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U. S. Nuclear Regulatory Commission
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Subject: Arkansas Nuclear One - Unit 1
Docket No. 50-313
License No. DPR-51
License Event Report 50-313/90-021-01

Gentlemen:

In accordance with 10CFR50.73(1)(2)(i)(A), enclosed is a supplement to the subject report concerning a reactor shutdown required by Technical Specifications due to an unisolable leak in a pressurizer nozzle which was caused by pure water stress corrosion cracking.

This supplement is being submitted to provide information regarding a change to original corrective actions resulting from subsequent evaluations.

Very truly yours,

Jerry W Yelverton
By Donald R Denton

JWY/KJM/jmt

Enclosure

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NRC FORM 366 (5-92)

U.S. NUCLEAR REGULATORY COMMISSION

LICENSEE EVENT REPORT (LER)

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

TITLE (4) Reactor Shutdown Required By Technical Specifications Due To Unisolable Leak In A Pressurizer Nozzle Which Was Caused By Pure Water Stress Corrosion Cracking

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 NAME	DOCKET NUMBER
12	22	90	90	--021--	01	02	17	94	FACILITY NAME	DOCKET NUMBER

OPERATING MODE (9)	N	THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR: (Check one or more) (11)			
POWER LEVEL (10)	000	20.402(b)	20.405(c)	50.73(a)(2)(iv)	70.71(b)
		20.405(a)(1)(i)	50.36(c)(1)	50.73(a)(2)(v)	70.71(c)
		20.405(a)(1)(ii)	50.36(c)(2)	50.73(a)(2)(vii)	OTHER
		20.405(a)(1)(iii)	X 50.73(a)(2)(i)	50.73(a)(2)(viii)(A)	Specify in
		20.405(a)(1)(iv)	50.73(a)(2)(ii)	50.73(a)(2)(viii)(B)	Abstract Below
		20.405(a)(1)(v)	50.73(a)(2)(iii)	50.73(a)(2)(x)	and in Text

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER (Include Area Code)
Kimberly J. Miller, Nuclear Safety and Licensing Specialist	501-964-8605

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
X	AB	PZR	B020	Y					

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 15 single-spaced typewritten lines) (16)

On December 22, 1990, maintenance personnel identified a potential Reactor Coolant System leak in the area of a pressurizer upper level instrumentation nozzle. An inspection was conducted which verified the existence of a very small leak at the nozzle. A Notification of Unusual Event was declared at 1011, and the plant was taken to cold shutdown. Subsequent inspection using Nondestructive Examination methods confirmed the existence of a small axial crack in the nozzle inner surface which extended to the annulus between the nozzle and the pressurizer shell and breached the outside diameter (OD) of the nozzle at the toe of the nozzle to vessel weld. Based on the location and orientation of the flaw, and industry experience, the most probable root cause was determined to be Pure Water Stress Corrosion Cracking. A temporary repair was completed which consisted of establishing the nozzle pressure boundary at the outside surface of the pressurizer and installing a new nozzle into the penetration from the shell OD. Subsequent evaluations submitted by letters dated December 20, 1991, and January 21, 1992 justified continued operation with the temporary repair in place. This is dependent upon future NDE inspection results consistent with NRC staff safety evaluation dated May 13, 1992.

A. At the time of this event, Arkansas Nuclear One, Unit One (ANO-1) was at a power level of approximately 10^{-8} amps (intermediate range). Reactor Coolant System (RCS) [AB] temperature was approximately 532 degrees and RCS pressure was 2150 psig. Low power physics testing was in progress.

B. Event Description

On December 22, 1990, after repairing a leak on a pressurizer level instrumentation isolation valve (RC-1002A), maintenance personnel identified a potential RCS leak in the area of the pressurizer upper level instrumentation nozzle. Although no visual indications were apparent, the noise level in the area indicated the presence of a leak. An inspection was conducted which verified the existence of a very small leak at the nozzle. Since this condition constituted an unisolable RCS pressure boundary leak, a Notification of Unusual Event (NUE) was declared at 1011, and a plant cooldown was initiated in accordance with Technical Specifications requirements. At 1949, the plant reached cold shutdown and the NUE was terminated.

Subsequent inspection using nondestructive examination (NDE) methods confirmed the existence of an axial crack in the nozzle inner surface starting about 0.2 inch from the inner end and extending for approximately 0.4 inch. The leak path was apparently through this crack, which is believed to extend to the annulus between the nozzle and the pressurizer shell, breaching the outside diameter (OD) of the nozzle at the toe of the nozzle to vessel weld. The indication on the OD of the nozzle was extremely small and closed up during cooldown, making it extremely difficult to locate.

