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 3

TITLE (4): Inadequate Verification Of ECCS Flow Due To Procedural Inadequacy

EVENT DATE (5) MONTH	DAY	YEAR	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	REPORT DATE (7) MONTH	DAY	YEAR	OTHER FACILITIES INVOLVED (8) FACILITY NAMES	DOCKET NUMBER(S)
0 9	2 5	8 6	8 6	— 0 4 4 —	0 0	1 0	2 4	8 6	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)
James D. Smith, Plant Operating Review Staff	6 1 5 870 - 6 6 7 2

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)

While both units were in mode 5 at zero percent power, a review was conducted of procedures used to meet technical specifications (TSs) surveillance requirements (SRs). On September 25, 1986, it was determined that Surveillance Instruction (SI)-137.3 for SR 4.4.6.2.1.C did not adequately ensure that the seal injection (EIIS System CB) flow rate was less than or equal to that which was assumed in the accident analysis. More specifically, SI-137.3 verified seal injection flow of 40 gpm at reactor coolant system (RCS) (EIIS System AC) pressure of 2235 $\pm$ 20 psig during normal operating configurations. However, Westinghouse has indicated that a 100 psi pressure differential in addition to the requirement for 40 gpm at normal RCS operating conditions is required to ensure an acceptable system resistance to limit emergency core cooling system flow through this pathway. Based upon Westinghouse identified acceptable criteria, nominal plant conditions with centrifugal charging pumps (CCPs) discharge pressure of at least 100 psi greater than the RCS pressure would have precluded any significant safety consequences as adequate CCP flow would have been available. The above described condition constitutes operation prohibited by TS and is reportable in accordance with 10 CFR 50.73, paragraph a.2.i.B, for both units 1 and 2. The cause for the inadequate SI has been determined to be inadequate preparation and review of the SI. Even though the SI complied with the explicit wording of the SR, it had not been verified that the SI ensured that the base of the TS was met.

SI-137.3 is being changed to ensure that an adequate reactor coolant pump seal injection flow path resistance is achieved and administrative controls will be established to ensure this resistance is maintained.

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

DESCRIPTION OF EVENT

With both units 1 and 2 in mode 5 at zero percent power (unit 1 was at 270 psig, 127 degrees F and unit 2 was at 5 psig, 140 degrees F), an ongoing review was conducted of plant surveillance instructions (SIs) used to complete technical specifications (TSs) surveillance requirements (SRs). On September 25, 1986, it was determined that SI-137.3, "Measurement of the Controlled Leakage to the Reactor Coolant Pump Seals," for SR 4.4.6.2.1.C did not ensure that the reactor coolant pump (RCP) seal injection (EIIS System CB) flow rate during a loss of coolant accident is less than or equal to that assumed for the accident analysis. SI-137.3 meets the explicit requirements of SR 4.4.6.2.1.C, i.e., measurement of the controlled leakage to the RCP seals when the reactor coolant system (RCS) (EIIS System AC) is 2235 $\pm$ 20 psig at least once every 31 days with the modulating valve (FCV-62-93) fully open to be less than or equal to 40 gpm. However, the intent of the SR is to ensure that the seal injection flow during the full spectrum of primary side line breaks is less than or equal to that which has been assumed by Westinghouse in the accident analysis. Westinghouse has indicated that the acceptable range for seal injection flow is 40 gpm at operating RCS pressures to 74 gpm at RCS pressures near atmospheric. SI-137.3 and/or applicable plant controls failed to ensure that:

1. The seal injection line provided a minimum acceptable resistance during the performance of SI-137.3.

2. Adequate administrative controls had been placed to ensure that this minimum acceptable resistance is maintained between performances of SI-137.3.

SI-137.3 is applicable to both units. No immediate operator action is required while both units are in mode 5.

CAUSE OF EVENT

SI-137.3 was inadequate in that preparation and previous reviews performed in conjunction with the SI did not adequately ensure that the intent of the SR, as described in the bases, was met. While the SI did meet the verbatim requirements of the TS and associated SR, lack of administrative controls and/or procedural requirements could have contributed to a condition which resulted in an operation prohibited by the intent of the TSs.

ANALYSIS OF EVENT

This condition is applicable to both units 1 and 2 and constitutes operation prohibited by TSs which is reportable in accordance with 10 CFR 50.73, paragraph a.2.i.B.

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

SI-137.3 measured the seal injection flow of 40 gpm at 2235 ± 20 psig. Westinghouse has indicated that a 100 psi pressure differential in addition to the requirement for 40 gpm at normal RCS operating conditions will ensure an acceptable resistance to limit emergency core cooling system flow through this pathway.

This event is not considered to have had any significant safety consequences on the plant for the following reasons:

1. Normal plant conditions while performing SI-137.3 would have likely given a pressure differential of greater than 100 psi as the centrifugal charging pump (CCP) discharge pressure is greater than 2400 psig.

2. SI-260.2, "BIT Cold Leg Injection Flow Balance, Pump Performance and Check Valve Test," verifies the pump capacity is within accident constraints for the large break LOCA. The CCPs' flow rate has previously been shown as more than adequate to meet ECCS assumptions (i.e., margin exists to minimum requirements).

CORRECTIVE ACTION

Westinghouse has confirmed that an acceptable seal injection configuration exists provided that a flow of less than or equal to 40 gpm is measured while the RCS is at 2235 ± 20 psig and a pressure drop of greater than or equal to 100 psi exists across the seal injection lines. SI-137.3 is being revised to reflect this pressure differential requirement. This will be completed before restart of either unit.

In addition, administrative controls will be placed on the seal injection line needle valves (VLV-62-556, -557, -558, and -559) to ensure that the seal injection line system overall resistance is not adversely affected following successful performance of SI-137.3. This will be completed before restart of either unit.

The SI review program currently in progress which identified this event will ensure the technical adequacy of this SI.

ADDITIONAL INFORMATION

Previous reports on items not performed as required by TSs - Thirteen - SQRO-50-327/86042, 86039, 86035, 86030, 86027, 86023, 86020, 86018, 86017, 86013, 86011, 86008, and 86007.

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TENNESSEE VALLEY AUTHORITY
Sequoyah Nuclear Plant
Post Office Box 2000
Soddy-Daisy, Tennessee 37379

October 24, 1986

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

Gentlemen:

TENNESSEE VALLEY AUTHORITY - SEQUOYAH NUCLEAR PLANT UNITS 1 AND 2 - DOCKET NOS. 50-327 AND 50-328 - FACILITY OPERATING LICENSE DPR-77 AND -79 - REPORTABLE OCCURRENCE REPORT SQRO-50-327/86044

The enclosed licensee event report provides details concerning inadequate verification of emergency core cooling system flow due to procedural inadequacy. This event is reported in accordance with 10 CFR 50.73, paragraph a.2.i.B.

Very truly yours,

TENNESSEE VALLEY AUTHORITY

P.R. Wallace

P. R. Wallace
Plant Manager

Enclosure
cc (Enclosure):

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NRC Inspector, Sequoyah Nuclear Plant

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