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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
Licensee Event Report 50-313/91-003-01

Gentlemen:

In accordance with 10CFR50.73(a)(2)(iv), enclosed is a supplement to the subject report concerning an automatic actuation of the Emergency Feedwater System.

This supplement is being submitted to provide updated information regarding the root cause of this event.

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)

FACILITY NAME (1) Arkansas Nuclear One, Unit One	DOCKET NUMBER (2)	PAGE (3)
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TITLE (4) Actuation of the Emergency Feedwater System During Plant Heatup due to Low Once Through Steam Generator Level Which Resulted From Inadequate Feedwater Flow Caused by a Degraded Auxiliary Feedwater Pump

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
0 4	2 1	9 1	9 1 --	0 0 3 --	0 1	0 3	2 3	9 3		0 5 0 0 0

OPERATING MODE (9): N

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

POWER LEVEL (10): 0 0 0

20.402(b)	20.405(c)	X 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)	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	
Richard H. Scheide, Nuclear Safety and Licensing Specialist	5 0 1	9 6 4 - 5 0 0 0

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	B A	I S V	V 0 8 5	Y					
X	B A	P	I 0 7 5	N					

SUPPLEMENT 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 fifteen single-space typewritten lines) (16)

On April 21, 1991, with the reactor subcritical, the Emergency Feedwater System (EFW) was automatically actuated due to a low level in the "A" Once Through Steam Generator (OTSG). At the time of the event, OTSG levels were being lowered to 30 inches in accordance with the "Plant Startup" procedure. However, when the "A" OTSG reached its programmed level, the auxiliary feedwater pump (P-75) was unable to maintain the required level. OTSG level continued to decrease until the EFW system automatically actuated (13 inches). The EFW system operated as designed and quickly returned the OTSGs to their programmed level. It was originally believed that the root cause of this event was a leaking feedwater recirculation isolation valve. However, subsequent disassembly and inspection of P-75 revealed that the cause of the event was a loose first stage channel ring bushing which effectively reduced the inlet area to the second stage impeller and resulted in decreased pumping capacity at increased flow rates. P-75 was repaired and satisfactorily tested during refueling outage 1R10 (February - May, 1992).

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

A. Plant Status

At the time of this event, Arkansas Nuclear One, Unit One (ANO-1) was preparing for startup following a maintenance outage. The reactor was subcritical with the Group I control rods withdrawn to their upper limits to establish immediate negative reactivity addition capability. Reactor Coolant System (RCS) [AB] pressure was approximately 2250 psig for an elevated system pressure walkdown and temperature was 538 degrees. Once Through Steam Generator (OTSG) pressure was approximately 900 psig.

B. Event Description

On April 21, 1991, at approximately 2150, the Emergency Feedwater System (EFW) [BA] was automatically actuated by the Emergency Feedwater Initiation and Control system (EFIC) in response to a low level in the 'A' OTSG.

The EFIC system monitors OTSG levels and pressures, Main Feedwater pump status, reactor coolant pump (RCP) status and Engineered Safeguards Actuation System [JE] channels 3 and 4 in order to initiate EFW or OTSG isolation should an actuation setpoint be reached. The EFW system, which includes a motor driven as well as a steam driven feedwater pump, is actuated to protect the reactor core from an overheating condition upon loss of main feedwater or RCP circulation. OTSG isolation is actuated to protect the core from an overcooling condition if a main steam line rupture should occur.

