REGULATORY INFORMATION DISTRIBUTION SYSTEM (RIDS)

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FACIL: 50-362 San Onofre Nuclear Station, Unit 3, Southern Californ 05000362
AUTH. NAME AUTHOR AFFILIATION
MORGAN, H. E. Southern California Edison Co.
RECIP. NAME RECIPIENT AFFILIATION

SUBJECT: LER 87-011-00: on 870621, reactor tripped on low steam generator water level. Caused by return of power to Instrument Bus 1. Complete insp & tightening of all bus bar & external connections conducted. W/870828 ltr.

DISTRIBUTION CODE: IE22D COPIES RECEIVED: LTR 1 ENCL 1 SIZE: 8
TITLE: 50.73 Licensee Event Report (LER), Incident Rpt, etc.

NOTES: ELD Chandler 1cy. 05000362

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	AEOD/DSP/ROAB	2	2	AEOD/DSP/TPAB	1	1
	DEDRO	1	1	NRR/DEST/ADS	1	0
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	NRR/DREP/RPB	2	2	NRR/PMAS/ILRB	1	1
	REG FILE 02	1	1	RES DEPY GI	1	1
	RES TELFORD, J	1	1	RES/DE/EIB	1	1
	RGN5 FILE 01	1	1			
EXTERNAL:	EG&G GROH, M	5	5	H ST LOBBY WARD	1	1
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(9/83)

U.S. NUCLEAR REGULATORY COMMISSION
APPROVED OMB NO. 3150-0104
EXPIRES: 8/31/85

LICENSEE EVENT REPORT (LER)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
SAN ONOFRE NUCLEAR GENERATING STATION, UNIT 3	0 5 0 0 0 3 6 2	1 OF 0 6

TITLE (4)
REACTOR TRIP ON LOW STEAM GENERATOR WATER LEVEL

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 6	2 1	8 7	8 7	0 1 1	0 1	0 8	2 8	8 7		0 5 0 0 0
										0 5 0 0 0

OPERATING MODE (9): 1
POWER LEVEL (10): 1 0 0

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)	[X] 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)
H. E. MORGAN, STATION MANAGER	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	MANUFAC-TURER	REPORTABLE TO NPRDS	CAUSE	SYSTEM	COMPONENT	MANUFAC-TURER	REPORTABLE TO NPRDS
X	B Q	F I T	F 1 8 0	Y					
X	B Q	S N B	P 0 2 9	Y					

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 June 21, 1987 at 0258, with Unit 3 in Mode 1 at 100% power, the reactor automatically tripped on low steam generator (SG) water level. The low SG water level was caused by an intermittent loss of power in one phase of a 120 VAC Non-1E Instrument Bus which resulted in the inability to control Main Feedwater and the consequent reduction in SG water level. Level continued to decrease to the low level reactor trip and emergency feedwater actuation setpoint. Following the reactor trip, 120 VAC Non-1E power returned and main feedwater flow resumed. The water level in SG E088 increased from the low level trip setpoint to above the high level alarm setpoint as operators were implementing their immediate post-trip actions in accordance with Emergency Operating Instructions (EOIs). This resulted in cooling down of the Reactor Coolant System (RCS) to below the Safety Injection Actuation Signal (SIAS) setpoint.

The 120 VAC power malfunction was determined to be due to a loose bolt connecting the "B" phase of Instrument Bus #1 to the main bus bars of the non-1E Uninterruptable Power Supply (UPS) main distribution switchboard, which resulted in intermittent loss of circuit continuity. This was evidenced by arcing and pitting at the connection, and confirmed by subsequent duplication of power interruptions when the assembly was manually moved. The loose connection was repaired and a complete inspection of all remaining UPS connections was completed. A complete inspection and tightening of all bus bar and external connections on the Non-1E instrument busses was conducted.

The Unit was stabilized in Mode 3 by operator action. The health and safety of plant personnel and the public was not affected by this event.

