REGULATORY INFORMATION DISTRIBUTION SYSTEM (RIDS)

ACCESSION NBR: 8706240374 DOC. DATE: 87/06/17 NOTARIZED: NO DOCKET #
FACIL: 50-361 San Onofre Nuclear Station, Unit 2, Southern Californ 05000361
AUTH. NAME AUTHOR AFFILIATION
MORGAN, H. E. Southern California Edison Co.
RECIP. NAME RECIPIENT AFFILIATION

SUBJECT: LER 86-022-01: on 860812, reactor trip occurred when RCS pressure reached core protection calculator auxiliary trip set point. Caused by intermittent failure of DC power supply filter capacitor. Capacitor replaced. W/870617 ltr.

DISTRIBUTION CODE: IE22D COPIES RECEIVED: LTR 1 ENCL 1 SIZE: 4
TITLE: 50.73 Licensee Event Report (LER), Incident Rpt, etc.

NOTES: ELD Chandler 1cy. 05000361

	RECIPIENT ID CODE/NAME	COPIES LTTR	COPIES ENCL	RECIPIENT ID CODE/NAME	COPIES LTTR	COPIES ENCL
	PD5 LA	1	1	PD5 PD	1	1
	ROOD, H	1	1			
INTERNAL:	ACRS MICHELSON	1	1	ACRS MOELLER	2	2
	AEOD/DOA	1	1	AEOD/DSP/ROAB	2	2
	AEOD/DSP/TPAB	1	1	DEDRO	1	1
	NRR/DEST/ADE	1	0	NRR/DEST/ADS	1	0
	NRR/DEST/CEB	1	1	NRR/DEST/ELB	1	1
	NRR/DEST/ICSB	1	1	NRR/DEST/MEB	1	1
	NRR/DEST/MTB	1	1	NRR/DEST/PSB	1	1
	NRR/DEST/RSB	1	1	NRR/DEST/SGB	1	1
	NRR/DLPQ/HFB	1	1	NRR/DLPQ/QAB	1	1
	NRR/DOEA/EAB	1	1	NRR/DREP/RAB	1	1
	NRR/DREP/RPB	2	2	NRR/PMAS/ILRB	1	1
	NRR/PMAS/PTSB	1	1	REG FILE 02	1	1
	RES DEPY GI	1	1	RGN5 FILE 01	1	1
EXTERNAL:	EG&G GROH, M	5	5	H ST LOBBY WARD	1	1
	LPDR	1	1	NRC PDR	1	1
	NSIC HARRIS, J	1	1	NSIC MAYS, G	1	1
NOTES:		1	1			

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NRC Form 366 (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 2	0 5 0 0 0 3 6 1	1 OF 0 3

TITLE (4)
REACTOR TRIP FOLLOWING MAIN STEAM ISOLATION SIGNAL

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 8	1 2	8 6	8 6	— 0 2 2	— 0 1	0 6	1 7	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)	[X] 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
H. E. MORGAN, STATION MANAGER	AREA CODE 7 1 4 368-6241

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	J E	J X	X 9 9 9	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 August 12, 1986 at 1330 with Unit 2 at 100% power, a reactor trip occurred when Reactor Coolant System (RCS) pressure reached the Core Protection Calculator (CPC) Auxiliary Trip set point of 2375 psia. The RCS pressure transient resulted from Main Steam Isolation Valve (MSIV) closure when the Main Steam Isolation System (MSIS) was actuated during surveillance testing of the MSIS Automatic Actuation Logic. The trip recovery proceeded normally and there were no safety consequences associated with this event.

The MSIS group actuation relays are maintained energized by current through two parallel circuits from the Engineered Safety Features Actuation System (ESFAS) trip initiation Solid State Relays (SSR). During the Technical Specification required surveillance test, one side of the parallel circuit is de-energized for both Trains "A" and "B" simultaneously while the other side remains energized. During this testing, de-energization of the SSR in the remaining parallel circuitry resulted in actuation of the MSIS. Following the occurrence, extensive inspection and testing of ESFAS circuitry was conducted, however, the condition could not be duplicated and a failed component could not be identified.

Through subsequent investigation, however, the cause of the occurrence was determined to be an intermittent failure of a DC power supply filter capacitor which had reached the end of its service life. All older similar capacitors will be replaced during the next refueling outage for each unit. Each channel's DC power supply output is being checked, at an interval capable of detecting incipient filter capacitor failures, prior to initiation of the monthly surveillance testing of the channel.

