

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

TITLE (4) REACTOR TRIP ON LOSS OF LOAD

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)
09	04	86	86	013	00	10	06	86			05000
THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more of the following) (11)											

OPERATING MODE (9) 1	20.402(b)	20.405(c)	X	50.73(a)(2)(iv)	73.71(b)
POWER LEVEL (10) 90	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: H. E. MORGAN, STATION MANAGER  
 TELEPHONE NUMBER: 714 368-6241

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)  
 YES (if yes, complete EXPECTED SUBMISSION DATE)  NO  
 EXPECTED SUBMISSION DATE (15)

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

On 9/4/86, at 1255, with Unit 3 at 90% power, a reactor trip on loss of load occurred when the power supply to the Turbine Protection Panel was momentarily interrupted, causing a Turbine Trip. The trip recovery proceeded normally, and there were no safety consequences associated with this event.

The loss of power occurred while manually aligning an alternate power supply to nonsafety-related 125 VDC bus 3D5. When circuit breaker 3D5-07 was closed, the adjacent circuit breaker, 3D5-06, opened, de-energizing the Turbine Protection Panel. Breaker 3D5-06 was removed for bench testing and no deficiencies in the breaker were identified. Based upon reconstruction of the event, it is believed that while 3D5-07 was being closed, the operator inadvertently made contact with the handle of 3D5-06, causing it to trip. The event was discussed during shift briefings with operating personnel to stress the need to use caution when closing circuit breakers to avoid overtravel.

During the trip recovery the power supply circuit breaker for DC Motor Operated Auxiliary Feedwater Valve 3HV-4706 tripped due to the breaker instantaneous current trip being set with insufficient margin to allow for variation in component operating parameters. All remaining components of the Auxiliary Feedwater System operated as designed.

The breaker for 3HV-4706 was replaced with one having a correct setpoint, and the power supply circuit breakers for the remaining DC Motor Operated valves in the Auxiliary Feedwater System were inspected and found to be correctly set.

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TEXT CONTINUATION

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SAN ONOFRE NUCLEAR GENERATING STATION, UNIT 3	0 5 0 0 0 3 6 2	8 6	- 0 1 3	- 0 0	0 2	OF	0 3

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

On September 4, 1986, at 1255, with Unit 3 at 90% power, the reactor tripped on loss of load from the Plant Protection System (EISS System Code JC) when the power supply to the Turbine Protection Panel (EISS System Code JJ) was momentarily interrupted, closing the Turbine stop and governor valves. As designed, an Emergency Feedwater Actuation Signal (EFAS) (EISS System Code BA) was initiated by the resulting low Steam Generator water levels. The trip recovery proceeded normally, and there were no safety consequences associated with this event.

The loss of power occurred while manually aligning an alternate power supply to nonsafety-related 125 VDC bus 3D5 (EISS System Code EI) (EISS Component Code BU) to allow maintenance to be performed on the associated battery charger, 3B005 (EISS Component Code BYC). When 125 VDC circuit breaker 3D5-07 (EISS Component Code 72) was closed, the adjacent circuit breaker, 3D5-06, opened, de-energizing 125 VDC Distribution Panel 3D5P4, (EISS Component Code PL) which supplies the Turbine Protection Panel. Although circuit breaker 3D5-06 was immediately reclosed, the momentary interruption in power caused the turbine to trip.

The handle of circuit breaker 3D5-07 is adjacent to the handle of 3D5-06, and when closing 3D5-07, it is moved towards the handle of 3D5-06. When closing the breaker, a firm force is required to charge the breaker trip spring, while only slight contact with a closed breaker handle can result in the breaker tripping open. Based upon reconstruction of the event, it is believed that in closing 3D5-07 the operator inadvertently made contact with the handle of 3D5-06, causing it to trip. Breaker 3D5-06 was removed for inspection, and bench tested. No deficiencies in the breaker were identified. Additionally, it was confirmed that a slight force on the handle can trip the breaker. The event was discussed during shift briefings with operating personnel to stress the need to use caution when closing circuit breakers to avoid overtravel.

During the recovery from the trip, circuit breaker MS-4706 (ITE Gould molded case circuit breaker), for Auxiliary Feedwater Valve 3HV-4706 (EISS System Code BA), tripped while the valve was being opened by EFAS. 3HV-4706 is a 125 VDC powered motor operated valve on the discharge of the Turbine Driven Auxiliary Feedwater Pump to Steam Generator E089. The redundant valve to Steam Generator E088, and both trains of A.C. motor operated auxiliary feedwater valves operated as designed during the event.

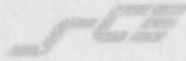
The MS-4706 circuit breaker was tested and it was found that the instantaneous current trip setpoint was about 14 amps, which was very close to the measured valve motor operator starting current. The instantaneous current trip setpoint, which was set during the breaker's initial start-up testing, should have been in the range of 22-24 amps. The procedure used during the start-up testing program was incorrectly applied in this instance, and resulted in the setting being adjusted to 14 amps. The valve had been successfully operated and tested on numerous occasions since start-up. On July 28, 1986, the valve motor operator was adjusted to increase the closing torque as a result of the Motor Operated Valve Analysis and Test System (MOVATS) activity required by IE Bulletin 85-03, which slightly increased the valve opening motor current. Although successful stroke testing had been conducted after completion of the adjustment, and again on August 24, 1986, the margin between the normal valve opening motor starting current and the trip setpoint was decreased such that nominal variations in DC bus voltage were the probable cause of the breaker trip on September 4.

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

The breaker was replaced with one having an instantaneous trip setpoint of 24 amps and the power supply circuit breakers for the remaining DC Motor Operated Auxiliary Feedwater valves were inspected and found to be correctly set. After removal, the original DC breaker tested satisfactorily throughout its adjustment range, confirming that the instantaneous current trip had been incorrectly set. A review of the start-up records for these valves determined that one other power supply circuit breaker had initially been set with a low trip setting, however, it had been subsequently reset. Additional circuit breaker setpoints will be checked in conjunction with the performance of MOVATS testing to assure that this is an isolated event. The test procedure for molded case circuit breakers is being upgraded to assure specific criteria for circuit breaker settings is included to reduce the potential for incorrect setpoint adjustments.



*Southern California Edison Company*

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October 6, 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. 86-013  
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,

*H.E. Morgan/mtdw*

Enclosure: LER No. 86-013

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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