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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 50-313/96-005-00

Gentlemen:

In accordance with 10CFR50.73(a)(2)(iv), enclosed is the subject report concerning an automatic reactor trip.

Very truly yours,

Dwight C. Mims

Dwight C. Mims
Director, Nuclear Safety

DCM/tfs

enclosure

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NRC FORM 366 (5-92)	U.S. NUCLEAR REGULATORY COMMISSION	APPROVED BY OMB NO. 3150-0104 EXPIRES 5/31/95
LICENSEE EVENT REPORT (LER)		ESTIMATED BURDEN PER RESPONSE TO COMPLY WITH THIS INFORMATION COLLECTION REQUEST: 50.0 HRS. FORWARD COMMENTS REGARDING BURDEN ESTIMATE TO THE INFORMATION AND RECORDS MANAGEMENT BRANCH (MNBB 7714), U.S. NUCLEAR REGULATORY COMMISSION, WASHINGTON, DC 20555-0001, AND TO THE PAPERWORK REDUCTION PROJECT (3150-0104), OFFICE OF MANAGEMENT AND BUDGET, WASHINGTON, DC 20503.

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
Arkansas Nuclear One - Unit 1	05000313	1 OF 7

TITLE (4) AUTOMATIC REACTOR TRIP AND ENGINEERED SAFETY FEATURES ACTUATIONS CAUSED BY FAILURE OF A SPEED SENSING PROBE IN THE CONTROL CIRCUITRY OF A MAIN FEED WATER PUMP TURBINE AND FAILURE OF A MAIN STEAM SAFETY VALVE TO RE-SEAT

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
05	19	96	96	005	00	06	18	96	FACILITY NAME	DOCKET NUMBER

OPERATING MODE (9)	4	THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR: (Check one or more) (11)						
			20.402(b)	20.405(c)	X	50.73(a)(2)(iv)		70.73(b)
POWER			20.405(a)(1)(i)	50.36(c)(1)		50.73(a)(2)(v)		70.73(c)
LEVEL (10)	100		20.405(a)(1)(ii)	50.36(c)(2)		50.73(a)(2)(vii)		OTHER
			20.405(a)(1)(iii)	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)
Thomas F. Scott, Nuclear Safety and Licensing Specialist	501-858-4623

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
B	JK	ST	L253	Y						
A	SB	RV	D243	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)

ANO-1 experienced a reactor trip on high Reactor Coolant System (RCS) pressure. The pressure increase was caused by reduced Main Feed Water (MFW) flow originating from a component failure affecting controls of one of the operating MFW pumps. After the reactor trip, the other MFW pump tripped on high discharge pressure. Emergency Feed Water (EFW) automatically actuated. As expected, several Main Steam Safety Valves (MSSVs) opened following the reactor trip. One MSSV failed to re-seat. This led to the manual start of one High Pressure Injection pump because of reduced pressurizer level and manual actuation of Main Steam Line Isolation of the affected Once Through Steam Generator (OTSG). The MSSV failed to re-seat because inadequate engagement between the valve release nut and a cotter pin used to lock it in place allowed the nut to rotate and engage the manual lift top lever. Secondary water inventory of the affected OTSG was depleted via the open MSSV. Isolation of the OTSG, which provides the primary source of main turbine gland sealing steam, caused degradation of main condenser vacuum. RCS temperature control was maintained by the Atmospheric Dump System on the other OTSG. The open MSSV was gagged shut. Water inventory was restored by EFW. The first MFW pump control anomaly was caused by a drop in power supply voltage due to a shorted speed sensor. The failed speed sensor was replaced. Modifications were made to fuse the speed sensors and correct the cause of the second MFW pump trip. A modification was made to the MSSVs (other than the one gagged) to minimize the possibility of future inadequate pin engagement.

A. Plant Status

At the time of this event, Arkansas Nuclear One Unit 1 (ANO-1) was operating in steady-state conditions at approximately 100 percent power with Reactor Coolant System (RCS) [AB] average temperature approximately 579 degrees.

B. Event Description

An automatic reactor trip on high RCS pressure occurred at 0312 hours on May 19, 1996, due to a reduction in Main Feed Water (MFW) [SJ] flow.

