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)
Palisades Nuclear Plant	0 5 0 0 0 2 5 5	1 OF 0 4

TITLE (4)
Primary Coolant System-Unidentified Leakage > 1 gpm

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 8	1 0	8 4	8 4	0 1 6	0 0	0 9	1 0	8 4	NA	0 5 0 0 0
									NA	0 5 0 0 0

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

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (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, 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)	[X] 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	
David W. Rogers, Technical Engineer; Palisades	6 1 6	7 6 4 - 8 9 1 3

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)

On August 10, 1984, with the Plant at 29% power, a primary coolant system (PCS) leak rate calculation identified excessive unidentified PCS leakage. The leak was discovered to be from a cracked socket weld on a PCS loop differential pressure sensing line. Following replacement, visual examination of the piping during pressure testing revealed PCS leakage from the identical location.

The occurrences were attributed to fatigue induced cracking of the socket weld. A support was added to the piping to inhibit the dynamic forces which resulted in the failures.

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

On August 10, 1984, at 0345, the results of a Primary Coolant System (PCS) [AB] leak rate calculation indicated an unidentified PCS leak rate of 1.98 gpm. Unidentified leakage in excess of 1 gpm is prohibited by Palisades Technical Specification 3.1.5(a). The Plant was operating at approximately 29% power at the time of the occurrence.

Investigation determined the source of the leakage to be from a cracked socket weld in a 3/4 inch PCS loop differential pressure (d/p) sensing line (See Figure 1) [PSP,PSF;AB]. The d/p sensing line is used to measure the pressure drop across the 'A' Steam Generator [SG;AB]. The outputs of the d/p transmitters [PDT;AB] on this, and similar d/p sensing lines, are also used to determine total core flow.

A Plant shutdown was initiated, with cold shutdown condition achieved at approximately 0200, August 11, 1984. The PCS was drained to facilitate replacement of the cracked socket weld. Subsequently, the elbow-to-pipe socket weld was removed, and replaced with a new pipe/elbow assembly. Pressure testing of the new assembly was scheduled to be completed when the Plant would again be at sufficient pressure for the test. Examination of the cracked weld, which was an original Plant weld, led to the conclusion that the failure was caused by the lack of sufficient weld penetration. Similar welds on the d/p sensing lines were visually inspected and found to be satisfactory. The weak weld was, therefore, considered an isolated occurrence.

Plant heat-up from cold shutdown was initiated on August 20, 1984. At 2001, on August 24, 1984, with the Plant in hot shutdown condition, visual inspection of the d/p sensing line revealed a leak of approximately 0.5 gpm at the same pipe-to-elbow socket weld. Plant cooldown was commenced, with the Plant again in cold shutdown condition at 0720 on August 26, 1984.

Investigation determined that a crack had developed in the socket weld/pipe interface on the underside of the pipe, in approximately the same location as the initial crack. Analysis of the cracked socket weld indicated that the weld itself was not defective. The weld was determined to have failed due to fatigue cracking caused by system vibration. In retrospect, the probable cause of the initial socket weld failure was the effect of the same fatigue phenomenon acting upon the imperfect weld, rather than an end-of-life failure of the imperfect weld, itself.

Continued investigation into the root cause of the occurrences determined that valves 1019A PC and 1019B PC [ISV;AB], which are immediately downstream of the cracked socket weld location, had been replaced during the 1983/1984 refueling outage. The replacemnet valves were heavier than the original valves by approximately 9 lbs. each (7.25 lbs vs 16 lbs). Design documentation completed at the time of the replacement indicated that system stresses would not be significantly higher due to the replacement valves. Justification for this conclusion was not provided with the design documentation, but, presumably, would have been based upon a static evaluation of the piping configuration. Subsequent investigation confirmed that the effect of the added weight on the system piping would not result in unacceptable static loading. When analyzed dynamically, however, the effect of the additional weight with normal system vibration induces motion in the cantilever-type configuration of sufficient magnitude to cause the observed fatigue failure in the socket weld area.

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The pipe/elbow assembly was again replaced with a new assembly. A support was designed and installed between valves 1019A PC and 1019B PC to inhibit the dynamic forces which resulted in the previous failures. Following dye-penetrant examination of the welds, the sensing line was pressure tested and visually inspected during Plant heat-up on September 1, 1984. The design change process will be reviewed to determine whether current methods of technical justification and verification are adequate, or if enhancements are necessary.

No threat to public health or safety resulted from either occurrence. Had the socket weld completely failed, causing the pipe to break free from the elbow, however, a small break LOCA condition would likely have resulted. A small break LOCA is an analyzed accident for the Palisades Plant. At higher power levels, the potential consequences resulting from a complete break would have been more significant. For the actual occurrences, however, with relatively minor PCS leakage from the cracked socket weld, alternate (higher) power levels would have had no significant effect.

