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 2	0 5 0 0 0 3 2 8	1 OF 0 7

TITLE (6)
Reactor Trip Resulting From Steam/Feedwater Flow Mismatch Coincident With Low Steam Generator Level Caused By A Missing Diode

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 6	0 6	8 8	8 8	0 2 7	0 1	0 7	2 9	8 8		0 5 0 0 0
										0 5 0 0 0

OPERATING MODE (9): 1

POWER LEVEL (10): 0 9 8

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)	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 (AREA CODE)
Don Siska K. W. Fenn, Plant Operations Review Staff	6 1 5 8 7 0 - 6 5 1 1

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)

This LER is being revised to update the corrective actions and to more accurately reflect the minimum reactor coolant system temperature that occurred following the reactor trip described herein. On June 6, 1988, at approximately 1415 EDT with unit 2 at 98 percent power, a reactor trip occurred from loop 4 steam/feedwater flow mismatch coincident with low steam generator (S/G) level. Instrument personnel were performing Surveillance Instruction (SI)-618, "Engineered Safety Features Actuation System Block Tests," to verify output continuity of the solid state protection system (SSPS) slave relays for feedwater isolation, turbine trip, and main feedwater (MFW) pump trip (train B). When test switch S801 in the safeguards test cabinet was depressed and released, the loop 4 MFW flow control valve (2-FCV-3-103) closed and feedwater was lost to this loop. As a result, S/G No. 4 level dropped, and the reactor tripped. The blocking circuitry of the above mentioned ESF functions incorporates test switch S801 and is designed to energize slave relay K601 and K621 while simultaneously providing a current path through diodes to block the final equipment actuation. It was discovered, subsequent to this trip, that the diode was missing in the blocking circuit for loop 4 feedwater flow control solenoid valve 2-FSV-3-103B. When the slave relays K601 and K621 were energized via the S801 switch, no current path was provided to maintain 2-FSV-3-103B energized and consequently the associated feedwater flow control valve 2-FCV-3-103 closed. A review of maintenance records did not identify any work associated with the missing diode, and past testing on this circuitry was always performed in a mode where the feedwater control valve would have already been closed. It is believed that the diode was most likely missing when the cabinet was received on site. The diode was replaced under Work Request (WR) B231541. The performance requirements of SI-618 were also reviewed, and it was determined that the performance of SI-618 is no longer required; thus, SI-618 was canceled. IE22 1/1

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	PAGE (3)
Sequoyah, Unit 2	0 5 0 0 0 3 2 8	8 8	— 0 2 7	— 0 1	0 2 OF 0 7

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

DESCRIPTION OF EVENT

This LER is being revised to update the corrective actions and to more accurately reflect the minimum reactor coolant system (RCS) (EIIS Code AB) average temperature (T-avg) that occurred following the reactor trip described herein. During a review of post-trip RCS temperatures to determine available shutdown margin (reference LER SQRO-50-328/88030), it was determined that the RCS average temperature had decreased to approximately 527 degrees F and not 510 degrees F, as previously reported.

On June 6, 1988, with unit 2 in mode 1 (98 percent power, 2234 psig, 576 degrees F), a reactor trip occurred at approximately 1415 EDT. The trip was caused by a main steam flow/feedwater flow mismatch coincident with low steam generator (S/G) water level in loop 4. Before the trip, instrument personnel were performing Surveillance Instruction (SI)-618, "Engineered Safety Features Actuation System Block Tests," which is performed to verify continuity of the final actuation circuitry for engineered safety features (ESF) (EIIS Code JE) components which cannot be actuated without causing plant upset or equipment damage. The specific ESF functions for which this procedure was being performed are (1) safety injection (EIIS Code BQ), (2) feedwater isolation (EIIS Code SJ), (3) phase "A" and "B" containment isolation (EIIS Code JM), and (4) steam line isolation (EIIS Code SB).

