Form 1062.01A

NRC Form 366
(9-83)

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LICENSEE EVENT REPORT (LER)

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

TITLE (4) Reactor Trip Caused by Turbine Generator Exciter Failure and Failure of Various Components in Main Feedwater System Results in Steam Generator Overfill

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|1	2|0	8|9	8|9	0|0|2	0|0	0|3	3|1	8|9	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) 1|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)

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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	T|L	E|X|C	W|1|2|0	N	B	S|J	|2|0		N
B	J|A			Y	X	J|K		L|2|5|3	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 1/20/89 a reactor trip occurred due to a failure in the main generator exciter. Following the trip, two reactor coolant pumps (RCPs) tripped on undervoltage when bus H1 failed to fast transfer to offsite power; the Main Feedwater (MFW) startup valves and low load control valves for both 'A' and 'B' MFW trains failed to close due to miswiring in the Integrated Control System; the 'B' MFW block valve failed to close due to an incorrect torque switch setting; MFW pump P1B failed to run back to the designed post-trip minimum speed due to an undetermined cause; and MFW pump P1A post-trip minimum speed was higher than designed minimum speed due to availability of steam from a Moisture Separator Reheater. The MFW system component failures caused an overfill of the 'B' once-through steam generator (SG). During the transient, operators initiated High Pressure Injection (HPI) to compensate for Reactor Coolant System (RCS) shrinkage caused by overcooling due to overfeeding the SGs. After securing HPI the combination of a partially failed open check valve and the RCP operating configuration allowed backleakage of reactor coolant into HPI system piping outside containment. During recovery from the transient, the Emergency Feedwater System automatically actuated on 'B' SG low level. A reactor building inspection following the trip revealed a small unisolatable leak in an RCS drain line and the unit was placed in cold shutdown. The component failures were corrected. Extensive reviews of the HPI system are being conducted to determine necessary corrective actions.

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1/1

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6) Year	Sequential Number	Revision Number	PAGE (3)
Arkansas Nuclear One, Unit One	0|5|0|0|0| 3| 1| 3	8| 9	0| 0| 2	0| 0	0|2|OF|1|2

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

A. Plant Status

At the time of the reactor trip, Arkansas Nuclear One - Unit 1 (ANO-1) was operating at 100 percent of full power. Reactor Coolant System (RCS) temperature was approximately 579 degrees Fahrenheit, RCS pressure was approximately 2155 psig, and pressurizer water level was 225 inches. Steam generator (SG) water levels were 81 percent in 'A' SG and 83 percent in 'B' SG on the operate range indicators.

B. Event Description

This event was a relatively complex transient beginning with an automatic reactor trip and resulting in an overfill of the 'B' Once-Through Steam Generator [AB-SG]. To aid in understanding this event, the following list describes the off-normal conditions which occurred subsequent to the reactor trip:

1. Reactor Coolant Pumps (RCPs) [AB-P] P32A and P32C tripped due to an undervoltage condition on 6900V AC bus H1. This bus did not "fast transfer" from the in-house power supply unit auxiliary transformer to off-site power supply Startup Transformer number one (SU1) when the generator lockout occurred. RCPs P32B and P32D remained running (one RCP per RCS loop). This condition did not impact the primary system response to the transient and required no operator action initially.

2. A "half trip" of the Emergency Feedwater Initiation and Control (EFIC) [BA] system occurred due to a low water level spike on 'B' SG sensed by EFIC channel 'C'. As the EFIC system requires trips on two of its four channels for automatic actuation, the half trip did not actuate any components.

3. Various Main Feedwater (MFW) [SJ] system anomalies occurred (reference Figure 2 for a simplified MFW valve configuration):

   a. The 'B' MFW block valve CV-2675 [SJ-20] did not fully close as designed.

   b. The 'A' and 'B' startup valves, CV-2623 and CV-2673 [SJ-FCV], respectively, and the 'A' and 'B' low load control valves, CV-2622 and CV-2672 [SJ-FCV] , respectively, failed to close as designed.

   c. The 'B' MFW pump P1B [SJ-P] failed to run back to minimum speed.

   d. The 'A' MFW pump P1A ran back to a minimum speed higher than the preset minimum speed. The post-reactor trip minimum speed for both turbine-driven pumps P1A and P1B is higher than the preset minimum speed set during startup because of the inventory of low pressure steam available to the pump turbine from the 'A' Moisture Separator Reheater, MSR E12A. During startup, the pump turbine runs on Main Steam (high pressure steam) until sufficient steam pressure is available from the MSR. (Four MSRs are located between the high pressure turbine and the two low pressure turbines.) When the minimum speed for the MFW pump turbine is set, only high pressure steam is available. However, after a reactor trip, low pressure steam from the MSR is still driving the MFW pump turbine until the MSR inventory is exhausted. This additional steam pressure after a reactor trip causes the MFW pump to run initially at a higher speed, resulting in a higher discharge pressure head. Normally this has no effect on the plant's post-trip response as feedwater flow to the SGs is controlled by the MFW valves, which are designed to close when the reactor trips and reopen only as necessary to maintain low level limits in the SGs. However, the combination of the higher pump speed and the valve failures during this transient produced an abnormal condition which affected the event.

