Form 1062.01A

NRC Form 366
(9-83)

U.S. Nuclear Regulatory Commission
Approved OMB No. 3150-0104
Expires: 8/31/85

LICENSEE EVENT REPORT (LER)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
Arkansas Nuclear One, Unit One	0 5 0 0 0 3 1 3	1 OF 0 6

TITLE (4) Reactor Trip on Low Reactor Coolant System Pressure Caused by Unplanned Regulating Rod Group Insertion Due to Control Rod Drive Circuitry Malfunction

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)
C 2	1 7	8 8	8 8	-- 0 0 3 --	0 0	0 3	1 8	8 8	N/A	0 5 0 0 0
									N/A	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 2 3

[ ] 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 (Snecify 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)
Patrick C. Rogers, Nuclear Safety and Licensing Specialist	5 0 1 9 6 4 - 3 1 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	A A	P M C	D 1 5 0	Y					
X	S J	C L U	L 2 0 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)

During reactor protection system (RPS) monthly surveillance testing, Group 7 control rods dropped into the core. An automatic RPS reactor trip subsequently occurred on low reactor coolant system pressure. Following the trip, 'A' main faedwater (MFW) block valve CV-2625 failed to close and both MFW pumps minimum speed was observed to be slightly higher than expected. These conditions resulted in an overfeed of the 'A' once through steam generator (OTSG). When 'A' OTSG water level reached 100% on the operating range level instrumentation, the overfeed was terminated by manually actuating emergency feedwater (EFW) and tripping the MFW pumps. Primary and secondary system responses post-trip were near normal with the exception of 'A' OTSG water level. An unidentified malfunction of the power sequencing programmer for Group 7 coupled with the conditions existing during RPS testing caused the Group 7 drop. CV-2625 failed to close because of a failure in the valve operator drive clutch. Steam sources which are available in the post-trip mode versus the mode existing when the MFW pump minimum speeds were originally set caused the higher than expected MFW pump speed. The programmer for Group 7 was replaced. A temporary modification was installed and a drive clutch adjustment made to CV-2625. Further inspection and repair of the drive clutch assembly for CV-2625 will be performed at the next outage of sufficient duration. An evaluation will be performed to determine improvements which could be made in setting MFW pump minimum speeds for acceptable post-trip response.

I. Description of Event

A. Unit Status

On 2/17/88, prior to the occurrence of this event, Arkansas Nuclear One, Unit One (ANO-1) was in steady state operation at approximately 80% reactor power with reactor coolant system (RCS) pressure of 2155 psig and RCS temperature of 579 degrees Fahrenheit.

B. Component Identification

This event involved a malfunction of a programmer used to sequence power to the six phases of the control rod drive (CRD) motors for regulating control rod Group 7. The programmer is a Model PS-70-1 supplied by Diamond Power Corporation (D150). EIIS Identifier = AA-PMC.

This event also involved a malfunction of a valve operator drive clutch for the 'A' train main feedwater (MFW) block valve, CV-2625. The clutch was manufactured by Airflex (no NPRDS code) and supplied by Limitorque Corporation (L200) as part of an SMB-4 valve operator. EIIS Identifier = SJ-CLU.

C. Sequence of Events

On 2/17/88 while in steady state operations at approximately 80% power, a monthly reactor protection system (RPS) surveillance test for channel 'C' was in progress. At 1255 hours the channel was tripped per the requirements of the test procedure. Approximately one minute and forty seconds after tripping RPS channel 'C', the Group 7 regulating control rods dropped into the core. Approximately twenty-two seconds later (1257 hours), with reactor power at approximately 23%, an automatic RPS reactor trip on low RCS pressure occurred.

Following the reactor trip, control room operators noted the 'A' once through steam generator (OTSG) water level was increasing. It was also observed that both MFW pumps were operating at a greater than expected post trip minimum speed and that MFW block valve (CV-2625) had failed to close resulting in excessive MFW supply (overfeed) to the 'A' OTSG. At approximately 1258 hours an operator was dispatched to manually close CV-2625. At approximately 1300 hours the 'A' OTSG water level had reached 100% on the operating range level instrumentation. The control room operators terminated the 'A' OTSG overfeed by manually actuating the emergency feedwater (EFW) system, to supply feedwater to 'B' OTSG, and tripping both operating MFW pumps. At approximately 1301 hours, manual closing of CV-2625 was completed. At approximately 1312 hours an auxiliary feedwater (AFW) pump, which is used for low power and startup operations, was placed in service. At approximately 1315 hours the EFW system was secured and aligned for automatic actuation.