At the time of the event, OTSG levels were being lowered from 185 inches to 30 inches in accordance with the "Plant Startup" procedure (OP1102.02). The auxiliary feedwater pump (P-75) was in service, being supplied by one condensate pump, and the startup valves (CV-2623 and CV-2673) were closed and in automatic control to allow them to open and control OTSG levels when the low level limits (30 inches) were reached. The 'B' OTSG reached its low level limit first and CV-2673 began controlling level at approximately 29 inches. At 2144, when the 'A' OTSG reached its low level limit, the licensed control board operator observed that the level continued to decrease below 30 inches. It was also determined that P-75 discharge pressure was less than OTSG pressures. At this time, additional operations personnel were dispatched to search for possible leaks in the condensate/feedwater system. OTSG levels continued to decrease until, at 2150, the EFW system was automatically actuated providing feedwater to the OTSGs. The introduction of cold feedwater to the OTSGs caused a decrease in pressure to approximately 850 psig. Upon observing the decreasing OTSG pressures, and in conjunction with other abnormal secondary system indications, the control board operator manually tripped the Group I control rods in consideration of the possibility of a main steam line break. The EFW system quickly returned the OTSGs to their programmed levels. The steam driven EFW

pump (P-7A) was secured at 2200, while the motor driven pump remained in service (P-7B) supplying the OTSGs. The auxiliary feedwater pump was secured and inspected to determine if it had incurred any damage. The inspection revealed no obvious damage to the pump. It was restarted at 2205. At 2210, after verifying that the auxiliary feedwater pump was operating normally, P-7B was secured and the EFIC system was reset. At 2237, the reactor trip was reset and plant startup was continued.

C. Root Cause

The cause of this event was originally determined to be leakage past the seat of FW-8A (feedwater recirculation line) which was closed at the time of this event, and through FW-9A, which was open, to the condenser (see attachment). This leakage, over a period of time when OTSG feedwater was not required, resulted in the feedwater line between CV-2623 and FW-7A becoming voided. When the 'A' OTSG reached its lower limit, CV-2623 began to open as designed. However, since flow was entering a partially voided line, the OTSG level continued to drop and CV-2623 continued to ramp open in response to the decreasing level. As CV-2623 continued to open, P-75 discharge pressure dropped below OTSG pressure and 'B' OTSG level began to decrease, resulting in the 'B' startup valve (CV-2673) beginning to open farther. It was believed that P-75 pumping into the voided feedwater line had resulted in the pump reaching a 'runout' condition which rendered it incapable of supplying feedwater to the OTSGs. However, subsequent evaluations performed during refueling outage 1R10 (February - May, 1992) determined that, although leakage past the seat of FW-8A was a contributing factor to this event, it was not the root cause. Disassembly and inspection of P-75 during 1R10 revealed that the first stage channel ring bushing was loose and had caused abnormal wear on the second stage impeller. Conversation with the pump vendor (Ingersol-Rand) verified that the loose bushing could move into the suction of the next stage impeller and effectively reduce the inlet area of that impeller. At increased flow rates, the restriction produced by the bushing could result in significant throttling. Therefore, it was concluded that the internal pump damage was the primary cause of this event.

D. Corrective Actions

P-75 was repaired prior to restart from 1R10. The channel ring bushing was tack welded in place to prevent it from loosening. P-75 was tested prior to restart from 1R10 and was subjected to conditions similar to those experienced during the event. No sudden decreases in flow capacity or other abnormal behavior was observed.

Additional corrective actions which were taken to aid in preventing the occurrence of similar events during future plant startups included:

° The 'Plant Startup' procedure (OP 1102.02) was revised to require that FW-9A and B be closed whenever FW-8A and B are required to be closed.

° The 'Plant Startup' procedure was also revised to include a requirement to verify the ability to feed the OTSGs with the auxiliary feedwater pump prior to reaching the low level limits when lowering OTSG levels during startup.

° A repetitive maintenance task was developed for FW-8A, FW-8B, FW-9A and FW-9B.

E. Safety Significance

The EFW system was actuated and operated as designed during this event. In addition, the reactor was subcritical at the time and no significant RCS perturbations resulted from this event. Therefore, there was no safety significance associated with the event.

F. Basis For Reportability

The automatic actuation of the EFW system as well as the manual tripping of the Group I control rods is reportable pursuant to 10CFR50.73(a)(2)(iv).

This event was also reported in accordance with 10CFR50.72 at 2345 on April 21, 1991.

G. Additional Information

There have been no previous similar events reported by ANO.

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

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

ATTACHMENT