TENNESSEE VALLEY AUTHORITY

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October 15, 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) 50-327/90021

The enclosed LER provides details of a Unit 1 reactor trip, which occurred on September 14, 1990. This trip was from a low steam generator water level, which was the result of a feedwater transient initiated by a vital inverter failure. This event is being reported in accordance with 10 CFR 50.73(a)(2)(iv) as an automatic reactor protection system actuation.

Very truly yours,

TENNESSEE VALLEY AUTHORITY

J. R. Bynum

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LICENSEE EVENT REPORT (LER)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
Sequoyah Nuclear Plant, Unit 1	0 5 0 0 0 3 2 7	1 OF 0 8

TITLE (4) Sequoyah Unit 1 reactor trip from low steam generator level as a result of a feedwater transient initiated by the failure of a vital inverter during a transfer from the maintenance of the normal power supply.

EVENT DAY (5) MONTH	DAY	YEAR	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	REPORT DATE (7) MONTH	DAY	YEAR	OTHER FACILITIES INVOLVED (8) FACILITY NAMES	DOCKET NUMBER(S)
										0 5 0 0 0
0 9	1 4	9 0	9 0	0 2 1	0 0	1 0	1 5	9 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)
Russell R. Thompson, Compliance Licensing	6 1 5 8 4 3 - 7 4 7 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
X	E F	I N V T	S 2 5 0	Y					

SUPPLEMENTAL REPORT EXPECTED (14)

X YES (If yes, complete EXPECTED SUBMISSION DATE) __ NO

EXPECTED SUBMISSION DATE (15): MONTH 0 2 DAY 0 1 YEAR 9 1

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

On September 14, 1990, with Unit 1 in Mode 1, a reactor trip occurred at 1613 Eastern daylight time. The trip was generated from a low-low steam generator water level signal in Loop 2. The low level was the result of a feedwater transient initiated by the failure of a vital inverter. The inverter failure occurred after the completion of maintenance activities on the inverter and during the transfer of the inverter from its maintenance power supply to its normal power supply. During the transfer, the inverter output voltage dropped to zero because of the random failure of the inverter's silicon-controlled rectifiers. This deenergized the 1-II vital instrument power board. The loss of power resulted in the main feedwater regulator valves closing and the main feedwater pumps dropping to minimum speed. This reduced feedwater flow to all four steam generators. Plant systems responded properly and the shutdown posed no danger to plant employees or the general public. The unit was stabilized in accordance with plant procedures. The vital inverter was repaired and returned to service on September 15, 1990.

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

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

DESCRIPTION OF EVENT

On September 14, 1990, with Unit 1 in Mode 1 (98 percent reactor power, 2235 pounds per square inch gauge [psig], and reactor coolant system [RCS] average temperature 577 degrees Fahrenheit [F]), a reactor trip occurred at 1613 Eastern daylight time (EDT). The trip signal was generated from a low-low steam generator water level signal in Loop 2 (EIIS Code JC), which was the result of a feedwater transient initiated by the failure of a vital inverter in the 120-volt (V), alternating current (ac) instrument power distribution system (EIIS Code EF). (Refer to Updated Final Safety Analysis Report [UFSAR] Figure 8.1.2-2.)

On September 14, 1990, two Transmission and Customer Service (T&CS) technicians were assigned to perform Work Request (WR) C011313 on Voltmeter 1-FI-220-QN2-E, which is located on 1-II vital inverter. The inverter is located on the Unit 1 side of Elevation 749 in the auxiliary building, 480V electric board room.

The inverter was removed from service at 1123 EDT for the replacement of the inverter output voltmeter. Removal of the inverter from service required entry in Technical Specification Limiting Condition for Operation Action 3.8.2.1. The 1-II 120V ac vital instrument power board, which is normally powered from the 1-II vital inverter, was transferred to its maintenance power source (1-B 120V ac instrument power distribution panel).

After completion of WR C011313 at 1530 EDT, the Unit 1 assistant shift operations supervisor (ASOS) proceeded to return the inverter to service using System Operating Instruction (SOI) 250.6, "120V AC Vital Instrument Power." During one of the recovery steps, the ASOS discovered the inverter precharge light would not illuminate as required by procedure. The T&CS engineer checked the precharge voltage with a voltmeter and found it to be correct at approximately 136V ac. A system engineer was notified to also verify correct precharge voltage. At 1600 EDT the precharge voltage was found to be approximately 130V ac. The system engineer determined that the precharge lamp was burned out and made the recommendation to proceed with the restoration of the inverter to service. The ASOS agreed, and the inverter energized as expected.