   e. The zero set on the 'B' SG operate range water level recorder was low by approximately 3 to 4 percent, causing the indicated level being used by the operators to be lower than the actual SG level. The level was verified during the post-trip review by comparisons with the 'B' SG EFIC high range level indication, which is considered to be more accurate after a reactor trip. (See Figure 3.)

The sequence of events for this transient follows. Figure 1 lists the time of each significant event with a brief description.

At approximately 2030 hours on January 20, 1989, the control room operators began observing occasional spikes in the indicated excitation voltage for the main generator exciter [TL-EXC]. Maintenance assistance was requested to determine the cause of the sporadic indication. Over the next hour and a half the spikes increased in frequency, size, and duration, until at 2158 hours the excitation voltage indicator pegged at maximum voltage and did not return on scale. An operator then switched the automatic voltage regulator off, and, at the same time, a generator lockout occurred. The generator lockout caused a main turbine trip, which in turn caused an anticipatory reactor trip. (Loss of the main turbine when the reactor is greater than 43 percent of full power initiates an anticipatory Reactor Protection System (RPS) [JC] trip.)

Following the trip, the operators observed that CV-2675, the 'B' MFW block valve, had failed to fully close and that pressurizer level was below the pressurizer heater cutoff point of 55 inches and falling rapidly. (The reactor trip signal should have driven the valve closed in approximately 19 seconds.) The SG overfeeding condition could be seen in both a level increase in 'B' SG (an alarm was received on 'B' SG high water level) and the pressurizer level decrease caused by the resultant RCS cooldown. Operators immediately provided additional RCS makeup by manually starting the standby High Pressure Injection (HPI) [BQ] pump P36A and opening two of the four HPI injection valves. Pressurizer level began increasing from a minimum indicated level of 30 inches. An attempt was made to manually close the 'B' MFW block valve. When the valve still did not close, the 'B' MFW isolation valve CV-2630 was manually closed. This action terminated the SG overfill at an indicated operate range level of approximately 95 percent (an actual level of 99 percent).

At approximately the same time that the 'B' MFW isolation valve was closed, MFW pump P1B speed began decreasing rapidly. About two minutes later the 'B' MFW block valve indicated closed. Observing that the block valve was now closed and 'B' SG water level was decreasing, the operator believed that the closure of the block valve would prevent further overfeeding and reopened the isolation valve. However, MFW pump P1A was continuing to feed 'A' SG through the failed open 'A' startup valve. With P1B speed decreasing, the suction pressure available to P1A increased. The higher suction pressure combined with the higher speed of P1A (above the preset minimum speed) increased the fill rate to 'A' SG. (The open low load control valves did not affect the transient as the low load block valves had closed as designed.)

As the operators observed the erratic operation of P1B, which had dropped to 925 rpm instead of running at its minimum speed of 3000 rpm, the MFW crossover valve CV-2827 was opened at 2200 hours, approximately two minutes after the trip. This action was performed in anticipation of stopping P1B, which did not have sufficient discharge pressure to feed the SG, and using P1A to feed both 'A' and 'B' SGs. However, with the crossover valve open, P1A was now overfeeding 'B' SG through the open 'B' startup valve CV-2673. The indicated level for 'B' SG began increasing from 87 percent (actual level 91 percent).

One minute after the crossover valve was opened, 'B' SG level peaked at 425 inches on the EFIC SG level indication, or greater than 100 percent actual operate range (Figure 3). The indicated operate range level during this overfill was stable at 97 percent. The level in 'B' SG then began to decrease, possibly due to the decreasing steam pressure to the P1A turbine caused by exhaustion of the MSR inventory. At 2202 hours operators manually tripped P1B.

About three and a half minutes after opening the crossover valve, operators observed that the startup valves and the low load control valves were still open. The indicated level in 'B' SG was 97 percent and stable and in 'A' SG was 84 percent and increasing. Operators manually closed both startup valves and both low load control valves while simultaneously closing both 'A' and 'B' MFW isolation valves. These actions terminated feedwater flow to both SGs. At this time the average RCS temperature (T_{ave}) reached a low point of 541 degrees (normal post-trip T_{ave} is 545 degrees).

