SOUTHERN CALIFORNIA
EDISON
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R. W. Krieger
Vice President
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U.S. Nuclear Regulatory Commission
Document Control Room
Washington, D.C. 20555

Subject: Docket No. 50-362
30 Day Report
Licensee Event Report No. 97-001
San Onofre Nuclear Generating Station, Unit 3

Pursuant to 10CFR 50.73(d), this submittal provides the required 30-day written Licensee Event Report (LER) for an occurrence involving reactor coolant system pressure boundary leakage. Neither the health nor the safety of plant personnel or the public was affected by this occurrence.

Sincerely,

RWKrieger

Enclosure: LER No. 97-001

cc: E. W. Merschoff, Regional Administrator, NRC Region IV
A. T. Howell, III, Director, Division of Reactor Safety, NRC Region IV
K. E. Perkins, Jr., Director, Walnut Creek Field Office, NRC Region IV
J. A. Sloan, NRC Senior Resident Inspector, San Onofre Units 2 & 3
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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|2	1 of 0 5

Title (4)

Reactor Coolant System Leakage - Instrument Thermowell Nozzles

EVENT DATE (5)			LER NUMBER (6)			REPORT DATE (7)			OTHER FACILITIES INVOLVED (8)	
Month	Day	Year	Year	Sequential Number	Revision Number	Month	Day	Year	Facility Names	Docket Number(s)
0 4	1 2	9 7	9 7	0 0 1	0 0	0 5	0 9	9 7	None	

OPERATING MODE (9)	2
POWER LEVEL (10)	0 0 0

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

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)	Other (Specify in Abstract below and in text) Voluntary
20.405(a)(1)(iii)	XX 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
R. W. Krieger, Vice President, Nuclear Generation	AREA CODE 7 1 4 3 6 8 - 6 2 5 5

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				

On April 11, 1997, plant operators began reducing reactor power for the Unit 3, Cycle 9 refueling outage. Between April 12-17, 1997, Edison inspected all Inconel Alloy 600 instrument nozzles in the RCS. During this inspection, Edison noted that four RCS nozzles had leaked during plant operation, and a fifth is suspected of leaking.

Technical Specification (TS) 3.4.13.a allows no pressure boundary leakage in Modes 1 through 4. If this LCO is not met, this TS requires the Unit to be in Mode 5 within 36 hours. Based on the existence of boric acid crystals around some of the leak locations, Edison believes it likely, that one or more of the leaks existed during plant operation (more than 36 hours prior to entering Mode 5). Consequently Edison is reporting these occurrences in accordance with 10CFR50.73(a)(2)(i).

Based on evaluation of the leak size, leak location, deposition of boric acid crystals, and previous experience with Inconel 600 nozzle leakage, Edison suspects the leakage to be from a crack through the nozzle in the heat affected zone (HAZ) of the partial penetration weld on each of the instrument nozzles. Historically, Edison has determined that similar cracks were caused by Primary Water Stress Corrosion Cracking (PWSCC) of alloy 600 type materials. Cracking of Inconel 600 material is well known and is likely the root cause of these leaks.

If, after further analysis, a cause other than provided here is found, this LER will be supplemented with additional details.

Edison will replace the outer half of the Inconel 600 material of the original nozzles with Inconel 690. Required welding will be completed in accordance with the ASME III, Class 1 welding guidelines. [Note: This repair is similar to that reported in LER 2-97-004.]

Description of Event:

Plant:	San Onofre Nuclear Generating Station Unit 3
Reactor Vendor:	Combustion Engineering
Event Date:	April 12, 1997
Event Time:	0542
Mode:	2
Power:	1E-7%
Temperature:	545 Deg F
Pressure:	2250 PSIA

On April 11, 1997, plant operators began reducing reactor power for the Unit 3, Cycle 9 refueling outage. Between April 12-17, 1997, Edison inspected all Inconel Alloy 600 instrument nozzles in the RCS [AB]. During this inspection, Edison noted that four of the five following RCS nozzles had leaked during plant operation, and the fifth is suspected of leaking:

