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CAUSE	SYSTEM	COMPON	ENT	MANUFAC TURER	REPORTABLE TO NPRDS		CAUS	E SYSTEM	COMPONENT	MANUFAC TURER	REPORTABLE TO NPRDS	
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a.	verm	ined th	did -	not adage	ately of	ddross t	(S1)-	13/.2,	Reactor	tion (TC	System Wate	er.
f	or ce	lculat	ine t	the prime	ry to e	aconderv	leak	rate i	n mode 1	when the	plant was	ant
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g	enera	tors (1	EIIS	Code SB)	and 50	0 gallon	s per	day fo	r any one	steam ge	enerator.	
A	iditi	onal in	nvest	tigation	reveale	d that t	he ful	l powe	r steam ge	enerator	water volu	ne
iı	corp	orated	into	the SI-	137.2 m	athemati	cal mo	del wa	s also inc	correct.	Α	
ma	odifi	cation	whic	ch occurr	ed befo	re origi	nal st	artup	changed th	ne steam	generator	
V	stor	; howe	ver,	the math	iematica.	1 model	was no	t revi	sed. Beca	ause of t	the incorrec	et
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T	le ev	ent wa	s car	ised by	in inade	quate in	struct	ion fo	r moscuri.	ne the pr	rimery to	
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1	imita	tions	on th	he primar	y to se	condary	leak r	ate du	ring plant	t operati	ion in mode	s 1
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W	ater	volume	s) fo	or measur	ing the	subject	leaka	ge dur	ing all mo	odes of 1	TS	
aj	pplic	abilit;	y. 1	The cause	of the	incorre	ct ste	am gen	erator was	ter volum	me has been	
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SI	afety	Analy	SIS H	Keport (H	SAR) fr	om being	revis	ed to	the correct	ct steam	generator	
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DESCRIPTION OF EVENT

On February 4, 1988, with units 1 and 2 in mode 5 (O percent power, 4 psig, 130 degrees F and O 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.

NRC Form 366A (9-83) LICENSEE E	VENT REPORT (LER) TEXT CONTINU	JATIO	N		U.S.	APP	ROVED	REGULATORY COMMISSION (ED OMB NO 3150-0104 (8/31/88							
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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

IS-831 LICENSEE EVE	LICENSEE EVENT REPORT (LER) TEXT CONTINUATION									
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TEXT (If more space is required, use additional NRC Form 3664'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 Soddy-Daisy, Tennessee 37379

February 26, 1988

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

Gentlemen:

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TENNESSEE VALLEY AUTHORITY - SEQUOYAH NUCLEAR PLANT UNIT 1 - DOCKET NO. 50-327 - FACILITY OPERATING LICENSE DPR-77 - REPORTABLE OCCURRENCE REPORT SQR0-50-327/88009

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

Plant Manager

Enclosure cc (Enclosure):

> J. Nelson Grace, Regional Administrator U. S. Nuclear Regulatory Commission Suite 2900 101 Marietta Street, NW Atlanta, Georgia 30323

Records Center Institute of Nuclear Power Operations Suite 1500 1100 Circle 75 Parkway Atlanta, Georgia 30339

NRC Inspector, Sequoyah Nuclear Plant

