NRC Form 366
(9-83)

U.S. NUCLEAR REGULATORY COMMISSION
APPROVED OMB NO. 3150-0104
EXPIRES: 8/31/85

LICENSEE EVENT REPORT (LER)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
Sequoyah, Unit 2	0 5 0 0 0 3 2 8	1 OF 0 4

TITLE (4)
Two Reactor Trips

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	1 5	8 5	8 5	0 0 4	0 0	0 3	1 3	8 5		0 5 0 0 0
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OPERATING MODE (9)	1	THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more of the following) (11)			
POWER LEVEL (10)	1 0 0	20.402(b)	20.405(c)	XX 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)	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
H. R. Rogers, Compliance Section Engineer	AREA CODE 6 1 5 — 8 7 0 - 6 1 4 7

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)		EXPECTED SUBMISSION DATE (15)	MONTH	DAY	YEAR
YES (If yes, complete EXPECTED SUBMISSION DATE)	XX NO				

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

Two reactor trips occurred on February 15, 1985, and February 17, 1985, respectively, due to lo-lo steam generator water level. All reactor protection and engineered safeguard systems performed as designed, and no anomalies were noted. There was no effect upon public health and safety for either event.

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

During normal power operations with unit 2 at 100 percent rated thermal power, a reactor trip occurred at 1511C on February 15, 1985, on lo-lo steam generator level in loop one. Upon the reactor trip, all reactor protection and engineered safety features performed as designed, and no anomalies were noted.

Investigation into the event revealed that power was lost to protection set I instrumentation, causing a reduction in feedwater flow to all steam generators. Feed pump speed control has input to a summing device for total steam flow. This signal can be manually selected at the main control board to be fed from either protection set I or II instrumentation. The normal operating position is protection set I, in which case, upon loss of power to protection set I, the steam flow signal went to a zero input to the feedwater pump speed controller causing the feedwater pump to reduce speed. This reduced pump speed resulted in a reduced feed flow to all steam generators when actual steam flow was still at full power rating. Also, loss of protection set I caused loops one and three feedwater regulator valves to fail closed. The final result was a reactor trip on lo-lo steam generator level in loop one.

The reason for the loss of protection set I power was a personnel error made while trying to remove vital inverter 2-I from service for maintenance. A fan had failed on the inverter, and Operations was removing the inverter from service to allow Electrical Maintenance to repair the fan. The assistant shift engineer (ASE) for unit 2 was taking the inverter out of service, using System Operating Instruction (SOI) 57.5A, "120-V AC Vital Instrument Power," which requires maintaining power to the 120-volt AC vital instrument power board using maintenance power. The 120-volt AC vital instrument power boards are normally fed from the 120-volt AC vital inverters. The maintenance power supply is from the 120-volt AC instrument power distribution panel 'A', unit 1, which is a nondivisional power source. To make this transfer, the unit was complying with action requirements of limiting condition for operation (LCO) 3.8.2.1. The ASE was in the process of making the transfer with the procedure in hand when, upon reading step 6 for removing inverter 2-I from service, he misinterpreted the wording. Step 6 reads to verify inverter sync supply light (blue) is on, and the ASE mistakenly read this to mean the sync light on the inverter itself, when actually it means the inverter sync supply light on the 120-volt AC vital instrument power board. When the ASE went to inverter 2-I, he saw a yellow sync light on instead of a blue one, and then he saw that inverter 1-I had a red sync light, so the ASE thought someone had inadvertently changed the lens covers on the lights.

While still at the inverter, the ASE went to step 7 which required moving the transfer switch on the 120-volt AC vital instrument power board 2-I from normal to alternate (maintenance) supply. However, another error was made when the ASE moved a transfer switch on the inverter from manual to bypass instead of a transfer switch on the 120-volt AC vital instrument power board. On step 8, which was performed correctly, the DC breaker was opened. At this time, an unusual noise was heard by the ASE, and upon calling the control room, he found that the unit had tripped.

Immediately upon the trip, the unit was stabilized with Tavg at 547 degrees F, and the operators complied with Abnormal Operating Instruction (AOI) 25.5 and emergency procedures E-0 and ES-0.1. Power was returned to the 120-volt AC vital instrument power board using maintenance supply, and all affected instrumentation was verified to be operable. The fan on the inverter was repaired, and the inverter was returned to service at 1728 CST on February 15, 1985.

