TENNESSEE VALLEY AUTHORITY

6N 38A Lookout Place
Chattanooga, Tennessee 37402-2801
July 2, 1990

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

Gentlemen:

TENNESSEE VALLEY AUTHORITY - SEQUOYAH NUCLEAR PLANT UNIT 1 - DOCKET NO. 50-327 - FACILITY OPERATING LICENSE DPR-77 - LICENSEE EVENT REPORT (LER) 50-327/90012

The enclosed LER provides details of a Unit 1 reactor trip caused by low-low steam generator level, which resulted from inadequate communication between control room operators and turbine building auxiliary unit operators. This event is being reported under 10 CFR 50.73(a)(2)(iv).

Very truly yours,

TENNESSEE VALLEY AUTHORITY

J. R. Bynum

J. R. Bynum, Vice President
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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 1

DOCKET NUMBER (2): 0|5|0|0|0|3|2|7

PAGE (3): 1 OF 0|8

TITLE (4) Sequoyah Unit 1 Reactor Trip Caused by Low-Low Steam Generator Level Resulting From Inadequate Communication Between Control Room Operators and Turbine Building Auxiliary Unit Operators

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|6	0|2	9|0	9|0	0|1|2	0|0	0|7	0|2	9|0		0|5|0|0|0|

OPERATING MODE (9): 1

POWER LEVEL (10): 0|1|1

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
G. A. Hipp, Compliance Licensing Engineer	6|1|5 8|4|3 - 7|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)	X NO				

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

On June 2, 1990, with Units 1 and 2 at approximately 11 and 100 percent power respectively, a reactor trip occurred on Unit 1 about 17 minutes after a generator/turbine trip had occurred as a result of electrical problems. Control room operators reduced reactor power and announced the turbine trip on the plant public access (PA) system. The plant was stabilizing as reactor power reached approximately 15 percent when main feedwater (MFW) flow was lost. Two auxiliary unit operators (AUOs) had misheard the PA announcement as "unit trip" rather than "turbine trip" and had isolated the steam supplies to the MFW pumps. Operators started the auxiliary feedwater pumps while continuing to reduce reactor power, but the reactor tripped on low-low steam generator level. The root cause of the reactor trip has been attributed to inadequate communication between control room operators and AUOs. As corrective action, Operations management has issued a night order clarifying what specific actions should be taken by AUOs following a reactor or turbine trip only with unit operator guidance. A related weakness in AUO training regarding actions to be taken following a turbine trip or reactor trip was identified and corrected during the investigation of this event. Operations will evaluate the adequacy of AUO procedural guidance and training for other events.

DESCRIPTION OF EVENT

On June 2, 1990, with Unit 1 in Mode 1 at approximately 11 percent power, 2,235 pounds per square inch gauge (psig), 554 degrees Fahrenheit (F), and Unit 2 in Mode 1 at 100 percent power, 2,235 psig, 578 degrees F, a reactor trip occurred on Unit 1 at 0802 Eastern daylight time (EDT). The trip resulted from a low-low steam generator (S/G) level signal (EIIS Code JC) from S/G No. 4 that was caused by inadequate communication between control room operators and turbine building (TB) auxiliary unit operators (AUOs).

Prior to the event, reactor and turbine load had been holding steady at approximately 22 percent in preparation for a main turbine overspeed test following the Unit 1 Cycle 4 refueling outage. At approximately 0745 EDT, a generator/turbine trip occurred as a result of operation of the main generator stator ground fault protective relay (EIIS Code RLY). This relay monitors stator winding insulation resistance to provide protection for the generator stator winding against ground faults. The relay can be sensitive to load changes when the unit is returned to service following a lengthy outage until any moisture on the winding has evaporated.

After the turbine tripped, the steam dump valves (EIIS Code SB) automatically actuated as designed to release main steam directly into the main condenser (EIIS Code SG). Referring to Abnormal Operating Instruction (AOI) 17, "Turbine and Generator Trips," control room operators began reducing the reactor power by inserting control rods, isolated S/G blowdown (EIIS Code WI), and announced the turbine trip on the plant public address (PA) system. Since the S/G levels began to swell as a result of the turbine trip and less feedwater would be needed at reduced power, the operating 1B main feedwater pump (MFP) (EIIS Code SJ) was placed in manual control, and its speed was reduced. The intention was to regulate feedwater flow to the four S/Gs primarily by varying the MFP speed rather than by manipulating the four S/G feedwater regulating valves (FRVs). The four FRVs could then be slowly closed in small increments as reactor power was reduced. This method worked well as reactor power was reduced until, at approximately 15 percent power, main feedwater (MFW) flow was lost.