During subsequent steps to transfer 1-II vital instrument power board from maintenance to normal (to 1-II inverter) the inverter failed, resulting in the 1-II vital instrument power board becoming deenergized. The ASOS immediately recognized the loss of voltage as indicated by a potential lamp and a voltmeter and transferred the 1-II 120V ac vital power board back to the maintenance supply. Subsequent evaluation determined that the inverter's bridge silicon-controlled rectifiers (SCRs) shorted, causing a short circuit of the direct-current (dc) source. The dc circuit fuses blew as a result of the short. This caused the ac output of the inverter to go to zero volts.

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

The loads that are supplied by the 1-II 120V ac vital instrument power board are shown on SQN UFSAR Figure 8.3.1-30. During the period that the 1-II 120V ac vital instrument power board was deenergized (approximately eight seconds), the following feedwater circuits and components were without power.

Process Control Group 2: Flow controller (FC) for Loop 2 regulator valve (references UFSAR Figure 10.4.7-5)

Process Protection Set II: Feedflow and steamflow inputs to main feedpump speed controller and regulator valve positioner (references UFSAR Section 7.7.1.7)

Plugmold Instrument Bus 2: (Panel 1-M-3) Flow indicating controllers (FICs) for Loops 1, 3, and 4 (references UFSAR Figures 10.4.7-5 and 6), and main feed pump turbine master speed controllers and speed indicating controllers.

With the loss of power to these components, the main feedpumps would have dropped to minimum speed decreasing feedwater pressure and flow. In addition, Loops 1, 3, and 4 regulator valves would have begun drifting closed and Loop 2 regulator valve would have begun driving closed. These actions would rapidly result in a loss of steam generator level with the reactor at full power.

After power was restored to the 1-II vital instrument power board, the Loops 1, 3, and 4 regulator valves began to reopen to restore feedwater flow to Loops 1, 3, and 4. The fact that these loops responded was because only the main control room FIC lost its power. The FIC for these loops is a one-to-one repeater of the flow controllers located in the process control groups for each loop. Because the flow controllers did not lose their power, the FICs responded to their inputs as soon as power was restored.

The Loop 2 regulator valve continued to close after power was restored due to the effect of dynamics associated with controller reset and possible electronic saturation of the controllers. During the power loss, the main control room FIC was powered but the flow controllers, feedflow, steam flow, and the level controller inputs for the Loop 2 FC were without power. During this time, the FIC tried to match the FC output (zero). Once power was restored, the protection set inputs for feedflow and steamflow did not return to normal for approximately 10 seconds because of Eagle 21 requiring up to 10 seconds to reinitialize the analog channels. This resulted in the Loop 2 regulator valve FC driving (reset action) to the fully-closed position until the inputs stabilized.

The main feedpump turbines continued to operate at minimum speed after power was restored due to the valves causing the speed program to see a high differential pressure. As a result, the levels in all four steam generators dropped from an initial level of approximately 44 percent to approximately 20 percent of narrow range within 25 seconds after the inverter failed. The Loop 2 steam generator water level reached the low-low trip setpoint of 13 percent of narrow range approximately 30 seconds after the inverter failed.

After the trip, Operations personnel responded using emergency procedures E-0, "Reactor Trip or Safety Injection," ES-0.1, "Reactor Trip Response," and General Operating Instruction (GOI)-3, "Plant Shutdown From Minimum Load to Cold Shutdown," to stabilize the unit. The plant steam dump control system (EIIS Code SE) operated as expected. The auxiliary feedwater (AFW) (EIIS Code BA) pumps started on a main feedwater (MFW) isolation which occurred as designed after the trip when Tavg decreased to 550 degrees F and began injecting greater than 440 gallons per minute (gal/min) to each steam generator. Manual control of AFW was taken when Tavg decreased to less than 547 degrees F in accordance with ES-0.1. Under manual control, the turbine-driven AFW pump (TDAFWP) was ramped to minimum speed, and the motor-driven AFW valves were placed in manual bypass to control flow to each steam generator at 120 gpm. The AFW level control valves (LCVs) performed as expected. However, Loop 2 steam generator was slower in recovering normal level because it had reached a lower level than Loops 1, 3, and 4 as a result of the Loop 2 process control circuit saturation effect coupled with controller reset after power was lost and then restored.

The inverter failure and subsequent reactor trip resulted in several other occurrences of lesser magnitude in the plant. These occurrences are summarized below along with actions taken to address them.

