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February 16, 2000

U.S. Nuclear Regulatory Commission
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10 CFR 50.73

Gentlemen:

TENNESSEE VALLEY AUTHORITY - SEQUOYAH NUCLEAR PLANT (SQN) UNIT 2 - DOCKET NO. 50-328 - FACILITY OPERATING LICENSES DPR-79 - LICENSEE EVENT REPORT (LER) 50-328/2000001

The enclosed report provides details concerning an automatic reactor trip as a result of a loss of power to a vital instrument power board and a safety injection actuation. This event is being reported, in accordance with 10 CFR 50.73(a)(2)(iv), as an event that resulted in an automatic actuation of engineered safety features including the reactor protection system and, in accordance with 10 CFR 50.73(a)(2)(i)(B), as an operation prohibited by technical specifications (TSs). Also, this report includes information required by TS Limiting Condition for Operation 3.5.2 Action b.

Sincerely,

M. Bajestani

Masoud Bajestani

Enclosure
cc: See page 2

IE22

Enclosure
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NRC FORM 366 (6-1998) U.S. NUCLEAR REGULATORY COMMISSION

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LICENSEE EVENT REPORT (LER)

(See reverse for required number of digits/characters for each block)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
Sequoyah Nuclear Plant (SQN) UNIT 2	05000328	1 OF 9

TITLE (4)

Reactor Trip Caused from a Low-Low Steam Generator Level resulting from a Static Switch Control Board Circuit Failure

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 NAME	DOCKET NUMBER
01	18	2000	2000	-- 001 --	00	02	16	2000	SQN Unit 1	05000327
									FACILITY NAME: NA	DOCKET NUMBER: 05000

OPERATING MODE (9)	1	THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more) (11)							
POWER LEVEL (10)	100		20.2201(b)		20.2203(a)(2)(v)	X	50.73(a)(2)(i)		50.73(a)(2)(viii)
			20.2203(a)(1)		20.2203(a)(3)(l)		50.73(a)(2)(ii)		50.73(a)(2)(x)
			20.2203(a)(2)(i)		20.2203(a)(3)(ii)		50.73(a)(2)(iii)		73.71
			20.2203(a)(2)(ii)		20.2203(a)(4)	X	50.73(a)(2)(iv)	X	OTHER
			20.2203(a)(2)(iii)		50.36(c)(1)		50.73(a)(2)(v)		Specify in Abstract below or in NRC Form 366A
			20.2203(a)(2)(iv)		50.36(c)(2)		50.73(a)(2)(vii)		

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER (Include Area Code)
J. W. Proffitt, Licensing Engineer	(423) 843-6651

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13)

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO EPIX		CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO EPIX
X	EF	CBD	S250	Y						
X	BA	BKR	AS04	Y						

SUPPLEMENTAL REPORT EXPECTED (14)				EXPECTED SUBMISSION DATE (15)	MONTH	DAY	YEAR
	YES (If yes, complete EXPECTED SUBMISSION DATE).	X	NO				

Abstract (Limit to 1400 paces, i.e., approximately 15 single-spaced typewritten lines) (16)

On January 18, 2000, at approximately 1051 Eastern standard time (EST), an automatic reactor trip occurred. The reactor trip was initiated as a result of a low-low steam generator (SG) level. A momentary loss of power to Vital Instrument Power Board 2-IV occurred when plant personnel were performing a modification to the plant that involved lifting a lead on a synchronization signal cable for the static switch on Vital Inverter 2-IV. The cause of the power loss was a circuit failure on the static switch control board. Additionally, an inadvertent safety injection (SI) actuation occurred on A-train. Investigation of the SI actuation indicates that this signal was caused by low SG pressure signals on Channels IV and II for SG No. 4. The Channel IV signal was attributed to the loss of power from Vital Instrument Power Board 2-IV and the Channel II signal was attributed to a noisy SG pressure transmitter. The control board to the static switch was replaced. This report includes information required by Technical Specification Limiting Condition for Operation 3.5.2 Action b for an emergency core cooling system injection.

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Sequoyah Nuclear Plant (SQN) Unit 2	05000328	2000 --	001 --	00	2 OF 9

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

I. PLANT CONDITION(S)

Unit 2 was in power operation at approximately 100 percent.

II. DESCRIPTION OF EVENT

A. Event:

On January 18, 2000, at approximately 1051 Eastern standard time (EST), an automatic reactor trip occurred. The reactor trip was initiated by a low-low steam generator (SG) level [EIIS Code AB]. A momentary loss of power to Vital Instrument Power Board 2-IV [EIIS Code EE] occurred when plant personnel were performing a modification to the plant that involved lifting a lead on a synchronization signal cable for the static switch on Vital Inverter 2-IV [EIIS Code EF]. The static switch control board circuit failed causing a transfer of the inverter power supply to a deenergized bus resulting in the loss of power. The circuit design should have prevented a transfer to a deenergized bus. The loss of power to the vital instrument board resulted in a loss of power to SG No. 4 feedwater regulating valve. The loss of power to the regulating valve caused the valve to go closed resulting in a loss of feedwater to the SG.