Specifically, section 13.0 of Data Sheet 2, "Block Test 9B/1 - Feedwater Isolation, Turbine Trip, and Main Feedwater Pump Trip, Train B, Cabinet R-53, Slave Relays K601, K620, K636, and K621," was being performed to check continuity on the output of the associated slave relays. This testing utilizes the safeguards test cabinet provided by Westinghouse as part of the plant solid state protection system (SSPS) (EIIS Code JG) and uses special test switches which block equipment actuation while testing of the ESF protective circuitry. When step 13.9 of Data Sheet 2 was performed, main feedwater (MFW) (EIIS Code SJ) flow control valve 2-FCV-3-103 closed, and feedwater flow was lost to steam generator (S/G) No. 4. As a result, the level in S/G No. 4 dropped to the low level setpoint (25 percent), and the reactor tripped on steam/feedwater flow mismatch coincident with low S/G level (loop 4). Step 13.9 (Data Sheet 2) of SI-618 requires the depressing and release of safeguards test switch S801 in the safeguards test panel (R-53). The performance of this step energizes slave relays K601 and K621 to simulate a turbine trip, a MFW trip, and a feedwater isolation. Simultaneous with the actuation of the slave relays, S801 also incorporates additional contacts to provide a current path through diodes in the safeguards test cabinet to block the actuation of the final equipment. Specifically, in the test that was being performed, the S801 contacts and the blocking circuit diodes are designed to provide a current path to maintain the feedwater flow control solenoid valves in an energized state and to prevent actuation of the feedwater flow control valves.

Immediately after the reactor trip, an investigation was initiated into the cause of the trip. During this investigation and troubleshooting under Work Request (WR) B231541, the diode (CR-9009) which is designed to provide a current path to the loop 4 flow control solenoid valve 2-FSV-3-103B, was found to be missing. With this diode missing, the current path to maintain 2-FSV-3-103B energized during this test was not provided. When step 13.9 was performed, the slave relays K601 and K621 actuated as designed, but the blocking function for 2-FSV-3-103B was not present. This caused 2-FSV-3-103B to deenergize, the associated feedwater flow control valve 2-FCV-3-103 to close, and the subsequent reactor trip.

SI-618 was being performed as a scheduled performance to satisfy a Final Safety Analysis Report (FSAR) quarterly schedule requirement. However, it was discovered during the post-trip investigation that this FSAR requirement had been deleted on revision 5 of the FSAR. The FSAR ESF testing requirements (section 7.3.2.2.5, page 7.3-14) now specify to refer to the plant technical specifications (TSs) for system test requirements and test frequencies.

Other anomalies noted subsequent to the reactor trip were as follows:

1. High pressure (HP) steam to moisture separator reheaters (MSRs) flow control valves 2-FCV-1-139 (MSR-C2) and 2-FCV-1-143 (MSR-B1) would not close while trying to isolate steam to the MSRs.

2. Condenser vacuum breaker was opened manually by the Turbine Building (TB) assistant unit operator (AUO) resulting in a loss of condenser vacuum and disabling of the steam dump valves. The steam dump valves are disabled via the C-9 control interlock when condenser vacuum is less than 17 inches of mercury. The vacuum breaker was opened because of a miscommunication between the TB assistant shift operations supervisor (ASOS) and the TB AUO. The high noise level in the area was a contributing factor in the miscommunication. The intent was to open the extraction low point drain valves using the handswitch near the condenser vacuum breaker.

3. Because the main control room operator was unaware of the vacuum breaker being opened locally, two vacuum pumps were turned off in an attempt to stabilize condenser vacuum.

4. The TB ASOS noted the condenser vacuum breaker open from the local handswitch and immediately closed the vacuum breaker. The vacuum breaker was open for approximately two minutes. Condenser vacuum restabilized, and the "B" vacuum pump was restarted.

