LICENSEE EVENT REPORT (LER)

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TITLE (4)
STEAM LINE FLOW ELEMENT SENSING LINE PINHOLE LEAK

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 NUMBERS(S)
0 8	0 5	9 3	9 3	0 2 9	0 0	1 1	0 5	9 3		0 5 0 0 0
										0 5 0 0 0

OPERATING MODE (9): 3

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

POWER LEVEL (10): 0 0 0

20.402(b)	20.405(C)	50.73(a)(2)(iv)	77.71(b)
20.405(a)(1)(i)	50.36(c)(1)	50.73(a)(2)(v)	73.73(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
C. L. Fies, Licensing Engineer	AREA CODE 5 0 9 3 7 7 - 4 1 4 7

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

SUPPLEMENTAL REPORT EXPECTED (14)

YES (If yes, complete EXPECTED SUBMISSION DATE) X NO

EXPECTED SUBMISSION DATE (15) MONTH DAY YEAR

ABSTRACT (16)

On August 5, 1993, with the plant in Mode 3 (Hot Shutdown) a system engineer discovered a small steam leak of reactor coolant located upstream of the "A" Inboard Main Steam Isolation Valve inside the Primary Containment. The steam flow was from an unisolatable pinhole leak at a flow element sensing line weld.

Control Room personnel immediately initiated a plant cooldown from Mode 3 to Mode 4 (Cold Shutdown) to allow repair of the leak.

The root cause of the steam leak was a weld defect. A defect introduced into the root of the weld during installation served as the initiation point with subsequent crack propagation due to fatigue.

The weld crack was repaired on August 6, 1993.

This event posed no threat to the safety of the public or plant personnel.

Plant Conditions

Power Level - 0%
Plant Mode - 3 (Hot Shutdown)

Event Description

On August 5, 1993, with the plant in Mode 3 (Hot Shutdown), a system engineer discovered a small unisolatable steam leak of primary coolant. The leak was discovered during ongoing work associated with recovery from a reactor scram (see LER 93-027). The leak was located in the Containment Drywell upstream of the "A" Main Steam Isolation Valve (MSIV), MS-V-22A. The steam flow was from an unisolatable pinhole leak emanating from the "A" Main Steam Line Flow Element, MS-FE-5A, sensing line weld.

Immediate Corrective Actions

On August 5, 1993, at 0846 hours, Control Room personnel initiated a plant cooldown from Mode 3 to Mode 4 (Cold Shutdown) to maintain compliance with Technical Specifications associated with PRESSURE BOUNDARY LEAKAGE in Modes 1, 2, or 3.

Further Evaluation, Root Cause, and Corrective Action

Further Evaluation

1. On August 5, 1993, at approximately 0838 hours, this event was reported to the NRC by telephone in accordance with 10CFR50.72(b)(2)(i). This event is also reportable under 10CFR50.73(a)(2)(i)(A), "The completion of any nuclear plant shutdown required by the plant's Technical Specifications." The WNP-2 Technical Specifications do not permit any reactor coolant pressure boundary leakage.

2. An Engineering review of the instrument line calculation was completed. Stresses were calculated to be well below the ASME Code allowables for all deadweight, thermal, and dynamic loading conditions.

3. The weld record for this weld was reviewed and no discrepancies were identified.

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4. Materials and Welding personnel performed a failure analysis on the weld and determined that the cracking had initiated at an undetectable construction defect at the root of the weld. The propagation of the crack from the root was attributed to fatigue. WNP-2 has had fatigue failures of socket welds in the past. The stress concentrations in a socket weld are at the root of the weld and the toe of the weld. If an anomaly exists at the root of the weld, the cyclic loading, if high enough, will tend to propagate the defect. If no anomalies exist at the root of the weld, the cyclic loading, if high enough, will initiate cracking at the toe of the weld. In this case, the root defect, which was not detectable by the required surface examinations, had propagated by fatigue to the weld surface.

5. No intergranular stress corrosion cracking was identified at this weld joint.

Root Cause

The root cause of the steam leak was a weld defect. A defect introduced into the root of the weld during installation acted as an initiation point for the fatigue failure.

Further Corrective Action

1. The weld crack was repaired in accordance with Maintenance Work Request AP4900 and ASME Section XI Plan 2-0975 on August 6, 1993.

2. Engineering has an on going program for identifying candidates for fatigue cracking on the small break LOCA boundaries, with the main emphasis on the primary coolant/containment pressure boundary. The program, however, focuses on high probability failure locations. Socket welded process piping similar to this failure have not historically been a problem area. Cantilevered socket welded vent, drain, and test connections continue to be replaced on a priority basis during annual outages.

Safety Significance

The steam leak was very small and it was concluded the weld defect did not challenge plant safety in that it represented a leakage well within the ability to provide makeup of primary coolant inventory. In addition, the steam plume did not challenge safety-related equipment. Plant records documenting drywell floor drain leakage from August 2, 1993, to August 6, 1993, report zero leakage confirming the character of the leak. Leak before break was demonstrated and if the crack had opened up during further plant operation the unidentifiable leak rate would have eventually increased identifying a problem within the containment.

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Similar events

The Supply System has had other small bore fatigue failures associated with socket welded vent, drain and test connections which are a cantilever beam type design as reported in LERs 90-028 and 91-030.. These, as mentioned above, are being addressed under an ongoing engineering program. There have been only two other instrumentation line failures inside containment, one failure mechanism was indeterminate and the other was due to intergranular stress corrosion. These two failures were not reportable as LERs because they were found during plant outages.

EIIS Information

Text Reference	EIIS Reference System	EIIS Reference Component
Main Steam Isolation Valve	SB	V
Primary Containment	BT	-
Steam Line Flow Element, MS-FE-5A	SB	FE