When the turbine trip occurred at 0745 EDT, the four TB AUOs on duty heard the steam dump valves actuate. A control room operator telephoned the TB operations office and told one of the AUOs that a turbine trip had occurred. The operator then directed the AUO to isolate the Nos. 1 and 2 feedwater heaters (EIIS Code SJ). This AUO told a second AUO who was in the office that a turbine trip had occurred. These two AUOs then proceeded to isolate the feedwater heaters as directed. The plant PA system announcement of the turbine trip was made at the same time. However, the announcement the other two AUOs thought they heard was "unit trip" rather than "turbine trip." Believing the reactor had tripped, these two AUOs proceeded to close the high pressure steam isolation valves on 1A and 1B MFPs. One AUO noticed the slower than normal speed of the operating MFP.

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TEXT CONTINUATION

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Sequoyah Nuclear Plant Unit 1	0|5|0|0|0|3|2|7	9|0	-- 0|1|2 --	0|0	0|3 OF 0|8

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

DESCRIPTION OF EVENT (Continued)

Because the MFP trip occurs with a reactor trip, the AUO assumed the slow pump speed confirmed the reactor trip. If the reactor had been tripped, these actions would have been appropriate. However, in this case, closing the MFP steam isolation valves caused a loss of MFW flow as reactor power reached approximately 15 percent.

Referencing AOI-16, "Loss of Main Feedwater," as the MFW flow was lost, the control room operator tried to manually speed up the 1B MFP and also tried to place the 1A MFP in service. When these efforts failed, all three auxiliary feedwater pumps (AFWPs) (EIIS Code BA) were started with maximum discharge flow. However, the loss of MFW flow resulted in the No. 4 S/G level reaching the reactor protection system (EIIS Code JC) normal containment low-low level setpoint of 13 percent of the narrow range at approximately 0757 EDT. The consequent reactor trip is delayed for power levels below 50 percent power by the Eagle 21 trip time delay feature. The time delay is proportional to power level; at 15 percent power, the time delay is approximately four minutes. Reduction of reactor power level by insertion of control rods was continued with the intention to reach three percent power where S/G levels could be sustained by the AFWP flow. However, before the AFWPs could reestablish S/G levels, the trip delay time expired, and the reactor tripped at 0802 EDT.

After the reactor trip, operators responded using emergency procedures 1-E-0, "Reactor Trip or Safety Injection," and 1-ES-0.1, "Reactor Trip Response," and General Operating Instruction (GOI) 3, "Plant Shutdown From Minimum Load to Cold Shutdown," to stabilize the unit.

Following the turbine trip, the steam dump valves had modulated to control reactor coolant system (RCS) average temperature (Tavg) at the programmed reference temperature. After MFW flow was lost, S/G levels fell rapidly and the AFWPs pumped a large amount of cold AFW into the S/Gs to try to restore the S/G levels and avoid a reactor trip. As a consequence of the cold AFW flow, RCS Tavg dropped and was below 547 degrees F at the time of the reactor trip. AFW flow was throttled when temperature dropped below 547 degrees F to minimize the RCS overcooling, but the subsequent reactor trip caused the Tavg to continue to drop. Following the reactor trip, emergency boration was started as prescribed by 1-ES-0.1 when RCS Tavg continued to decrease to 540 degrees F. Initially, problems were encountered with receiving adequate boration flow. However, these problems were resolved when the "A" boric acid tank recirculation line was isolated. The boric acid tank had been recirculating prior to the event and until the line was isolated, continued recirculating part of the boric acid transfer pump flow that was needed for emergency boration. Main steam isolation valves were closed in accordance with 1-ES-0.1 when RCS Tavg dropped below 530 degrees F. A minimum RCS Tavg of 522 degrees F was reached before recovering and stabilizing at 548 degrees F. The plant response during and after the trip is further discussed later in this report. Overall, plant systems responded properly, and the shutdown posed no danger to plant employees or the general public.

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

CAUSE OF EVENT

The root cause of the reactor trip has been attributed to inadequate communication between control room operators and TB AUOs. The failure to distinguish adequately between "turbine trip" and "unit trip" led to the inappropriate isolation of high pressure steam to the MFPs, which resulted in the reactor trip on low-low S/G level.

