



SOUTHERN CALIFORNIA
EDISON

An EDISON INTERNATIONAL Company

R. W. Krieger
Vice President
Nuclear Generation

April 2, 1997

U.S. Nuclear Regulatory Commission
Attention: Document Control Room
Washington, D.C. 20555

Subject: Docket No. 50-361
Voluntary Report
Licensee Event Report No. 97-004
San Onofre Nuclear Generating Station, Unit 2

This submittal provides a voluntary written report regarding reactor coolant system leakage from the pressurizer thermowell nozzle. Neither the health nor the safety of plant personnel or the public was affected by this occurrence.

Sincerely,

R. Walden for R.W. Krieger

Enclosure: LER No. 97-004

cc: E. W. Merschoff, Regional Administrator, NRC Region IV
A. T. Howell, III, Director, Division of Reactor Projects, 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) SAN ONOFRE NUCLEAR GENERATING STATION, UNIT 2												Docket Number (2) 0 5 0 0 0 3 6 1				Page (3) 1 of 0 4		
Title (4) Reactor Coolant System Leakage - Pressurizer Thermowell																		
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 3	0 3	9 7	9 7	---	0 0 4	---	0 0	0 4	0 2	9 7	None							
OPERATING MODE (9) 4			THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10CFR (Check one or more of the following) (11)															
POWER LEVEL (10) 0 0 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)				<input checked="" type="checkbox"/> Other (Specify in			
			20.405(a)(1)(iii)				50.73(a)(2)(i)				50.73(a)(2)(viii)(A)				Abstract below and			
			20.405(a)(1)(iv)				50.73(a)(2)(ii)				50.73(a)(2)(viii)(B)				in text)			
20.405(a)(1)(v)				50.73(a)(2)(iii)				50.73(a)(2)(x)				Voluntary						
LICENSEE CONTACT FOR THIS LER (12)																		
Name R. W. Krieger, Vice President, Nuclear Generation												TELEPHONE NUMBER 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	////////							
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SUPPLEMENTAL REPORT EXPECTED (14)												Expected Submission Date (15)		Month Day Year				
Yes (If yes, complete EXPECTED SUBMISSION DATE)												X NO						

At about 2045 on March 3, 1997, an Operator making a routine Mode 4 inspection inside containment noted steam emanating from the pressurizer. At that time, Unit 2 was in Mode 4 at 0% power.

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. Operators returned the Unit to Mode 5 at 1041 on March 4, 1997 (about 14 hours after discovery). Because Unit 2 was returned to Mode 5 within 36 hours of the onset of pressure boundary leakage, the TS requirements were satisfied and a report to the NRC is not required. Edison is voluntarily reporting this occurrence due to potential industry interest and for NRC reference.

Edison concluded the leak was caused by Primary Water Stress Corrosion Cracking (PWSCC) of alloy 600 type materials of the pressurizer liquid temperature thermowell nozzle. The nozzle was removed and replaced with Inconel 690.

The crack was oriented parallel to the long axis of the nozzle. Consequently, a complete circumferential failure of an RCS instrument nozzle is not believed to be a credible event. Nevertheless, the consequences of such a failure would be bounded by the small break loss of coolant accident (SBLOCA) analyzed in the UFSAR. The actual leak rate during this occurrence was about 0.76 gpm of water. Consequently, this event had minimal safety significance.

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Description of Event:

Plant: San Onofre Nuclear Generating Station Unit 2
Reactor Vendor: Combustion Engineering
Event Date: March 3, 1997
Event Time: 2045
Mode: 4
Power: 000%
Temperature: 279 Deg F
Pressure: 2250 PSIA

On March 2, 1997, Operators began moving Unit 2 from Mode 5 to Mode 4 at the end of a refueling outage. At about 2045 the following day, an Operator (utility, licensed) noted steam emanating from the pressurizer [AB] during a routine Mode 4 inspection inside containment. (This leak was later determined to be from a crack in the heat affected zone (HAZ) of the weld for a pressurizer temperature instrument nozzle). At that time, Unit 2 was in Mode 4 at 0% power. 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. In response to the observed steam leak, Operators returned the Unit to Mode 5 at 1041 on March 4, 1997 (about 14 hours after discovery).

The affected pressurizer nozzle was visually inspected during the last (U2C9) refueling outage (see Additional Information, Item 1, below), and no evidence of leakage was observed. About 6 hours before discovery of the leak, an Operator (utility, non-licensed) completed a VT-2 (visual) inspection of another component near the pressurizer temperature nozzle; however, there was no pressure boundary leakage observed at that time. Subsequent examination of residue from near the leak provided no compelling evidence that the leak existed prior to the VT-2 inspection. Edison concluded the leak occurred sometime between the VT-2 examination and its discovery. Because Unit 2 was returned to Mode 5 within 36 hours of the VT-2 inspection, the TS requirements were satisfied and a report to the NRC is not required. Edison is voluntarily reporting this occurrence due to potential industry interest and for NRC reference.

