
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