At 2201 hours operators had observed pressurizer level at 63 inches and closed the two HPI valves previously opened. Approximately 10 to 15 minutes after HPI was secured, a fire alarm was received in the Upper North Piping Penetration Room. This room contains the HPI piping that penetrates the Reactor Building. An operator was dispatched to investigate. The cause of the alarm was found to be smoke coming from combustible material which was in contact with the HPI piping. This piping had been exposed to high temperature RCS water because HPI check valve MU-34B had failed to reseat when HPI flow was stopped. A differential pressure existed between an HPI line which injected into an RCS cold leg with a running RCP and an HPI line associated with a tripped RCP. This pressure differential caused reverse flow through the failed open check valve into an HPI crossover line and back into the RCS. The reverse flow was apparently stopped approximately six hours later when the tripped RCPs were restarted. Additional details related to this incident will not be discussed further in this report as it is the subject of a separate Licensee Event Report, 50-313/89-004-00.

At 2238 hours, 40 minutes after the reactor trip, the 'A' and 'B' MFW isolation valves were reopened as SG levels decreased to approximately 100 inches on the startup range level indication. The startup valves and P1A were being controlled in manual due to the previous control problems.

At 2259 hours the level in 'B' OTSG was observed decreasing to the EFIC low level actuation setpoint of 14.5 inches. The operator began incrementally opening 'B' startup valve as the 'B' SG level decreased. However, he had not verified the discharge pressure on P1A and was unaware that the discharge head was insufficient to supply enough flow to the SG to maintain level. By the time this was recognized and pump speed was increased, level could not be recovered before the EFIC actuation setpoint was reached. Emergency Feedwater was briefly injected into 'B' SG before the two EFW pumps were secured approximately 20 seconds after the actuation. At 2305 hours the auxiliary (startup) feedwater pump was started to supply feedwater to both SGs, and at 2315 hours P1A was secured.

A routine post-trip Reactor Building walkdown was conducted and identified a possible non-isolable RCS strength boundary leak on January 21 at 0458 hours. This leak was located upstream of manual valve RBD-8B, a drain valve on the RCP P32B suction line, and was identified by a small buildup of boron crystals and a damp spot on the floor underneath. After removal of the insulation, this was confirmed to be from an elbow weld in the 1½-inch drain line. A cooldown to cold shutdown was initiated. Since Technical Specifications (TS) do not allow any leakage through an RCS strength boundary, this leak was considered to be greater than TS limits and a Notification of Unusual Event (NUE) was declared accordingly at 1258 hours on January 21.

The Decay Heat Removal system was placed in operation at 1245 hours on January 22. Cold shutdown conditions were established at 1730 hours and the NUE was terminated.

C. Safety Significance

The load rejection from 100 percent power and operation with one RCP per loop are both analyzed plant conditions and presented no safety concerns in themselves.

The improper response of secondary system components caused an abnormal post-trip response requiring significant operator intervention to stabilize the plant in a hot shutdown condition. However, the actions required are encompassed by the Emergency Operating Procedures and the operators are trained in the use of those procedures. Those actions were carried out adequately to place the plant in a safe condition.

The effects of the overfill of 'B' SG were evaluated by Babcock and Wilcox and determined to be covered by existing stress analyses for the steam generators.

The non-isolable RCS leak was not large enough to have been indicated by RCS leak rates calculated before the plant trip or by the normally in-service leak detection equipment. However, the fact that this leak was in an RCS strength boundary and was caused by a welding flaw does make this condition significant.

D. Root Cause

Post-trip investigations were conducted to determine the following:

1. the initial cause of the reactor trip (root cause of the exciter voltage spiking);
2. the cause of the failure of bus H1 to complete a fast transfer to SU1;
3. the reason for the erratic post-reactor trip response of P1B;
4. the reason CV-2675, 'B' MFW block valve, failed to fully close;
5. the reason the startup valves and low load valves failed to close;
6. if the EFIC half-trip indicated an equipment malfunction; and
7. the cause of the RCS leak upstream of RBD-8B.

1. Initial Cause of the Reactor Trip

Inspection of the exciter stator after the trip revealed a broken connection strap between poles which are components of the AC exciter field. Arcing had occurred inside the exciter, and the intermittent voltage fluctuations were generated each time the circuit was broken and reestablished across the separation in the pole piece. As contact was lost, the subsequent drop in generator voltage would create a large increase in exciter voltage demand. When contact was made once again, the demand would quickly return to normal. Each time contact was lost, arcing further deteriorated the edges of the break. Finally contact could no longer be established and a generator lockout on loss of field was initiated.