Cause of the Event:

The leak was determined to be from a crack in the heat affected zone (HAZ) of the weld for pressurizer liquid temperature instrument 2TE0101 nozzle. See Figure 1. Edison has concluded the crack was caused by Primary Water Stress Corrosion Cracking (PWSCC) of alloy 600 type materials. Cracking of Inconel 600 material is well known and was the root cause of this leak. The crack initiated on the inside diameter of the nozzle in the HAZ (not the weld).

Corrective Actions:

2TE0101 was originally designed and installed as a one piece nozzle made of Inconel 600 and SA-182, F316 stainless steel, welded with a J-Groove weld on the inside of the pressurizer (see Figure 1).

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Edison replaced the outer half of the Inconel 600 material of the original nozzle with Inconel 690. Required welding was completed in accordance with ASME III, Class 1 welding guidelines. [Note: This repair was similar to that reported in LER 3-95-001; see additional information section below].

Safety Significance of the Event:

Edison confirmed the crack noted above was oriented parallel to the long axis of the nozzle. Consequently, 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. The actual leak rate during this occurrence was about 0.76 gpm. Consequently, this event had minimal safety significance.

Additional Information:

In the past three years, Edison has reported two similar 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 PWSSC. As a result, the pressurizer nozzles are scheduled for routine inspection during every refueling outage.
2. LER 3-96-004 reported RCS leakage from a broken thermowell. In that event, the RCS leakage was caused by a failed thermowell, a cause not present in this case.

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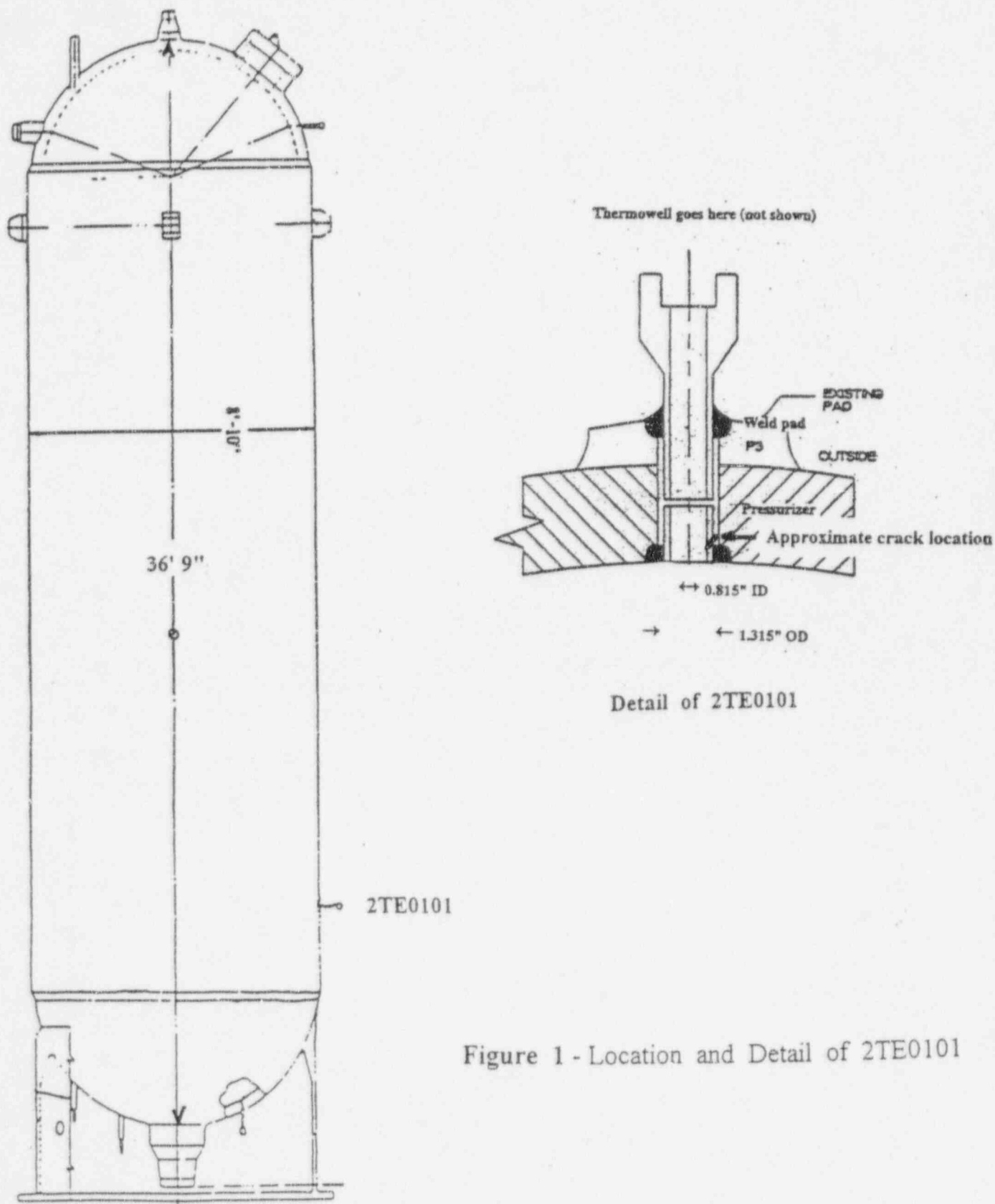


Figure 1 - Location and Detail of 2TE0101