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December 4, 2002

U. S. Nuclear Regulatory Commission
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Subject: Arkansas Nuclear One - Unit - 1
Docket No. 50-313
License No. DPR-51
Licensee Event Report 50-313/2002-004-00

Dear Sir or Madam:

In accordance with 10CFR50.73(a)(2)(ii)(A), enclosed is the subject report concerning Reactor Coolant System pressure boundary leakage. The enclosure contains no commitments.

Sincerely,

Sherrie R. Cotton

Sherrie R. Cotton
Director, Nuclear Safety Assurance

SRC/tfs

enclosure

IE22

cc: Mr. Ellis W. Merschoff
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NRC FORM 366 (1-2001)

U.S. NUCLEAR REGULATORY COMMISSION

APPROVED BY OMB NO. 3150-0104 EXPIRES 6-30-2001

LICENSEE EVENT REPORT (LER)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
Arkansas Nuclear One - Unit 1	05000313	1 OF 5

TITLE (4) Reactor Coolant System Pressure Boundary Leakage from a Fatigue Stress Crack of a Weld Connecting a Drain Line to a High Pressure Injection Line

EVENT DATE (5)			LER NUMBER (6)			REPORT DATE (7)			OTHER FACILITIES INVOLVED (8)	
MO	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REV NO	MO	DAY	YEAR	FACILITY NAME	DOCKET NUMBER
10	05	2002	2002	004	00	12	04	2002	FACILITY NAME	DOCKET NUMBER

OPERATING MODE (9)	3
POWER LEVEL (10)	000

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

	20.2201(b)		20.2203(a)(3)(i)		50.73(a)(2)(i)(C)		50.73(a)(2)(vii)
	20.2201(d)		20.2203(a)(3)(ii)	X	50.73(a)(2)(ii)(A)		50.73(a)(2)(viii)(A)
	20.2203(a)(1)		20.2203(a)(4)		50.73(a)(2)(ii)(B)		50.73(a)(2)(viii)(B)
	20.2203(a)(2)(i)		50.36(c)(1)(i)(A)		50.73(a)(2)(iii)		50.73(a)(2)(ix)(A)
	20.2203(a)(2)(ii)		50.36(c)(1)(ii)(A)		50.73(a)(2)(iv)(A)		50.73(a)(2)(x)
	20.2203(a)(2)(iii)		50.36(c)(2)		50.73(a)(2)(v)(A)		73.71(a)(4)
	20.2203(a)(2)(iv)		50.46(a)(3)(ii)		50.73(a)(2)(v)(B)		73.71(a)(5)
	20.2203(a)(2)(v)		50.73(a)(2)(i)(A)		50.73(a)(2)(v)(C)		OTHER
	20.2203(a)(2)(vi)		50.73(a)(2)(i)(B)		50.73(a)(2)(v)(D)		Specify in Abstract or NRC Form 366A

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER (Include Area Code)
Thomas F. Scott, Nuclear Safety and Licensing Specialist	479-858-4623

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

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

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

ABSTRACT (16)

During inspections in the Reactor Building at the start of a scheduled refueling outage, a leak was discovered from a weld on a drain connection to a High Pressure Injection (HPI) line. The weld was removed and sent to an off-site laboratory. This evaluation determined that the cracked weld was from high cycle, low stress fatigue. Weld defects were present and were a major contributor to crack initiation. The weld had been in service for 12 years. The weld was replaced with an enhanced configuration to provide greater vibration resistance. Inspections of similar welds in the HPI lines revealed no other leakage indications. Vibration measurements during shutdown conditions, including HPI Pump full flow testing, indicated acceptable levels. Additional vibration measurements are planned. An evaluation concluded that the HPI System would have been able to provide adequate flow to the Reactor Vessel for potential accident scenarios even if the weld had failed completely.

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A. Plant Status

At the time this condition was discovered, Arkansas Nuclear One Unit 1 (ANO-1) was in Mode 3 (Hot Standby) conditions.