5. Unit operators noted No. 3 S/G level rising and reverted to manual control of both motor-driven (MD) and turbine-driven (TD) auxiliary feedwater (AFW) level control valves (2-LCV-3-148 and 2-LCV-3-172) to S/G No. 3. The AFW level control valves are designed to control level at 33 percent when in automatic. S/G level increased to 54 percent before terminating the operation of the TDAFW pump to prevent S/G overfill. It was later determined that 2-LCV-3-172 (S/G No. 3 TDAFW level control valve) was not closing completely on demand.

6. S/G No. 2 power operated relief valve, 2-PCV-1-12 opened prematurely at approximately 1005 psig. The design setpoint is 1040 psig.

None of the above anomalies prevented the operators from recovering safely from the reactor trip. The reactor trip recovery was initiated using emergency instructions E-0, "Reactor Trip or Safety Injection," and ES-0.1, "Reactor Trip Response."

CAUSE OF EVENT

This reactor trip was a result of loop 4 steam flow/feedwater flow mismatch coincident with low S/G level. The steam/flow mismatch and low S/G level were both caused by solenoid valve 2-FSV-3-103B being deenergized and the subsequent closure of loop 4 feedwater flow control 2-FCV-3-103. The solenoid valve 2-FSV-3-103B deenergized during performance of SI-618 because of a missing diode (CR-9009) in the blocking circuit in safeguards test cabinet 2-R-53 (train B). Maintenance records and past performances of SI-618 were reviewed, and it was determined that the diode had most likely been missing since the cabinet was supplied by Westinghouse. The past performances of SI-618 and the pre-operational tests on this cabinet did not check the actuation of the solenoid valves specifically, and these tests were always run during a mode in which the feedwater flow control valves would have already been closed.

Causes of the other anomalies subsequent to the reactor trip are as follows:

1. Flow control valves 2-FCV-1-139 and -143 would not close completely because of the torque switch settings being too low to close off against the high pressure steam.

2. The condenser vacuum breaker was inadvertently opened because of a miscommunication between the TB ASOS and AUO.

3. S/G level in loop 3 could not be maintained with the TDAFW pump running because of the level control valve 2-LC-3-172 leaking through. The valve was leaking through because of a stem maladjustment.

4. S/G power operated relief valve 2-PCV-1-12 opened prematurely because of an out of calibration condition on the pressure switch (2-PS-1-13) which opens the valve on a high pressure condition.

ANALYSIS OF EVENT

This event is being reported in accordance with 10 CFR 50.73, paragraph a.2.iv, as an event which resulted in the automatic actuation of the reactor protection system.

The safety-related equipment required to mitigate the reactor trip operated as designed. The ESF actuation logic performed as designed without the blocking circuit functioning properly by deenergizing the solenoid valve 2-FSV-3-103B and closing feedwater flow control valve 2-FCV-3-103. The reactor trip logic actuated as designed by causing a reactor trip (reactor trip breakers opened and all control rods dropped to bottom position) from the steam flow/feedwater flow mismatch coincident with low S/G level. The feedwater isolation on all loops occurred as designed on a reactor trip coincident with RCS low average temperature (low Tavg) of 554 degrees F. Pressurizer pressure decreased to approximately 2040 psig and pressurizer level decreased to approximately 22 percent level. The RCS average temperature (i.e., the average of the T-avg's from each of the four RCS loops) decreased to approximately 527 degrees F. Therefore, the required safety-related functions performed as designed and the safety of the plant and public was not compromised. If an actual accident condition had occurred causing loss of feedwater to S/G No. 4, the reactor would have shut down as designed.

CORRECTIVE ACTION

Immediate corrective actions were to initiate reactor trip recovery using emergency procedures E-0 and ES-0.1. An immediate investigation was also initiated into the cause of the reactor trip and to determine any corrective actions necessary before returning the unit to operation.