At approximately 0311 hours, control oil pressure for the "A" Main Feed Water Pump (MFWP) turbine decreased causing pump speed and feed water flow to decrease. The feed water cross-over valve remained in its normal closed position because no MFWP trip signal was present. The reduction in flow caused the Integrated Control System (ICS) [JA] to demand maximum flow from both MFW loops. The ICS maximum demand signal was incorrectly interpreted as failed by the MFWP control system. This caused "B" MFWP controls to shift to the "Diagnostic-Manual" mode. While in this mode, the MFWP control system is being directed by what it considers the last valid signal and does not respond to additional ICS signals. The controls for "B" MFWP remained in "Diagnostic-Manual" and maintained the pump at its maximum speed. RCS pressure began increasing due to the decreased heat removal from degraded "A" MFWP flow. The Control Room Operator observed the increasing RCS pressure and attempted to manually trip the reactor. The manual trip was sensed 0.2 seconds following an automatic trip on high RCS pressure at 0312, less than one minute after the condition was initiated. All control rods inserted with acceptable insertion times.

Approximately four seconds after the trip, Emergency Feed Water (EFW) [BA] actuated on a sensed low water level in "B" Once Through Steam Generator (OTSG) [AB]. The "B" MFWP, which was holding at full speed in the "Diagnostic-Manual" mode, tripped on high discharge pressure approximately 14 seconds after the reactor trip when its associated MFW block valve closed as designed in response to the reactor trip. The fault in the "A" MFWP control system cleared and the pump controls attempted to respond to the high demand signals being generated by the ICS. As a result, the "A" MFWP turbine tripped on mechanical over speed approximately 37 seconds after the reactor trip. OTSG inventory was subsequently maintained by EFW with both trains functioning properly.

Pressure in "A" OTSG did not reach the setpoints of the Main Steam Safety Valves (MSSVs) [SB] because of the reduced water inventory as a result of the initiating control problems of "A" MFWP. Steam pressure in "B" OTSG was sufficient to cause six of the eight MSSVs to open. MSSVs lifting following a reactor trip is an expected response for ANO-1. One of these valves, PSV-2685, failed to re-seat. This caused an accelerated cooldown of the RCS. When pressurizer level fell below 30 inches, one High Pressure Injection (HPI) [BQ] pump was manually started in accordance with Emergency Operating Procedure (EOP) guidance at 0318 hours to assist the running Make Up (MU) [CB] pump in maintaining RCS inventory. The minimum pressurizer level of 12 inches occurred at 0319. The HPI pump was stopped at 0327.

At 0328, after trying unsuccessfully to re-seat the MSSV, Operators manually initiated Main Steam Line Isolation (MSLI) [JB] of "B" OTSG to stop the cool down transient. Both actions, attempting to re-seat the MSSV and isolation of the OTSG, were performed using EOP guidance. At 0330, a Notification of Unusual Event (NUE) was declared based upon the uncontrolled depressurization of "B" OTSG and its EFW supply was manually isolated. The secondary side of "B" OTSG began to dry via the open MSSV. During the blow down, the RCS cool down rate remained within analysis and Technical Specification limits. RCS average temperature remained above 520 degrees. Because of the lack of steam in "B" OTSG, the shell cooled to approximately 74 degrees below RCS temperature. This exceeded the tube-to-shell temperature difference (tubes hotter) of 60 degrees recommended by the vendor. Both of these conditions were evaluated by the vendor, Framatome Technology, Inc. (FTI), with regard to impact to the OTSG and reactor vessel. Effects of the transient were determined to be bounded by limits of existing analyses.

The isolation of "B" OTSG, which provides the only source, other than the startup boiler, of gland seal steam for the main turbine, resulted in degradation of vacuum in the main condenser. RCS temperature control was shifted through the Atmospheric Dump Valve (ADV) on "A" OTSG beginning at approximately 0348. Steam pressure was controlled by the modulating motor-operated ADV isolation valve per existing procedural guidance until the startup boiler was available to supply gland seal steam. Gland seal steam was restored at 0445, and condenser vacuum was restored at 0549.

A gagging device was installed on the open MSSV at 0853. Restoration of water level in "B" OTSG using EFW began at 0916. The MSLI was cleared, normal feed water established to both OTSGs, and the plant restored to normal hot shut down conditions. The NUE was terminated at 1304 on May 19, 1996. ANO-1 remained in hot shutdown conditions while the transient was evaluated and repairs and testing were completed. The reactor was critical at 0440 on May 24, 1996, and full power was reached at 1933 on May 25, 1996.

