NRC Form 366
(9-83)

U.S. NUCLEAR REGULATORY COMMISSION
APPROVED OMB NO. 3150-0104
EXPIRES: 8/31/88

LICENSEE EVENT REPORT (LER)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
Joseph M. Farley - Unit 2	0 5 0 0 0 3 6 4	1 OF 0 5

TITLE (4)

Unit Shut Down Due To Pressure Boundary Leakage

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 5 0 0 0
1 2	0 9	8 7	8 7	— 0 1 0 —	0 0	0 1	0 6	8 8		0 5 0 0 0

OPERATING MODE (9): 3

POWER LEVEL (10): 0 0 0

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)	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)
J. D. Woodard, General Manager - Nuclear Plant	2 0 5 8 9 9 - 5 1 5 6

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)

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

EXPECTED SUBMISSION DATE (15): MONTH 0 3 DAY 1 5 YEAR 8 3

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

At 2255 on 12-~~8~~ 9-87, with the unit at 33% power following a refueling outage, it was observed that the containment cooler drain pot levels were abnormally high. A reactor coolant system (RCS) leakage calculation confirmed that the RCS unidentified leakage had increased. A containment entry was made and a leak was identified in the vicinity of the B loop resistance temperature detector (RTD) manifold. A unit shutdown was made to repair the leak. After shutdown and upon closer examination, the pressure boundary leakage was identified to be from the RCS loop B cold leg safety injection line between a check valve and the RCS loop.

The leak resulted from a through wall defect in a welded joint between a long radius elbow and a straight section of pipe. The section of piping containing the defect has been replaced. Preliminary results of the metallurgical evaluation of the failed joint have identified a fatigue mechanism as the cause for crack initiation and propagation. Further investigation and analysis are being performed to determine the source of the fatigue.

A review of construction radiographs and nondestructive examinations (ultrasonic tests and radiographic tests) performed as a result of this event revealed that no problems exist on similar piping welds in the cold leg injection lines of either unit.

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

Plant and System Identification:

Westinghouse - Pressurized Water Reactor
Energy Industry Identification System codes are identified in the text as [XX].

Summary of Event

At 1050 on 12-9-87, while the unit was in Mode 3 (Hot Standby), pressure boundary leakage of slightly less than one gallon per minute was discovered in a section of safety injection piping [BP, BQ] on the reactor coolant system (RCS) [AB] loop B cold leg. The unit had been shut down to investigate leakage suspected at a mechanical joint on the B loop resistance temperature detector (RTD) manifold [JC, JD].

Description of Event

At 2255 on 12-8-87, with the unit at 33% power following a refueling outage, it was observed that the containment cooler drain pot levels were abnormally high. An RCS leakage calculation confirmed that the RCS unidentified leakage had increased. At 0030 on 12-9-87, a containment entry was made and a steam leak in the vicinity of the B loop RTD manifold was observed. This leak was suspected to be at one of several mechanical joints in the RTD manifold. A unit shutdown was made to repair the leak. The unit was shut down (reactor trip breakers open) by 1016 on 12-9-87. After shutdown and upon closer examination, it was determined that the RTD manifold was not leaking but pressure boundary leakage was identified on the RCS loop B cold leg safety injection line between a check valve and the RCS loop.

The unit was placed in Mode 5 (Cold Shutdown) in accordance with Technical Specification action statement requirement 3.4.7.2.a. Cold Shutdown was entered at 1916 on 12-9-87.

Cause of Event

The leak resulted from a through wall defect in a welded joint between a long radius elbow and a straight section of pipe. A section of piping with the defect intact was removed and sent to the hot cell facility at the Westinghouse R&D Center for failure analysis. Preliminary results from the metallurgical evaluation of the failed joint have identified a fatigue mechanism as the cause for crack initiation and propagation. Potential sources of cyclic loading include vibration and thermal cycling. Although the apparent cause of the leak is thermal fatigue, further analyses are being performed to support a conclusion.

Reportability Analysis and Safety Assessment

This event is reportable because the unit was shut down (entered Cold Shutdown) in accordance with the requirements of Technical Specification 3.4.7.2. The RCS leakage was well within the makeup capability of the normal charging pump and there was no radiological release associated with this event. Therefore, the health and safety of the public were not affected.

