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Dr. Raymond W. Waldo
Vice President - Nuclear
San Onofre

August 24, 2007

U.S. Nuclear Regulatory Commission
Document Control Desk
Washington, D.C. 20555-0001

Subject: **Docket No. 50-361**
Licensee Event Report No. 2007-001-01
San Onofre Nuclear Generating Station, Unit 2

Reference: Ray Waldo (SCE) letter to NRC Document Control Desk, "Licensee Event Reports Nos. 2007-001 and 2007-002", dated August 17, 2007

Dear Sir or Madam:

This submittal provides revision 1 to Licensee Event Report (LER) 2007-001, which describes the loss of Instrument Air pressure and resulting manual trip of the reactor on June 20, 2007. Southern California Edison has updated the Safety Significance section of this LER to include the Probability Risk Assessment (PRA) for this event. This event did not affect the health and safety of either plant personnel or the public.

If you require any additional information, please contact me.

Sincerely,

Raymond Waldo

Unit 2 LER No. 2007-001-01

cc: B. S. Mallett, NRC Regional Administrator, Region IV
C. C. Osterholtz, NRC Senior Resident Inspector, San Onofre Units 2 & 3

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NRC FORM 366 (7-2001) U.S. NUCLEAR REGULATORY COMMISSION	APPROVED BY OMB: NO. 3150-0104 EXPIRES: 06/30/2007
LICENSEE EVENT REPORT (LER) (See reverse for required number of digits/characters for each block)	Estimated burden per response to comply with this mandatory information collection request: 50 hours. Reported lessons learned are incorporated into the licensing process and fed back to industry. Send comments regarding burden estimate to the Records Management Branch (T-6 E6), U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, or by internet e-mail to bjs@nrc.gov, and to the Desk Officer, Office of Information and Regulatory Affairs, NEOB-10202 (3150-0104), Office of Management and Budget, Washington, DC 20503. If a means used to impose information collection does not display a currently valid OMB control number, the NRC may not conduct or sponsor, and a person is not required to respond to, the information collection.

1. FACILITY NAME	2. DOCKET NUMBER	3. PAGE
San Onofre Nuclear Generating Station (SONGS) Unit 2	05000361	1 OF 4

4. TITLE
Instrument Air system failure results in Manual Reactor Trip

5. EVENT DATE			6. LER NUMBER			7. REPORT DATE			8. OTHER FACILITIES INVOLVED	
MO	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REV NO	MO	DAY	YEAR	FACILITY NAME	DOCKET NUMBER
									None	
6	20	2007	2007-001-01			08	20	2007	FACILITY NAME	DOCKET NUMBER

9. OPERATING MODE	1	11. THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check all that apply)							
			20.2201(b)		20.2203(a)(3)(ii)		50.73(a)(2)(ii)(B)		50.73(a)(2)(ix)(A)
10. POWER LEVEL	96		20.2201(d)		20.2203(a)(4)		50.73(a)(2)(iii)		50.73(a)(2)(x)
			20.2203(a)(1)		50.36(c)(1)(i)(A)	X	50.73(a)(2)(iv)(A)		73.71(a)(4)
			20.2203(a)(2)(i)		50.36(c)(1)(ii)(A)		50.73(a)(2)(v)(A)		73.71(a)(5)
			20.2203(a)(2)(ii)		50.36(c)(2)		50.73(a)(2)(v)(B)		OTHER Specify in Abstract below or in NRC Form 366A
			20.2203(a)(2)(iii)		50.46(a)(3)(ii)		50.73(a)(2)(v)(C)		
			20.2203(a)(2)(iv)		50.73(a)(2)(i)(A)		50.73(a)(2)(v)(D)		
			20.2203(a)(2)(v)		50.73(a)(2)(i)(B)		50.73(a)(2)(vii)		
			20.2203(a)(2)(vi)		50.73(a)(2)(i)(C)		50.73(a)(2)(viii)(A)		
			20.2203(a)(3)(i)		50.73(a)(2)(ii)(A)		50.73(a)(2)(viii)(B)		

12. LICENSEE CONTACT FOR THIS LER	
NAME	TELEPHONE NUMBER (Include Area Code)
R. W. Waldo, VP Generation	949-368-8725

13. COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT

CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX	CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX

14. SUPPLEMENTAL REPORT EXPECTED				15. EXPECTED SUBMISSION DATE	MONTH	DAY	YEAR
	YES (If yes, complete EXPECTED SUBMISSION DATE)	X	NO				

16. ABSTRACT (Limit to 1400 spaces, i.e., approximately 15 single-spaced typewritten lines)

On June 20, 2007, at about 2240 PDT, Unit 2 was operating in Mode 1 at about 96 percent power when a line in the instrument air system separated at a soldered connection. The resulting loss of Instrument Air (IA) pressure caused the loss of control of the Steam Generator Feedwater Regulating valves. One steam generator level rose uncontrolled necessitating the control room Operators to trip the reactor manually at about 2250 PDT, June 20, 2007. Operators subsequently manually tripped the Main Feedwater pumps to stop excess feedwater to the steam generators and actuated the Auxiliary Feedwater System. As designed, loss of IA system pressure isolates cooling water to the normal containment coolers and Operators manually started the Containment Emergency Cooling Units as a conservative measure.

