Tennessee Valley Authority, Post Office Box 2000, Soddy-Daisy, Tennessee 37379

J. L. Wilson
Vice President, Sequoyah Nuclear Plant

August 17, 1992

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D.C. 20555

Gentlemen:

TENNESSEE VALLEY AUTHORITY - SEQUOYAH NUCLEAR PLANT UNIT 2 - DOCKET NO. 50-328 - FACILITY OPERATING LICENSE DPR-79 - LICENSEE EVENT REPORT (LER) 50-328/92010

The enclosed LER provides details concerning the inoperability of the B-Train residual heat removal pump because of a mislaid wire on a flow switch. This event is being reported in accordance with 10 CFR 50.73(a)(2)(i)(B) as an operation prohibited by technical specifications and also in accordance with 10 CFR 50.73(a)(2)(ii)(A) as a condition that was outside the design basis of the plant.

Sincerely,

JLWilson

J. L. Wilson

Enclosure
cc: See page 2

JE22 1/1

cc (Enclosure):

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

Mr. D. E. LaBarge, Project Manager
U.S. Nuclear Regulatory Commission
One White Flint, North
11555 Rockville Pike
Rockville, Maryland 20852

NRC Resident Inspector
Sequoyah Nuclear Plant
2600 Igou Ferry Road
Soddy-Daisy, Tennessee 37379

Mr. B. A. Wilson, Project Chief
U.S. Nuclear Regulatory Commission
Region II
101 Marietta Street, NW, Suite 2900
Atlanta, Georgia 30323

NRC Form 366 (6-89) | U.S. NUCLEAR REGULATORY COMMISSION | Approved OMB No. 3150-0104 Expires 4/30/92

LICENSEE EVENT REPORT (LER)

FACILITY NAME (1): Sequoyah Nuclear Plant, Unit 2

DOCKET NUMBER (2): 0|5|0|0|0|3 2 8

PAGE (3): 1 OF 07

TITLE (4) Residual Heat Removal Pump Inoperable due to a Miswired Flow Switch for the Miniflow Valve

EVENT DAY (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|5|0|0|0|
0 7	1 7	9 2	9 2	0 1 0	0 0	0 8	1 7	9 2		0|5|0|0|0|

OPERATING MODE (9): 1

POWER LEVEL (10): 1 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)	XX 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
C. H. Whittemore, Compliance Licensing	AREA CODE 6 1 5 | 8 4 3 - 7 2 1 0

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) | X NO

EXPECTED SUBMISSION DATE (15): MONTH | DAY | YEAR

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

On July 17, 1992, with Unit 2 in Mode 1 at 100 percent power operations, personnel performing a surveillance instruction identified a Residual Heat Removal (RHR) Pump 2B-B miniflow valve to be malfunctioning. Operations personnel declared the RHR pump inoperable, and Limiting Conditions for Operation (LCOs) 3.5.2 and 3.6.2.1 were entered at 1100 Eastern daylight time (EDT) on July 17, 1992. An investigation determined the problem to be an incorrectly terminated wire on the flow switch. The wire was correctly terminated and the flowswitch was functionally tested and returned to service. LCOs 3.5.2 and 3.6.2.1 were exited at 2249 EDT on July 17, 1992. A subsequent investigation into the event identified the root cause of the mislaid wire as being inattention to detail with an inadequate second-party verification. Maintenance personnel have been briefed on specific problems identified in this event. A less than adequate post maintenance test (PMT) also contributed to the event. On July 28, 1992, during the review of the event by the Plant Event Review Panel (PERP), it was discovered that a potential issue existed involving the RHR systems being outside of design basis of the plant. A one-hour telephone call notifying NRC of the issue was made at 1928 EDT on July 28, 1992.

I. PLANT CONDITIONS

Unit 2 was operating at approximately 100-percent reactor thermal power.

II. DESCRIPTION OF EVENTS

A. Event

On July 17, 1992, with Unit 2 in Mode 1 and 100-percent power, Operations personnel performing a quarterly residual heat removal (RHR) pump surveillance instruction, identified the 2B-B RHR (EIIS Code BP) pump (EIIS Code P) miniflow valve (EIIS Code FCV) to be malfunctioning. The miniflow valve was cycling open and closed instead of remaining open. Operations personnel declared the RHR pump inoperable, and Limiting Condition for Operation (LCOs) 3.5.2 and 3.6.2.1 were entered at 1100 Eastern daylight time (EDT). An investigation revealed the flow switch for the miniflow valve had been miswired on July 1, 1992. It should be noted that between July 1 and July 17, 1992, there were 10 instances where Train A safety equipment, i.e., centrifugal charging pump (CCP), safety-injection pump, diesel generator (D/G), and 6.9 kilovolt shutdown boards were inoperable for short periods of time. With the exception of two instances that are described in the following paragraph, the periods of inoperability were of short duration.