The connection straps for the cables from the voltage regulator to the AC exciter field were lacking micarta blocks where the cables were bolted. Without this micarta blocking for support, the connection strap is susceptible to fatigue over time or to being bent during maintenance activities. Other connection straps in the exciter field have this support blocking. The opposite end of the broken connection strap showed evidence of fatigue.

2. Failure of Bus H1 to Fast Transfer

The fast transfer circuit for this bus was functionally tested by requiring the bus to fast transfer from Startup Transformer number two (SU2) to SU1. No problems were noted.

Because of the voltage oscillations occurring at the time of the trip, the synchronization (sync)-check relay [25] was suspected of not allowing the fast transfer. The function of this relay is to permit a fast transfer only if the frequency difference between the existing bus voltage and the SU1 voltage is negligible or zero. Upon loss of synchronism, the relay opens its contact. After synchronism returns, there is a time delay before the contact resets (closes). The length of the time delay is determined by the relay's time dial setting (TDS).

During the inspection of the sync-check relay, the TDS was found to be set at 10, which corresponds to a delay time of approximately 20 seconds. Sync-check relays for the 4160V buses and for the Arkansas Nuclear One - Unit Two 4160V buses and 6900V buses have a TDS of 2, which is approximately 4 seconds. Although initially the longer reset time was suspected of affecting the bus transfer, subsequent conversations with General Electric Company confirmed that this was not a probable cause for the failure to fast transfer. However, the relays for the ANO-1 6900V buses have been reset to a TDS value of 2.

This sync-check relay was also found to have a thicker stationary contact than a similar relay on another feeder breaker for the bus. Conversations with General Electric Company established that these relays are made with four sizes of stationary contacts - 5 mils, 7 mils, 12 mils, and 16 mils - and this relay should use the 5 mil size. Actual measurements showed that the relay had a 16 mil contact. This may have affected the ability of the relay to perform during the voltage oscillations, but was not considered to have contributed significantly to the event.

The most likely cause of the failure to fast transfer was the voltage oscillations occurring at the time of the trip. The sync-check relay for bus H1 apparently detected an out-of-sync condition with SU1 and did not allow the fast transfer.

3. P1B Erratic Response to Reactor Trip

Initially P1B failed to automatically run back to minimum speed after the reactor trip. About one and a half minutes after the reactor trip, P1B speed dropped rapidly to approximately 900 rpm instead of its preset minimum speed value of 3000 rpm. The pump was manually tripped approximately four minutes following the reactor trip, after operators opened the MFW crossover valve to allow P1A to feed both SGs.

The Main Feedwater Pumps are controlled by a Lovejoy Control System [JK-L253], which receives input signals from the Integrated Control System (ICS) [JA]. When a reactor trip occurs, the Lovejoy Control System receives a signal from the Rapid Feedwater Reduction (RFR) circuit of ICS. The RFR signal is designed to reduce pump speed demand immediately upon a reactor trip instead of allowing the reduced ICS demand for feedwater to run the pump back; the response of the pump to a reactor trip is faster with the RFR signal.

Initially the P1B control system failure was suspected to be in the Track and Hold circuit of the Lovejoy Control System. Track and Hold is designed to track the demand signal to the pump and hold the demand at that point if the input signal should fail. However, this circuit was tested and no problems were found. Proper operation of the entire control system was tested by simulating the pre-reactor trip conditions of the pump and simulating simultaneous reactor trip and RFR signals. The control system responded as designed. Exhaustive testing was performed using steam supplied from the startup boiler to actually run the pump and simulating the reactor trip and RFR signals; no anomalies were found. Testing and functional checks verified the initiating signals and the expected response of P1B to be in accordance with the Lovejoy Control System design. No cause for the abnormal response of P1B after the reactor trip could be found.

4. 'B' MFW Block Valve Failure to Fully Close

The MFW block valves close at two different speeds, either fast speed or slow speed. During a normal plant shutdown, the valve receives a signal to close in slow speed based on the feedwater demand. However, on a reactor trip the valve receives a signal to close in fast speed based on the reactor trip signal. In fast speed the valve closes in 19 seconds; slow speed requires 127 seconds.

During this event the block valve did receive the signal to close in fast speed as the actual closing time was 90 seconds. However, the failure of P1B to run back to its designed minimum speed created an abnormally large pressure differential across the valve, causing the valve motor operator to reach its closing torque limit and stop. When the 'B' MFW isolation valve was closed and P1B speed dropped, the pressure differential across the block valve was reduced and the valve closed. The developed pressure differential across the block valve has been estimated to be greater than 300 psid at the time the valve attempted to close; the closing torque switch setting was found to be slightly less. The torque switch for the valve had been set during the previous outage (1R8) based on settings provided by engineering personnel. However, the torque switch setting was based on pressure differential calculations which had assumed the normal operating value of 70 psid for the valve, which was not indicative of the worst case conditions. Therefore the calculated value for the torque switch setting was too low. (This valve, even though not safety-related, is included in the ANO-1 valve testing program.)

