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 2

TITLE (4)
REACTOR POWER INCREASE

EVENT DATE (5) MONTH	DAY	YEAR	LER NUMBER (6) YEAR	SEQ. NUMBER	REV. NUMBER	REPORT DATE (7) MONTH	DAY	YEAR	OTHER FACILITIES INVOLVED (8) FACILITY NAMES	DOCKET NUMBER(S)
										0 5 0 0 0
0 7	0 9	8 4	8 4	0 2 9	0 0	0 8	1 4	8 4		0 5 0 0 0

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more of the following) (11)

OPERATING MODE (9): 1				
POWER LEVEL (10): 1 0 0	20.402(b)	20.405(c)	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)	Voluntary Report
	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	MANUFAC-TURER	REPORTABLE TO NPRDS	CAUSE	SYSTEM	COMPONENT	MANUFAC-TURER	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)

This report is submitted to provide information concerning a partial loss of extraction steam feedwater heating which resulted in a reactor power increase above rated power. On July 9, 1984, at approximately 1415 with Unit 3 in Mode 1 at 100% power, during an adjustment to second point heater 3E-038, a high level excursion occurred resulting in the closure of extraction steam valves 3HV-8808 and 3HV-8800 and isolating the second and first point heaters from the steam supply. Due to steam loss, colder feedwater to the steam generators resulted in a reactor power increase. Calibrated reactor power increased to about 104%. No Pre-Trips were actuated, since uncalibrated power used to produce the Pre-Trip signal remained below 102.5%. Turbine power was reduced, and reactor power was restored to 100% within 30 minutes.

While the actions taken to reduce power were timely and adequate, the importance of reducing power promptly has been re-emphasized to control operators, and technicians have been re-instructed to notify the control room immediately when equipment is actuated. In addition, while Alarm Response Procedure SO23-5-2.11 is adequate, relative to procedural actions, it will be changed to include loss of feedwater heating as one of the potential causes of Linear Power Level Hi Channel Pre-Trip. Additionally, NRC guidance on "timely manner" in reducing reactor power is being reviewed for applicability.

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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 3	0 5 0 0 0 3 6 2	8 4	0 2 9	0 0	0 2 OF 0 2

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

This report is submitted to provide information concerning a partial loss of extraction steam feedwater heating which resulted in a reactor power increase above rated power. On July 9, 1984, at approximately 1415 with Unit 3 in Mode 1 at 100% power, annunciator CR53A46 (EIIS Component Function Identifier ANN) alarmed due to high level in the second point heater 3E-038 (EIIS Component Code HX). A technician started lowering the level setpoint on level controller 3LC-3632 (EIIS Component Function Identifier LC). At this point, a high level excursion occurred and the extraction steam isolation valves 3HV-8808 and 3HV-8800 (EIIS Component Function Identifier ISV) received a high level extraction steam isolation valve closure signal, and the valves shut. The temperature of the feedwater entering the steam generators decreased, resulting in the reduction of Reactor Coolant System (EIIS System Identifier AB) cold leg temperature (Tc) and causing calibrated reactor power (corrected for the decrease in cold leg temperature) to rise to about 104% (3525 megawatts thermal). No Pre-Trips were actuated, since uncalibrated power as measured by the excore nuclear instruments which are used to produce the Pre-Trip signal, remained below the Pre-Trip setting discussed below.

Operators noted the closure of 3HV-8808 and 3HV-8800, and began to reduce turbine output to raise Tc. Reactor power was restored to 100% at 1445. Extraction steam isolation valves 3HV-8808 and 3HV-8800 were fully reopened at 1614 and 1628, respectively, and the turbine load was returned to 100% power. A similar occurrence at Unit 2 on February 1984 was previously reported in LER 84-010.

While the actions taken to reduce power to within the allowed 100% during this event, were timely and adequate, we have taken further action in re-emphasizing to control operators the importance of promptly reducing power when a minor increase such as reported herein occurs. Technicians were also re-instructed in the need to notify the control room immediately when equipment is actuated during performance of their work. Additionally, a review of our Alarm Response Procedure indicates that the procedural actions are adequate; however, we will change SO23-5-2.11 to include a loss of feedwater heating as a potential cause for Linear Power Level Hi Channel Pre-Trip. In order to determine whether additional actions are required, NRC guidance on "timely manner" in reducing reactor power is being reviewed for applicability. The results of the review will be discussed with the Resident Inspector.

This power transient was within the design basis of the plant. Plant overpower trips and Pre-Trips are set at 110% and 102.5% rated power, respectively. There was no safety significance to this power increase and there are no credible circumstances under which this event could have been more severe.

Southern California Edison Company

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J. G. HAYNES
STATION MANAGER

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August 14, 1984

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

Subject: Docket No. 50-362
30-Day Report
Licensee Event Report No. 84-029
San Onofre Nuclear Generating Station, Unit 3

This submittal provides an informational Licensee Event Report (LER) for an occurrence involving a partial loss of extraction steam feedwater heating which resulted in a calibrated reactor power increase of 4% above rated power. Neither the health and safety of plant personnel nor the public were affected by this event.

If you require any additional information, please so advise.

Sincerely,

VG Haynes

Enclosure: LER No. 84-029

cc: A. E. Chaffee (USNRC 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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