B. Event Description

Reactor Coolant System (RCS) [AB] pressure boundary leakage was discovered from a weld on a drain connection to a High Pressure Injection (HPI) [BQ] line.

On October 5, 2002, during inspections in the Reactor Building at the start of a scheduled refueling outage, a leaking socket weld was discovered at the connection of a three-quarter inch drain line from a two and one-half inch HPI supply line to one of the four RCS cold leg discharge pipes. Between the leak location and the RCS piping, there is a normally open manual isolation valve and a check valve. The leakage was estimated to be approximately 0.2 gpm.

C. Root Cause

The leaking weld was removed and sent to an off-site laboratory. Metallographic and fractographic evaluations revealed that the root cause of the cracking was high cycle, low stress fatigue. The source of the fatigue loading was mechanical vibration or internal pressure pulsations. The presence of a few toe cracks indicated that the predominant loads were mechanical bending. Weld defects (lack of fusion and weld shrinkage), were present and played a major role in crack initiation. No evidence of any aggressive species that could cause stress corrosion cracking was found. The weld had been in service for 12 years. The drain line containing the weld was installed as part of a plant modification to improve HPI check valve performance. The crack was determined to have become through-wall during the most recent operating cycle.

D. Corrective Actions

The leaking weld was replaced with an enhanced design configuration. The new configuration of the drain piping is shorter than the failed line to remove its vibration frequency farther from the known Reactor Coolant Pump (RCP) driving frequencies. The replacement welds are "2T" fillet welds incorporating supplemental fabrication requirements to provide greater vibration resistance.

All four HPI lines were inspected and no other through-wall leakage was found.

Vibration levels were measured at the location of the failed weld and the other HPI lines during shutdown conditions with two RCPs running. These measurements confirmed that vibration levels were low. Additional vibration

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information was obtained during HPI Pump full flow testing and confirmed that the testing had not caused the weld failure. Additional vibration measurements of the HPI line are planned.

Liquid Penetrant (PT) testing was performed on HPI line vent and drain connection welds that have not been enhanced. Results met all ASME acceptance criteria.

The configurations of other HPI line vent and drain connections in the Reactor Building were evaluated. Most of the welds in locations equivalent to the leaking weld had been enhanced at some time in the past to provide more resistance to failure.

A review of maintenance history identified no prior RCS pressure boundary leakage from the HPI lines.

E. Safety Significance

The total unidentified RCS leak rate just before the start of the refueling outage was 0.284 gpm. This value is significantly less than the one gpm allowed by Technical Specifications. The time at which the leakage originated could not be conclusively determined; however, a review of records indicated that it could have started as early as April 2002.

The HPI system injects into each of the four RCS cold legs with flow from each of the supplying HPI pumps. At the point when HPI is injecting into the RCS, the HPI supply lines are pressurized to a slightly higher pressure than the RCS. During a design basis event, this pressurization of the HPI line could have caused the drain line to sever. In this case, a large part of the flow from this line would be spilled out into the Reactor Building. Even if all the HPI flow from the affected line spilled out into the Reactor Building, flow from the other three lines would be available.

Based on a review of relevant historical information for the previous three years, all the HPI flow from the two designated pumps, emergency power, and associated cooling systems would have been available, as required by Technical Specifications, during a demand situation. Of all licensing basis transient analyses, the Small Break Loss Of Coolant Accident (SBLOCA) analysis is most affected by reduction in expected HPI flow. The Large Break Loss Of Coolant Accident (LBLOCA) analysis does not credit HPI flow. If the HPI drain line weld had completely failed during power operation, at the worst, only one of the four HPI lines into the four RCS cold legs would have been unavailable. Since the lines contain valves that are pre-throttled, only a portion of the flow in the affected line would be lost. The pre-throttled HPI line break flow is a conservative estimate of how much of this flow could be expected to make it into the Reactor Vessel for core cooling. The analysis indicates that this amount of HPI flow ranges from about 39 percent of the available train's flow at high RCS pressures after Engineered Safeguards Actuation System (ESAS) [JE] actuation to about 59 percent at low pressures.