The following corrective actions were completed before unit 2 entered mode 2 (startup):

1. The missing diode (CR-9009) in unit 2, train "B," safeguards test cabinet (2-R-53) was replaced under WR B231541.

2. S/G No. 3 level control valve (2-LCV-3-172) was repaired under WR B261181 to allow valve to close completely.

3. Valves 2-FCV-1-139 and -143 were both retested utilizing the motor-operated valve analysis test system (MOVATS) to allow the valves to stroke closed completely (WRs B253211 and B751298, respectively).

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	PAGE (3)
Sequoyah, Unit 2	0 5 0 0 0 3 2 8	8 8	— 0 2 7	— 0 1	0 6 OF 0 7

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

4. Power operated relief valve 2-PCV-1-12 opening prematurely was corrected by recalibrating the pressure switch (2-PS-1-13) which actuates the valve (WR B751300).

Other corrective actions which have been completed are:

1. A maintenance search was performed to determine the cause of the missing diode, but no records were found relating to the diode. The diode was most likely missing before TVA received the cabinet.

2. A review of past SI-618 performances and testing on the safeguards test cabinet was performed. This testing was never performed before this event in a mode where the feedwater water control valves would have been observed to close.

3. SI-618 performance requirements were reviewed and determined not to be required as a result of the FSAR revision 5 (section 7.3.2.2.5, page 7.3-14), and the SQN Technical Specification (TS) change No. 67 approved as follows:

   Unit 1 - Amendment No. 47 issued by NRC on September 17, 1986, as revision 51 to TS.

   Unit 2 - Amendment No. 39 issued by NRC on September 17, 1986, as revision 39 to TS.

   The present revision of TSs requires channel functional testing for ESF instrumentation at least once every 18 months. This testing is satisfied by SI-26 series of instructions and by SI-166.3, "Full Stroking of Category 'A' and 'B' Valves During Cold Shutdown." Therefore, SI-618 has been canceled.

Corrective actions still to be performed as a result of this event are:

1. A clear plastic cover will be installed over the condenser vacuum breaker local handswitch to deter inadvertent actuations. This cover will be installed by September 7, 1988.

2. An evaluation will be performed on other local handswitches in the Turbine Building which are vital to plant operations and install clear plastic covers, if necessary, to deter inadvertent actuations. This will be complete by September 7, 1988.

3. The main control room (MCR) controls for the condenser vacuum breaker will be evaluated to determine if a design change is appropriate to prevent the local handswitch from overriding the handswitch in the MCR. This evaluation will be complete by September 7, 1988.

4. The repeat back of verbal communications as addressed in Administrative Instruction (AI)-30, "Nuclear Plant Conduct of Operations," will be reemphasized to Operations personnel during operations requalifications. This will be complete by September 9, 1988.

ADDITIONAL INFORMATION

This was the third reactor trip since unit 2 startup from an extended shutdown of approximately 2 1/2 years (8/85 - 5/88). The first two reactor trips are detailed in LERs SQRO-50-328/88023 and 88024. The first two reactor trips were caused by an equipment failure and procedural noncompliance, respectively.

COMMITMENTS

1. A clear plastic cover will be installed over the local handswitch for the condenser vacuum breaker by September 7, 1988. (Systems Engineering)

2. Perform an evaluation on other Turbine Building local handswitches and install plastic covers as necessary by September 7, 1988. (Systems Engineering)

3. MCR controls for the condenser vacuum breaker will be evaluated by September 7, 1988. (Systems Engineering)

4. Reemphasize to Operations personnel the need for repeat back of verbal instructions by September 9, 1988. (Operations)

0993Q

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

July 29, 1988

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-50-328/88027 REVISION 1

The enclosed licensee event report has been revised to update the corrective actions and to more accurately reflect the minimum reactor coolant system average temperature that occurred following the reactor trip of June 6, 1988. This event was originally reported in accordance with 10 CFR 50.73, paragraph a.2.iv, on June 23, 1988.

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

IE22
1/1