A Unit 1 containment ventilation isolation, an auxiliary building isolation, and a Control Building isolation occurred within seconds previous to the reactor trip. These isolations were caused as a result of power failures on the associated radiation monitors. The isolations were verified to have occurred as designed, and the affected systems were subsequently realigned to normal in accordance with the appropriate procedures.

Subsequent to the reactor trip the balance-of-plant operator observed Loop 2 steam generator Power-Operated Relief Valve (PORV) 1-PCV-1-12 remaining open. The PORV had opened when steam pressure increased following the turbine trip. The operator adjusted 1-PIC-1-13 from 85 percent to approximately 89 percent before the PORV closed. Subsequent maintenance troubleshooting determined that the output voltage from 1-PIC-1-13 was low. The controller was recalibrated and returned to service.

The No. 3 heater drain tank (HDT) LCVs 1-LCV-6-106 A and B appeared to exhibit erratic performance during the transient. The valves were discovered to be binding, and were repaired prior to returning the unit to service.

Main Bank Transformer 1A sudden pressure relay actuated three minutes after the reactor trip causing a high pressure fire protection sprinkler actuation signal to be generated. However, the sprinkler system was isolated for maintenance on leaking valves and did not actuate.

The reactor trip Bypass Breaker A was observed to be blowing fuses after the reactor trip. Electrical Engineering and Electrical Maintenance investigated the cause and found the inertia latch on the breaker was not tripping. The breaker was serviced, and Maintenance Instruction 10.9.2, "Bypass Trip Breaker Type DB50 Inspection Associated with System 99," was performed to verify proper breaker operation.

The Unit 2 main turbine oil tank sprinkler system actuated during the undervoltage condition on the 1-II 120V ac vital instrument power board. The turbine building auxiliary unit operator verified that the sprinkler actuation was not needed and isolated the sprinkler system. The Fire Protection Unit was notified and a fire watch was established.

The control room operator inadvertently reset the "first-out" annunciator when clearing his board to ensure he was not missing new alarms. No one noticed which annunciator had been lit. Subsequently, the investigation team used the sequence of events recorder to determine the event sequence.

During the posttrip review of the sequence of events computer printout and the postmortem computer printout, the posttrip review team members determined that the P-250 computer did not provide a complete sequence of events. Numerous bistable trips and resets from Protection Set II were not recorded, one of which was reactor trip breaker operation. The ramifications of this aspect of the event are still being examined.

CAUSE OF EVENT

The root cause of the reactor trip is attributed to the random failure of the 1-II vital inverter SCRs. The consequent failure of this inverter initiated the feedwater transient, which resulted in the low-low level in Loop 2 steam generator. After replacing the four SCRs in the inverter, the inverter was returned to service. The failure of the SCR is considered to be a random failure unrelated to the maintenance activity performed.

ANALYSIS OF EVENT

This 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 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

Prior to the event, RCS pressure was at or near 2235 psig. When the reactor trip occurred, the pressurizer pressure dropped to approximately 2050 psig within one minute. Pressure recovered to 2250 psig within the following 30 minutes. The decrease in pressure can be attributed to the cooldown. As discussed in the UFSAR Section 15.2.8, the AFW system was capable of removing enough residual heat to prevent overpressurization of the RCS. During the cooldown transient, pressurizer pressure decreased from approximately 2235 to 2050; pressure was returned to normal in approximately 45 minutes.

RCS Temperature

Pretrip average temperature (Tavg) was at 577 degrees F. Posttrip Tavg declined to approximately 545 degrees F. The operator worked within the guidelines of ES-0.1 to control overcooling. Tavg stabilized at approximately 548 degrees F.

Heatup and Cooldown Limits

Technical specifications limit the RCS cooldown rate to 100 degrees F in any one hour time period. This limit was not exceeded. The cooldown rate in this case was 33 degrees F per hour (578 to 545 degrees F in the first hour). No heatup was experienced during the event.

Feedwater Flow

MFW flow was reduced on Loops 1, 3, and 4 by the loss of power to the steam generator level controllers. Power was returned to the level controllers in approximately nine seconds and water level recovery was initiated by the controllers. As a result of the power loss, Steam Generator 2 regulator valve began reducing the water level to match the level programmed value of 33 percent. However, the water level in Steam Generator 2 did not recover sufficiently after power was returned to the controllers to prevent the trip. The reason for this is believed to be caused by Loop 2 process control circuit saturation and the effect of dynamics associated with controller reset after power was lost and then restored.

