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 3	0 5 0 0 0 3 6 2	1 OF 0 5

TITLE (4)
REACTOR TRIP - RPS COMPONENT FAILURES

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 3	2 9	8 5	8 5	— 0 1 0	— 0 0	0 4	2 9	8 5		0 5 0 0 0
										0 5 0 0 0

OPERATING MODE (9): 1
POWER LEVEL (10): 0 9 6

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)	X	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	
J. G. HAYNES, STATION MANAGER	7 1 4	4 9 2 - 7 7 0 0

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
B	J C	I M O D	E 1 4 6	Y					
B	J C	R L Y	E 1 4 6	Y					

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 3/29/85, at 2108, with Unit 3 in Mode 1 at 96% power, the reactor tripped due to apparent low flow through Steam Generator (S/G) E-088. The trip occurred, even though Control Room indications showed only that a Channel 'C' low flow trip had actuated, which would not complete the two-out-of-four reactor trip logic. With some exceptions, control and safety systems were verified to have functioned properly and major plant parameter responses were satisfactory.

The reactor trip was determined to be the result of two failures in the Reactor Protection System circuitry. A failure of a relay from the Channel 'A' S/G low flow bistable caused a Channel 'A' trip prior to the Channel 'C' trip. This was unknown to the Control Room, since it was downstream of the S/G low flow bistable and alarm/indication signal. Reactor trip logic was completed when Channel 'C' tripped due to failure of a setpoint card. Both of these components were replaced.

This submittal also provides the report pursuant to Limiting Condition for Operation 3.4.7, Action Statement 'd', for RCS specific activity exceeding 1.0 microcurie/gram Dose Equivalent I-131, which was caused by iodine spiking following the shutdown.

There are no reasonable or credible circumstances which could have increased the severity of this event.

IE22
1/1

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

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

On March 29, 1985, at 2108, with Unit 3 in Mode 1 at 96 percent reactor power, the reactor (EIIS Component Code RCI) tripped due to apparent low flow through Steam Generator (S/G)(EIIS Component Code SG) E-088. The trip occurred, even though, Control Room indications showed only that a Channel 'C' low flow trip had actuated, which does not by itself complete the two-out-of-four reactor trip logic. Post trip investigations revealed a failed relay (EIIS Component Code RLY) in the Channel 'A' trip logic circuit, which actuates the Channel 'A' low flow trip, but did not give an indication in the Control Room. The Channel 'A' failure had occurred prior to the Channel 'C' trip and therefore, upon the Channel 'C' trip, the two-out-of-four reactor trip logic was completed.

An emergency plant shutdown was performed per Emergency Operating Instruction (EOI) SO23-12-1, "Standard Post Trip Actions." Following the reactor trip EOI SO23-12-2, "Reactor Trip Recovery," was performed upon completion of SO23-12-1. All four reactor coolant pumps (EIIS Component Code P) remained in operation and all steam generator primary differential pressures appeared to be normal during the entire event.

The reactor trip was caused by two equipment failures in the Reactor Protection System (RPS) (EIIS System Code JC) circuitry. The first failure occurred in a relay from the Channel 'A' S/G low flow bistable to the associated contacts in its coincidence matrices. The relay remained energized but failed to keep the S/G low flow bistable contact shut on the Channel 'A' side of each coincidence matrix ladder. Since the failure occurred downstream of the S/G low flow bistable and alarm/indication signal, no alarms or indications were present to indicate a problem. The failed Channel 'A' coincidence matrix relay was replaced.

The second failure occurred in an operational amplifier in the Channel 'C' rate limited variable setpoint card (EIIS Component Code IMOD). This card provides a minimum flow signal to the RPS bistable comparator for S/G low flow to compare it to actual S/G primary differential pressure. This card failure resulted in a Channel 'C' S/G low flow bistable trip, completing the two-out-of-four reactor trip logic. The faulty card was replaced.

Four unrelated items were identified following the trip:

1. Power to Bus 3A08 (EIIS System Code EA) was temporarily lost because it failed to transfer to the reserve auxiliary transformer. The bus was re-energized by aligning it to the reserve auxiliary transformer supply breaker manually. The failure of Bus 3A08, a non-safety-related bus, to transfer from the unit to the reserve auxiliary transformer was caused by a failure in the HFA coil for Breaker 3A0804. A piece of molded plastic broke off the coil insulation and fell into the coil, preventing it from latching up. The failure of this coil is identical to that described in IE Bulletin 84-02 for safety-related HFA relay coils, all of which have been replaced at SONGS. The application of non-safety-related HFA relays is presently being reviewed to determine if replacement of the coils is warranted. The damaged coil was subsequently replaced with an upgraded coil.

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

2. Immediately after the bus was re-energized, both Main Feedwater Pump Turbines (EIIS Component Code TRB), 3K005 and 3K006 tripped due to high vibration. 3K006 was reset and main feedwater restored. 3K005 was left off since it was not required. Maintenance testing of 3K005 and 3K006 was performed, however, the cause of the vibration trip could not be determined.

