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December 4, 2002

U. S. Nuclear Regulatory Commission
Document Control Desk
Mail Station OP1-17
Washington, DC 20555

Subject: Arkansas Nuclear One - Unit - 1
Docket No. 50-313
License No. DPR-51
Licensee Event Report 50-313/2002-003-00

Dear Sir or Madam:

In accordance with 10CFR50.73(a)(2)(ii)(A), enclosed is the subject report concerning leakage through the Reactor Coolant System pressure boundary. 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)	Estimated burden per response to comply with this mandatory information collection request: 50 hours. Send comments regarding burden estimate to the Records Management Branch (T-6 E6), U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, and to the Desk Officer, Office of Information and Regulatory Affairs, NEOB-10202 (3150-0104), Office of Management and Budget, Washington, DC 20503.

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TITLE (4) Reactor Coolant System Pressure Boundary Leakage from a Crack in a Control Rod Drive Mechanism Nozzle Reactor Vessel Head Penetration Weld due to Previous Weld Repair Method

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	07	2002	2002	003	00	12	04	2002	FACILITY NAME	DOCKET NUMBER

OPERATING MODE (9)	5
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
B	AB	NZL	B015	Y						

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

ABSTRACT (16)

Indication of boric acid was noted in the area of one Control Rod Drive Mechanism (CRDM) nozzle on the Reactor Vessel (RV) head during visual inspection at the start of a scheduled refueling outage. Other nozzles did not exhibit indications of leakage. Non-destructive examination (NDE) found indications of cracking in the nozzle that had resulted in Reactor Coolant System (RCS) pressure boundary leakage. NDE of all nozzles revealed indications in six others (not through-wall) and a likely porosity weld defect in another. Other than the porosity, all indications were from primary water stress corrosion cracking (PWSCC). The leaking nozzle had been repaired during the previous refueling outage. The root cause of the leakage was attributed to the design of the previous repair method that did not isolate all PWSCC susceptible material from the Reactor Coolant System. All nozzles with indications were repaired with improved design methods. The leakage did not result in wastage of the RV head. A new RV head fabricated with nozzle material and welds that are significantly more resistant to PWSCC has been purchased for installation in a future outage.

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A. Plant Status

At the time this condition was discovered, Arkansas Nuclear One Unit 1 (ANO-1) was in Mode 5 (Cold Shutdown) at the start of a scheduled refueling outage.

B. Event Description

Reactor Coolant System (RCS) [AB] pressure boundary leakage occurred from a crack in a Control Rod Drive Mechanism (CRDM) [AA] nozzle that penetrates the Reactor Vessel (RV) [AB] head.

On October 7, 2002, following shutdown for a scheduled refueling outage, routine inspections of the RCS were conducted for evidence of boron in accordance with NRC Generic Letter 88-05, "Boric Acid Corrosion of Carbon Steel Reactor Pressure Boundary Components in PWR Plants," NRC Bulletin 2001-01, "Circumferential Cracking of Reactor Pressure Vessel Head Penetration Nozzles," and NRC Bulletin 2002-01, "Reactor Pressure Vessel Head Degradation and Reactor Coolant Pressure Boundary Integrity." These inspections revealed boric acid residue on the down hill side of CRDM nozzle #56 extending approximately 180 degrees in the annulus area with a small nodule of boric acid crystals in the area of the J-groove weld that joins the nozzle to the RV head. The inspection was performed with remote video equipment while the head was still on the RV. This nozzle had been repaired during the previous refueling outage due to pressure boundary leakage attributed to primary water stress corrosion cracking (PWSCC).

Between October 10 and October 27, 2002, all CRDM nozzles were examined by ultrasonic testing (UT) and three nozzles were tested by liquid penetrant (PT). UT and PT indications in nozzle #56 were detected just outside the previous weld repair zone. Six other nozzles had relevant indications requiring repair. These nozzles were #3, 6, 15, 17, 33, and 54. Nozzle #68 did not have indications in the nozzle but did have an indication in the weld, a likely porosity weld defect. There was no evidence that any of these seven nozzles had a leak path.

C. Root Cause

The repair of nozzle #56 during the previous refueling outage employed the embedded flaw repair as allowed by Section XI of the ASME Code. Post repair examination showed that the flaw was isolated per design and that there were no recordable PT indications remaining in the weld. Also, fracture mechanics analysis indicated that the remaining axial flaw on the nozzle outside diameter (OD) surface above the J-groove weld would not propagate by fatigue. Thus, that repair met ASME Code requirements. The embedded flaw repair method that was used on nozzle #56 is unique to Babcock and Wilcox (B&W) plants. In that repair, the flaw was ground out up to, but not greater than, the original weld butter material for the J-groove weld. This depth was specified in order to preclude having to use a temper bead weld repair. The

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weld metal used to re-fill the cavity was Alloy 152, which was deposited by the shielded metal arc welding (SMAW) process. Because of its high chromium content, Alloy 152 weld metal is highly resistant to PWSCC. The weld repair zone was on the nozzle OD and in the J-groove Alloy 182 weld in the shape of an arc for slightly less than 90 degrees of the circumference. The outward radial edge of the arc extended approximately to the butter region of the J-groove weld. Both ends of the weld cavity where the weld metal was deposited were rounded. While the weld repair zone encapsulated the through-wall weld defect, it did not cover the entire wetted surface of the Alloy 182 J-groove weld. It is estimated that over three-fourths of the J-groove weld was still exposed after the repair. The requirements for the repair of nozzle #56 did not provide for remediation of the J-weld metal. As the repair requirements for nozzle #56 were being prepared, it was known from industry experience that the best approach for repairing the J-groove weld would be to include a complete weld overlay of the existing Alloy 182 J-groove weld. This approach would isolate the PWSCC susceptible weld material from the RCS, thus minimizing or eliminating the possibility for PWSCC in the future. Just prior to the previous ANO-1 refueling outage, industry welder resources were strained due to repairs at other licensees. Also during that time period, an automated welding process for applying weld overlays was not available. For these reasons, weld overlay was not performed. Water jet conditioning was also considered for mitigation of the nozzle ID surface. The water jet conditioning remediation method can relieve tensile stresses and induce compressive stresses on the nozzle inside diameter (ID). Since the flaws noted on nozzle #56 were associated with the nozzle OD and J-groove weld only, there were no apparent benefits from pursuing water jet remediation. The root cause of the leakage from CRDM nozzle #56 is attributed to the design of the previous repair method that did not isolate all PWSCC susceptible material from the RCS.