Prior to 1R8, a review process had been initiated in which operations personnel review the pressure differential calculations; however, the calculations for the 'B' MFW block valve had not been forwarded for review. The operations review includes an assessment of the assumptions used in the calculations.

5. Startup Valves and Low Load Control Valves Failure to Close

The RFR circuit of ICS sends a signal to the normally open train 'A' and 'B' startup valves and low load control valves when a reactor trip occurs. A large negative signal from the RFR circuit is input to the valve controllers, which drives them closed. The valves are released when SG level reaches 30 inches and are then allowed to control SG level at low level limits (approximately 27 inches on the startup range level indication).

During the last refueling outage, the RFR circuitry to these four feedwater valves was modified as part of an ICS simplification design change. A wiring error during this modification resulted in the failure of the RFR signal to reach the valve controllers. The same wiring error occurred in both loop 'A' and loop 'B' of the RFR circuitry. The valves responded to the loss of an input signal by oscillating near mid-scale on valve demand until manually closed by the operator. The low load control valve failures did not affect the transient because the low load block valves closed as designed on the reactor trip. The close signal to these block valves comes directly from the reactor trip signal and does not depend on the RFR circuit.

6. EFIC Half Trip on 'B' SG Low Level in EFIC Channel 'C'

Each of the four EFIC channels contains a 'B' SG low level initiate bistable. Previous inadvertent trips of these bistables have occurred due to post-trip pressure waves in the steam generator which last approximately half a second. A time delay of two seconds had been incorporated previously into the EFIC design to prevent an unwarranted EFW actuation. This instrument channel from the transmitter associated with the low level indication was checked and no malfunctions were found; the time delay setting was verified to be set at two seconds. No equipment malfunctions were indicated. (The transmitter had recently been calibrated and was not calibrated again as part of this investigation.)

7. RCS Leak Upstream of Valve RBD-8B

The location of the leak was determined to be a weld upstream of RBD-8B, a normally closed drain valve on the RCP P32B suction line. The leak was apparently caused by a minor welding flaw in combination with thermal cycling, which created a crack that propagated through the full thickness of the fillet weld. Although this type of crack normally appears at the toe of the weld, in this case the crack appeared at the surface in the transverse direction. The crack ran circumferentially around the weld. Examination of the weld indicated that the welder had left a cold lap in an area where the welding was stopped.

E. Basis for Reportability

This report contains three events which are considered to be reportable under 10CFR50.73(a)(2)(iv), an event or condition that resulted in manual or automatic actuation of any Engineered Safeguards Feature, including the RPS: the automatic reactor trip caused by the generator loss of field, the initiation of additional makeup using the HPI system, and the automatic actuation of EFIC on low 'B' SG level.

This event was reported to the NRC Operations Center in accordance with 10CFR50.72(b)(2)(ii) at 2235 hours on January 20. The EFIC actuation was reported at 0202 hours on January 21.

F. Corrective Actions

The affected poles in the exciter field were replaced and all the pole connections were removed, inspected, and reinstalled in accordance with directions from the exciter vendor, Westinghouse. Also, blocking was added to provide additional support to the pole connections that failed.

The bus H1 sync-check relay 16-mil-thick stationary contact has been replaced by a 5-mil-thick contact. Also, the sync-check relays for the ANO-1 6900V buses have been reset to a TDS of 2.

The operate range level recorders for both 'A' and 'B' SG were calibrated to ensure that the zero set is as accurate as possible.

The availability of low pressure steam from the MSR to the MFW pump turbines and its effect on pump speed after a reactor trip had been previously recognized as a concern (see Licensee Event Report 50-313/88-003-00). At that time engineering evaluations were initiated to consider possible alternatives. No acceptable resolution has yet been found, and engineering is continuing efforts to successfully resolve this concern.

The miswiring in the RFR circuit which caused the failure of the startup valves and low load valves to perform as designed was corrected. Continuity checks and functional tests were performed on the circuit. Reviews were performed of the Design Change Package (DCP) used to install the modification which caused the wiring error, as well as the work plan used for post-modification testing, to determine if any other ICS modifications performed at that time needed additional testing. All other areas of the DCP were determined to have been functionally tested after the installation was finished; however, certain areas were functionally tested again after the miswiring was corrected as an additional verification measure.