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FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6) YEAR	SEQ. NUMBER	REV. NUMBER	PAGE (3)
SAN ONOFRE NUCLEAR GENERATING STATION, UNIT 2	0 5 0 0 0 3 6 1	8 6	– 0 2 2	– 0 1	0 2 OF 0 3

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

On August 12, 1986, at 1330 with the Unit 2 reactor at 100% power, both Main Steam Isolation Valves (MSIV) (EIIS System Code SB) (EIIS Component Code ISV) and Main Feedwater Isolation Valves (MFIV) (EIIS System Code SJ) (EIIS Component Code ISV) closed, and the reactor tripped when the Reactor Coolant System (RCS) pressure reached the Core Protection Calculator (EIIS System Code JC) Auxiliary Trip set point of 2375 psia. The MSIV/MFIV closure occurred during Engineered Safety Features Actuation System (ESFAS) surveillance testing. RCS pressure peaked at 2478 psia, which is below the pressurizer safety valve lift pressure. Main Steam Safety Valves opened and an Emergency Feedwater Actuation Signal (EIIS System Code BA) was initiated on low Steam Generator water level. The trip recovery proceeded normally and there were no safety consequences associated with this event.

Actuation of an Engineered Safety Feature (ESF) requires interruption of DC power to actuation relays. Such power interruption is designed to take place only when initiation relays are actuated by two of the four ESFAS sensor channels. Initiation relays are actuated in pairs, one associated with each train of the ESF in such a manner that one of the two trip paths in each train is actuated. Unless initiation relays associated with the remaining trip path are also actuated, the surveillance testing should not result in actuation of the ESF.

During performance of the aforementioned monthly channel functional testing of the Main Steam Isolation System (MSIS) (EIIS System Code JE) Automatic Actuation Logic circuitry, required by Technical Specification Table 4.3-2, operation of each initiation relay was verified. Following testing and reset of one pair of initiation relays, a second pair of initiation relays were actuated in accordance with the test procedure. At this time, the first pair of relays spuriously actuated resulting in the MSIV closure.

Following the Unit Trip, the MSIS Automatic Actuation Logic circuitry was extensively tested and no discrepancies were found. The surveillance testing sequence leading to the trip was repeated four times, and the trip could not be duplicated. All other Unit 2 ESFAS functions and channels containing similar trip path circuitry were likewise tested with similar results.

Continued investigation into the cause of the event has determined that a 12 volt DC power supply (EIIS Component Code JX) had failed, such that it intermittently provided DC power with an excessive AC ripple, to one of the six trip path matrix relays. During periods when such rippled DC power was being provided to the initiation relays, the relays would occasionally open, thus de-energizing one of the two trip paths and resulting in partial actuation of an ESF train.

Analysis of the failed power supply determined that such rippled DC voltage resulted from a corroded, and intermittently open, filter capacitor (EIIS Component Code CAP) internal lead. This capacitor and those installed in other ESFAS and identical Plant Protection System (PPS) power supplies, have a median useful life of at least fifteen years.

There have been no subsequent filter capacitor failures in ESFAS or PPS power supplies.

Because this type of filter capacitor is installed in other ESFAS and PPS power supplies, the following corrective actions are being implemented:

1. Each PPS channels' power supply output is being checked for indication of excessively rippled DC output (i.e., filter capacitor failure), prior to performing the 31 day interval channel functional test for that channel. The interval between these checks will be increased as data is developed on the rate at which filter capacitors degrade as they approach the end of their useful life.

2. All filter capacitors of this type in Units 2 and 3 PPS and ESFAS power supplies that are more than five years old will be replaced during the next refueling outage for the unit. Subsequently, these filter capacitors will be replaced at approximately ten year intervals.

Failure analysis of the trip path matrix relays subjected to the intermittent inadequately smoothed DC power indicates that they are undamaged and remain serviceable.

Since all safety systems performed as designed, there was no impact on the health and safety of plant personnel or the public as a result of this event.

Following the main steam isolation and during completion of the aforementioned ESFAS testing, the MSIVs were closed, requiring the use of the Atmospheric Dump Valves (ADV) to dissipate reactor decay heat during Hot Shutdown operation. During previous ADV operation in June 1986, several of the ADV noise suppression muffler studs broke and were ejected from the ADV exhaust stacks creating a significant personnel safety hazard in the area surrounding the ADVs. All personnel were, therefore, evacuated from potentially hazardous stud impact areas at 1445 on August 12, 1986. Consequently, hourly fire watch patrols pursuant to Technical Specification 3.3.3.7, 3.7.8.2 and 3.7.9 were terminated for areas in which access can only be made via the hazardous areas. On August 14, 1986 at 0208, these hourly fire watch patrols were re-established when ADV operation was no longer necessary. The existing ADV muffler studs were replaced with new studs and installed with a higher pre-load.

Southern California Edison Company

SAN ONOFRE NUCLEAR GENERATING STATION

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H. E. MORGAN
STATION MANAGER

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June 17, 1987

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

Subject: Docket No. 50-361
30-Day Report
Licensee Event Report No. 86-022, Rev. 1
San Onofre Nuclear Generating Station, Unit 2

Reference: Letter H. E. Morgan (SCE) to USNRC Document Control Desk, dated September 11, 1986

The referenced letter provided the required 30-day Licensee Event Report (LER) for an occurrence involving the Reactor Protection System. This submittal provides additional information regarding the results of an investigation into the cause of this event.

If you require any additional information, please so advise.

Sincerely,

HEMorgan

Enclosure: LER No. 86-022, 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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