Corrective Action

Cooldown and depressurization in accordance witn Unit Operating Procedures was commenced to Cold Shutdown. A containment sump level watch was maintained during cooldown and depressurization to determine if RCS leakage increased in magnitude.

An inservice inspection ultrasonic test (UT) was performed in accordance with the applicable ASME code requirement in April of 1986 on this weld with no reportable indications observed. Using the same examination technique, no reportable indication was identified on the cracked weld. An enhanced UT technique was used to identify the crack. All subsequent UT examinations were made using this enhanced technique.

A review of construction radiographs was performed with no cause for the defect being found. The section of the piping containing the defect has been replaced. Ultrasonic tests were performed on corresponding welds on the cold leg injection lines of both units using the enhanced technique and no similar indications were identified. Radiography was performed to resolve any question from the review of the construction radiographs and the ultrasonic inspections.

A stress analysis verification was performed and a review of the design analysis of record for the affected piping was conducted. This review did not reveal any significant discrepancies between applicable design input data and the actual design input data used in the analyses.

A walkdown was performed on piping inside the bioshield on both Unit 1 and Unit 2. The purpose of this walkdown was to check the general condition of the piping and supports for evidence of interference with the thermal movement of the lines. The attributes considered during the walkdowns were:

1. clearances around piping and supports
2. snubber visual inspection
3. pipe geometry was compared to isometric
4. pipe supports location, type, direction and condition
5. any unusual condition

No discrepancies or unusual conditions which could have caused the failure were identified during the piping walkdown.

Based on the stress analysis verification and the walkdown effort, no design deficiency was found which could have resulted in the pipe failure.

Temporary instrumentation has been installed to allow temperature and vibration data to be collected during startup and for some period after achieving steady state full power operation. The following instrumentation has been installed on the pipe in which the defect occurred:

1. Five RTDs were installed circumferentially on the pipe between the check valve and the RCS loop.

2. Five additional RTDs were installed circumferentially around the pipe upstream of the check valve.

3. Two accelerometers were installed to monitor the horizontal movement at the elbow where the defect had occurred.

4. Three accelerometers were installed upstream of the check valve. Two of these accelerometers monitor the horizontal movement while the remaining accelerometer monitors the vertical movement.

As a reference, instrumentation was also installed on the safety injection line to the C RCS loop. Loop C was chosen to be instrumented because of its design similarities to loop B.

1. Two RTDs (one on top and one on bottom of the pipe) were installed downstream of the check valve.

2. Two RTDs (one on top and one on bottom of the pipe) were installed upstream of the check valve.

3. Accelerometers were installed in an arrangement similar to those on the B loop.

A review of data collected during startup following the repair revealed thermal stratification and cyclic temperature changes due to slight leakage through the boron injection tank bypass line through the B cold leg injection line. This stratification and thermal cycling has been stopped by diverting the leakage away from the injection line. A thermal stratification and cyclic temperature change of a lower magnitude and reduced frequency is still indicated in the B cold leg injection line upstream of the check valve (opposite side of the check valve from the crack). This condition has been evaluated to be acceptable at least in the short term. No abnormal vibration was observed on the line where the defect occurred. Further data analysis is being performed.

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

In addition, visual walkdowns of affected sections of B loop piping were performed during heatup in preparation for plant startup. The walkdowns verified that the piping was moving as predicted and was not binding. Although the apparent cause of the leak is thermal fatigue, further analyses are being performed to support a conclusion.

Additional Information

This event would not have been more severe if it had occurred under different operating conditions. No components failed during this event. Upon satisfactory testing, the unit returned to power operation on 12-27-87 at 1341.

Upon completion of the review of this event, a supplemental report will be submitted.

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R. P. McDonald
Senior Vice President

Alabama Power
the southern electric system

January 6, 1988

Docket No. 50-364

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

Dear Sir:

Joseph M. Farley Nuclear Plant - Unit 2
Licensee Event Report No. LER 87-010-00

Joseph M. Farley Nuclear Plant, Unit 2, Licensee Event Report No. LER 87-010-00 is being submitted in accordance with 10CFR50.73.

If you have any questions, please advise.

Respectfully submitted,

R. P. McDonald

RPM/JAR:dst-D-LER

Enclosure

cc: IE, Region II

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