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 5

TITLE (4)
Reactor Trip

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 1	1 2	8 5	8 5	— 0 0 2 —	0 0	0 2	1 1	8 5		0 5 0 0 0
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THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: *(Check one or more of the following)* (11)

OPERATING MODE (9): 1
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: AREA CODE	
H. R. Rogers, Compliance Section Engineer	6 1 5	8 7 0 - 6 1 4 6

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

CAUSE	SYSTEM	COMPONENT	MANUFAC-TURER	REPORTABLE TO NPRDS	CAUSE	SYSTEM	COMPONENT	MANUFAC-TURER	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)

On 01/12/85, unit 2 experienced a reactor trip on lo-lo steam generator level while the unit was at 100% power. Following the automatic trip, it was noted by the reactor operator that train A reactor trip breaker had failed to open automatically. The breaker was opened manually from the main control board. All other reactor protection and engineered safeguard features operated as expected and there was no effect upon public health and safety.

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

Unit 2 was operating at 100% power on 01/12/85 when a reactor trip occurred at 0329 CST. Prior to the trip at approximately 0320 CST, a steam generator level deviation alarm was received, and the operator noted that steam generator level in loop 3 had increased from 44% to 49%. As the operator proceeded to stabilize the steam generator level, he observed an increase in flow from the number 3 heater drain tank pumps to the condensate system upstream of the number 3 heaters. The unit assistant shift engineer immediately went to the turbine building to observe the level control valves on the discharge of the number 3 heater drain tank (HDT) pumps. During this time at approximately 0327 CST, all three number 3 HDT pumps tripped off and within a few seconds the control room shift engineer restarted pumps 'A' and 'B' to help stabilize condensate and feedwater flow. However, due to these unstable flow conditions, the 'A' main feedwater pump tripped on low seal injection water pressure to the main feedwater pump (MFP). The main feedwater pump seal injection water is fed from two pumps taking suction from the condensate system. Unstable flow in the condensate system due to the loss of the number 3 HDT pumps caused a momentary low seal injection water pressure. The 'A' main feedwater pump tripped at 0328 CST. The operator immediately took manual control of the 'B' MFP and increased its speed to provide sufficient flow to maintain steam generator levels. Simultaneously, a turbine runback was initiated to reduce load to prevent a reactor trip on low steam generator levels. The operator actions to prevent a trip were unsuccessful, and the unit automatically tripped at 0329 CST on lo-lo level in steam generator number 3. The operator followed the immediate operator actions in Emergency Procedure E-0, "Reactor Trip or Safety Injection," and, in his verification of control room indicator lights that both train 'A' and 'B' reactor trip breakers opened, he noted that the train 'A' breaker failed to open from the automatic signal. The train 'B' breaker did open on the automatic signal from the solid state protection system logic and all rods inserted as designed. Immediately, within 5 seconds of the automatic trip, the operator tripped the train 'A' breaker using the manual handswitch on the main control board. All other reactor protection and engineered safeguard features operated as expected upon receiving the automatic trip.

An investigation into the secondary side problems leading to the reactor trip were made, and it was found that one HDT pump discharge valve, LCV-6-106A, had failed. Upon disassembly of the 10-inch Masoneilan 10000 Series (Model 37-10134) valve, it was found that the stainless steel stem had broken loose from the stainless steel plug causing a partial blockage of its flow path. A parallel valve in the flow path from the pumps was also disassembled, and no problems were found. A review was made of recent maintenance history of these valves, and similar failures had occurred the week prior to the trip. On 01/03/85, valve LCV-6-106A experienced a similar failure and a new stem was installed and welded to the plug (normally, the plug and stem are threaded and pinned together). On 01/09/85, another failure occurred, and again the stem was found to be broken just above the weld at the plug. A new stem was installed and welded to the plug with a dye-penetrate test performed on the weld with satisfactory results. On 01/11/85, a different type failure occurred in that the stem started rotating and broke the feedback arm from the stem to the valve position controller causing a loss of automatic control of the valve. The feedback arm was repaired, and the valve returned to service. In all the above incidents, valve LCV-6-106B experienced no failures.

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

Several actions were taken following the 01/12/85 failure. The broken welded stems were sent to TVA central lab facilities for a failure analysis and detailed measurements were taken of the failed valve components (LCV-6-106A) and compared with the nonfailed valve (LCV-6-106B). It was found that the 01/03/85 failure resulted from a broken pin which connected the stem and the plug with the threads on the plug end of the stem being worn smooth. The 01/09/85 failure resulted from an inadequate weld due to incompatible materials of the stem and plug with similar results on the 01/12/85 failure. The following conclusions were made as a result of the analysis: (1) the principal failure mode was fatigue cracking, (2) incompatability of weld metals where the stem and plug were welded resulting in an accelerating factor in both failures, and (3) excessive valve bushing clearances between the stem and the bushing allowed plug movement and produced alternating stresses necessary for the fatigue failure.

The following immediate corrective actions were taken by Mechanical Maintenance personnel to return the valves to service: (1) a new stem was made of a material compatible for welding with the plug, (2) the plug was modified to accept a larger diameter stem to provide additional strength, and was pinned and not welded, and (3) a new stem bushing was made from 4130 steel with clearances as recommended by the manufacturer. The modifications were made to both the LCV-6-106A and -106B valves. Long-term corrective actions include replacing the plug, stem, and bushing with new parts from the manufacturer at the next refueling outages. This will be done on both units 1 and 2 and will include an inspection of the valve for excessive wear and clearances between the stem, plug, and bushings. Other items being considered are studies of operational conditions that would produce excessive valve vibration which might result in wear on valve parts. This will include looking at excessive valve operator movement as a contributor to stem failures. It is believed that these actions will preclude similar failures in the future.