3. A fire suppression deluge valve (EIIS Component Code V) was activated in the vicinity of 3K006 immediately after the main feedwater pump turbine tripped. The deluge valve was reset. Our investigation into this matter is continuing.

4. Condensate Pump 3P-051's discharge overboarded below its normal trip setpoint. The Condensate System (EIIS System Code KD) was returned to normal lineup with Condensate Pump 3P-053 secured. The overboarding of Condensate Pump 3P-051 was due to an incorrect trip setpoint of 0.39 micromhos being present in the conductivity indicating transmitter. The setpoint was adjusted to its normal value of 1.5 micromhos. Additionally, since Channel 'A' S/G low level trip exhibited sluggish response, the instrument sensing line was blown down. Since this was the second occurrence of clogged steam generator sensing lines at this site (see also LER 2-84-043, Docket No. 361), all Unit 2 and 3 steam generator level sensing lines will be scheduled to be blown down during the next outage of sufficient duration.

There was no safety significance to this event, since RPS failures resulting in a reactor trip of this nature are bounded by analyses contained in the FSAR. Neither the health and safety of plant personnel nor the health and safety of the public was affected by this event.

On March 29, 1985, at 2312, with Unit 3 in Mode 3, Reactor Coolant System (RCS) (EIIS System Code AB) sample analysis indicated that RCS specific activity exceeded 1.0 microcurie/gram Dose Equivalent (DE) I-131. RCS specific activity was reduced to less than 1.0 microcurie/gram DE I-131 by purification flow at 0900 on March 30, 1985. This event was an indication of iodine spiking. Similar occurrences were previously reported in LER's 83-111, 84-005, 84-013, 84-015, 84-023, 84-037, 84-038, 84-039 and 85-001.

Pursuant to Limiting Condition for Operation (LCO) 3.4.7, Action Statement 'd', this submittal also provides the required 30-day written report for the iodine spiking following the shutdown. Additional information, required by LCO 3.4.7, Action Statement 'd', is provided in the tables below. Although the unit has a degasification path which operates continuously and takes pressurizer steam, condenses it and directs it to Liquid Radwaste, degassing history is not applicable, because this system reduces the noble gas content of the RCS but has no effect on Iodine.

CLEANUP FLOW HISTORY

Period	Average Cleanup Flow (gpm)
3-27-85, 2312 to 3-29-85, 2108	88.6
3-29-85, 2108 to 3-29-85, 2118	131.0
3-29-85, 2118 to 3-29-85, 2215	88.6
3-29-85, 2215 to 3-29-85, 2220	131.0
3-29-85, 2220 to 3-29-85, 2235	88.7
3-29-85, 2235 to 3-29-85, 2240	129.0
3-29-85, 2240 to 3-29-85, 2250	88.7
3-29-85, 2250 to 3-29-85, 2255	130.0
3-29-85, 2255 to 3-29-85, 2312	88.6

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

REACTOR POWER HISTORY

Period	Reactor Power
3-27-85, 2312 to 3-28-85, 2000	100% Rated Power
3-28-85, 2000 to 3-28-85, 2300	100% to 85%
3-28-85, 2300 to 3-29-85, 0600	85%
3-29-85, 0600 to 3-29-85, 0700	85% to 97%
3-29-85, 0700 to 3-29-85, 0900	97%
3-29-85, 0900 to 3-29-85, 1100	97% to 90%
3-29-85, 1100 to 3-29-85, 1300	90%
3-29-85, 1300 to 3-29-85, 1500	90% to 96%
3-29-85, 1500 to 3-29-85, 2108	96%
3-29-85, 2108 to 3-29-85, 2312	0%

REACTOR COOLANT SYSTEM SPECIFIC ACTIVITY

Date/Time of Sample	DEI-131 (uCi/gram)
3-29-85, 2312	2.05
3-30-85, 0200	1.96
3-30-85, 0600	1.27
3-30-85, 0900	0.88

NRC Form 366A
(9/83)

LICENSEE EVENT REPORT (LER) TEXT CONTINUATION

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

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6) YEAR	SEQ. NUMBER	REV. NUMBER	PAGE (3)
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TEXT (If more space is required, use additional NRC Form 366A's) (17)

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CECOR2.0 HZ 1209        SJCIF036                          EDIT= 23  04/04/85  19103119  PAGE = 57
                  AXIALLY  INTEGRATED  AND  PEAK  OUTPUT ASSEMBLY EXPOSURE EDITS
    FORMAT OF ASSEMBLY IN CORE MAP
    ASSEMBLY NUMBER - BATCH NUMBER
    INTEGRATED BOX EXPOSURE IN  10**003MWD/T
    MAXIMUM  BOX  EXPOSURE  IN  10**003MWD/T
    LOCATION OF MAX. ASS. EXP. IN 0/0 HEIGHT