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At approximately 0250 on June 21, 1987, with Unit 3 in Mode 1 at 100% power, a momentary interruption of instrumentation electrical power occurred, affecting the #2 main Feedwater Control System (FWCS) (EIIS System Code JB) which serves Steam Generator (SG) E088 (EIIS Component Code SG). This interruption resulted in the Control Room annunciation of a spurious Reactor Trip Override (RTO) alarm for E088, which then promptly cleared. The RTO circuitry automatically reduces main feedwater flow to the SGs following a reactor trip. Although this momentary interruption did not result in reduction in flow, discussions were immediately held among the Control Room operators and the Shift Superintendent. These discussions concluded that, in the event of a sustained spurious RTO signal, manual control should immediately be taken of the feedwater flow to SG E088. This manual control appeared to have been successful when a spurious RTO signal had been received for a brief period on Unit 3 approximately one month earlier. (Investigation with the unit on-line following the earlier spurious RTO signal had not been successful in locating the cause, however supplemental monitoring and recording instrumentation had been maintained in place to assist in locating the cause if it should reoccur. The more thorough investigation conducted with the unit off-line, as described below, concluded that both the earlier event, and the June 21, 1987, event were caused by the same problem.)

At approximately 0257, a large number of Control Room alarms associated with a partial loss of power to 120 VAC Non-1E Instrument Bus No.1 (EIIS Component Code EBU)(EIIS System Code EE), occurred. Subsequent analysis shows that power to the "B" phase of the bus was interrupted and returned three times during an initial 80 second period. The first interruption was for 6 seconds, followed by a return of power for 2 seconds. The second interruption was for 4 seconds. The third interruption began 36 seconds after the first interruption began, and it lasted for 30 seconds.

FWCS #2 is powered from the "B" phase of Instrument Bus #1, and the only symptom noted in the Control Room resulting from the initial, momentary power interruption at 0250 had been the spurious RTO signal for SG E088. However, all feedwater pump and valve controllers for both steam generators, as well as other non-safety-related equipment, control instrumentation and Control Room indication are powered from this source. Therefore, during each interruption of power in the "B" phase of Instrument Bus #1, air-operated feedwater regulating valves (EIIS System Code SJ)(EIIS Component Code FCV), and bypasses, associated with both steam generators, moved toward the fully closed position, and Main Feedwater Pump (MFWP)(EIIS System Code SJ)(EIIS Component Code P) speed decreased for both pumps.

As they had planned in connection with receipt of a sustained spurious RTO signal, the operators took manual control of the SG E088 feedwater regulating valves and their bypasses, and attempted to maintain SG level. Manual control was also taken of MFWP P062 speed, based on the continuing decrease in level in both steam generators. However, as the extent of the intermittent loss of power precluded either automatic or manual control of both trains of feedwater, and resulted in slow-down of both MFWPs, level could not be maintained, and the reactor tripped on low SG level 12 seconds after the end of the third power interruption. As a result of low water level in both steam generators, the Auxiliary Feedwater (AFW) System (EIIS System Code BA) started and all three pumps, and associated valves, operated normally. At this time, feedwater control to E088 and MFWP P062 speed were in manual, as the operators had been attempting to maintain levels above the reactor trip setpoint (i.e., Valve position and pump speed were set for maximum feedwater flow). It is apparent from review of post trip computer printouts and chart recordings that MFWP P063 had also either been placed in manual, or had its bias setting increased, prior to the reactor trip.

Upon reactor trip, valid RTO signals were generated and responded to by the feedwater controls for SG E089. However, as the "B" phase of Instrument Bus #1 now remained energized for a considerable period, both MFWPs regained speed and began refilling SG E088 through the regulating and bypass valves which were open and in manual control. The Control Room operators did not continue monitoring SG level as they performed immediate post-reactor trip actions and, therefore, did not recognize that level was increasing due to resumed main feedwater flow.

Approximately 20 seconds after the reactor trip, the feedwater control station was checked in accordance with standard post-trip actions. Since a "Loss of Main Feedwater" was the event that had occurred, the check was focused on verifying proper performance of the auxiliary feedwater system (adequate flow to both steam generators) and the subtle variance in main feedwater controller status (automatic demand signal indicators downscale and manual demand indicators upscale) was not detected. At this time, an excessively high SG level had not yet occurred and the safety functions were confirmed as being implemented. Satisfied, the operator continued with other standard post trip actions and returned his attention to the main feedwater control station only after it was too late to mitigate the transient.

At 85 seconds following the reactor trip, a Safety Injection Actuation Signal (SIAS) (EIIS System Code JE) was generated as a result of decreasing Reactor Coolant System (RCS) (EIIS System Code AB) temperature and pressure. This decrease was primarily caused by the continuing increase in SG E088 level which remained unmonitored as the operators now began their post-SIAS immediate actions.