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LICENSEE EVENT REPORT (LER) TEXT CONTINUATION

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

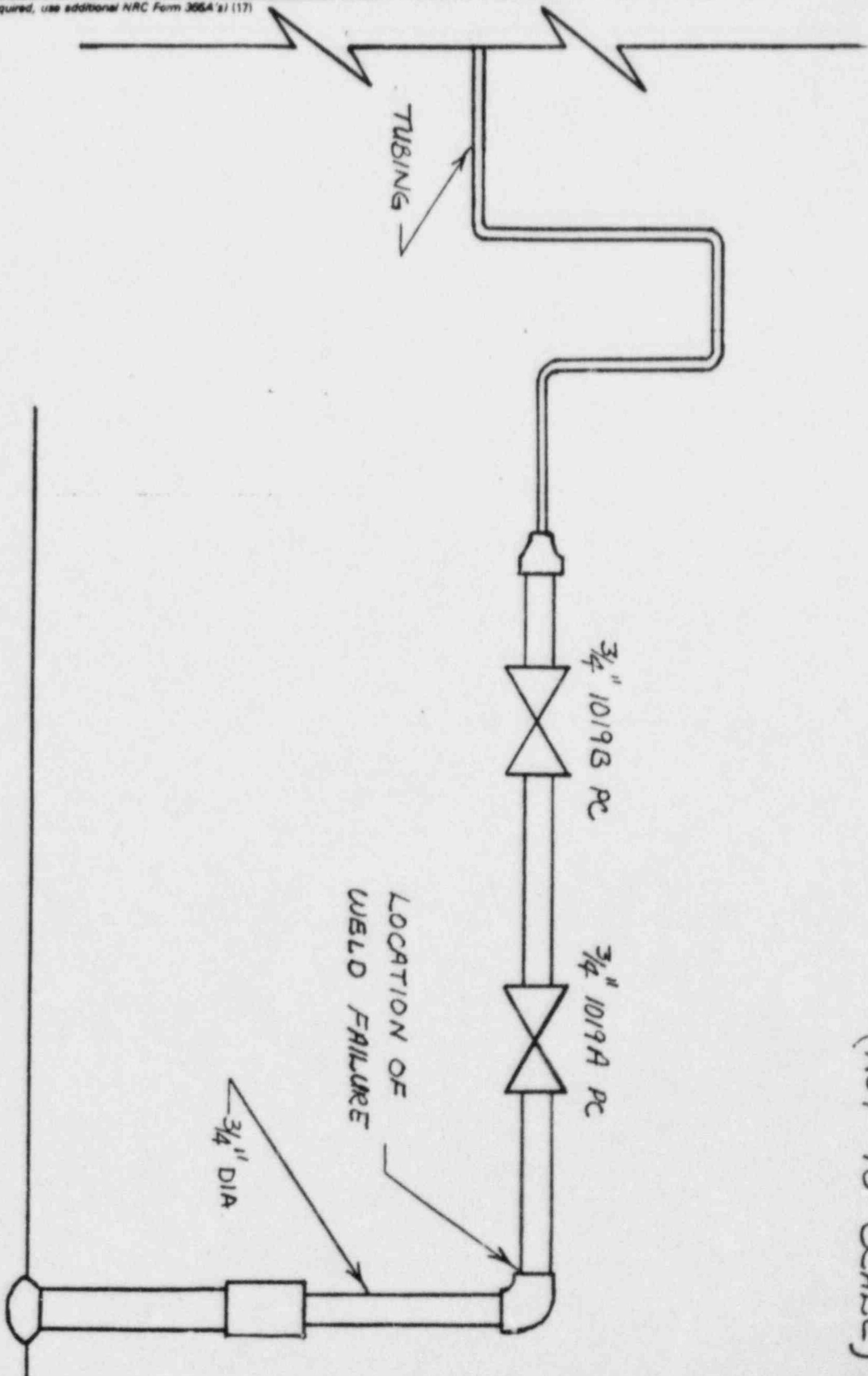

FIGURE 1

(NOT TO SCALE)

NRC FORM 366A
(9-83)

Consumers
Power
Company

General Offices: 1945 West Parnall Road, Jackson, MI 49201 • (517) 788-0550

September 10, 1984

US Nuclear Regulatory Commission
Document Control Desk
Washington, DC 20555

DOCKET 50-255 - LICENSE DPR-20 -
PALISADES PLANT - LICENSEE EVENT REPORT 84-016
(PCS UNIDENTIFIED LEAKAGE >1 GPM)

Attached please find Licensee Event Report 84-016 (PCS Unidentified Leakage >1 gpm) which is reportable to the NRC per 10 CFR 50.73(a)(2)(i) and 50.73(a)(2)(ii).

Vernon R Badini
Thomas C ~~Bordine~~
Staff Licensing Engineer

CC Administrator, Region III, USNRC
Director, Office of Nuclear Reactor Regulation
NRC Resident Inspector - Palisades

Attachment

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