The AFW system performed as designed. Flow to the steam generators from AFW continued at greater than 440 gal/min for each steam generator as expected while steam generator levels remained below 33 percent. Manual control of AFW was taken by the operators in accordance with ES-0.1.

Steam Flow

Steam flow pretrip was at an expected value of 14.6 x 10E06 pounds per hour and dropped rapidly upon the reactor trip. Following the trip, flow continued to the steam dumps and the TDAFWP from Steam Generator 1 until the TDAFW pump was removed from service. Steam also continued through the steam generator PORV, which remained open longer than intended.

Steam Pressure

Pretrip steam generator pressure was constant at 865 psig. Posttrip, steam generator pressure rose to 1025 psig within one minute, then slowly decreased due to the cooling affect of AFW. The lowest steam generator pressure during the transient is 921 psig. Steam pressure returned to no-load pressure, and Tavg returned to 548 degrees F.

Steam Generator Level

Prior to the event, all steam generator levels varied at or near 44 percent. The Steam Generator 2 level dropped after the inverter failed, as previously discussed. After the reactor tripped, AFW flow was started to recover the levels. Steam Generators 1, 3, and 4 water levels recovered to approximately 35 percent to 40 percent within 1.5 hours. Steam Generator 2 water level recovered to approximately 20 percent in about 2.5 hours. Reasons for the longer recovery period for Steam Generator 2 are provided below.

Regulation Valve 1-FCV-3-48 was experiencing control air leakage prior to the reactor trip. The valve was tested for response and the leakage was determined to not have an effect. The inverter failure resulted in this valve driving closed instead of drifting closed like the other three loops' regulating valves. In addition, Steam Generator 2 PORV remained open longer than intended. A combination of the above resulted in additional mass loss in Steam Generator 2 prior to and following the reactor trip. Therefore, a longer period of time was required to recover the Steam Generator 2 water level by AFW.

Containment Pressure and Temperature/Radiation

No perturbations were observed in containment pressure, temperature, or radiation, but a containment ventilation isolation (CVI) occurred from loss of power to the radiation monitors.

Pressurizer Level

Pressurizer level was constant at 58 percent pretrip. Response of the pressurizer level to the transient closely parallels that of RCS pressure and temperature. Level dropped due to cooling of the RCS.

At approximately 30 minutes into this event, pressure was stabilized per program at approximately 24 percent. Pressurizer level reacted within the bounds of a loss of feedwater flow event as described in the UFSAR.

Shutdown Margin

Pretrip, the reactor was operating above the minimum insertion limits, and by definition, adequate shutdown margin was available. Following the trip, expected cooldown occurred as has been previously discussed. Adequate shutdown margin was maintained in accordance with ES-0.1 and 1-SI-NXX-000-002.0, "Shutdown Margin."

Reactor Power

The reactor was operating at 98 percent before the trip occurred. UFSAR analysis for loss of normal feed flow assumes 102 percent rated thermal power. Upon receipt of the trip signal the shutdown and control banks dropped into the core, and reactor power rapidly decreased as expected.

CORRECTIVE ACTION

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

The vital inverter was repaired and returned to service at 2044 EDT on September 15, 1990. The failure of the SCRs is considered to be a random failure and, accordingly, does not warrant additional recurrence control. The criteria and procedures for transferring vital inverters from maintenance to normal power supplies were reviewed for adequacy. No procedural problems were identified. Additionally, the Loop 2 steam generator PORV setpoint controller was also subsequently recalibrated. The air leak on Steam Generator 2 main feedwater regulator valve was also repaired. Other minor equipment anomalies noted during or after the trip on the steam dump control system, heater drain tank level control system, and the A bypass reactor trip breaker were corrected before the equipment was returned to service. TVA is also evaluating the resetting the first-out annunciator prior to identification of the first-out alarm from a human performance standpoint. As a result of the sudden pressure relay actuation on Main Bank Transformer 1A, the spare transformer was placed in service on September 16, 1990.

A follow-up investigation of P-250 computer performance and capabilities is being conducted under a condition adverse to quality report. A revision of this LER will be issued to update corrective action when that investigation is completed.

ADDITIONAL INFORMATION

There have been no previous reports of reactor trips resulting from the failure of a vital inverter.

COMMITMENT

A follow-up investigation of P-250 computer performance and capabilities is being conducted under a condition adverse to quality report. A revision of this LER will be issued to update corrective action when that investigation is completed.

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