In addition to the reactor trip signal, an inadvertent safety injection (SI) actuation occurred on A-train. Investigation of the SI actuation indicates that this signal was caused by low SG pressure signals on Channels IV and II for SG No. 4. The Channel IV signal was attributed to the loss of power from Vital Instrument Power Board 2-IV and the Channel II signal was attributed to a noisy SG pressure transmitter.

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

None.

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

C. Dates and Approximate Times of Major Occurrences:

January 18, 2000, at 1051 EST — A momentary loss of power to Vital Instrument Power Board 2-IV occurred when plant personnel were performing a modification to the plant that involved lifting a lead on a synchronization signal cable for the static switch on Vital Inverter 2-IV.

at 1051 EST — A reactor trip occurred with a turbine trip and a subsequent SI on A-Train. The main control room operators took appropriate actions to manually start and align the appropriate B-Train equipment and stabilized the reactor in Mode 3 (hot standby).

at 1051 EST — Operations entered the following limiting condition for operations (LCOs): LCO 3.7.1.2, Action A for the 2A auxiliary feedwater (AFW) pump motor trip, LCO 3.5.5 for the refueling water storage tank (RWST) being below the required level as a result of an SI actuation, LCOs 3.6.4.1.a and 3.3.3.1 for the containment radiation monitors isolating as a result of the containment ventilation isolation, LCO 3.0.3 for both trains of SI automatic logic being disabled because of the SI actuation signal being reset.

at 1051 EST — Operations entered LCO 3.0.3 on Unit 1 for the emergency gas treatment system (EGTS) being aligned only to Unit 2 as a result of the SI actuation.

at 1258 EST — Operations exited LCOs 3.6.4.1.a and 3.3.3.1 after returning the containment radiation monitors to service.

at 1310 EST	Operations exited LCO 3.5.5 following restoring the RWST above the required level.
at 1423 EST	The reactor trip breakers were cycled enabling the automatic SI actuation signal. Operations exited LCO 3.0.3 on Unit 2.
at 1454 EST	Operations returned EGTS to standby and exited LCO 3.0.3 on Unit 1.
January 19, 2000, at 0220 EST	Operations exited LCO 3.7.1.2, Action A when the 2A AFW pump was returned to service and declared operable.
at 1827 EST	Operations entered LCO 3.8.2.1.b when Vital Inverter 2-IV was placed in bypass for troubleshooting.
at 2310 EST	Operations exited LCO 3.8.2.1.b when testing of Vital Inverter 2-IV was completed and the inverter was declared operable.

D. Other Systems or Secondary Functions Affected:

None.

E. Method of Discovery:

The reactor and turbine trips were annunciated on the main control room panels.

F. Operator Actions:

Control room operators responded to the reactor and turbine trips as prescribed by emergency procedures. They promptly diagnosed the condition and took appropriate actions to manually start and align the appropriate B-Train equipment and stabilize and maintain the unit in a safe condition. Operations entered the applicable LCOs

associated with the loss of power to the vital board and the SI actuation. It was noted that the operators failed to start an essential raw cooling water (ERCW) pump as required by the emergency procedures. Three ERCW pumps were running and a fourth pump should have been verified running by procedure.

G. **Safety System Responses:**

The reactor protection systems, including feedwater isolation and AFW start, responded to the trip as expected, with the loss of Vital Instrument Power Board 2-IV. The A-Train equipment started as designed except the 2A-A AFW pump failed to start as a result of the breaker failing to close. The B-Train equipment was manually started by the operators except for the B-Train ERCW pump. The SI signal initiated a Phase A signal and started the diesel generators. The low SG pressure initiated the signal to close the main steam isolation valves (MSIVs), as designed.

III. CAUSE OF THE EVENT

A. **Immediate Cause:**

The immediate cause of the unit trip was the initiation of a reactor trip signal when the feedwater regulating valves closed resulting in a loss of feedwater to the No. 4 SG. The reactor trip signal was initiated as a result of a low-low SG level.

B. **Root Cause:**

The root cause of the unit trip was the failure of a circuit board on the static switch on Vital Inverter 2-IV. The inverter control circuit failed causing the loss of power to Vital Instrument Board 2-IV.

