TVA

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Ken Powers
Vice President, Sequoyah Nuclear Plant

August 12, 1994

U.S. Nuclear Regulatory Commission
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Gentlemen:

TENNESSEE VALLEY AUTHORITY - SEQUOYAH NUCLEAR PLANT UNIT 1 - DOCKET NO. 50-327 - FACILITY OPERATING LICENSE DPR-77 - LICENSEE EVENT REPORT (LER) 50-327/94011

The enclosed LER provides details concerning a manual reactor trip and a subsequent inadvertent feedwater isolation. This event is being reported in accordance with 10 CFR 50.73(a)(2)(iv) as a condition that resulted in a manual and automatic actuation of engineered safety features, including the reactor protection system.

Sincerely,

Ken Powers

Ken Powers

Enclosure
cc: See page 2

180102

JE22

cc (Enclosure):

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U.S. NUCLEAR REGULATORY COMMISSION

Approved OMB No. 3150-0104
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LICENSEE EVENT REPORT (LER)

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

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

PAGE (3): 1 OF 0 6

TITLE (4): Manual Reactor Trip and Subsequent Inadvertent Feedwater Isolation

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 7	1 5	9 4	9 4	0 1 1	0 0	0 8	1 2	9 4		0 5 0 0 0
										0 5 0 0 0

OPERATING MODE (9): 1

POWER LEVEL (10): 2 5

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: J. Bajraszewski, Compliance Licensing

TELEPHONE NUMBER: AREA CODE 6 1 5 | 8 4 3 - 7 7 4 9

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	S B	P I C	F 1 8 0	YES					

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 15, 1994, at 1352 Eastern daylight time (EDT), with Unit 1 at approximately 25 percent power and with the turbine offline, a manual reactor trip was initiated. As designed, the reactor trip initiated a feedwater isolation (FWI). Subsequent to the trip, at 1558 EDT, a second FWI occurred. The trip was initiated in accordance with procedures as a result of the loss of a reactor coolant pump (RCP). During implementation of a dispatch switching order (clearance tagging) for main generator voltage regulator repair, an individual inadvertently removed bus-side fuses instead of line-side fuses to the 1D 6.9-kilovolt unit board. The bus fuses provide power to the board's loss-of-voltage relays. When the fuses were removed, a loss of voltage was sensed, resulting in the tripping of motor breakers supplied by the board, including the Loop 4 RCP motor breaker. The condition was caused by personnel error. The individual performing the clearance did not realize that the wrong fuse compartment was accessed until after the fuses were removed. The appropriate disciplinary action was taken with the involved individual. The second FWI appears to have been caused by erratic operation of the Loop 4 steam generator power-operated atmospheric relief valve. Troubleshooting identified a problem with the valve controller. The controller was repaired and returned to service.

I. PLANT CONDITIONS

Unit 1 was in Mode 1 at approximately 25 percent; the turbine was off-line in preparation for main generator voltage regulator repair.

II. DESCRIPTION OF EVENT

A. Event

On July 15, 1994, at 1352 Eastern daylight time (EDT), a manual reactor trip was initiated. As designed, the reactor trip initiated a feedwater isolation (FWI). Subsequent to the trip, at 1558 EDT, a second FWI occurred. The trip was initiated in accordance with procedures as a result of the loss of a reactor coolant pump (RCP) (EIIS Code AB). During implementation of dispatch switching orders (clearance tagging) for main generator voltage regulator (EIIS Code TL) repair, an individual inadvertently removed bus-side fuses instead of line-side fuses to the 1D 6.9-kilovolt (kV) unit board (EIIS Code EA). The bus fuses provide power to the board's loss-of-voltage relays. When the fuses were removed, a loss of voltage was sensed, resulting in the tripping of motor breakers supplied by the board, including the Loop 4 RCP motor breaker. The 1D unit board remained energized throughout the event. Upon hearing the actuation of various turbine building equipment, the individual realized the error and reinstalled the fuses. Subsequent to the trip, Operations personnel reset the FWI associated with the trip to allow shutdown of the turbine-driven auxiliary feedwater pump. Following the FWI reset, a second FWI occurred because of reactor coolant average temperature (T_{avg}) decreasing below 550 degrees Fahrenheit (F). The FWI signal is initiated upon a decrease in reactor coolant below 550 degrees F coincident with open reactor trip breakers. The temperature decrease was attributed to erratic operation of the Loop 4 steam generator (S/G) power-operated atmospheric relief valve (ARV)(EIIS Code SB).

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

The turbine was removed from service for repair of the main generator voltage regulator before the event because of a failed pulse generator card. Although the voltage regulator contained redundant systems, plant management made a conservative decision to replace both firing circuit drawers.

C. Dates and Approximate Times of Major Occurrences

July 15, 1994 at 1101 EDT	The Unit 1 main turbine/generator was taken off-line. The reactor was at approximately 20 percent power.

at 1351 EDT	An assistant shift operations supervisor (ASOS) was implementing a dispatch switching order for the main generator voltage regulator. The ASOS removed bus fuses on the 6.9-kV 1D unit board. Main control room annunciators indicated the motor trips for the Loop 4 reactor coolant pump (RCP) and the 1C condenser circulating water pump.
at 1352 EDT	The control room operators initiated a manual reactor trip in accordance with procedure. A FWI occurred because of the low T_{avg} coincident with a reactor trip as designed.
at 1446 EDT	The No. 4 RCP was returned to service.
at 1543 EDT	The FWI signal was reset in preparation for removing the turbine-driven auxiliary feedwater pump from service.
at 1548 EDT	A second FWI occurred as a result of reactor coolant T_{avg} dropping below 550 degrees F coincident with open reactor trip breakers. The temperature drop was attributed to the lack of modulation control of the Loop 4 S/G power-operated ARV, the valve was found in the full open position.
at 1900 EDT	The unit was stabilized in Mode 3 (hot standby).

