

LICENSEE EVENT REPORT (LER)

FACILITY NAME (1) SAN ONOFRE NUCLEAR GENERATING STATION, UNIT 3	DOCKET NUMBER (2) 0 5 0 0 0 3 6 2	PAGE (3) 1 OF 0 3
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TITLE (4)
UNIT 3 TRIP DUE TO TURBINE TRIP

EVENT DATE (5)			LER NUMBER (6)			REPORT DATE (7)			OTHER FACILITIES INVOLVED (8)		
MONTH	DAY	YEAR	YEAR	SEQ. NUMBER	REV. NUMBER	MONTH	DAY	YEAR	FACILITY NAMES		DOCKET NUMBER(S)
0 1	1 2	8 6	8 6	0 0 1	0 0	0 2	1 1	8 6			0 5 0 0 0
0 5 0 0 0											

OPERATING MODE (9) 1	THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more of the following) (11)									
POWER LEVEL (10) 0 4 4	<input type="checkbox"/> 20.402(b)	<input type="checkbox"/> 20.405(c)	<input checked="" type="checkbox"/> 50.73(a)(2)(iv)	<input type="checkbox"/> 73.71(b)						
	<input type="checkbox"/> 20.405(a)(1)(i)	<input type="checkbox"/> 50.36(c)(1)	<input type="checkbox"/> 50.73(a)(2)(v)	<input type="checkbox"/> 73.71(c)						
	<input type="checkbox"/> 20.405(a)(1)(ii)	<input type="checkbox"/> 50.36(c)(2)	<input type="checkbox"/> 50.73(a)(2)(vii)	OTHER (Specify in Abstract below and in Text, NRC Form 366A)						
	<input type="checkbox"/> 20.405(a)(1)(iii)	<input type="checkbox"/> 50.73(a)(2)(i)	<input type="checkbox"/> 50.73(a)(2)(viii)(A)							
	<input type="checkbox"/> 20.405(a)(1)(iv)	<input type="checkbox"/> 50.73(a)(2)(ii)	<input type="checkbox"/> 50.73(a)(2)(viii)(B)							
<input type="checkbox"/> 20.405(a)(1)(v)	<input type="checkbox"/> 50.73(a)(2)(iii)	<input type="checkbox"/> 50.73(a)(2)(x)								

LICENSEE CONTACT FOR THIS LER (12)		TELEPHONE NUMBER	
NAME H. E. MORGAN, STATION MANAGER		AREA CODE 7 1 4	3 6 8 - 6 2 4 1

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

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NRC	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NRC

SUPPLEMENTAL REPORT EXPECTED (14)			EXPECTED SUBMISSION DATE (15)	MONTH	DAY	YEAR
<input type="checkbox"/> YES (If yes, complete EXPECTED SUBMISSION DATE)	<input checked="" type="checkbox"/> NO					

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

On 1/12/86 at 1735 with Unit 3 in Mode 1 at 44% power, and a power increase in progress, the Main Turbine tripped when a Loss of Load signal was generated by the Turbine Supervisory System (TSS). The turbine trip initiated a reactor trip according to design.

The TSS Loss of Load signal was erroneously generated as a result of an improper switch alignment which prevented input of existing generator load to the TSS.

The root cause of the event was the failure of an operator to inform the Control Room Supervisor of a perceived discrepancy in a switching order and a failure to confirm plant configuration with as-built drawings.

In addition to disciplinary action taken with respect to the operators involved, corrective action to prevent recurrence includes: a required reading assignment for all operators reviewing the event; revising Operating Instructions S023-6-5 "Main and Auxiliary Transformer Operation;" issuing a memorandum to all operations supervisors emphasizing "CRS Involvement in Control Room Activities;" providing signage to label the location of switches; incorporation of lessons learned from the event into the appropriate training programs.

Insofar as the Turbine Supervisory System is a nonsafety-related system and all safety-related systems responded to the turbine trip in accordance with design, there was no safety significance to this event.

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TEXT CONTINUATION

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

On 1/12/86 at 1735 with Unit 3 in Mode 1 at 44% power, and a power increase in progress, the Main Turbine (EIIS System Code TA)(EIIS Component Code TRB) tripped when a Loss of Load signal was generated by the Turbine Supervisory System (TSS) (EIIS System Code IT). According to design, the turbine trip initiated a reactor trip.