IV. ANALYSIS OF THE EVENT

When the reactor trip occurred, the pressurizer level was at approximately 60 percent. Following the reactor trip, pressurizer level decreased to approximately 45 percent and then increased off-scale following the SI actuation on low SG pressure. Operator action terminated the SI before the pressurizer reached a water solid condition. Pressurizer level was returned within the range of the level instrumentation with a continuing trend to its no-load value. Pressurizer pressure was maintained within the safety analysis limits. The reactor coolant system (RCS) average temperature decreased from the normal operating value of approximately 578 degrees Fahrenheit (F) to approximately 530 degrees F following the reactor trip. The loss of nuclear heat generation, SI flow, the introduction of cold AFW, and the opening of the atmospheric relief valves to remove decay heat resulted in the cooldown. Following operator actions to terminate SI flow and control heat removal through the secondary, RCS temperature was restored to its no-load value of approximately 550 degrees F. Reactor coolant temperatures remained within those evaluated in the safety analysis report.

Following the reactor trip, steam pressure increased to approximately 1040 pounds per square inch gauge (psig) resulting in steam being released through the SG atmospheric relief valves (ARVs). The steam dumps were not available because the MSIVs were closed. This contributed to the lifting of the ARVS. The introduction of SI flow, cold AFW, and opening of the ARVs reduced steam pressure to approximately 830 psig. Pressure in the SGs then increased to near the 1025 psig setpoint of the SG ARVs as decay heat generation and secondary side heat removal became matched. The technical specifications and Final Safety Analysis Report (FSAR) requirements on SG pressure were not challenged.

The failure of the 2A-A AFW pump to start was evaluated and determined not to affect flow to the SGs. AFW flow to the SGs increased rapidly after start of the turbine-driven AFW pump and the 2B-B pump. Adequate AFW flow was delivered by the operating pumps to remove decay heat and ensure that the technical specifications and FSAR requirements were met.

The failure to start an ERCW pump was evaluated and determined not to affect heat removal capabilities of the safety systems. Three ERCW pumps were running and providing sufficient cooling water flow.

The plant safety systems responses during and after the unit trip were bounded by the responses described in the FSAR. Therefore, this event did not adversely affect the health and safety of plant personnel or the general public.

V. CORRECTIVE ACTIONS

A. Immediate Corrective Actions:

The emergency procedures were entered as a result of the reactor trip and SI actuation. The operators quickly identified the SI actuation as inadvertent and followed the emergency procedures for unit recovery. In addition, operator response effectively handled the unexpected system responses during the unit trip. The operators stabilized the unit in a safe condition (Mode 3).

B. Corrective Actions to Prevent Recurrence:

The control board on the static switch was replaced. Testing of the new control board was completed to ensure that the board would function properly. The 120-VAC Vital Instrument Power Board 2-IV was returned to operable status.

An analysis of the control board circuit is being performed to determine the reason for the failure.

The SG pressure transmitter that indicated a low steam pressure was replaced.

VI. ADDITIONAL INFORMATION

A. Failed Components:

An AFW pump breaker [Asea Brown Boveri (ABB), Model No. ABB 7.5 HK 1000-1200] failed to function as required during the event. The breaker was replaced and testing is being performed to determine why the breaker failed.

The Static Switch control board on the 2-IV inverter (Solid State Controls, Model No. 80-9212635-90) failed, resulting in the loss of power to Vital Instrument Power Board 2-IV.

B. Previous LERs on Similar Events:

A review of previous reportable events identified three previous events (LER 50-327/1998003, LER 50-327/90021, and LER 50-327/1998001) involving the vital inverters.

LER 50-327/90021 involved an inverter failure during 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 failure of the inverter's silicon-controlled rectifiers.

LER 50-327/1998001 was a failure of an alternate feeder breaker to a shutdown board. The loss of the shutdown board caused a voltage transient tripping Vital Inverter 1-I.

LER 50-327/1998003 involved an inverter component failure in the bridge circuit. This caused the inverter to trip resulting in a loss of power to Vital Instrument Power Board 1-IV.

Although these previous events were associated with vital inverter problems, the corrective actions could not have prevented this event.

C. **Additional Information:**

Technical Specification LCO 3.5.2, Action b requires reporting the total actuation cycles to date following an ECCS actuation and the current value of the usage factor for each affected SI nozzle whenever its value exceeds 0.70. There have been 12 actuation cycles to date and the usage value is less than 0.70.

During the review of previous similar events, it was identified that the ECCS actuation cycles and usage value was not reported in LER 50-327/92011 dated May 28, 1992, and 50-328/92011 dated September 21, 1992. The Unit 1 event (LER 50-327/92011) total actuation cycles was 7 and the usage value was less than 0.70. The total actuation cycles for the Unit 2 event (LER 50-328/92011) was 11 and the usage value was less than 0.70.

D. **Safety System Functional Failure:**

This event did not result in a safety system functional failure in accordance with NEI 99-02.

VII. COMMITMENTS

None.