Engineering has revised the pressure differential calculations for the 'A' and 'B' MFW block valves and the 'A' and 'B' low load block valves. The revised maximum pressure differential for these valves is now 800 psid. This calculation has also been reviewed and approved by Operations, and this review process will be continued. The torque switch settings for the MFW block valves and the low load block valves have been adjusted based on the revised calculation.

The initial indication in the leaking weld upstream of RBD-8B was ground to determine if the weld was cracked or if the leak was through a pin hole. After verifying that the weld was cracked, the section of piping containing the weld was removed to facilitate the examination of the faulty weld. A new section of piping was prefabricated and welded into place. Also, the rest of the welds between RBD-8B and the RCS cold leg were inspected. No other defects were identified through either visual examination or liquid penetrant examination.

This event has been evaluated based on the Human Performance Evaluation System to determine factors which affected the performance of the operators. Recommended actions for future operations training have been developed and will be evaluated for implementation. These recommendations include emphasizing the use of the EFIC steam generator level indicators after a reactor trip and increasing operator awareness of potential inaccuracies in any indication as it approaches its maximum or minimum value.

G. Additional Information

A previous event involving a SG overfill caused by a MFW block valve failure to close and the elevated post-reactor trip minimum speed of the 'A' MFW pump was reported in Licensee Event Report 50-313/88-003-00.

Energy Industry Identification System codes are identified in the text in brackets [XX].

FIGURE 1

SEQUENCE OF EVENTS

TIME	EVENT
1-20-89	
20:30	Exciter voltage regulator swings first observed by operations personnel - Electrical maintenance personnel called to investigate problem.
21:58	Exciter failure - broken/burned pole piece connector
21:58:11	Automatic Voltage Regulator turned off
21:58:11	Turbine solenoid trip from generator lockout
21:58:11	Reactor Trip
21:58:15	Undervoltage on 'H1' bus
21:58:15	RCPs P32A and P32C trip on undervoltage
21:58:15	EFIC "half-trip" due to low level spike in 'B' SG
21:58:15	'B' MFW block valve CV-2675 fails to fully close
21:58:15	'A' & 'B' trains startup & low load feedwater control valves fail to close
21:58:15	'B' MFW Pump fails to run back to minimum speed
21:58	Immediate actions of Emergency Operating Procedure performed
21:59	Operators observe pressurizer level below the pressurizer heater cutoff point of 55 inches and falling rapidly
21:59:08	ES standby HPI Pump started and HPI initiated (P36A, CV-1219, CV-1220)
21:59:23	Pressurizer level begins to increase from 30 inches as indicated on control room (CR) level recorder
21:59:25	'B' SG high level alarm at 92%
21:59:38	'B' MFW isolation valve manually closed
21:59:40	'B' MFW block valve indicates closed
21:59:40	'B' MFW Pump running back
21:59:44	'B' SG level decreasing from 99% (95% on CR level recorder)
22:??:??	'B' MFW isolation valve manually reopened
22:00:18	MFW Crossover Valve opened
22:00:25	'B' SG level increasing from 91% level (87% on CR level recorder)
22:01:04	'B' SG level > 100% on operate range (CR level recorder stable at 97%)
22:01:22	'B' SG level begins decreasing
22:01:33	Pressurizer level at 63 inches

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

22:01:33 HPI secured: CV-1219 and CV-1220 closed

22:01:40 'B' SG level on scale at 100% and decreasing

22:02:33 'B' MFW Pump secured

22:03:53 'A' and 'B' MFW isolation valves manually closed

22:03:53 'A' and 'B' trains of startup and low load control valves manually closed

22:??:?? Fire alarm in Upper North Piping Penetration Room; operator dispatched to investigate

22:38 'A' and 'B' MFW isolation valves reopened as steam generator levels decreased to approximately 100 inches

22:59:41 EFIC actuation on Low Level in 'B' SG

23:00:00 EFW pumps secured

23:05 Auxiliary feedwater pump started

23:15 'A' MFW Pump secured

<u>1-21-89</u>

03:54 RCP P32C restarted

03:56 RCP P32A restarted

04:58 Reactor building walkdown identifies RCS leakages including possible strength boundary leakage

12:58 NUE declared based on RCS leakage greater than Technical Specification limits

13:08 Commenced plant cooldown to cold shut down

<u>1-22-89</u>

17:30 Cold shutdown conditions established, NUE terminated

NRC Form 366A
(9-83)