Approximately 45 seconds following SIAS, the "B" phase of Instrument Bus #1 was interrupted for the fourth and final time during the transient. This time its power was restored after 12 seconds, and about 45 seconds later the MFWPs tripped due to high vibration. It is likely that the high vibration was caused by the rapid increase in pump speed following the return of power to the instrument bus and/or the transients caused by valve cycling. Subsequent MFWP performance has been monitored and no increase in vibration has been observed.

As RCS pressure continued to decrease below the SIAS setpoint, the operators monitored the narrow range pressure indication which they believed remained on-scale above 1,500 psia and, therefore, above the High Pressure Safety Injection (HPSI) pump (EIIS System Code BQ)(EIIS Component Code P) shutoff head. In fact, subsequent analysis of recorded data demonstrated that RCS pressure had decreased to a minimum value of 1,247 psia, and HPSI flow had occurred to the RCS for a period of about 90 seconds, commencing 200 seconds following the reactor trip. Maximum flow reached approximately 100 gpm through each of the four injection legs. Since the operators believed that RCS pressure remained on-scale by narrow range indication, and above the HPSI pump shutoff head, they did not monitor indicated safety injection flow continuously and did not observe in the Control Room that it had occurred. An Unusual Event was, therefore, not declared.

Pressurizer (EIIS System Code AB)(EIIS Component Code PZR) level was reduced below the lower level indication, and subsequent analysis indicated that it had decreased to approximately the level of the pressurizer surge line nozzle before it began increasing. Indicated RCS temperature decreased to 495 degrees F. Based on the maximum change in cold leg temperatures during the first hour following the reactor trip, the maximum RCS cooldown rate was calculated to be 62 degrees F per hour. The maximum pressurizer cooldown rate was 82 degrees F per hour. Minimum calculated margin to saturation in the core was approximately 64 degrees F. Subsequent calculations have demonstrated that, for a period of approximately 30 seconds, a very small bubble may have existed in the reactor vessel head. The size of the bubble would have been sufficiently small such that the Reactor Vessel Level Indicating System (EIIS System Code JD) would not have indicated any voided volume.

RCS temperature and pressure were stabilized, normal levels were established in the pressurizer and steam generators, and the unit remained in Mode 3 for the next four days during which the event was carefully analyzed and corrective action to prevent recurrence was identified and taken. The results of this effort are summarized as follows:

Loss of Electrical Power to Non-1E Instrumentation

With the benefit of the supplemental monitoring and recording instrumentation that had been installed earlier, the intermittent loss of electrical power was traced to a loose connection at a bolt connecting the "B" phase of the Instrument Bus #1 supply breaker (EIIS Component Code BKR) to the main bus bars of the Non-1E Uninterruptable Power Supply (UPS) (EIIS System Code EE)(EIIS Component Code UJX) main distribution switchboard. The bolt was found backed out 5 to 6 turns. It was located in an area which had not been inspected following the momentary loss of power a month earlier, due to concern that the inspection itself would result in a spurious unit trip.

When the breaker was moved by hand within the switchboard, the "B" phase voltage on Instrument Bus #1 decreased to zero, while the "A" and "C" phase voltages remained normal, duplicating indications observed and recorded during the event. Evidence of high temperature was observed at the associated "C" phase connection as well, and, although not visually loose, discoloration and charring indicated poor electrical contact. The breaker and the "C" phase bus bar were replaced, and a complete inspection and tightening of all bus bar and external connections on the non-1E instrument busses was conducted.

As long-term corrective action, the identification of all equipment which can result in a unit trip has been ongoing at San Onofre Units 2 and 3. Consideration has been given in the past to providing a redundant power supply to the Foxboro rack which provides control signals to feedwater flow controllers, feedwater control valves and feedwater pump speed. As a result of this event, additional consideration of such a design change is being given. All future work on this important to safety but non-safety related equipment will be accomplished with the same controls as are applied to safety-related work. This should minimize the likelihood that electrical connections will be left uncompleted, as apparently occurred in this case.