Based on the evidence obtained from non-destructive examination (NDE), the similarity of the cracking in the six other cracked CRDM nozzles in ANO-1 is consistent with the PWSCC cracking found at plants of a similar design. The large database of CRDM nozzle cracking from PWSCC in these plants is sufficient evidence for concluding that the cracks in ANO-1 CRDM nozzles #3, 6, 15, 17, 33, and 54, as well as the re-cracking of CRDM nozzle #56 in an area around the previous repair, was due to PWSCC. The cracking of nozzle #56 was affected by residual stresses from the previous repair.

D. Corrective Actions

CRDM nozzle #56 was repaired with a technique that consisted of removing the portion of the nozzle that extends below the surface of the RV head. The nozzle first received a roll expansion into the RV head base material. The lower portion of the nozzle was then removed to a depth above the existing J-groove partial penetration weld. A new pressure boundary weld was installed between the shortened nozzle and the inside bore of the RV head base at the mid-wall location of the head. With this repair, the original J-groove weld was not part of the new pressure boundary weld. This weld was deposited with Alloy 52 weld metal that is resistant to PWSCC. To preclude cracking of the roll area above the new pressure boundary weld, the water jet conditioning process was applied to this area. This method prevents

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recurrence of cracking of the previous repair. Nozzles #3, 6, 15, 17, and 33 were also repaired using this same process.

Unsuccessful attempts were made to remove the flaw, most likely a porosity weld defect, from the weld on nozzle #68. CRDM nozzles #54 and 68 were repaired by using the embedded flaw and weld overlay repair method deposited with Alloy 52 weld material. For these repairs, precautions were taken to ensure that the weld overlay covered the entire exposed original weld and nozzle material, thus precluding the possibility of re-cracking of these areas.

ANO has purchased a new RV head fabricated with nozzle material and welds that are significantly more resistant to PWSCC. It will be installed during a future outage.

E. Safety Significance

The total unidentified RCS leak rate just before the start of the refueling outage was 0.284 gpm. This value was significantly less than the one gpm allowed by Technical Specifications. While the contribution of the CRDM nozzle leakage to this value cannot be quantified, the measurement provides a bounding value.

The quantity of boron present on the head was estimated to be no more than a few ounces. The boric acid did not show any discoloration, which would indicate that there was no significant corrosion to the carbon steel head occurring. After inspection of the boric acid, the RV head was cleaned of boric acid from around nozzle #56 and the annulus area was inspected for potential wastage. The carbon steel interface around the annulus did not show any noticeable degradation or loss of metal. The nozzle annulus as found configuration was essentially the same as non-leaking nozzles.

A previous safety assessment of PWSCC cracking of CRDM nozzles determined that axial flaws with configurations similar to those found during the current outage do not promote catastrophic failure of the nozzle. Leak rates from these cracks are low and can be detected by visual examination before there is a risk of failure. Crack growth into the low alloy steel RV head is not expected due to the low susceptibility of this material to stress corrosion cracking. Boric acid corrosion rates of the RV head are so low that safe operation of the plant would not have been affected before the leakage was detected by routine visual inspection. The safety assessment also concluded that it is not probable that complete failure of the CRDM nozzle weld could result in a small break LOCA or control rod ejection accident.

Therefore, the overall safety significance of this condition was determined to be minimal. There was no actual impact on the public health and safety due to this condition.

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F. Basis for Reportability

The crack in the CRDM nozzle resulted in RCS pressure boundary leakage and constituted a degradation of one of the plant's principal safety barriers. This condition is being reported pursuant to 10CFR50.73(a)(2)(ii)(A). This condition was reported to the NRC Operations Center pursuant to 10CFR 50.72(b)(3)(ii)(A) at 1138 CDT on October 7, 2002

G. Additional Information

ANO has previously reported as Licensee Event Reports (LERs) six conditions involving RCS pressure boundary leakage attributed to PWSCC of Alloy 600 material. In LER 50-313/90-021-00 (letter 1CAN019112) dated January 21, 1991, ANO-1 reported leakage from an Alloy 600 Pressurizer level sensing nozzle. In LER 50-313/2000-003-00 (letter 1CAN030001) dated March 16, 2000, ANO-1 reported leaking welds for RCS hot leg level instrumentation nozzles. In LER 50-313/2001-002-00 (letter 1CAN050101) dated May 8, 2001, ANO-1 reported a leaking CRDM nozzle. In LER 50-368/87-003-01 (letter 2CAN088801) dated August 12, 1988, ANO-2 reported leaking Pressurizer heater sheaths. In LER 50-368/2000-001-00 (letter 2CAN080011) dated August 29, 2000, ANO-2 reported leaking Pressurizer heater sleeves and an RCS resistance temperature detector nozzle. In LER 50-368/2002-001-00 (letter 2CAN060201) dated June 12, 2002, ANO-2 reported leaking Pressurizer heater sleeves. Corrective actions for these conditions were not intended to prevent recurrence of PWSCC in Alloy 600 material that is subject to this failure mechanism.

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