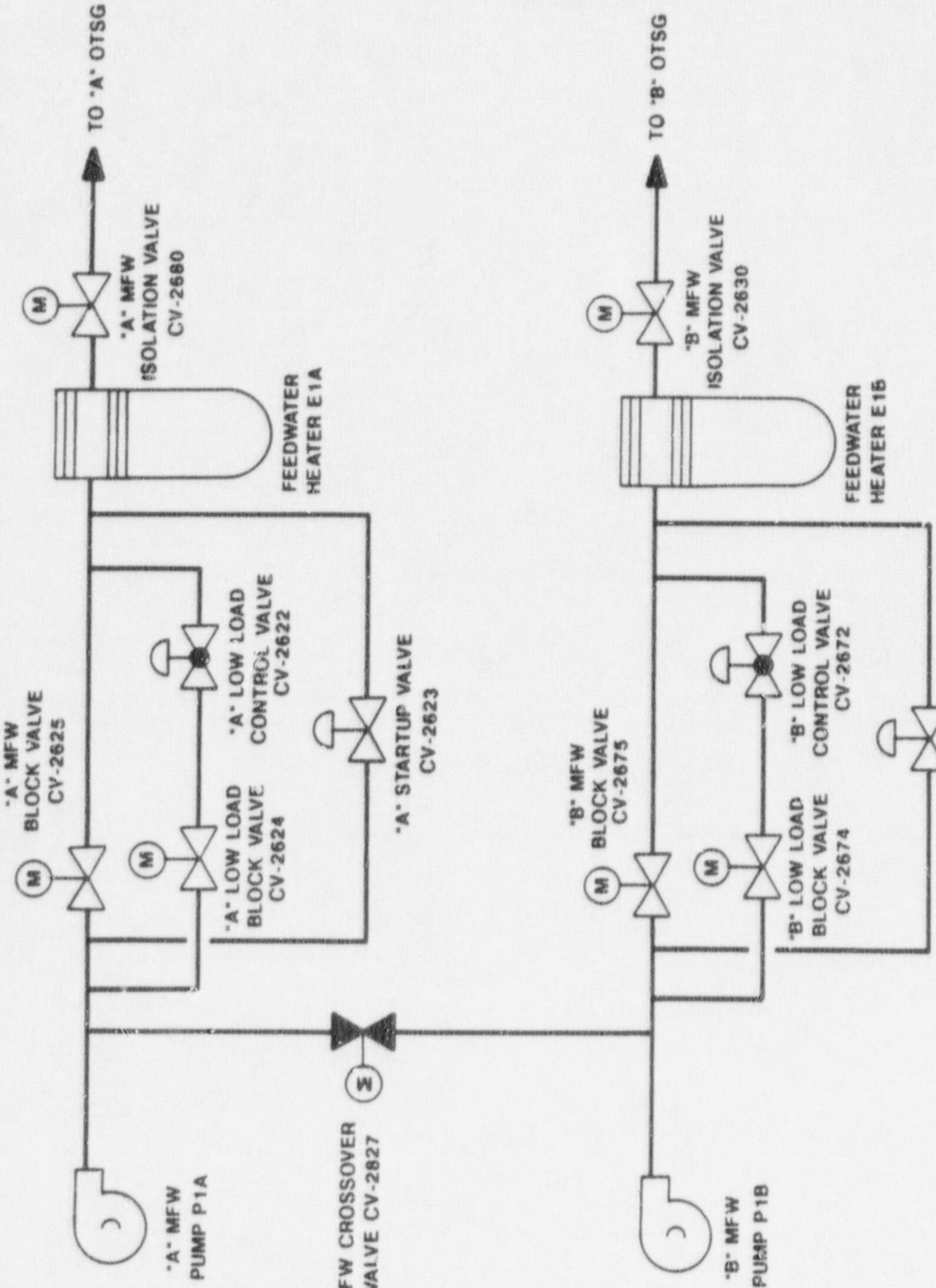
Form 1062.01B
U.S. Nuclear Regulatory Commission
Approved OMB No. 3150-0104
Expires: 8/31/85

LICENSEE EVENT REPORT (LER) TEXT CONTINUATION

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6) Year	Sequential Number	Revision Number	PAGE (3)
Arkansas Nuclear One, Unit One	0|5|0|0|0| 3| 1| 3	8| 9	-- 0| 0| 2 --	0| 0	1|1 OF 1|2