                         1-05    2-05    3-05    4-05
                        5.990   7.774   7.760   5.976
                        7.344   9.493   9.477   7.308
                       34.000  34.000  34.000  32.000

         5-05    6-05    7-07    8-07    9-04   10-07   11-07   12-05   13-05
        5.769   7.798   9.348  11.057   9.026  11.048   9.305   7.718   5.626
        7.143   9.563  11.578  13.682  12.237  13.630  11.476   9.398   6.895
       36.000  34.000  32.000  34.000  34.000  34.000  32.000  34.000  42.000

   14-05   15-06   16-04   17-02   18-04   19-02   20-04   21-02   22-04   23-06   24-05
   6.299   9.568   9.716   9.901  10.462  10.303  10.850   9.877   9.641   9.558   6.229
   7.677  11.011  12.019  12.132  13.433  12.651  13.402  12.069  11.855  11.567   7.561
  36.000  36.000  36.000  34.000  34.000  34.000  34.000  34.000  34.000  36.000  34.000

   25-05   26-02   27-04   28-02   29-04   30-02   31-04   32-02   33-04   34-02   35-04   36-02   37-05
   6.387   8.892   9.798   9.999  11.081  10.601  11.203  10.601  11.029   9.916   9.690   8.888   6.323
   7.836   9.909  12.122  12.231  13.406  12.939  13.062  12.929  13.515  12.123  11.916   9.751   7.711
  34.000  36.000  36.000  36.000  36.000  36.000  36.000  34.000  36.000  34.000  34.000  34.000  34.000

   38-35   39-36   40-34   41-02   42-04   43-02   44-04   45-02   46-04   47-02   48-04   49-02   50-04   51-06   52-05
 [illegible] 9.538 9.888  10.014  11.156  10.743  11.584  10.809  11.567  10.764  11.075   9.900   9.777   9.430 [illegible]
 [illegible] 11.651 12.153 12.216  13.657  13.027  14.181  13.255  14.166  12.994  13.544  12.057  11.972  11.520 [illegible]
 [illegible] 36.000 36.053 36.000  36.000  36.000  34.000  34.000  34.000  36.000  35.000  36.000  36.000  36.000 [illegible]

[illegible — assemblies 53-05 through 193]

   194-05  195-06  196-04  197-02  198-04  199-02  200-04  201-02  202-04  203-06  204-05
    6.261   9.538   9.698   9.904  10.886  10.337  10.862   9.863   9.650   9.482   6.235
    7.613  11.643  11.888  12.052  13.365  12.607  13.341  12.009  11.828  11.569   7.534
   36.100  34.000  34.000  34.000  34.000  34.000  36.000  36.000  36.001  36.000  36.000

         205-05  206-05  207-07  208-07  209-04  210-07  211-07  212-05  213-05
          5.721   7.754   9.275  11.069   9.079  11.071   9.319   7.732   5.676
          6.973   9.410  11.354  13.546  12.159  13.569  11.435   9.391   6.899
         34.000  34.000  34.000  34.000  34.000  34.000  34.000  34.000  32.000

                         214-05  215-05  216-05  217-05
                          5.945   7.791   7.793   5.990
                          7.256   9.488   9.426   7.278
                         34.000  34.000  34.000  34.000

MAXIMUM INTEGRATED ASSEMBLY EXPOSURE IS   0.1208160+05 MWD/T IN ASSEMBLY 110
MAXIMUM PEAK AXIAL EXPOSURE IS   0.1460780D+05 MWD/T, OCCURRING AT  36.00 0/0 OF THE CORE HEIGHT IN ASSEMBLY 108
CORE AVERAGE EXPOSURE IS   0.9983230+04 MWD/T

-------------------- BATCH AVERAGE EXPOSURES --------------------
BATCH NUMBER          BATCH NAME          AVERAGED EXPOSURE (MWD/T)
     1                    A1                      11.196
     2                    A2                      10.178
     3                    B1                      11.038
     4                    B2                      10.657
     5                    C                        8.792
     6                    C*                    [illegible]
     7                    C+                      10.358
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Southern California Edison Company

SCE

SAN ONOFRE NUCLEAR GENERATING STATION
P.O. BOX 128
SAN CLEMENTE, CALIFORNIA 92672

J. G. HAYNES
STATION MANAGER

TELEPHONE
(714) 492-7700

April 29, 1985

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

Dear Sir:

Subject: Docket No. 50-362
30-Day Report
Licensee Event Report No. 85-010
San Onofre Nuclear Generating Station, Unit 3

Pursuant to 10 CFR 50.73(a)(2)(iv) and Limiting Condition for Operation (LCO) 3.4.7, Action Statement 'd' of Appendix A, Technical Specifications to Facility Operating License NPF-15 for San Onofre Unit 3, 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,

J. G. Haynes

Enclosure: LER No. 85-010

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

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

Institute of Nuclear Power Operations (INPO)

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