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This condition would have different ramifications for SBLOCAs with breaks at different locations. For all SBLOCA analyses, a single failure of one train of ESAS and a loss of the motor-driven Emergency Feedwater (EFW) [BA] Pump is postulated through the failure of one Emergency Diesel Generator (EDG) [EK]. Since both HPI trains were available during previous periods of plant operation, more HPI flow would have been available than is predicted in any of the SBLOCA analyses.

For the analysis of RCS pump discharge breaks, less than 70 percent of one train of HPI flow is credited with making it into the Reactor Vessel based on system pressure. One train is unavailable due to postulated EDG single failure and 30 percent of the HPI flow is assumed to go out the break. Since both HPI pumps would have been available during previous operation, more than 70 percent of one train's flow would have always been available to the core for cooling from the initiation of ESAS. Therefore, the analysis condition is bounded at all times for all cold leg pump discharge break sizes.

For the analysis of Core Flood Tank (CFT) [BP] line breaks, HPI flow is the only makeup source initially available until pressure decreases to CFT injection and Low Pressure (LPI) [BP] shutoff head. Using the HPI line break assumptions for the HPI flow into the Reactor Vessel, at least as much HPI flow would be available for core cooling as assumed in the analysis. The CFT line break results show that at the time HPI flow is assumed to initiate, RCS pressure has dropped to about 1100 psia. At this pressure the HPI flow into the Reactor Vessel from both HPI loops with one of the four HPI lines completely severed (conservative for the faulted condition of concern), over 420 gpm would be available to the core. The analysis assumed that at this pressure less than 380 gpm would be available. Therefore, more flow than what was assumed in the CFT line break analysis would have been available in the faulted condition.

A third SBLOCA scenario is the break of one of the four HPI injection lines injecting into the four RCS cold legs. Current Emergency Operating Procedure (EOP) action for a SBLOCA is to throttle HPI flow in the highest flow loop to within 20 gpm of the lowest flow loop for more balanced flow from the four loops. This action limits the effect of a break in one of the HPI lines if it were to occur. It would also limit the flow out the leaking line by increasing that line's resistance. For the HPI line break, analysis assumes this occurs at 10 minutes into the transient. As discussed before, pre-throttling also limits the flow from the leaking and/or broken HPI line even before the HPI line flows are balanced. If an HPI line break were to have occurred in one of the other HPI lines, since two HPI trains of flow were available, and the flow out the severed HPI drain line would be less than out the broken line, the HPI flow into the vessel would have been greater than assumed in the analysis. Therefore the faulted condition would have been bounded by analysis for the HPI line break.

Based on the availability of both HPI trains and the pre-throttled HPI injection lines, the leakage from the HPI drain line weld did not place the plant in an unanalyzed condition that significantly degraded plant safety and existing safety analyses bounded the as-found condition. Therefore, this condition is considered to have had minimal actual safety significance.

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F. Basis for Reportability

When it was determined that the leak location constituted RCS pressure boundary leakage per the 10CFR50.2 definition, a notification was made to the NRC Operations Center in accordance with 10CFR50.72(b)(3)(ii)(A) at 1420 CDT on October 8, 2002. This report is being submitted in accordance with 10CFR50.73(a)(2)(ii)(A) as a serious degradation of one of the plant's principal safety barriers.

G. Additional Information

RCS pressure boundary leaks from socket weld failures were reported in Licensee Event Report (LER) 50-313/89-010-00 (ANO letter 1CAN068909) dated June 16, 1989, for an RCS cold leg drain valve and LER 50-368/88-011-00 (ANO letter 2CAN098802) dated September 8, 1988, for an RCP seal cavity pressure sensing instrument line. In November 1989, a cracked weld was discovered on a vent line from an HPI line. This condition was not reported in an LER because the leak was isolated.

In 1990, ANO made changes to the welding process used for vent and drain socket welds. The current process greatly reduces the likelihood of incomplete fusion and penetration, slag inclusions, and porosity in socket weld joints.

Energy Industry Identification System (EIIS) codes are identified in the text as [XX].