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

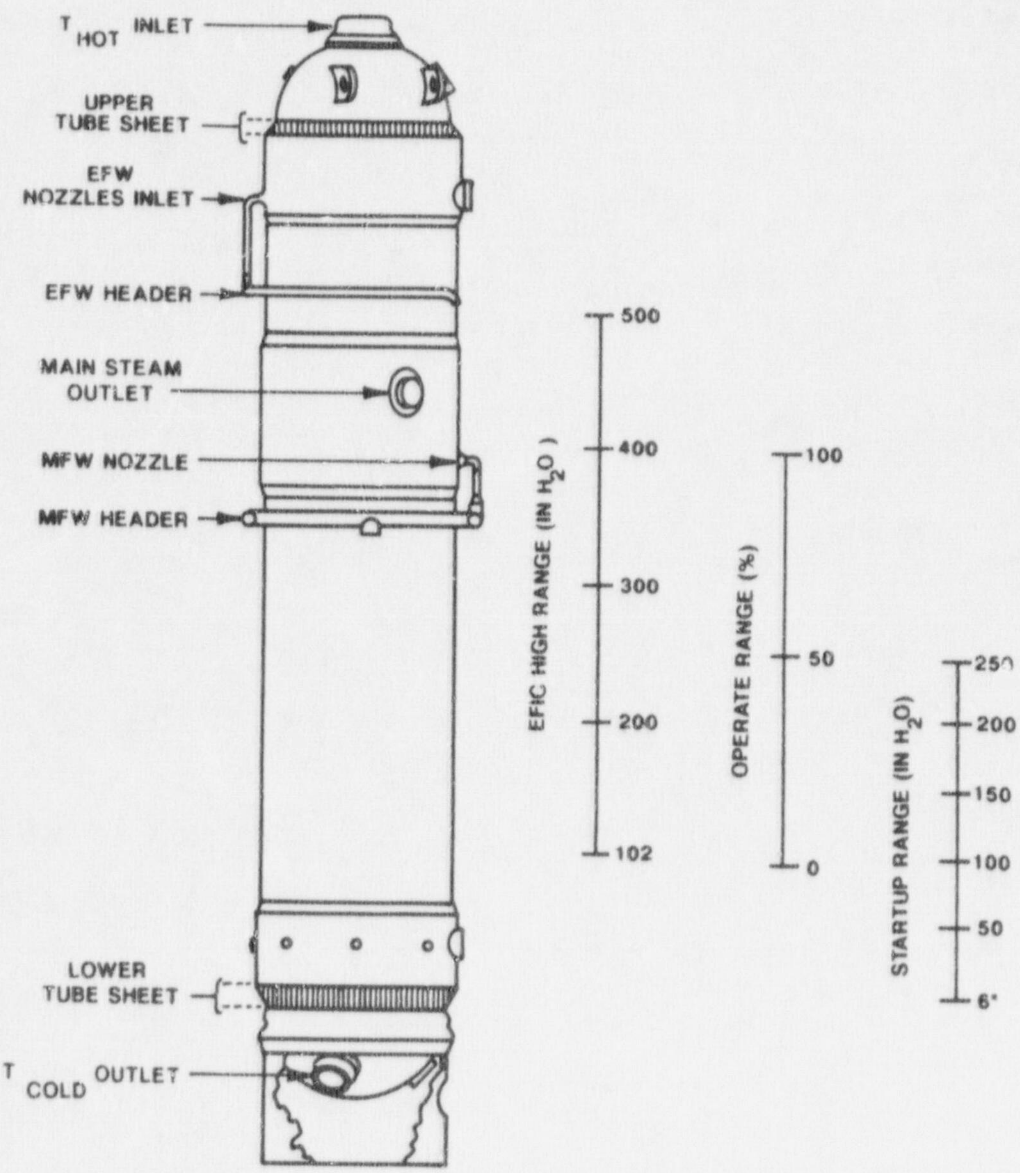
FIGURE 2



FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6) Year	Sequential Number	Revision Number	PAGE (3)
Arkansas Nuclear One, Unit One	0|5|0|0|0| 3| 1| 3	8| 9	0| 0| 2	0| 0	1|2 OF 1|2

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

FIGURE 3



STEAM GENERATOR LEVEL INDICATORS

ARKANSAS POWER & LIGHT COMPANY
March 31, 1989

1CANØ38906

U. S. Nuclear Regulatory Commission
Document Control Desk
Mail Station P1-137
Washington, D. C. 20555

SUBJECT: Arkansas Nuclear One - Unit 1
Docket No. 50-313
License No. DPR-51
Licensee Event Report No. 313/89-002-00

Gentlemen:

In accordance with 10CFR50.73(a)(2)(iv), attached is the subject report concerning a reactor trip caused by the failure of the turbine generator exciter and the failure of various components in the main feedwater system resulting in a steam generator overfill.

Very truly yours,

JMLevine

J. M. Levine
Executive Director,
Nuclear Operations

JML:JDJ:vgh
attachments

c w/att: Regional Administrator
Region IV
U. S. Nuclear Regulatory Commission
611 Ryan Plaza Drive, Suite 1000
Arlington, TX 76011

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