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 5 0 0 0 |
| 0 8 | 1 2 | 8 6 | 8 6 | 0 2 2 | 0 0 | 0 9 | 1 1 | 8 6 | | 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 (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 | MANUFACTURER | REPORTABLE TO NPRDS | CAUSE | SYSTEM | COMPONENT | MANUFACTURER | REPORTABLE TO NPRDS |
|---|---|---|---|---|---|---|---|---|---|
| | | | | | | | | | |
| | | | | | | | | | |

SUPPLEMENTAL REPORT EXPECTED (14)

| | | EXPECTED SUBMISSION DATE (15) | MONTH | DAY | YEAR |
|---|---|---|---|---|---|
| YES (If yes, complete EXPECTED SUBMISSION DATE) | X NO | | | | |

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 setpoint 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. Extensive inspection and testing of ESFAS circuitry was conducted, however, the condition could not be duplicated and a failed component could not be identified. When the cause of the failure is identified, a supplemental LER will be submitted.

Data from other ESFAS partial actuations is being reviewed and investigation into possible causes is continuing in association with Combustion Engineering.




IE22
1/1

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|---|---|---|---|---|---|
| SAN ONOFRE NUCLEAR GENERATING STATION, UNIT 2 | 0 5 0 0 0 3 6 1 | 8 6 | - 0 2 2 | - 0 0 | 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 setpoint 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. When the cause of the failure is identified, a supplemental LER will be submitted. Since this event, similar partial initiations of other ESFAS functional units with identical circuitry have also occurred. In these instances, replacement of coincident logic relay boards has rectified the partial initiation condition.

Current investigation of this circuitry is focusing on the possibility that degradation of individual components on such boards may result in intermittent or spurious initiation relay actuation. SCE and Combustion Engineering are also reviewing the surveillance testing procedures and the design of the actuation initiation circuitry in a further attempt to determine the cause and corrective action. In order to reduce the magnitude of the transients resulting from such a trip, consideration is also being given to conducting required ESFAS surveillance testing at reduced power until a cause has been identified and corrected.

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.

NRC Form 366A
(9/83)

# LICENSEE EVENT REPORT (LER) TEXT CONTINUATION

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

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 will be replaced with a higher pre-load to eliminate future failures.

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

September 11, 1986

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
San Onofre Nuclear Generating Station, Unit 2

Pursuant to 10 CFR 50.73(a)(2)(i and iv), this submittal provides the required 30-day written Licensee Event Report (LER) for an occurrence involving 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,

HE Morgan

Enclosure: LER No. 86-022

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)

IE22
1/1