B. Inoperable Structures, Components, or Systems That Contributed to the Event

On July 8, 1992, D/G 2A-A was inoperable for 17 hours.

On July 9, 1992, CCP 2A-A was inoperable for six hours.

C. Dates and Approximate Times of Major Occurrences

June 30, 1992 0600 EDT	Flowswitch quarterly preventive maintenance (PM) was started.
June 30, 1992 0820 EDT	A work request (WR) was written to replace a flowswitch when a problem was found that prevented calibration and testing.
July 1, 1992 0627 EDT	A WR was completed (flowswitch replaced).
July 1, 1992 0730 EDT	A PM was completed and the RHR pump was declared operable.
July 8, 1992 0600 EDT	Diesel Generator (D/G) 2A-A was inoperable - LCO 3.8.1.1 was entered.

July 8, 1992 2301 EDT	D/G 2A-A was operable - LCO 3.8.1.1 was exited.
July 9, 1992 1841 EDT	CCP 2A-A was inoperable for maintenance, LCOs 3.5.2, 3.1.2.4, and 3.1.2.2 were entered.
July 10, 1992 0059 EDT	CCP 2A-A was operable, and LCOs 3.5.2, 3.1.2.4, and 3.1.2.2 were exited.
July 17, 1992 1100 EDT	Quarterly operability surveillance instruction test for RHR pump 2B-B identifies miniflow valve cycling open and closed. LCOs 3.5.2 and 3.6.2.1 were entered.
July 17, 1992 1830 EDT	Miniflow valve flowswitch was found to be miswired - the wiring was corrected.
July 17, 1992 2249 EDT	LCOs 3.5.2.1 and 3.6.2.1 were exited for 2B-B RHR pump.
July 18, 1992 0015 EDT	The wiring on Unit 1 Train A and both trains of Unit 2 RHR pump miniflow switches was verified as correct.
July 28, 1992 1928 EDT	Following management's review of the event in the Plant Event Review Panel (PERP) meeting, NRC was notified of the condition under 10 CFR 50.72 as potentially having placed the plant outside of design basis, because of Train A safety equipment and/or components out of service between July 1 and July 17, 1992.

D. Other Systems or Secondary Functions Affected

None.

E. Method of Discovery

Operations personnel performing a quarterly operability test on the 2B-B RHR pump identified the abnormal operation of the miniflow valve. Investigation into the cause of the abnormal operation of the valve revealed the flowswitch that controls the miniflow valve had a field wire incorrectly terminated.

F. Operator Actions

Operations personnel identified that the miniflow valve was malfunctioning and took appropriate action by declaring the 2B-B RHR pump inoperable and for entering LCOs 3.5.2 and 3.6.2.1. A WR was initiated to investigate and troubleshoot the cause. After corrective action was concluded and the miniflow valve was functionally verified as being able to perform its intended function, LCOs 3.5.2 and 3.6.2.1 were exited.

G. Safety System Response

No safety system responses were required.

III. CAUSE OF EVENT

A. Immediate Cause

The immediate cause of this event was the incorrectly terminated wire for the miniflow valve, which rendered the 2B-B RHR pump inoperable. The inoperability of opposite train equipment contributed to the event.

B. Root Cause

There were three root causes for the event:

1. Inadequate self-checking and inattention to detail was the cause for the craftmen to incorrectly terminate the field wire. There was only one wire removed and reterminated during the July 1, 1992, flowswitch calibration PM.

2. Secondary-party verification was not effectively implemented. The verifier did not identify that the field wire was terminated on the correct terminal. The terminal block was correctly labeled and the label corresponded to the procedure and drawing. The wire was misterminated on a terminal that was not labeled.

3. A third root cause for this event was that the postmaintenance test (PMT) for the maintenance activity was ineffective. The WR did not clearly specify requirements necessary to verify that the miniflow valve functioned properly after the flowswitch was replaced in conjunction with the PM. The PMT as stated in the WR was to properly calibrate and functionally check the flowswitch. The ambiguity in the PMT led the craftsmen to belief that a system functional test or independent verification was not required.