Cause of Steam Generator Overfilling

The cause of the SG overfill was the return of power to Instrument Bus #1, with feedwater controllers in manual and set for high flow, concurrent with operators shifting their focus to respond to the reactor trip. As immediate corrective action, all shifts were briefed on the cause of the SG overfilling, and its consequences, and advised that operators should not divert their attention to other activities, while leaving a control station in manual with significant demand settings in place. Operators will be advised of the need to insert a minimum manual signal into those controllers in "Manual" at the time a reactor trip occurs and their attention must be focused on post trip activities. This advisement will also be formally communicated to all operations personnel.

As longer term corrective action, the generic application of this principle is being included in the operator training program. (Also, the evaluation of the event is being presented to the Combustion Engineering Owner's Group for their information.)

Consequences of Steam Generator Overfilling

All affected steam lines and equipment were inspected and evaluated to ensure against any damage resulting from water ingress. There was no indication of any main or auxiliary feedwater pump turbine damage attributed to water carryover.

Monitoring of RCS Pressure

As immediate corrective action, all shifts were reminded that the narrow range pressure instrument in the Control Room should not be used below approximately 1,550 psia. and that the wide range trend recorder located nearby should be referred to any time safety injection is initiated for confirmation of indicated RCS pressure. Prior to this event, a design change had been planned to add wide range pressurizer pressure indication, along with existing narrow range indication, at the pressurizer pressure control station. As a result of this event, consideration will be given to provide for recording of the wide range pressure signal at the pressurizer control station as well, and the schedule for this additional indication will be accelerated, if practical.

Safety Injection System Operation

Post-trip review of computer records did not provide indication of flow through loop 2B. Subsequent investigation included flow testing and calibration checks of the flow transmitter (EIIS Component Code FIT). The problem was traced to a zero-offset in the flow transmitter (Foxboro, Model NE13DH). The flow transmitter was then replaced with an in-kind spare. Flow testing of loop 2B demonstrated satisfactory performance with the new transmitter in place. Our investigation into the cause of the zero-offset is continuing, and if the cause can be identified, a supplement to this report will be submitted, as appropriate.

In addition, during subsequent stroke testing of Emergency Core Cooling System (ECCS) (EIIS System Code BP) snubbers (EIIS Component Code SNB), four small Pacific Scientific snubbers associated with HPSI pumps P-017 and P-019 discharge lines were found frozen, and were replaced. Previous failure mode analyses performed for similar ECCS snubber failures on Units 2 and 3, most recently reported in the March 1987 submittal of Licensee Event Report (LER) 85-049 Rev. 1 (Docket No. 50-361), has demonstrated that the snubber failures were most likely due to hydraulic transients which can result from the rapid closure of certain stop/check valves (EIIS Component Code SHV). Although the stop/check valve discs are intended to seat closed by gravity when their associated ECCS pump flow is stopped, it is possible that they not do so until reverse flow from other operating pumps on the common discharge header results in increased differential pressure across the disc, causing it to rapidly seat. As reported in LER 85-049, Rev. 1, procedures for system operation had been revised to preclude rapid stop/check valve closure. SCE's evaluation of the adequacy of this corrective action to preclude ECCS hydraulic transients is continuing, and any further corrective actions resulting from this evaluation will be reported in a supplement to LER 85-049.

These snubber failures were all bounded by the engineering analysis performed for the previous snubber failures. Based on that analysis, it was demonstrated that no damage had occurred to piping or other components and that the system remained functionally operable and would have performed its intended safety function during a Design Basis Earthquake (DBE).

Southern California Edison Company

SAN ONOFRE NUCLEAR GENERATING STATION

P. O. BOX 128

SAN CLEMENTE, CALIFORNIA 92672

H. E. MORGAN
STATION MANAGER

TELEPHONE
(714) 368-6241

August 28, 1987

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

Subject: Docket No. 50-362
Supplemental Report
Licensee Event Report No. 87-011, Revision 1
San Onofre Nuclear Generating Station, Unit 3

Reference: Letter, H. E. Morgan (SCE) to USNRC Document Control Desk, dated July 21, 1987.

The referenced letter provided the required 30-day written Licensee Event Report (LER) for an occurrence involving an actuation of the Reactor Protection System. Enclosed is a revision to this LER, which includes additional relevant information for completeness.

If you require any additional information, please so advise.

Sincerely,

HEMorgan

Enclosure: LER No. 87-011, Rev. 1

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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