- RCS Hot Leg Loop 2 Narrow Range Temperature sensor
- RCS Hot Leg Loop 2 Temperature sensor
- Steam Generator (E088) Channel C RCS Differential Pressure sensor
- RCS Loop 1 Hot Leg spare thermowell
- RCS Cold Leg Loop 2B Narrow Range Temperature sensor

Technical Specification (TS) 3.4.13.a allows no pressure boundary leakage in Modes 1 through 4. If this LCO is not met, this TS requires the Unit to be in Mode 5 within 36 hours. Based on the existence of boric acid crystals around some of the leak locations, Edison believes it likely that one or more of the leaks existed during plant operation (i.e., more than 36 hours prior to entering Mode 5). Consequently, Edison is reporting these occurrences in accordance with 10CFR50.73(a)(2)(i).

Cause of the Event:

Based on evaluation of the leak size, leak location, deposition of boric acid crystals, and previous experience with Inconel 600 nozzle leakage, Edison suspects the leakage is from a crack through the nozzle in the heat affected zone (HAZ) of the partial penetration weld on each of the instrument nozzles. Historically, Edison has determined that similar cracks were caused by Primary Water Stress Corrosion Cracking (PWSCC) of alloy 600 type materials. Therefore, cracking of Inconel 600 material is well known, and is believed to be the root cause of the leaks.

If, after further analysis, a different cause is found, this LER will be supplemented with additional details.

Corrective Actions:

Four of these nozzles (temperature measurement) were originally designed and installed as a one piece nozzle made of Inconel 600 welded with a J-Groove weld on the inside of the RCS piping (see Figures 1-5). The pressure measurement nozzle differs from the temperature nozzles in that it has a SA-182, F316 stainless steel socket butt welded to the Inconel 600 nozzle shank.

Edison will replace the outer half of the Inconel 600 material of the original nozzles with Inconel 690. Required welding will be completed in accordance with the ASME III, Class 1 welding guidelines. [Note: This repair is similar to that reported in LER 2-97-004; see additional information section below.]

Safety Significance of the Event:

Based on the suspected location of the cracks and previous plant experience, a complete circumferential failure of an RCS instrument nozzle (about 1.315 inch diameter) is not believed to be a credible event. Nevertheless, Edison has verified that the consequences of such a failure would be bounded by the small break loss of coolant accident (SBLOCA) analyzed in the UFSAR (multiple simultaneous failures of instrument nozzles is not credible). The actual leak rate through these cracks was not measurable. Consequently, this event had minimal safety significance.

Additional Information:

In the past three years, Edison has submitted the following three LERs on RCS leakage events:

1. LER 3-95-001 reported RCS nozzle leakage caused by PWSCC of alloy 600 type materials. As detailed in Report No. 90022, "Susceptibility of Reactor Coolant System Alloy 600 Nozzles To Primary Water Stress Corrosion Cracking and Replacement Program," Rev. 1, dated February 9, 1995, Edison developed a nozzle inspection program based on the nozzle's susceptibility to PWSCC. As a result, all Reactor Coolant System Inconel Alloy 600 nozzles are scheduled for routine inspection during every refueling outage, which resulted in identification of the leaks reported herein.

2. LER 2-97-004 reported leakage from the pressurizer liquid temperature thermowell nozzle. Edison concluded the leak was caused by Primary Water Stress Corrosion Cracking (PWSCC) of alloy 600 type materials. The outer half of the nozzle was removed and replaced with Inconel 690.

3. LER 3-96-004 reported RCS leakage from a broken thermowell, a cause not present in any of the occurrences reported herein. That thermowell failure mechanism appeared to fatigue, not PWSCC.