The unit remained in hot shutdown for repairs to the MFW block valve and investigation into the cause of the inadvertent control rod insertion. The unit was restarted and synchronized to the grid at 0222 hours on 2/19/88.

II. Event Analysis

A. Event Cause

The leadscrew-type CRD mechanism motor stator assembly contains six stator coils (designated as phases A, B, C, AA, BB, CC). The six stator coil phases are sequentially energized to produce a rotational drive force for the motor's rotor assembly. This rotary motion is translated into linear motion of the leadscrew attached to a control rod. For each regulating control rod group, a CRD system programmer provides the sequencing commands for energizing the six stator coil phases. The CRD system programmers are powered with redundant (main and secondary) power supplies. The programmers contain a single, rotating optical disc with redundant (main and secondary) light sources, photo detectors, and gating circuits which provide the sequencing commands to energize the six stator coil phases.

While performing the RPS monthly surveillance test, channel 'C' was tripped per the requirements of the test procedure. RPS channel 'C' controls and trips the main power supply to the CRD programmers for the regulating control rod groups, including Group 7. Although not known at the time of the event, a malfunction had occurred in the output of the secondary side of the Group 7 CRD programmer which would normally provide redundant sequencing commands for energizing the Group 7 stator coils' phase 'A'. With the 'C' RPS channel in a tripped condition for surveillance testing and a malfunction in the secondary side output for phase 'A' of the Group 7 programmer, the power to the Group 7 CRD stator coils' phase 'A' was not available.

The Integrated Control System (ICS) was operating in the automatic mode during the surveillance test. In this mode of operation, one method used to compensate for minor system variations, e.g., Tave control, is regulating control rod movement. During the surveillance test, the ICS called for Group 7 movement. When the Group 7 CRD stator coils' phase 'A' should have sequenced to be energized, the phase was not energized. As a result, Group 7 control rods began to drop into the core. As the programmer continued to sequence the other stator coils' phases, because of the ICS control rod motion demand, the Group 7 CRD mechanisms engaged the control rods then disengaged the control rods when the programmer again sequenced the Group 7 stator coils' phase 'A'. This engaging and disengaging of the Group 7 control rods is termed a ratchet trip. The ratchet trip insertion of the Group 7 control rods caused RCS pressure to decrease to the RPS trip setpoint for low RCS pressure producing an automatic reactor trip.

The ICS includes a rapid feedwater reduction (RFR) circuit designed to prevent MFW flow to the OTSGs if sufficient water level is present under post-trip conditions. Following a reactor trip, the RFR circuitry is designed to decrease both MFW pumps' speed to a preset minimum speed. This minimum speed was previously set at 3100 rpm during hot shutdown conditions. At hot shutdown conditions only high pressure steam is available to the MFW pump turbine for speed setting. Following the reactor trip on 2/17/88, the MFW pumps' minimum speed was observed to be 400 to 600 rpm higher than the previously set minimum speed. This was due to the availability of additional post-trip low pressure steam from the moisture-separator/reheaters which is not available when the minimum speed is originally set in hot shutdown conditions.

The RFR circuitry is also designed to close both MFW block valves following a reactor trip. However, a drive clutch failure in the valve operator for CV-2625 prevented the valve from closing following the reactor trip. CV-2625 is operated by a Limitorque SMB-4 operator which uses drive clutch assemblies manufactured by Airflex. The drive clutch assemblies utilize electrical coils to shift the motor operator between two sets of gears to allow for fast or slow speed operation of the valve. On a reactor trip, the valve should close in the fast speed. Other valve movement should be in the slow speed.

During the transient, the 'B' OTSG water level decreased as expected. Since the 'B' MFW block valve closed as designed, a proper water level for the 'B' OTSG could be maintained even with a higher MFW pump minimum speed. However, due to the failure of the 'A' MFW block valve to close following the reactor trip, the high MFW pump minimum speed caused feedwater flow to continue to the 'A' OTSG.