Southern California Edison(SCE) reported this occurrence to the NRC Operations Center at 0024 PDT on June 21, 2007 (NRC Event Log Number 43435). SCE is providing this follow-up written report in accordance with 10CFR50.73(a)(2)(iv)(A).

The soldered connection failed due to improper fit and slow acting corrosion. SCE inspected the IA system piping and installed pipe clamps to strengthen piping joints where needed to add additional margin.

Plant:	San Onofre Nuclear Generating Station (SONGS) Unit 2
Discovery Date:	June 20, 2007
Reactor Vendor:	Combustion Engineering
Mode:	Mode 1 – Power Operation
Power:	96 percent

Background

San Onofre Units 2 and 3 share a common instrument air (IA) [LD] system that pneumatically operates components in the plant. The IA system was designed as a Quality Class III (non-safety-related) system and is necessary for plant startup and power operation. Safety Related Equipment supplied by the IA system is designed to fail to their safety function positions on a loss of air pressure. This design allows a non-safety-related system to support safety-related components.

The IA system is provided with three 100 percent capacity air compressors powered from different power supplies. The IA system can be supplemented with air from Service Air [LF] and an independent backup nitrogen supply in case of system failure. Although the IA system is common to both Units 2 and 3, check valves isolate the unit with a break when a high change in pressure occurs. Capacity of the IA system is such that normal system operation could continue with a break in a 1" or smaller IA line.

Description of Event

On June 20, 2007, at 2240 PDT, Unit 2 was operating in Mode 1 at about 96 percent power when a 3" diameter copper line in the instrument air system separated at a soldered connection joint. The loss of Instrument Air pressure caused the feedwater bypass valves to close. The resulting decrease in steam generator levels caused the controller to increase the speed (flow) of the main feedwater pumps. Due to the differences in the positions of the Main Feedwater Regulating valves that lock as-is on loss of instrument air, the steam generators filled unequally. When Steam Generator E088 level reached about 85 percent, the Control Room Operators tripped the reactor. Operators subsequently manually tripped the Main Feedwater pumps to stop excess feedwater to the steam generators and actuated the Auxiliary Feedwater System. Operators manually initiated the Emergency Feedwater Actuation Signals [JE] to start the Auxiliary Feedwater System [SA]. As designed, the loss of IA system pressure also resulted in the isolation of cooling water to the normal containment coolers. Therefore, operators manually started the Containment Emergency Cooling Units as a conservative measure.

The IA back-up system and check valves sustained IA service to Unit 3 until the break could be completely isolated.

Southern California Edison reported this occurrence to the NRC Operations Center at 0024 PDT on June 21, 2007 (NRC Event Log Number 43435), in accordance with 10CFR50.72(b)(2)(iv)(B) for actuation or the Reactor Protection System, and 50.72(b)(3)(iv)(A) for valid actuation of the Emergency Feedwater and Containment Emergency Cooling Unit. SCE is providing this follow-up written report in accordance with 10CFR50.73(a)(2)(iv)(A).

Cause of Event

The cause of this event is loss of instrument air system pressure when the instrument air system separated at a soldered connection in a section of 3" diameter piping. The connection failed because (1) a weak solder joint, and (2) corrosion of the solder joint.

This section of pipe and the failed solder joint were installed during original plant construction (~1980). The solder connection was weak because the gap between the tube and the coupling was too large. The larger gap did not allow for the capillary action necessary to provide an even distribution of the melted solder and the solder pooled at the bottom of the coupling. The larger gap also allowed solder flux to remain in the solder, which lead to slow acting corrosion of the solder joint. When corrosion had sufficiently reduced the solder joint strength, the connection separated.

Corrective Actions

- SCE replaced the Instrument Air solder joint that had separated and the leaking joint adjoining it.
- SCE inspected all soldered joints on piping greater than one inch in the IA system on Units 2 and 3 (approximately 818 joints). Thirty-two (32) clamps were installed on joints indicating leakage to add additional margin.
- SCE identified the other system containing solder joints at SONGS (Domestic Water System, up to 6" diameter). Given that there is no indication of water leakage, SCE concluded that Domestic Water System can be eliminated from the Extent of Condition scope.

Additional corrective actions will be implemented in accordance with SONGS Corrective Action Program.

Safety Significance

The Instrument air system is a non-safety-related system. Safety Related Equipment supplied by the IA system are designed to fail to their safety function position on a loss of air pressure. In response to this manual reactor trip reported in this LER, plant equipment responded as required. This occurrence remained bounded by the Updated Final Safety Analysis Report evaluation of a loss of instrument air event and did not affect the health and safety of either plant personnel or the public.

An assessment of the conditional core damage probability (CCDP) and the conditional large early release probability (CLERP) for the June 20, 2007 event determined that the Unit 2 CCDP and CLERP were 3.3E-6 and 1.9E-7, respectively. The assessment was based on the reported actual component unavailabilities, system alignments and operating conditions that existed at the time of the event.

Additional Information:

During this event, there was a separate Technical Specification violation described in LER 2007-002.

In the past three years, there have been no other occurrences of a failed solder connection resulting in a reactor trip.