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

A review of the incident by plant management has concluded that the event was primarily caused by personnel error in that the ASE failed to clearly understand a procedure before proceeding with work activities. A secondary cause is contributed to the procedure being used, in that better wording could have been used to better identify the location of the inverter sync supply light. The following corrective actions were taken to prevent recurrence of the incident: (1) the ASE was counselled concerning this and other procedures on making sure he fully understands them before proceeding with work, (2) disciplinary action will be taken against the individual involved, and (3) the procedure is being revised to clarify step 6 and to require double verification of each step.

A careful evaluation of the event was made, including effects it could have had upon other equipment in the plant, and it was determined that the unit was safe for restart. The reactor returned critical at 0513 CST on February 16, 1985, and proceeded to 30 percent reactor power without incident.

While operating at 30 percent power, a second reactor trip occurred at 0243 CST on February 17, 1985, due to lo-lo steam generator level in loop two. During this trip, all reactor protection and engineered safeguard features operated as expected.

Prior to the event, at 0233 CST, a low electrohydraulic control (EHC) system level and reserve oil pump running alarm was received in the main control room, and an ASE and an assistant unit operator (AUO) were dispatched to investigate the problem. At 0235 CST, a load reduction was started at 2 percent per minute. The AUO notified the control room at 0236 CST of the loss of EHC fluid and that a turbine trip could be expected. At 0239 CST, the 'A' main feed pump tripped on low seal injection pressure, causing a turbine trip. During the time the main feed pump was lost, the steam generator controls were in manual, and with the loss of feedwater flow, auxiliary feedwater was unable to maintain levels at this power level and subsequently, a lo-lo steam generator level reactor trip occurred at 0243 CST. Following the reactor trip, the unit stabilized in mode 3 at 547 degrees F.

The main feed pump trip which was caused by low seal injection pressure was attributed to condensate feedwater flow fluctuations while the operator was trying to match Tavg-Tref during the load reduction with steam generator controls in manual. Since seal injection water for the main feed pump is taken from the condensate feedwater flow paths, the seal injection system pressure experienced a momentary drop during the transient.

An investigation into the loss of EHC fluid revealed that an inboard filter cartridge cover from the 'A' EHC pump failed, dumping approximately 80 gallons of EHC fluid on the floor. The cartridge cover is threaded and screws into the cartridge housing and sees system pressures fluctuating from 1600 to 2100 psi. The top half of the cover was found to be broken, leaving the threaded portion in the housing. The threaded portion was removed from the housing, and an examination of the housing showed no signs of damage.

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The failed filter cartridge cover was sent to TVA central lab for testing, and results of the testing indicate that the failure was not caused by system pressure fluctuations, but were due to overstressing which most probably occurred from overtightening of the cover. There were no records of any one specific maintenance item done recently which would have caused such an event. Therefore, no exact time could be found which caused the failure. It is suspected that the overstressed condition occurred over a period of time and not from one event. The covers are made of an aluminum alloy material and were supplied as part of the EHC system package by Westinghouse. This has been the only cartridge cover failure at Sequoyah Nuclear Plant in approximately five years of operation.

The following corrective actions have been taken to prevent further recurrence: (1) a new cover was made of an aluminum material 2024T4 as recommended by TVA metallurgists, and the new cover was installed and the system tested with satisfactory results, (2) two new covers will be ordered from Westinghouse and will be maintained as spares, (3) signs will be placed on top of the EHC control block at the filter cartridges instructing personnel not to overtorque the covers, and (4) efforts are being made to obtain from Westinghouse torque values recommended when replacing these covers.

After repairs were made to the EHC system, an evaluation was completed which concluded that the reactor was safe for restart. The reactor went critical at 0514 CST on February 17, 1985, and returned to power without further incident.

There was no effect upon public health and safety for either event, and these are the third and fourth reactor trips for unit 2 for 1985.

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

March 13, 1985

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

Gentlemen:

TENNESSEE VALLEY AUTHORITY - SEQUOYAH NUCLEAR PLANT UNIT 2 - DOCKET NO. 50-328 - FACILITY OPERATING LICENSE DPR-79 - REPORTABLE OCCURRENCE REPORT SQRO-50-328/85004

The enclosed licensee event report provides details concerning two reactor trips occurring on February 15, 1985, and February 17, 1985, respectively. This event is reported in accordance with 10 CFR 50.73, paragraph a.2.iv.

Very truly yours,

TENNESSEE VALLEY AUTHORITY

P. R. Wallace
Plant Manager

Enclosure
cc (Enclosure):

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