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U. S. Nuclear Regulatory Commission
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SUBJECT: Arkansas Nuclear One - Unit 2
Docket No. 50-368
License No. NPF-6
Licensee Event Report 50-368/92-002-01

Gentlemen:

In accordance with 10CFR50.73(a)(2)(i)(B) and 10CFR50.73(a)(2)(ii) enclosed is the subject report concerning steam generator tube surveillance. This supplement is being submitted to document the preliminary results of destructive examinations of steam generator tubes samples that altered the potential safety significance of the inadequate surveillance. This report will be revised if the final examination results are significantly different from the preliminary information.

Very truly yours,

James J. Fisicaro

James J. Fisicaro
Director, Licensing

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NRC Form 366
(6-89)

U.S. Nuclear Regulatory Commission
Approved OMB No. 3150-0104
Expires: 4/30/92

LICENSEE EVENT REPORT (LER)

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TITLE (4) Eddy Current Analysis Personnel Error Results in Inadequate Steam Generator Tube Technical Specification Surveillance

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 3	1 7	9 2	9 2	0 0 2	0 1	0 8	0 7	9 2		0 5 0 0 0
										0 5 0 0 0

OPERATING MODE (9): 5

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)	X 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
Thomas F. Scott, Nuclear Safety and Licensing Specialist	Area Code 501 964-5000

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
E	A B	S G	C 4 9 0	Y					

SUPPLEMENT REPORT EXPECTED (14)

Yes (If yes, complete Expected Submission Date) X No

EXPECTED SUBMISSION DATE (15): Month | Day | Year

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

On March 9, 1992, a primary-to-secondary leak in "A" steam generator was detected. The plant was shutdown with the leak rate at approximately 0.25 gpm, half of the Technical Specification limit. Subsequent testing identified the leaking tube which was plugged and stabilized. A review of eddy current data from the previous refueling outage in 1991 revealed that an indication had been present that if analyzed correctly would have required further evaluation. Failure to adequately complete the steam generator surveillance required by Technical Specifications was determined to have been caused by a cognitive personnel error on the part of two independent analysts who evaluated the eddy current test results from "A" steam generator. A review of the data from the prior inspection revealed six additional indications in "A" steam generator that should have received further analysis. Eddy current inspection in the area of interest in both steam generators has been completed. Tubes identified as being defective were sleeved or plugged and stabilized prior to plant startup. Preliminary destructive examination results of three tubes samples that were removed from the steam generators indicated that they were degraded beyond the minimum strength required to maintain adequate structural margins for accident conditions.

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

A. Plant Status

At the time the inadequate surveillance was discovered, Arkansas Nuclear One Unit 2 (ANO-2) was in cold shutdown conditions with Reactor Coolant System (RCS) [AB] temperature approximately 103 degrees and pressure approximately 15 psia.

B. Event Description

At 1230 on March 9, 1992, ANO-2 was operating at approximately 100 percent power when an alarm was received from the condenser vacuum pump discharge radiation monitor. Primary-to-secondary leakage was estimated to be approximately 0.25 gpm. The leakage rate was confirmed by three different methods-argon, tritium, and RCS inventory balance. Even though the leakage was below the 0.5 gpm limit for continuous operation contained in Technical Specifications, a decision was made to shutdown and locate the leak since leakage was above the unit's administrative limit of 0.1 gpm. The administrative limit is the point at which Operations personnel are directed to notify management of the condition and begin preparations for plant shutdown. Plant shutdown started at 1900. The reactor was shutdown at 2021, and the plant reached cold shutdown conditions at 0730 on March 10, 1992. Subsequent helium pressure testing located the leak in the hot leg side of tube 67-109 of "A" steam generator on March 15, 1992. The leak was confirmed by eddy current testing (ECT) using both a bobbin coil and motorized rotating pancake coil (MRPC). The defect in the tube was at the top of the tubesheet (estimated to be approximately 0.19 inch above the tubesheet) and had a circumferential orientation. A review of the bobbin coil ECT data obtained from this tube during the last refueling outage in 1991 was performed. On March 17, 1992, it was determined that an indication had been present for this tube in the location of the defect at the time of the last inspection. Although the through-wall depth could not be determined from the bobbin coil data, the indication was judged to have bee sufficiently significant so as to have required further evaluation by other methods following the prior testing.

C. Root Cause

The root cause for not having adequately analyzed Unit 2 steam generator eddy current indications during the 1991 refueling outage was determined to be cognitive error on the part of two independent analysts employed by Westinghouse. The analysts were performing this service under a contract to ANO.

Contributing causes were:

1. A lack of training for the eddy current analysts in site specific guidelines which incorporate both damage mechanisms specific to ANO and those known to have occurred at other sites.

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2. The lack of a requirement for performance demonstration testing of the analysts using actual historical data.