Primary system (i.e., RCS) responses were near normal for expected post-trip conditions. The secondary system responses were normal with the exception of the higher than normal post-trip 'A' OTSG water level. The control room operators were aware of the overfeed condition and were monitoring plant conditions as required by the emergency operating procedures. Since primary system parameters were not adversely affected by the overfeed condition, the control room operators did not terminate the overfeed condition until the 'A' OTSG water level reached 100% on the operating range. To terminate the overfeed condition EFW was manually actuated to provide feedwater to the 'B' OTSG and both operating MFW pumps were tripped. EFW system high range water level instrumentation, which envelopes the operating range water level instrumentation range (see Figure 1), was also available (although not on the control room front panels) and was being monitored by the control room operators. The 'A' OTSG water level remained well within the EFW high range water level instrumentation range.

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

The safety significance of this event can be related to the post-trip primary and secondary systems responses. Primary system response was near normal after the reactor trip. The RPS initiated an automatic reactor trip as designed at an RCS pressure of 1800 psig. Following the trip, RCS pressure subsequently decreased to only approximately 1750 psig. This pressure is well above the safety injection setpoint of approximately 1526 psig. Additionally, RCS temperatures were not significantly affected and pressurizer level remained on scale throughout the event. These responses demonstrated that the affects of the overfeed of 'A' OTSG on primary system parameters was negligible. With regard to secondary system parameters, the responses were normal with the exception of 'A' OTSG water level. Neither OTSGs' steam pressures increased high enough to cause lifting of the main steam safety valves. OTSG 'A' water level did not increase to a point where water entered the main steam line. Therefore, based on overall plant response to the transient, the safety significance of this event was considered to be minimal.

B. Root Cause

The root cause of the reactor trip was a ratchet trip of control rod Group 7 due to a malfunction in the secondary side of the Group 7 CRD programmer. After removal and replacement, a failure analysis of the programmer was performed by Babcock & Wilcox . Diagnostic testing of the programmer in a special circuit analyzer could not reproduce a malfunction, therefore, the exact failure mechanism of the programmer could not be determined.

The cause of the continued feed to the 'A' OTSG was a higher than previously set post-trip minimum MFW pump speed combined with the open MFW block valve. The root cause of the MFW pump minimum speed being higher than set was due to the additional steam source from the moisture separator reheaters to the MFW pump turbine in the post-trip mode versus the mode existing during minimum speed setting which uses high pressure steam only. The root cause of the failure of CV-2625 to close was not conclusively determined but is believed to be related to the settings associated with some components of the drive clutch assemblies for the valve operator.

C. Basis For Reportability

This event resulted in an unplanned automatic actuation of the RPS and a manual actuation of the EFW system and is therefore reportable under the provisions of 10CFR50.73(a)(2)(iv). This event was reported per 10CFR50.72(b)(2)(ii) at 1352 hours on 2/17/88.

III. Corrective Action

A. Immediate

Immediate actions were to verify that post-trip system responses were satisfactory. Upon noting CV-2625 had not closed and the level in 'A' OTSG was increasing, operations personnel were dispatched to manually close CV-2625. To terminate the overfeed to 'A' OTSG, EFW was manually actuated to control 'B' OTSG water level and both MFW pumps were tripped when the 'A' OTSG level reached 100% on the operating range water level instrumentation. The auxiliary feedwater system (AFW) was placed into service to maintain OTSG water levels for hot shutdown conditions, and the EFW system was secured and aligned for automatic actuation.

B. Subsequent

The regulating control rod Group 7 CRD programmer was replaced and proper operation of the programmer and the Group 7 CRD circuitry was verified after repairs. Proper control rod movement was verified for the Group 7 control rods.

B. Subsequent (Cont'd)

The RPS monthly surveillance test procedures were modified to require placing the CRD system in manual to prevent automatic movement of control rods by the ICS during RPS surveillance testing. Additionally, the test procedures were revised to require verification of the programmer secondary power prior to removing primary power during testing.

Adjustments of the drive clutch for CV-2625 were made. A temporary modification was installed to allow only high speed operation of CV-2625. Proper operation of the valve was verified by stroke testing.

C. Future

Inspections will be performed on the drive clutch assemblies in the valve operator for CV-2625 during the next outage of sufficient duration. Drive clutch replacement or necessary repairs will be performed at this time.

An engineering evaluation will be performed to determine the feasibility of a potential design improvement for setting the MFW pumps minimum speed to ensure proper post-trip response in the event that MFW valves do not close as required.