The cause of the turbine trip was actuation of the generator stator ground fault protective relay believed to have resulted from residual moisture from injection of fresh hydrogen following the lengthy Cycle 4 refueling outage. Subsequent generator stator winding insulation resistance readings indicated no problems with the generator winding. The subject relay is a very sensitive device by design and did not function in an anomolous manner. Monitoring of the relay may have detected approach to setpoint or potential for inadvertent actuation.

ANALYSIS OF EVENT

The event is being reported in accordance with 10 CFR 50.73(a)(2)(iv) as a reactor protection system actuation that was not part of a preplanned sequence. As shown by the following discussion of plant response during and after the trip, plant systems and parameters behaved in a manner consistent with the responses described in the SQN Updated Final Safety Analysis Report (UFSAR). Consequently, it can be concluded that there were no adverse consequences to the health and safety of plant personnel or the general public as a result of this event.

RCS Pressure

Initial RCS pressure was 2,235 psig. When the trip occurred, the RCS pressure dropped to 2,150 psig and later dropped as low as 2,100 psig before returning to 2,235 psig. These pressures are consistent with the expected range of values shown in UFSAR for this type of event. Overall, RCS pressure responded as expected.

RCS Temperature

Initial RCS temperature was 554 degrees F. After the reactor trip, Tavg dropped as low as 522 degrees F before recovering and stabilizing at 548 degrees F. Following the turbine trip, the steam dump valves modulated to control RCS Tavg at the programmed reference temperature. After the MFW flow was lost, S/G levels dropped rapidly and the AFWPs pumped a large amount of cold AFW into the S/Gs to try to restore S/G levels. The cold AFW flow caused RCS Tavg to drop below 547 degrees F, at which time the AFW flow was throttled back to minimize overcooling in accordance with 1-ES-0.1. Following the reactor trip, emergency boration was initiated when RCS Tavg decreased to less than 540 degrees F, and MSIVs were isolated when RCS Tavg went below 530 degrees F. The Tavg drop to 522 degrees F was caused by the large amounts of cold auxiliary feedwater added to the S/Gs prior to the reactor trip in an attempt to recover S/G levels with the MFPs out of service.

ANALYSIS OF EVENT (Continued)

Pressurizer Level

Initial pressurizer level was 30 percent. During the event, the level fell as low as 15 percent before recovering. The level stabilized within the limits of the control system and within the bounds of UFSAR accident analyses.

Forced/Natural Circulation

All four reactor coolant pumps remained in operation for the duration of the event. Consequently, no UFSAR assumptions were challenged.

Containment Pressure, Temperature, and Radiation

No perturbations were observed in containment environmental parameters during the event. Consequently, no technical specification (TS) requirements or UFSAR assumptions were challenged.

Heatup/Cooldown Limits

TSs limit the RCS cooldown rate to 100 degrees F in any one hour period. Based upon strip chart recorder traces, this cooldown limit was not exceeded during this event. No heatup was experienced during the event.

Reactor Power

Prior to the reactor trip, reactor power was approximately 11 percent rated thermal power and decreasing. After the trip, power decreased as expected.

Steam Pressure

Before the turbine trip, S/G pressures were being maintained at 1002 psig. After the turbine trip, S/G pressure increased to 1,005 psig and was controlled by steam dump valve modulation. Following the loss of MFW and addition of cold AFW to the S/Gs, the steam pressure fell as saturation pressure followed S/G average temperature. Following the reactor trip, steam pressure stabilized at no-load pressure as RCS Tavg stabilized at 548 degrees F. No TS requirements or UFSAR accident analyses were challenged.

Feedwater Flow

Feedwater flow was steady at 25 percent flow prior to the turbine trip ith all four feedwater regulatory bypass valves in automatic and the four FRVs throttled to 25 percent in manual control. Following the turbine trip, reactor power was reduced to 15 percent to stabilize conditions while dumping steam to the main condenser. As reactor power was decreased, feedwater flow was decreased by the

ANALYSIS OF EVENT (Continued)

operator taking manual control of 1B MFP. Manual control of the pump, in conjunction with approximately a five percent manual closure of the FRVs, resulted in what the operators felt was a stabilized system. However, when the steam supplies to the MFPs were isolated, the result was decreasing feedwater flow and S/G levels. Attempts to manually load the MFPs were unsuccessful.

