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)
Pilgrim Nuclear Power Station - Unit No. 1	0 5 0 0 0 2 9 2	1 OF 0 3

TITLE (4)
Weld Leak on Reactor Water Level Instrument Line

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	1 6	8 6	8 6	— 0 0 6	— 0 0	0 4	1 5	8 6		0 5 0 0 0
										0 5 0 0 0

OPERATING MODE (9)	N	THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more of the following) (11)							
POWER LEVEL (10)	0 8 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)		OTHER (Specify in Abstract below and in Text, NRC Form 366A)
			20.405(a)(1)(iii)	X	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)
Paul Hamilton - Sr. Plant Engineer	6 1 7 7 4 6 - 7 9 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
B	J C	P S F	B 1 3 0	Y					

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 March 16, 1986, at 0115 hours, the reactor was shutdown in response to an increase in unidentified drywell leakage. Cause of the leakage was a cracked weld at a 2-inch by 1-inch reducing coupling that joins the 1-inch reactor water level instrument line to reactor vessel penetration N-16A. Root cause of the weld crack was an incomplete root pass penetration on the pipe to coupling socket weld combined with thermally induced stresses.

Corrective action was to repair and sleeve the affected area of piping and reduce the thermal stress. To preclude recurrence the other 3 instrument line socket welds were inspected to ensure similar problems did not exist.

During the repair work it was discovered that the N-16A penetration is installed backwards. This means that Inconel cladding was shop welded to the 304 stainless steel safe end using Inconel filler; and the Inconel end of the penetration was field welded to 304 stainless steel piping using 308 filler materials. Although the field weld is difficult to perform, radiography indicates the weld is sound. The long term corrective action is to replace the 304 to inconel field weld during the next refueling outage. The penetration orientation is not related to the cracked weld discussed above. Other similar penetrations were inspected and found to be installed correctly.

NRC Form 366
(9-83)

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On March 15, 1986, during steady state operation at approximately 0800 hours, an increase in unidentified drywell leakage from 0.3 gpm to approximately 2.0 gpm was observed by Operations personnel. In response, to investigate cause of the increased leakage, a controlled shutdown was initiated at 0820 and completed on March 16, 1986, at 0115 hrs.

During the shutdown, a drywell inspection was performed. During this inspection a leak was observed in the 2" x 1" reducing coupling socket weld that joins the 1-inch reactor water level instrument line to reactor vessel penetration N-16A. The leak resulted from a crack in the fillet weld on the 2-inch side of the coupling and extended about 120 degrees around the circumference.

In the event the cracked weld had failed completely, a 2-inch unisolable leak would have occurred approximately 70-inches above the top of active fuel. A 2-inch leak is within the capabilities of normal or emergency makeup sources to the reactor and is within the safety analysis of Pilgrim's Final Safety Analysis Report. Both the normal and emergency makeup sources, i.e.. Feedwater, High Pressure Coolant Injection, Low Pressure Coolant Injection, and Core Spray were operable when the leak was discovered.

Root cause analysis of this event identified an incomplete root pass penetration on the 2-inch pipe-to-coupling socket weld. Also contributing to the failure was a "bound" pipe guide and some missing thermal insulation which had apparently been inadvertently removed previously. The bound guide resulted from extraneous wire which was wrapped around piping within the common guide assembly. It is postulated that stresses resulting from the "bound" guide and missing insulation coupled with the incomplete root pass caused the failure. Inter-granular Stress Corrosion Cracking (IGSCC) was ruled out as a factor contributing to the cracked weld.

Corrective action was to repair and sleeve the affected area of piping in accordance with Temporary Modification (TM) 86-11. Resolution of the "bound up" pipe guide was to remove the previously discussed wire and restore the guide to the original design configuration and to reinsulate. The other piping in the bound guide was evaluated and design parameters/stresses were found to be within acceptable limits. To preclude recurrence the other 3 reactor water level instrument lines were inspected to ensure that insulation was in place, pipe guides were not bound, and that the other three 2-inch pipe-to-coupling socket welds were sound via liquid penetrant testing. No similar problems were noted.

During the repair, an unrelated problem was discovered in that the N-16A penetration is installed backwards. The penetration is a shop fabrication of inconel and 304 stainless steel which is machined to fit tightly through a penetration in the reactor vessel wall. The Inconel end would be shop welded to the Inconel vessel cladding while the stainless end is designed to

facilitate field welding to stainless steel pipe. The reversal means the 304 stainless end was welded to the Inconel cladding in the shop using Inconel weld wire, a procedure used elsewhere in the vessel. The 304 stainless steel pipe was therefore, field welded to an Inconel penetration using 308 filler wire--a difficult weld. However, original acceptance radiography and radiography taken during the investigation indicate the field weld is sound. As a precaution long term corrective action will be to remove the Inconel-to-304 stainless steel field weld when the sleeve is replaced during RFO 7. The replacement will consist of a shop fabricated assembly with a 304-stainless-steel-to-Inconel transition. This will permit an Inconel-to-Inconel field weld at the penetration. (There are five similar penetrations in Pilgrim's reactor vessel; all were inspected and found to be installed correctly.) The reversed installation of the penetration is not related to the cracked weld discussed above. The reversed penetration was manufactured by Combustion Engineering and supplied by General Electric Co.

Boston Edison Nuclear Engineering Department and General Electric Company have analyzed these problems. Safety Evaluation (SE) 1942 was issued to justify continued operation with the reversed penetration and the sleeve over the cracked weld.

On April 1, 1986, the unit was returned to service. A search of LER records identified no previous events of a similar nature. The EIIS System and component codes are JC and PSF respectively.

BOSTON EDISON COMPANY
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WILLIAM D. HARRINGTON
SENIOR VICE PRESIDENT
NUCLEAR

April 15, 1986
BECo Ltr. #86-045

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

Docket No. 50-293
License No. DPR-35

Dear Sir:

The attached Licensee Event Report 86-006-00, "Weld Leak on Reactor Water Level Instrument Line" is hereby submitted in accordance with the requirements of 10CFR50.73.

If there are any questions on this subject, please do not hesitate to contact me.

Respectfully submitted,

W D Harrington

W. D. Harrington

PJH/vp

Enclosure: LER 86-006-00

xc: Dr. Thomas E. Murley
Regional Administrator, Region I
U.S. Nuclear Regulatory Commission
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