C. Root Cause

The cause of the initiating event was a component failure in the "A" MFWP control system. Reduced voltage on the turbine control system 24 volt power supply bus resulted from a short circuit in one of the pump speed sensing probes. The speed probe provides speed indication and an interlock for the turning gear but is not used for speed control. It is a potted assembly, and it has not been disassembled to evaluate the specific failure mechanism. According to the probe vendor, this was the first known instance of this probe failing with a short circuit. Previous failures involved an open circuit condition. An open circuit would not have caused an adverse impact upon the MFWP control system. An upgrade to the feed pump control system was installed in 1995 for the purpose of improving system response to a MFWP trip. The equipment design specifications required failure immunity and redundancy to ensure a highly fault tolerant system. Although separate fusing of non-critical components was not explicitly stated in the design specification, failure to provide such protection was inconsistent with good design practice. This is considered to be the root cause.

The probable cause for the second MFWP inappropriately transferring to manual was inappropriate equipment specification. Signal failure detection logic in the MFWP controls requires that the input exceed either a maximum or minimum value while changing in excess of a rate of change setpoint. Recorded data indicate that input rates of change were less than half of the required rate. Troubleshooting revealed that the final output device in the ICS, a signal limiter, was causing "ringing" (noise) on the MFWP speed demand input which likely caused the measured rate of change to exceed the required value. The signal limiter, which was added several years ago, is inappropriate for this application. An evaluation of the post-modification testing following the 1995 upgrade concluded that a sound approach was applied and the testing provided reasonable assurance that the logic of the control system was functional. While more elaborate testing may have identified the problem with the signal limiter, proper response to an actual ICS input was verified for the full range of the input signal. Testing also confirmed proper turbine response (entry into "Diagnostic-Manual") on rapid off-scale high and low movements of the test signal both with and without a reactor trip.

On the ANO-1 MSSVs, the spindle is a threaded extension of the valve stem that is located above the valve body. At the upper part of the spindle, a release nut is threaded on to the spindle. The release nut is prevented from rotating on the spindle by a lock (cotter) pin that is installed through a slot in the release nut and a hole in the spindle. The release nut slot is open at the upper end. The release nut serves as a leverage point for the top lever which is a part of the manual lift mechanism. During MSSV setpoint testing, the release nut is removed to allow installation of the test device. Because of incomplete engagement between the pin and the nut during the most recent installation activity, the nut vibrated and rotated down the spindle while the MSSV was open following the reactor trip. Contact between the release nut and the top lever prevented the valve from re-seating. The release nut was unable to turn because the entire valve spring load was wedging the top lever against the bottom of the release nut, preventing the valve from seating. The lift lever pin was removed and the top lever forced from under the release nut in order to provide a clearance between the top lever and release nut. Movement of the top lever allowed the release nut to turn. The cotter pin and release nut were then removed from the spindle. After removing the release nut, PSV-2685 was closed and gagged. The root cause for incomplete engagement was determined to be "personnel work practices; document use practices; documents not followed correctly." The procedure for release nut installation was not followed correctly by ANO-1 Mechanical Maintenance personnel following the last surveillance testing of the MSSVs. A contributing factor to the MSSV failing to re-seat was determined to be an inadequate original design of the release nut. The release nut slot is approximately 0.40 inches high. This results in a very small area for the cotter pin and nut engagement to occur.

D. Corrective Actions

The speed sensing probe and the speed monitor module that was damaged by the probe failure were replaced. A fuse was also added to the digital speed monitor circuit for fault protection of the control power supplies of both MFWPs as a reliability enhancement. Changes were also made to the control setpoints to prevent undesirable transfers to the "Diagnostic-Manual" mode that isolated input signals from the ICS.

The 15 ANO-1 MSSVs (excluding PSV-2685 that remained gagged) were inspected. Cotter pins for two other valves were found not engaged in the release nuts. These valves were determined to have been operable since the release nuts could not be rotated due to the cotter pin ends being engaged on the nuts. Six valves had the pins partially engaged at the top end of the release nut slot. Seven valves were found with the cotter pins fully engaged. A modification was installed to replace the 15 MSSV release nuts with a "taller" nut with a slot dimension increased to 0.75 inches to significantly minimize the possibility of future instances of inadequate cotter pin engagement.