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(9-83)

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LICENSEE EVENT REPORT (LER) TEXT CONTINUATION

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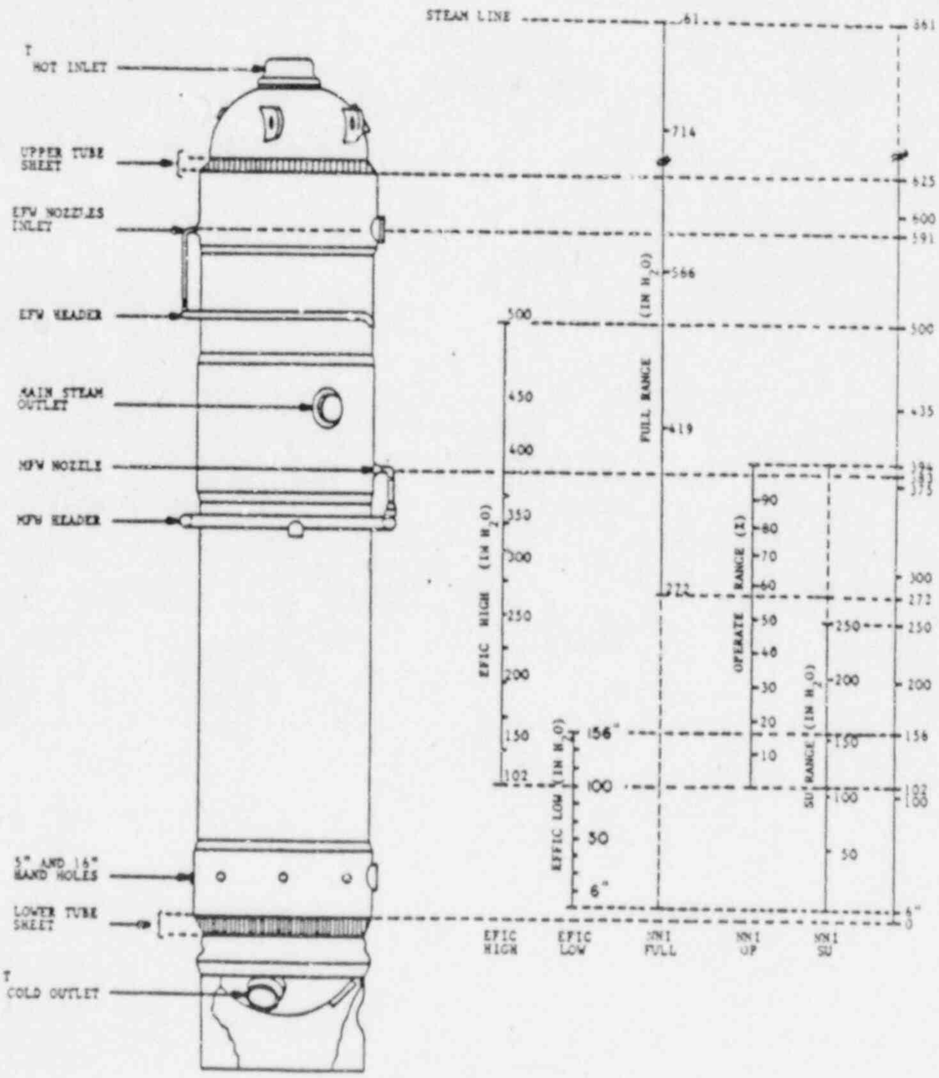
TEXT (If more space is required, use additional NRC Form 366A's) (17)

FIGURE 1



ARKANSAS POWER & LIGHT COMPANY
March 18, 1988

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U. S. Nuclear Regulatory Commission
Document Control Desk
Washington, D.C. 20555

SUBJECT: Arkansas Nuclear One - Unit 1
Docket No. 50-313
License No. DPR-51
Licensee Event Report 313/88-003-00

Gentlemen:

In accordance with 10CFR50.73(a)(2)(iv), attached is the subject report concerning a reactor trip on low reactor coolant system pressure caused by an unplanned regulating control rod group insertion due to a control rod drive circuitry malfunction.

Very truly yours,

JM Levine

J. M. Levine
Executive Director,
Nuclear Operations

JML:DJM:dm
attachment

cc: Regional Administrator
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