A detailed investigation was also made into the failure of train 'A' reactor trip breaker to open on an automatic signal from the solid state protection system (SSPS). With Electrical and Instrument Maintenance working together, troubleshooting was divided into three areas with these being (1) the SSPS output components, (2) the cabling and connections between the SSPS and the reactor (Rx) trip switch gear, and (3) the Rx trip breaker. In troubleshooting areas (1) and (2) above, Rx trip breaker train 'B' was relocated in the train 'A' compartment. This would allow for testing and inspection of the train 'A' breaker in the as-found condition.

With the train 'B' breaker installed and closed in the train 'A' compartment, the instrument mechanics initiated a trip signal from the SSPS cabinets. The train 'B' breaker, which had tripped earlier in its own compartment, failed to open in the train 'A' compartment indicating that the problem was associated with the SSPS trip signal output. It was found that the output of the undervoltage card would not deenergize, thus holding the undervoltage trip coil energized and preventing the breaker from opening. Maintenance request (MR) A298460 was initiated and train 'A' SSPS was removed from service using procedure TI-52, and the reactor trip undervoltage card was replaced. While using a volt meter across the undervoltage output, an automatic trip signal was injected through the system test circuit, and the new undervoltage card was verified to deenergize. The old undervoltage card was then reinstalled and tested with the output voltage failing to deenergize, thus verifying the point of failure in the SSPS system. The new undervoltage card was reinstalled, reverified to be operable, including closing of the Rx trip breaker and verifying that it would trip automatically

from the train 'A' SSPS output signal. IMI-99-FT-19, "Automatic Test of Reactor Trip Breakers," was performed to verify operability and the train returned to service per TI-52.

Additionally, to verify the SSPS input relays, the bistables from steam generator lo-lo level were tripped in every combination to ensure all logic was functionally acceptable, and no deficiencies were noted.

On January 14, 1985, troubleshooting of the defective undervoltage card, serial number 0101, confirmed that transistor Q3 (output transistor on Westinghouse Drawing 6058090) had failed. The failure was an emitter to collector short which prevented the output voltage from deenergizing. The transistor was replaced, and the undervoltage card was retested using a card test box and procedure 0-SMI-99-1, and all functions of the card were verified operable. The failure of transistor Q3 probably occurred on 12/29/84 during the performance of IMI-99-FT-18, "Manual Test of Reactor Trip Breaker." The test was being performed as a startup requirement as a result of a unit trip which occurred on that date (reference LER 2-84021). During the test, a digital volt meter was inadvertently used on the current scale to measure voltage across the undervoltage coil. The low resistance of the current scale could have drawn enough current to fail transistor Q3.

Immediate corrective action was to replace the failed undervoltage card and verify through testing satisfactory operability. To prevent further recurrence, a note was sent to all Instrument Maintenance foremen to ensure that the automatic test of reactor trip breakers, FT-19, is performed after the manual test, FT-18. This will ensure the breakers and the undervoltage cards are fully operable before returning the system to service. Further, FT-18 and FT-19 have been revised and combined into one instruction which is structured to perform the automatic test after the manual test. The use of a meter to measure the voltage of the undervoltage coil has been discontinued. These measures will reduce the possibility of inadvertently damaging the output transistor on the SSPS undervoltage card and will also reduce the overall time needed to perform the breaker test. This revision was completed on 01/25/85.

An inspection was also performed on the train 'A' breaker in the as-found condition by Electrical Maintenance. The Westinghouse DB-50 breaker was checked for smooth manual closing, alignment trip bar movement, adequate contact surfaces, and overall condition in accordance with Maintenance Instruction MI-10.9. The undervoltage trip assembly linkage was lubricated with the recommended Westinghouse lubricant, and the linkage was exercised manually to allow penetration of the lubricant. A direct current (DC) power supply was connected to contact points 11 and 12, and rated voltage was applied to the undervoltage coil. The voltage was slowly lowered until the breaker tripped at approximately 19 volts DC. This was repeated with the same results. This voltage is within the range of the acceptance criteria as set forth by the Westinghouse Owners Group for actuation of the undervoltage relay. There were no deficiencies found with the DB-50 breaker assembly. A response time test of the reactor trip breaker from logic through the breaker was performed after all maintenance activities were complete and was found to be .08 second which is well within the 0.2 second allowable. Major maintenance to the breaker will continue to be performed using MI-10.9. An overall timing test is presently required by MI-10.9 as postmaintenance testing. This will ensure that the breaker and automatic logic is operable after breaker maintenance.

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After the replacement of the SSPS undervoltage card, verification of proper breaker operation, and repair of the condensate system number 3 HDT level control valves, an evaluation of the unit by Operations personnel and plant management concluded that the unit was safe for restart. The unit returned critical at 2156 CST on 01/12/85 without incident. There was no effect upon public health and safety, and this was the first trip on unit 2 for 1985.

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

February 11, 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-50-328/85002

The enclosed licensee event report provides details concerning a reactor trip which occurred on January 12, 1985. This event is reported in accordance with 10 CFR 50.73, paragraph a.2.iv.

Very truly yours,

TENNESSEE VALLEY AUTHORITY

P.R. Wallace

P. R. Wallace
Plant Manager

Enclosure
cc (Enclosure):

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