C. Root Cause

Based on the location of the nozzle flaw, its axial orientation, similar indications at other nuclear utilities and information supplied by the pressurizer vendor (Babcock and Wilcox), it was determined that the most probable cause of the crack was Pure Water Stress Corrosion Cracking (PWSCC). PWSCC refers to intergranular stress corrosion cracking in the primary water environment of pressurized water reactors (PWR). Laboratory and service experience indicates that this cracking can be hastened at elevated temperatures, which is believed to be the reason the majority of the PWR nozzle failures have occurred in pressurizers. The evidence also suggests that certain of the product forms of Inconel Alloy 600, of which the ANO-1 pressurizer nozzles are made, are susceptible to PWSCC. A conclusive determination of the root cause could not be completed at this time because the portion of the nozzle containing the crack remains in the pressurizer and is unavailable for analysis.

D. Corrective Actions

A temporary repair was completed which consisted of establishing the level instrumentation nozzle pressure boundary at the outside surface of the pressurizer shell by depositing a weld pad on the shell OD around the nozzle penetration. A partial penetration weld preparation was formed in the pad and a new nozzle was installed into the penetration from the shell OD leaving a small gap between the original nozzle and the new nozzle. A portion of the original nozzle will remain in place until a permanent repair is determined necessary.

Evaluations were completed by B & W Nuclear Services in 1990 and 1991 which concluded that the structural integrity of the pressurizer would not be jeopardized by the existing nozzle repair for at least one fuel cycle. Further evaluations, the results of which were submitted by letters 1CAN129104 and 1CAN019201 dated December 20, 1991, and January 21, 1992, respectively, concluded that deferral of permanent repairs would be justified for at least one additional fuel cycle (1R11), and possibly several fuel cycles could be justified pending development of an NDE inspection technique for the carbon steel vessel wall or further corrosion test which would confirm that corrosion is not a safety concern for the pressurizer vessel (gap region of the nozzle exposed to boric acid). NRC staff safety evaluation dated May 13, 1992, concurred with this position provided the NDE technique was demonstrated to be effective in evaluating pressurizer base metal corrosion. The NDE technique was developed and successfully demonstrated during 1R10 and 1R11. No corrosion indication was found in the annulus area on the vessel wall. ANO letter 1CAN089302 dated August 5, 1993, summarized the technique involved and updated commitments from the aforementioned correspondence.

A visual inspection of the repaired pressurizer nozzle and other nozzles on the vessel was conducted during hot shutdown conditions prior to startup after repairs were completed. No leakage was observed. As committed in the corrective actions, these nozzles were again visually inspected for degradation at hot shutdown conditions on April 9, 1991, just prior to the cooldown for the 1M91 outage and again, no leakage was observed. Visual inspections were also completed during 1R10 and 1R11 on the nozzle repair with no leakage observed.

As discussed in 1CAN089302, if inspections continue to show no detectable corrosion or minimal corrosion, the inspection frequency will be reevaluated and further repair will be deferred until it is found necessary or prudent to modify the existing nozzle repair. Design Change Package development and implementation are also being deferred accordingly.

Initially, ANO intended to attempt to save the portion of the nozzle remaining in the pressurizer containing the crack for further metallurgical evaluations when additional repairs were made. Since it was subsequently determined that additional repair may not be necessary for several cycles and the exact nature of the repair and methodology to be used have not been determined, it is not known if the metallurgical evaluation on the nozzle portion would be possible. However, the portion of the nozzle which was removed was sent to B & W Alliance Research Center for further metallurgical evaluations. The results of the tests as reported in "Examination of Alloy 600 From ANO-1 Level Sensing Nozzle" concluded that the nozzle was moderately susceptible to PWSCC. Based on the above reasoning, ANO will not be saving the damaged portion of the nozzle.

E. Safety Significance

The safety significance of this condition is lessened by the fact that the unisolable RCS leak which resulted from the crack in the pressurizer nozzle was extremely small and did not cause a noticeable degradation of RCS pressure or result in any significant loss of inventory from the RCS.

Industry experience documents that failure of Alloy 600 components due to PWSCC occurs as a result of the propagation of axial cracks and that no such failures have been attributed to circumferential crack propagation. Therefore, considering the inherent toughness of Alloy 600 and the location of the nozzle within the pressurizer shell, a catastrophic failure of the nozzle was not likely to occur if ANO-1 had returned to power without identifying the leak. The "leak before break" mode of failure which is characteristic of PWSCC would have facilitated detection of the leak prior to its becoming a significant safety concern.

F. Basis for Reportability

This event is considered reportable pursuant to 10CFR50.73(a)(2)(i)(A) because the identification of an unisolable RCS leak necessitated the initiation and completion of a shutdown as required by the plants Technical Specifications.

G. Additional Information

A similar event in which stress corrosion cracking resulted in an unisolable RCS leak was reported in LER 50-368/87-003-00.

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