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

TITLE (4): Loss Of Residual Heat Removal During Pump Swap Over

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)
1 0	0 9	8 5	8 5	0 4 0	0 0	1 1	0 7	8 5		0 5 0 0 0
										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)	XX	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	
Mike R. Cooper, Compliance Section Engineer	6 1 5	8 7 0 - 6 7 6 6

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)

On October 9, 1985, at 1807 CST during cold shutdown, swap over from "B" train to "A" train residual heat removal (RHR) resulted in both trains becoming inoperable due to air injection into the suction of the pumps. This required both pumps to be vented and required RCS level to be raised from 695'1" to 695'5" to prevent a possible recurrence of the vortex problem. Suction for RHR comes from the loop 4 hot leg which has a center line of 695'5". The cause for the loss of flow can be attributed to the additional suction caused by placing the standby RHR pump in-service coupled with the low RCS level of 695'1". System Operating Instruction (SOI)-74, "Residual Heat Removal System," is being revised to change the lower RCS operating limit from 695'0" to 695'6" and will require the operating pump to be removed from service prior to starting the standby pump. There were no safety implications to the public because the unit was in cold shutdown with only a 0.2 degrees F rise in RCS temperature resulting from the event. Technical Specification 3.4.1.4 action (b) says that ". . . with no RHR loops in operation, suspend all operations involving a reduction in boron concentration of the Reactor Coolant System." At the time of this event, the chemical volume control system (CVCS) (makeup system) was tagged out of service; therefore, no violations of technical specifications occurred.

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

On October 9, 1985, at 1807 CST during cold shutdown (mode 5) operations with reactor coolant system (RCS) temperature at 100 degrees F, a swap over from "B" train residual heat removal (RHR) cooling to "A" train RHR cooling was performed. This was accomplished by placing the "A" train pump (A-A RHR pump) in-service, and then removing the "B" train pump (B-B RHR pump) from service. After the swap over, RHR "A" train flow was observed to be "zero" and pump amperage indicated approximately 10 amps. The A-A RHR pump mini flow recirculation valve 1-FCV-74-12 was also observed to be full open. The B-B RHR pump was immediately placed back in-service (and exhibited normal operating indications), and the A-A RHR pump was removed from service. Concurrently with these actions, the Auxiliary Building assistant unit operator (AUO) was notified and verified proper "A" train alignment. After completing the verification and finding no deviations, the AUO was instructed to vent the A-A RHR pump which he did via the local pressure indicator (PI)-74-6 drain valve. After completion of the venting, at approximately 1842 CST, B-B RHR pump was stopped, and the A-A RHR pump was started. Again, the same parameters as described earlier existed on the A-A RHR pump. A swap over back to the B-B RHR pump was made, but the condition then also existed on B-B RHR pump. Both pumps were immediately stopped.

Following a brief discussion with the unit senior reactor operator (SRO) and outage coordinator (also an SRO), it was decided that the level in the RCS was too low to accommodate the additional demand of placing both RHR pumps in-service simultaneously, thus accounting for the loss of suction to the RHR pumps. At 1845 CST, after verifying all personnel were clear of No. 2 steam generator primary side (in which maintenance was in progress at that time), valve 1-FCV-63-1 from the refueling water storage tank (RWST) was cracked open. The RCS level was increased from 695'1" to 695'5". The B-B RHR pump was vented via PI-74-18 drain valve, and the pump was started at 1850 CST. The RHR flow and pump amperage appeared to be normal. The A-A RHR pump casing was then vented, the B-B pump was stopped, and the A-A RHR pump was placed in-service. The RHR flow and pump amperage remained normal.

The root cause for this event can be attributed to procedure inadequacy of SOI-68.1C, "Draining Reactor Coolant System," and the failure to have a section in SOI-74, "Residual Heat Removal System," to contain instructions for RHR pump swap over while on RHR cooling with RCS level in the loops. Other conditions which contributed to the event were (1) the lack of normal letdown and charging since the chemical volume control system (CVCS) was out of service, (2) apprehension by operations to use the 20-inch header from the RWST with a motor-operated valve for makeup since the potential existed of raising the RCS level into the No. 2 steam generator where personnel were performing tube plugging and eddy current testing, (3) the occurrence of a gradual reduction to 695'1" in RCS level over the three weeks which the makeup/letdown system was out of service, and (4) startups of the standby RHR pump being made prior to shutdown of the operating RHR pump.

A review of RCS temperature monitoring charts revealed only a 0.2 degrees F increase in RCS temperature due to this event.

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

Corrective Actions Taken

1. Pumps were immediately shutoff and vented upon detection of air binding.
2. RCS level was increased to 695'5" via RWST to prevent suction problems.
3. Proper valve alignment was verified when the problem occurred.

Corrective Action Planned

1. All operators will be made aware of this event and associated corrective actions.
2. Section will be added to SOI-74 for RHR pump swap over with reactor coolant level drained down to hot legs which will include a minimum RCS level greater than or equal to 695'6"and shutdown of the operating RHR pump prior to starting the shutdown pump.
3. Change SOI-68.1C step "X" caution from 695'0" to 695'6".

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

November 7, 1985

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

Gentlemen:

TENNESSEE VALLEY AUTHORITY - SEQUOYAH NUCLEAR PLANT UNIT 1 - DOCKET NO. 50-327 - FACILITY OPERATING LICENSE DPR-77 - REPORTABLE OCCURRENCE REPORT SQRO-50-327/85040

The enclosed licensee event report provides details concerning both trains of residual heat removal (RHR) system being inoperable due to suction problems when operating at loop 4 hot leg center line level. This event is reported in accordance with 10 CFR 50.73, paragraph a.2.v.

Very truly yours,

TENNESSEE VALLEY AUTHORITY

P.R. Wallace

P. R. Wallace
Plant Manager

Enclosure
cc (Enclosure):

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