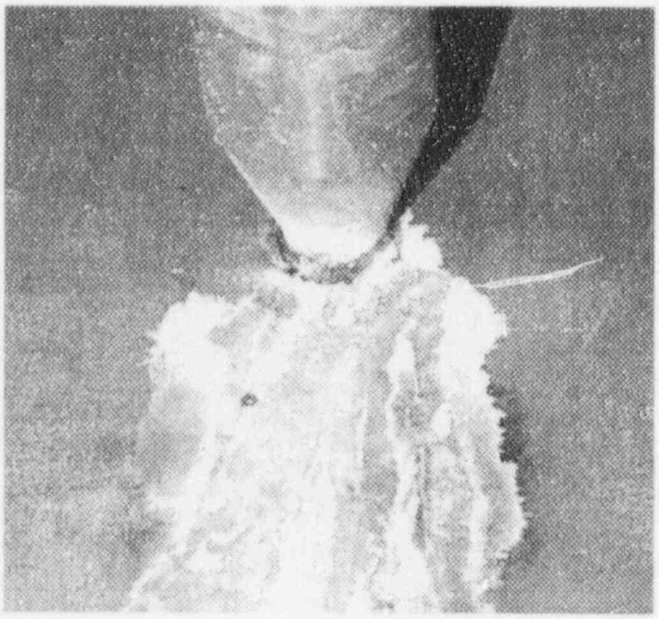

Figure 1
Hot Leg Loop 2 Narrow Range Temperature

Figure 2
Hot Leg Loop 2 Temperature

Figure 3
Steam Generator Channel C Differential Pressure

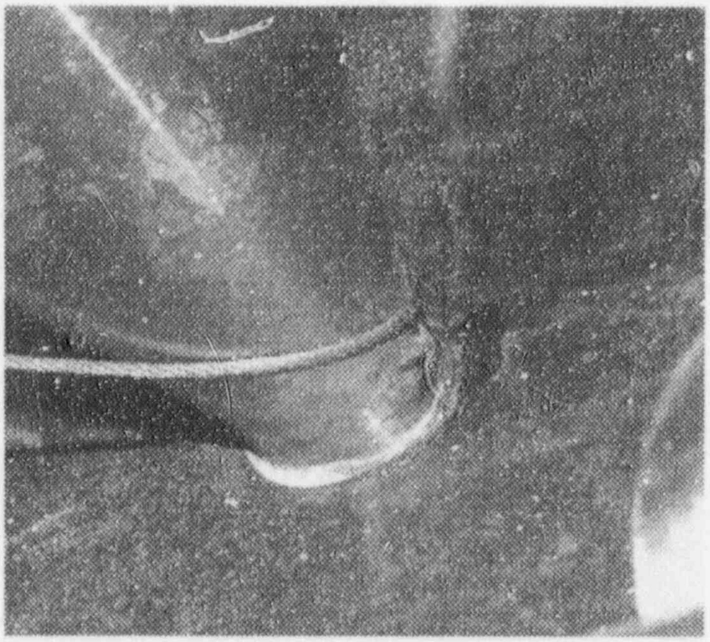

Figure 4
Hot Leg Loop 1 Spare Thermowell

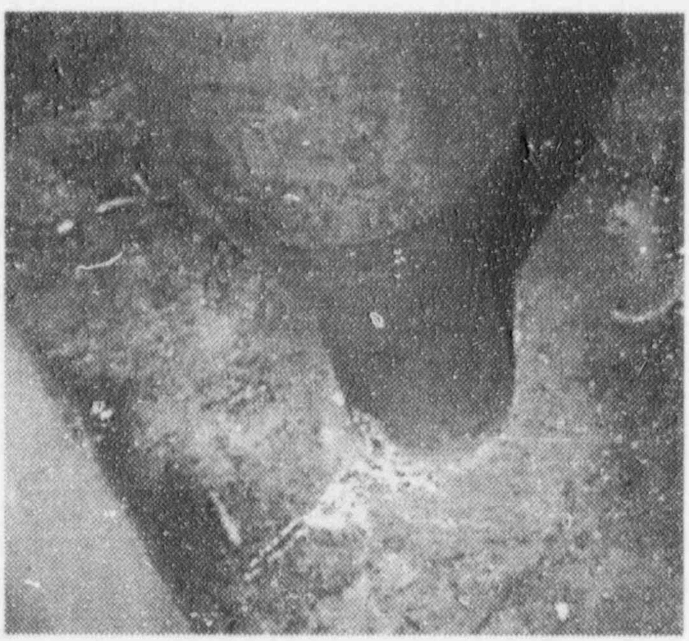

Figure 5
Cold Leg Loop 2 Narrow Range Temperature