IV. ANALYSIS OF EVENT

This event involves a wiring error that resulted in the miniflow recirculation valve cycling when the valve should have remained open.

The flowswitch that was miswired controls closure of the recirculation valve when the RHR pump discharge exceeds a setpoint of approximately 1,250 gallons per minute (gpm). (This setpoint accounts for instrument inaccuracies.) The basis for the valve closure is to ensure adequate flow goes to the core whenever reactor coolant system (RCS) pressures are low enough to allow RHR to inject.

The design logic requires the valve to be open at 500 gpm (decreasing) through the pump to protect the pump from heating damage, and for the valve to close at 1,500 gpm (increasing) to assure adequate flow to the reactor core for accident mitigation. The recirculation valve, which is motor operated, is part of the safety injection logic; therefore, it does not use thermal overloads. The actuator motor is rated for intermittent duty and can fail after approximately fifteen minutes of continuous operation. The pump recirculation requirement of 500 gpm is a continuous operation value. The continuous cycling of the valve ramped the flow from zero to approximately 750 gpm with each valve cycle. This may meet the cooling requirements for continuous flow through the pump, but the action puts a thrust cycle on the pump impeller and motor bearings that creates additional wear on the pump.

During an accident situation, the pump normally would be in recirculation mode during the injection phase of the accident. The pump is then used for net positive suction head (NPSH) boost during the recirculation phase until the RCS pressure drops below the pump deadhead pressure. With the recirculation valve open, the pump would operate normally and complete the accident mitigation task as designed.

The worst-case scenario involves a small break loss of coolant accident with the miniflow-valve motor failing in the fully closed position. Failing in the closed position, the RHR pump is subject to overheating and ultimate failure. This scenario, coupled with opposite train safety component unavailability, results in a condition outside design basis.

Further investigation and computer-simulated scenarios revealed that no damage would result from the valve cycling for approximately 25 minutes. It is fully expected that operators in the main control room would detect the abnormal operation from annunciators signaling the rapid change of position of the valve, and the fluctuation of the motor amperage. Upon detection, the RHR pump would then be turned off. This expectation was demonstrated by submitting the problem to operators during requalification training. These simulations did not cycle the miniflow valve, stopping the RHR pump relied on normal SI termination criteria

contained in emergency procedures. The times ranged between 21 and 25 minutes before the RHR pump was removed from service. Therefore, the added indications of position status lights and motor amps should prompt the operators to earlier intervention.

V. CORRECTIVE ACTIONS

A. Immediate Corrective Actions

Operations personnel immediately entered LCOs 3.5.2 and 3.6.2.1 for Unit 2.

Operations personnel exited LCOs 3.5.2 and 3.6.2.1 for Unit 2 after the misplaced wire was correctly terminated and the functional test verified the miniflow valve performed as designed.

B. Corrective Actions to Prevent Recurrence

1. Wiring on the other miniflow switches for Unit 1 and Unit 2 was checked and verified as being correctly terminated.

2. The instrument PMs data packages associated with the RHR miniflow valve switches have been revised to require independent verification for wire connections and also for jumpers.

3. Maintenance craftsmen, planners, and procedure writers have been briefed on this event with an emphasis on the need for an adequate PMT or specifying an independent verification in lieu of a PMT.

4. Maintenance planners will be trained on the proper way to specify acceptance criteria for verifying that components can perform their intended functions. This will be accomplished by September 14, 1992.

VI. ADDITIONAL INFORMATION

A. Failed Components

None.

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	PAGE (3)
Sequoyah Nucl Plant, Unit 2	0 5 0 0 0 3 2 8	9 2	0 1 0	0 0	0 7 OF 0 7

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

B. Previous Similar Events

A review of the licensee event report data base was conducted to identify any previous or similar events, and if so, to determine if corrective actions had been unsuccessful in preventing recurrence. Several events were identified that were caused by or had contributing factors similar to those noted in the investigation of this event, i.e., inattention to detail, inadequate verification, and inadequate PMT. Actions have been taken in response to previous events to ensure that expectations of management were clearly conveyed, understood, and concurred with by working-level personnel. Following this event, an independent team was assembled to evaluate the verification and PMT processes and their implementation. Corrective actions from this evaluation will be pursued as part of the overall SQN performance improvement efforts.

VII. COMMITMENT

Maintenance planners will be trained on the proper way to specify acceptance criteria for verifying that components can perform their intended functions. This will be accomplished by September 30, 1992.