The TSS Loss of Load trip circuitry compares reactor power and secondary system heat removal capacity. When reactor power is more than 10% above the secondary system heat removal capacity a turbine trip signal is generated. The secondary system heat removal capacity is a function of the number of Steam Bypass Control System (SBCS)(EIIS System Code JI) valves in service, and the main generator load. The turbine Loss of Load trip signal was generated because the generator load to the Loss of Load trip circuit appeared to be zero, rather than the actual generator load of 400 MWe, due to a switch located in a turbine panel being left in the open position. In the absence of any generator output signal, total secondary system heat removal capacity calculated by the TSS is dependent upon the availability of the turbine bypass valve capacity. With three of the four SBCS valves in service (HV-8423 was out of service) the SBCS is capable of removing the equivalent of approximately 34% reactor power. When reactor power reached 44% (10% greater than secondary system heat removal capacity) a turbine Loss of Load trip signal was therefore generated and tripped the main turbine.

The Plant Protection System (PPS) (EIIS System Code JC) reactor Loss of Load bistables are normally set at 55% power which would preclude a turbine initiated reactor trip at 44% power. However, due to SBCS valve HV-8423 being out of service the PPS reactor Loss of Load bistables were set to 44%, whereby a turbine trip initiated a reactor trip.

There were no Engineered Safety Features Actuation System (ESFAS) (EIIS System Code JE) actuations, and all systems performed as anticipated during a reactor trip.

The cause of the misaligned switch was operator error. In preparation for maintenance to be performed during the Unit 3 refueling outage the Main Generator and Transformer System (EIIS System Code TB) was properly removed from service in accordance with Operating Instruction S023-6.5, "Main and Auxiliary Transformer Operation." Upon returning the Main Generator and Transformer System to operation a switching order was prepared which included closing six switches associated with the Main Generator and TSS. A qualified Plant Equipment Operator (PEO) was assigned as the "electrical switcher" and a second PEO was assigned as "electrical checker." During the performance of carrying out the switching order the PEOs could only find five of the six switches, and they believed the switching order was in error. After discussing the situation with the common Control Operator (CO), the CO also believed the switching order was in error. The CO and the PEOs did not refer to the appropriate drawings to determine the as built plant configuration. The Control Room Supervisor (CRS) was not informed of the apparent discrepancy. The sixth switch was located in a Turbine Panel (on the lower level of the Control Building) and is the generator load input to TSS panel 3L-183.

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TEXT CONTINUATION

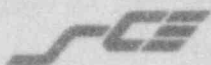
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TEXT (If more space is required, use additional NRC Form 366A's) (17)

The root cause of the event was the failure of an operator to inform the Control Room Supervisor of a perceived discrepancy in a switching order and a failure to confirm plant configuration with as-built drawings.

In addition to disciplinary action taken against the operators involved, corrective action to prevent recurrence includes; a required reading assignment for all operators reviewing the event and its root cause; revising Operating Instruction S023-6-5 "Main and Auxiliary Transformer Operation" to explicitly define the location of all switches in the Main Generator and Transformer System; issuing a memorandum to all operations supervisors emphasizing "CRS Involvement in Control Room Activities;" providing signage to label the location of switches associated with the Main Generator TSS to assist in operator performance of such alignments; incorporation of lessons learned from the event into the appropriate training programs.

Insofar as the Turbine Supervisory System is a nonsafety-related system and all safety-related systems responded to the turbine trip in accordance with design, there was no safety significance to this event.



Southern California Edison Company

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H. E. MORGAN
STATION MANAGER

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February 11, 1986

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

Subject: Docket No. 50-362
30-Day Report
Licensee Event Report No. 86-001
San Onofre Nuclear Generating Station, Unit 3

Pursuant to 10 CFR 50.73(a)(2)(iv), this submittal provides the required 30-day written Licensee Event Report (LER) for an occurrence involving an actuation of the Reactor Protection System. Neither the health and safety of plant personnel nor the health and safety of the public was affected by this event.

If you require any additional information, please so advise.

Sincerely,

Enclosure: LER No. 86-001

cc: F. R. Huey (USNRC Senior Resident Inspector, Units 1, 2 and 3)

J. B. Martin (Regional Administrator, USNRC Region V)

Institute of Nuclear Power Operations (INPO)

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