D. Other Systems or Secondary Functions Affected

None.

E. Method of Discovery

The reactor trip was manually initiated by the main control room operators as a result of the loss of an RCP. The undervoltage condition of the 1D 6.9-kV unit board, the loss of the RCP, and FWIs were annunciated on the main control room panels.

F. Operator Actions

Control room operators responded as prescribed by emergency procedures. They promptly diagnosed the plant condition and took the actions necessary to stabilize the unit in a safe condition and maintained the unit in the hot standby mode (Mode 3).

G. Safety System Response

The plant responded to the unit trip as designed. After the reactor trip, T_{avg} dropped to a minimum of approximately 535 degrees F. The additional cooldown was a result of cooler main feedwater that existed before the trip because the turbine and extraction steam had been removed from service in conjunction with lower core residual heat. The low residual heat was the result of the trip occurring from 25 percent reactor power, the core being near beginning-of-life, and the lower mechanical heat because the Loop 4 RCP was not running. T_{avg} ultimately stabilized a no-load condition (547 degrees F). Pressurizer level dropped rapidly from approximately 33 percent before the trip to approximately 17 percent following the reactor trip because of the introduction of auxiliary feedwater and the opening of the S/G power-operated ARVs. A letdown isolation occurred, as designed, as a result of pressurizer level dropping to 17 percent. Operations personnel reestablished letdown flow, and pressurizer level was stabilized at approximately 35 percent. The unit trip was initiated as a result of the loss of the Loop 4 RCP. The loss of forced circulation in one reactor coolant loop at the reduced reactor power level was well within the analysis boundaries of the Final Safety Analysis Report (FSAR). No safety system responses were required as a result of the second FWI; however, equipment receiving the signal responded as designed.

II. CAUSE OF EVENT

A. Immediate Cause

The immediate cause of the reactor trip was the loss of the Loop 4 RCP. By procedure, operators were required to trip the reactor.

The immediate cause of the second FWI was a decrease in reactor coolant temperature below 550 degrees F coincident with open reactor trip breakers. The temperature decrease was attributed to the erratic operation of the Loop 4 S/G power-operated ARV.

B. Root Cause

The root cause of the event was personnel error. An ASOS was assigned the duties of performing a switching order to support maintenance work on the main generator voltage regulator. The individual accessed the wrong 6.9-kV unit board and incorrectly removed the wrong fuses, resulting in the loss of the Loop 4 RCP. The individual performing the clearance did not realize that the wrong fuse compartment was accessed until after the fuses were removed. The individual failed to self-check and use the stop, think, act, review (STAR) methodology. Upon hearing the actuation of plant equipment to the reactor trip, he realized the error, reinstalled the fuses, and notified control room personnel of his actions.

The root cause of the second FWI was attributed to erratic operation of the Loop 4 S/G power-operated ARV. Valve operation was erratic because of a failed pressure-indicating controller. The controller was repaired and returned to service.

C. Contributing Factors

Investigation of the reactor trip event identified two contributing factors. The first contributing factor was that the level of detail contained in the switching order did not provide specific location information of components, such as compartment numbers, for the individual to use in performance of the switching order. The second contributing factor was that labeling of the fuse locations for various feeder lines and buses could be enhanced to better identify the component. Although these factors would not have prevented the event, they would have provided additional information to the individual and established additional error prevention barriers to assist the individual in performing the work. These areas are being reviewed for potential enhancements.

Investigation identified a contributing factor to the second FWI. The reactor coolant system was operating very close to the temperature setpoint for the FWI initiating logic. Therefore, after reseting the initial FWI signal, with T_{avg} just above 550 degrees, the potential existed for an FWI. Operations' procedures are being reviewed for enhancement to provide guidance for FWI reset following post-trip recovery.

IV. ANALYSIS OF EVENT

Plant responses during and after the unit trip were consistent with responses described in the FSAR and, accordingly, the event did not adversely affect the health and safety of plant personnel or the general public.

V. CORRECTIVE ACTION

A. Immediate Corrective Action

Control room personnel responded as prescribed by emergency procedures. They promptly diagnosed the plant condition and took the action necessary to stabilize the unit in a safe condition.

B. Corrective Action to Prevent Recurrence

The appropriate disciplinary action was taken with the involved individual.

Troubleshooting of the Loop 4 S/G power-operated ARV identified the failed components. The failed components were replaced and the controller was tested, calibrated, and returned to service.

VI. ADDITIONAL INFORMATION

A. Failed Components

Four transistors in the main amplifier circuit and the automatic/manual switch were replaced in the Foxboro Company pressure indicating controller, Model No. 62HB-5E-OH, for the Loop 4 S/G power-operated ARV.

B. Previous Similar Events

A review of previous similar events identified one similar event (LER 50-328/90017) that resulted from the loss of a unit board and loss of the associated RCP. In that event, the loss of the unit board was caused by sticking contacts on a fast transfer relay. The actions taken for that event would not have prevented the event described by this LER. The search also identified three previous events (50-327/93003, 94005, and 94008) associated with reactor trips as a result of personnel error because of a failure to adequately apply the STAR process. The Operations Improvement Plan was initiated, in part, by these events and addresses personnel performance and management expectations. Management has placed and will continue to place emphasis on the proper application of the self-check methodology through stand-down meetings, safety meetings, coaching/counseling, and monitoring and by holding individuals accountable for their actions.

VII. COMMITMENTS

None.