To restore decreasing S/G levels, the operators initiated AFW flow. A large quantity of cold AFW flow was injected into the S/Gs because of the relatively high reactor power level at the time. Before the S/G levels could be restored however, the Eagle 21 trip time delay expired, and the reactor tripped. The large quantity of cold AFW flow injected just prior to the reactor trip while trying to recover the S/G levels was the cause of the excessive RCS cooldown following the trip.

Steam Flow

Steam flow decreased after the turbine trip and was controlled by modulation of the steam dump valves. When RCS Tavg dropped below 547 degrees F following the loss of MFW flow, the steam dump valves modulated closed. The steam dump valves remained closed until after the reactor tripped and RCS Tavg stabilized at 548 degrees F. The steam dump valves then again modulated to maintain the reference temperature. The steam flow response was bounded by UFSAR accident analyses.

S/G Level

Prior to the event, levels in all four S/Gs were steady at 44 percent. The S/G levels dropped rapidly after MFW flow was lost. The reactor tripped after the low-low level trip setpoint of 13 percent was reached and the trip time delay corresponding to 15 percent power had expired. The S/G levels decreased below the 13 percent trip setpoint prior to the actual reactor trip. AFW flow was unable to restore the levels before the trip time delay expired. AFW flow was throttled when RCS Tavg fell below 547 degrees F. When RCS Tavg had been recovered to 547 degrees F after the reactor trip, AFW flow was increased and normal S/G levels were restored.

Shutdown Margin

Prior to the trip, the reactor was operating with control rods above the minimum insertion limits. Thus, by definition, adequate shutdown margin was available. Following the trip, RCS cooldown occurred as previously described. Adequate shutdown margin was maintained by conformance to 1-ES-0.1 guidelines regarding emergency boration. After the reactor was stabilized, shutdown margin was verified by performance of Surveillance Instruction 38, "Shutdown Margin." No TS or accident analysis assumptions were violated.

NRC Form 366A (6-89)

U.S. NUCLEAR REGULATORY COMMISSION

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CORRECTIVE ACTION

The immediate action taken by the operators was to stabilize the unit in accordance with the governing instructions. A posttrip review team was assembled and an assessment of the cause of the trip and response of the unit was begun.

Several corrective actions have also been implemented as recurrence controls. Operations management has issued a night order clarifying what specific actions should be taken by AUOs following a reactor or turbine trip only with unit operator guidance. This subject has also been discussed with each crew at shift turnover prior to assuming shift. To prevent recurrence of the emergency boration flow problem initially encountered, Operations will revise AOI-34, "Emergency Boration," 1-ES-0.1, and 2-ES-0.1 to have the boric acid tank recirculation line isolated upon initiation of emergency boration. These revisions will be completed by September 1, 1990.

While not contributing directly to this event, a related weakness in AUO training was identified in the investigation following the event. The AUOs were trained to secure steam loads, including steam to the MFPs, following a reactor trip to minimize RCS cooldown. Because of relatively little experience with turbine trips below 50 percent power that did not result in a reactor trip, the TB AUOs were conditioned to react to a turbine trip and reactor trip similarly. However, it is inappropriate to assume a reactor trip will occur after a turbine trip with power at less than 50 percent. Consequently, these facts have been discussed with the AUOs to ensure appropriate posttrip actions are taken. Additionally, Operations will evaluate the adequacy of AUO procedural guidance and training with respect to what independent actions should be taken and the communication of these actions for other events. This evaluation will be completed by September 1, 1990.

Increased monitoring following future refueling outages is expected to minimize the potential for inadvertent generator stator ground fault protection relay actuation.

ADDITIONAL INFORMATION

There have been 17 previously submitted LERs since 1984 reporting reactor trips that occurred as the result of low-low S/G level. However, none of these previous events were the result of loss of MFW flow due to inadequate communication between control room operators and AUOs.

NRC Form 366A
(6-89)

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

COMMITMENTS

1. Operations will revise AOI-34, 1-ES-0.1, and 2-ES-0.1 to have the boric acid tank recirculation line isolated upon initiation of emergency boration. These revisions will be completed by September 1, 1990.

2. Operations will evaluate the adequacy of AUO procedural guidance and training with respect to what independent actions should be taken and the communication of these actions for other events. This evaluation will be completed by September 1, 1990.

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