3. The location of the indications in the "explansion" (explosive transition) region of the tube where the tube reduces in diameter at the upper edge of the tube sheet. Because of interference from the tube sheet, the roll transition, and deposits of iron and copper on the secondary side, eddy current signals in this area are more difficult to analyze.

D. Corrective Actions

Westinghouse was notified of the deficient analysis so that appropriate corrective action can be taken concerning the specific analysts who were involved.

Work was initiated in the latter part of 1991 to develop a required testing and training program for ANO-2 eddy current analysts. This program will be in place prior to the start of the next ANO-2 refueling outage. This is expected to help prevent recurrence of the analysis deficiency.

Training and testing of eddy current analysts involved in the 1992 ANO-1 refueling outage, who were provided by another contractor, was completed prior to identification of the inadequate ANO-2 Surveillance to prevent two of the contributing causes identified above. Due to the timing of the ANO-2 forced outage, formal training guidelines and performance demonstration testing requirements had not been completely developed. To address this concern, the lead Level III analyst provided information to individual analysts describing the specific damage mechanism and signal characteristics in the area of interest. This instruction is believed to be sufficient to prevent overlooking potential indications near the tube sheet upper face during the Unit 2 steam generator outage.

Tube 67-109 was plugged and stabilized. The stabilizer will restrict the tube from causing any damage to adjacent tubes if it should become completely severed due to vibration during plant operation.

Data from "A" steam generator bobbin coil eddy current analysis from the 1991 outage was reanalyzed to confirm that analysis problems were limited to the region in the area of the tube sheet upper face on the hot leg side. Six other indications were identified that should have received more analysis during that outage. Each of the seven indications had been reviewed and not identified by two independent analysts.

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

A 100 percent inspection of both steam generator hot leg tubes has been performed using the MRPC eddy current test method. The MRPC method provides greater detail of the tube surface, thus providing better indication of the integrity of the primary-to-secondary boundary. The testing was limited to the area of interest, approximately two inches above and below the tubesheet upper face. Additionally, twenty percent of the tubes in the sludge pile region of "A" steam generator cold leg were similarly eddy current tested. There were no defective tubes identified.

Tubes which were identified as having confirmed indications were sleeved or plugged and stabilized prior to startup from the outage. A Technical Specification change to allow ANO-2 to sleeve steam generator tubes was approved by the NRC.

Tube samples were removed for additional analysis. Based on preliminary examination results, three tube samples with circumferential cracks from the sludge pile region of the hot legs appear to have exceeded the calculated maximum degradation allowed to maintain the safety margins required by Regulatory Guide 1.121, "Bases for Plugging Degraded PWR Steam Generator Tubes". The ANO-2 limit is 77 percent throughwall, based on an axial extent of the degradation of 0.25 inch maximum. The actual ANO-2 cracks in the three samples were found to be greater than or equal to 85 percent throughwall average. This information was evaluated and determined not to pose a current operational concern based on the inspections and corrective actions during the steam generator repair outage. Integrity of ANO-2 steam generator tubing will be evaluated and determined to be acceptable prior to restart from the refueling outage that is currently scheduled to start in September 1992.

E. Safety Significance

The indications in the steam generator tubes that were not completely evaluated had a circumferential orientation at the tube sheet upper face. Continued operation with defects of this configuration increased the risk of a tube failing in such a manner as to damage adjacent tubes. The manner in which this particular tube indication propagated allowed sufficient time for actions to be taken to place the plant in a safe condition under normal operational loads before progressing to a point where further damage could possibly have resulted. Preliminary examination results of the samples of tubes removed from the steam generators revealed that under accident loading conditions there would not have been sufficient structural margins available to ensure that a tube rupture would not occur. The safety significance of this event is reduced by there being several methods of detecting and monitoring small primary-to-secondary leakage and the fact that the plant was shutdown while leakage was approximately half of the Technical Specification limit for continuous operation.

F. Basis For Reportability

Technical Specification 4.4.5.2 for steam generator tube surveillance requires that "the inspected tubes shall be verified acceptable per the acceptance criteria of specification 4.4.5.4". Since steam generator tubes had eddy current indications during the last inspection that were not verified to be acceptable per the appropriate criteria, this represents a condition prohibited by Technical Specifications reportable per 10CFR50.73(a)(2)(i)(B).

The preliminary destructive examination results of the tube samples revealed that the inadequate surveillance resulted in Unit 2 having been operated with a principal safety barrier (steam generator tubes) seriously degraded. This condition is reportable per 10CFR50.73(a)(2)(ii). A report was made per 10CFR50.72(b)(1)(ii) at 1155 on July 15, 1992 when the condition was discovered.

G. Additional Information

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

There have been no similar events reported as Licensee Event Reports at ANO.