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 2

TITLE (4)
REACTOR TRIP ON LPD/DNBR

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	2 6	8 4	8 4	— 0 5 0 —	0 0	0 9	2 5	8 4		0 5 0 0 0

OPERATING MODE (9): 1
POWER LEVEL (10): 0 1 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	
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
X	E D	B K R	G 1 8 7	N					

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 26, 1984 at 1816, with Unit 2 at 10% power following power reduction because of a turbine trip due to loss of condenser vacuum, the reactor tripped in response to Low Departure from Nucleate Boiling Ratio (DNBR) and high Local Power Density (LPD) trip signal on Channels B and C of the Reactor Protection System (RPS). These RPS trip signals were due to a Core Protection Calculator Auxiliary Trip created by the conservative algorithums being used by the Core Protection Calculators which assume a power level of 20% whenever the actual power level is below 20%. This caused these calculations to become overly conservative and not indicative of the actual plant conditions. The Technical Specification requires operation within the limits of these Core Protection Calculator calculations for Mode 1, at 20% power and above, only. When the trip occurred the actual DNBRs and LPDs were well within their allowable limits. All eight Reactor Trip Breakers opened and all 91 Control Element Assemblies (CEA) inserted fully. All other systems and components functioned properly during this event.

There are no reasonable or credible alternative conditions under which this event would have been more severe.

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 4	- 0 5 0	- 0 0	0 2 OF 0 2

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

On August 26, 1984, with Unit 2 at 100%, a loss of the AC Lube Oil Pump (P-121)(EIIS Component Code P) which is the backup lube oil pump to the Feedwater Pump-Turbine (K005)(EIIS Component Code TRB) occurred due to a trip of the feeder breaker (2B1401) (EIIS Component Code BKR) for 480 VAC load center (2B14). AC Lube Oil Pump (P-123) (EIIS Component Code P) and the backup pump (P-121) were both in service due to an existing oil leak at K005. When P-121 tripped, P-123 was unable to maintain the required flow and tripped on thermal overload. The complete loss of lube oil caused K005 to trip out of service. A reduction in the condenser vacuum level (EIIS System Code SH) followed, which tripped the turbine (EIIS Component Code TRB).

A rapid power reduction was initiated and at approximately 10% power the unit began to stabilize. Xenon levels had increased due to the rapid power reduction which resulted in the initiation of dilution of the Reactor Coolant System and Group 6 Control Element Assemblies (CEA) (EIIS Component Code ROD) withdrawal in an attempt to maintain criticality. A Power Flux Transient (PFT) toward the top of the core resulted. This was caused predominantly from the Xenon redistribution, but was enhanced by the CEA withdrawal and decreasing power level.

At 1816, with the unit at 10% power, the reactor (EIIS Component Code RCT) tripped in response to low Departure from Nucleate Boiling Ratio (DNBR) and high Local Power Density (LPD) trip signals on Channels B and C of the Reactor Protection System (RPS)(EIIS System Code JC). These RPS trip signals were due to a Core Protection Calculator (CPC) (EIIS Component Code CPU) auxiliary trip resulting from a Hot Pin Axial Shape Index which had exceeded the trip limit. When the trip occurred, the actual DNBRs and LPDs were not outside their allowable range as specified in Section 2.2.1 of the Technical Specification. The CPC software which provides for the calculation of DNBR/LPD also provides auxiliary trips for cases where single key parameters exceed their allowable range. The CPC calculation uses an assumed power distribution and power level of 20% whenever the actual power level is less than 20%. Operating below 20% power can cause some of the internal parameters of these calculations to be outside their expected range, which can cause these calculations to become invalid and not indicative of the actual plant conditions. The Technical Specifications require operation within the limits of these CPC calculations for Mode 1, at 20% power and above, only. During this power reduction Xenon affects had produced a power distribution which was very different from that assumed by the calculation. With these conservative algorithms being used, a calculated Hot Pin Axial Shape Index outside the allowable range resulted and caused the auxiliary trip. All eight Reactor Trip Breakers (EIIS Component Code BKR) opened and all ninety-one CEAs inserted fully. All other systems and components functioned properly during this event.

The feeder breaker, which initiated this event, has a Brown Boveri Solid State Trip Device. It was determined during the startup testing program for Unit 2 that devices of this type, which have serial numbers below 28300, are subject to spurious trips and are being replaced. These devices have been replaced in all applicable safety-related 1E breakers and are being replaced in applicable non-1E breakers as the equipment becomes available for both Units 2 and 3. Therefore, the feeder breaker (2B1401) for 480 VAC load center (2B14) has had its trip device replaced, the oil leak at K005 has been repaired and all systems and components involved have been verified as operable.

There are no reasonable or credible alternative conditions under which this event would have been more severe.

Southern California Edison Company

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

J. G. HAYNES
STATION MANAGER

TELEPHONE
(714) 492-7700

September 25, 1984

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

Subject: Docket No. 50-361
30-Day Report
Licensee Event Report No. 84-050
San Onofre Nuclear Generating Station, Unit 2

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 actuation of the Reactor Protection System. The health and safety of plant personnel or the public were not affected by this event.

If you require any additional information, please so advise.

Sincerely,

Vbr Haynes

Enclosure: LER 84-050

cc: A. E. Chaffee (USNRC Senior Resident Inspector, Units 1, 2 and 3)
J. P. Stewart (USNRC Resident Inspector, Units 2 and 3)

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

Institute of Nuclear Power Operations (INPO)

IE22
1/1