The ANO-1 maintenance procedure for release nut installation requires, "replace the release nut, flat side down, and temporarily install the cap and lever in order to adjust the release nut position. The bottom of the release nut should clear the top of the lever by 1/16 to 1/8 inches. Remove the lever and cap. Insert a new stainless steel cotter pin through the release nut slots and spindle and spread the cotter pin ends." These requirements came directly from the vendor technical manual. During discussions with the MSSV vendor, Dresser, it was discovered that the 1/8 inch dimension was not a functional or practical requirement because holes in the spindles are not drilled in the same locations on all spindles provided by Dresser. The maintenance procedure was changed to indicate that the 1/16 inch was a minimum clearance. Reference to the maximum clearance was deleted. A caution regarding adequate cotter pin engagement was also added to the procedure.

An inspection of the installed ANO-1 pressurizer code safety valves, ANO-2 MSSVs, and spare ANO-2 pressurizer code safety valves identified no concerns similar to those associated with the failure of PSV-2685.

The ANO-1 Plant Manager has reviewed this event with Unit 1 Mechanical Maintenance personnel to emphasize the importance of procedural adherence. Similar discussions will be conducted with appropriate personnel from the Operations and Maintenance organizations of both units and Modifications personnel. These discussions will be completed by September 15, 1996. Additional initiatives previously implemented to address procedure usage issues are being utilized to validate the ability to use procedures as written in complete compliance with site directives and administrative procedural requirements. This includes observations of procedure usage by staff or craft personnel of all Maintenance disciplines to verify and validate proper use.

E. Safety Significance

Performance of Operations personnel in bringing the plant to a safe and stable condition was competent, professional, and produced satisfactory results. One OTSG and both trains of EFW remained available throughout the event. No other safety-related equipment potentially used for reactor core cooling or any other system potentially used to mitigate the effects of the MFWP unavailability was affected by the sequence of events. The open MSSV had the effect of removing heat from the RCS until "B" OTSG reached dry-out conditions. Each of these considerations provided mitigation to the safety significance of the event.

Considering the conditions that occurred during and following the trip, an evaluation determined that the Conditional Core Damage Probability (CCDP) was similar to that expected for industry loss of MFW events. The CCDP was estimated to be above the screening criterion for low risk events but within the lowest range of events analyzed in NUREG/CR-4674, "Precursors to Potential Severe Core Damage Accidents," i.e., between 1E-06 and 1E-05. The impact of the atypical elements of this transient are mitigated by:

- the Auxiliary Feed Water Pump was available and utilized (although its inventory source, the condenser hotwell, was temporarily in a limited capacity due to a partial loss of condenser vacuum);
- the EOPs provided for the ability for EFW "trickle feed" to the OTSG that was rendered temporarily unisolable by the open MSSV; and
- the RCS pressure reduction due to the slight over-cooling transient was limited to several hundred pounds above the safety systems actuation setpoint, essentially eliminating any potential for safeguards actuation induced primary safety valve lifting.

Although these mitigating factors possibly could be utilized analytically to reduce the CCDP below the screening criterion for low risk events, as a minimum they provide confirmation that this event was within the lowest range of events that are analyzed in the Accident Sequence Precursor Program. Therefore, this event is evaluated to have a low level of safety significance.

F. Basis for Reportability

The automatic reactor trip, automatic EFW actuation, manual HPI actuation, and manual MSLI actuation constitute events reportable in accordance with 10CFR50.73(a)(2)(iv) as Reactor Protection System or Engineered Safety Features actuations. This event was reported to the NRC Operations Center at 0402 on May 19, 1996, in accordance with 10 CFR50.72(a)(1)(i) for declaration of the NUE; 10CFR50.72(b)(1)(iv) for HPI injection into the RCS; and 10CFR50.72(b)(2)(ii) for the reactor trip and actuation of EFW, HPI, and MSLI. Updates were provided at 0600, 1324, and 2154. Termination of the NUE was reported during the 1324 update.

G. Additional Information

There was one previous similar event reported by ANO as a Licensee Event Report (LER). A Main Steam Safety Valve failing to re-seat because the cotter pin did not prevent the release nut from binding with the top lever was reported as part of LER 50-313/89-018-00 (letter 1CAN058915). The 1989 event was due to a missing pin, not one with inadequate engagement. The corrective action for that event was a change to the maintenance procedure to require initials verifying pin replacement.

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