

LICENSEE EVENT REPORT (LER)

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| FACILITY NAME (1) Duane Arnold Energy Center | DOCKET NUMBER (2) 0 5 0 0 0 0 3 3 1 | PAGE (3) 1 OF 0 5 |
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TITLE (4)
Manual Scram in Response to Feedwater Level Control Problems During Reactor Startup

| EVENT DATE (5) | | | LER NUMBER (6) | | | REPORT DATE (7) | | | OTHER FACILITIES INVOLVED (8) | | | | |
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| MONTH | DAY | YEAR | YEAR | SEQUENTIAL NUMBER | REVISION NUMBER | MONTH | DAY | YEAR | FACILITY NAMES | | | | |
| 0 | 6 | 1 | 3 | 8 | 6 | 8 | 6 | 0 | 1 | 7 | None | | |
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| OPERATING MODE (9) N | THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR 5: (Check one or more of the following) (11) | | | | | | | | | | | |
| POWER LEVEL (10) 0 0 5 | 20.402(b) | | | 20.406(c) | | | <input checked="" type="checkbox"/> 50.73(a)(2)(iv) | | | 73.71(b) | | |
| | 20.406(a)(1)(i) | | | 50.38(a)(1) | | | 50.73(a)(2)(v) | | | 73.71(e) | | |
| | 20.406(a)(1)(ii) | | | 50.38(a)(2) | | | 50.73(a)(2)(vii) | | | OTHER (Specify in Abstract below and in Text, NRC Form 366A) | | |
| | 20.406(a)(1)(iii) | | | 50.73(a)(2)(i) | | | 50.73(a)(2)(viii)(A) | | | | | |
| | 20.406(a)(1)(iv) | | | 50.73(a)(2)(ii) | | | 50.73(a)(2)(viii)(B) | | | | | |
| 20.406(a)(1)(v) | | | 50.73(a)(2)(iii) | | | 50.73(a)(2)(ix) | | | | | | |

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| LICENSEE CONTACT FOR THIS LER (12) Bradford N. Thomas, Technical Support Engineer | | | | | | | TELEPHONE NUMBER 3 1 9 8 5 1 - 7 3 0 9 | | | | |
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| COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13) | | | | | | | | | | |
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| CAUSE | SYSTEM | COMPONENT | MANUFACTURER | REPORTABLE TO NRC | | CAUSE | SYSTEM | COMPONENT | MANUFACTURER | REPORTABLE TO NRC |
| X | S, J | - F, C, V | C 6 0 0 | YES | | | | | | |
| X | S, J | - F, C, V | C 6 0 0 | YES | | | | | | |

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| SUPPLEMENTAL REPORT EXPECTED (14) | | | | EXPECTED SUBMISSION DATE (15) | | MONTH | DAY | YEAR |
| <input type="checkbox"/> YES (If yes, complete EXPECTED SUBMISSION DATE) <input checked="" type="checkbox"/> NO | | | | | | | | |

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

On June 13, 1986 the Duane Arnold Energy Center was being returned to service following a planned maintenance outage (6-8-86 through 6-13-86) for the electrical tie-in of the newly installed auxiliary transformer. At 0322 hours with reactor power at approximately 5%, a manual scram was initiated when reactor water level was approaching the low level scram setpoint (170" above top of active fuel) due to feedwater flow control problems. Containment Isolation Groups II, III, IV, and V subsequently initiated as expected at the 170" level. All equipment responded as designed except for the 'A' Control Rod Drive (CRD) pump which tripped following the manually initiated scram. This CRD pump trip was due to low pump suction pressure. Operators restarted the CRD pump at 0326 hours. Operators then secured the 'B' Feedwater Pump at 0327 hours. At 0331 hours the scram was reset as Reactor Water Level was being recovered with condensate and CRD flow. The root cause of the feedwater control problems was desiccant material restricting instrument air flow into the intake orifices of the Feedwater control valve positioners, which in turn caused the control valves to fluctuate in the near closed position. This condition in turn caused the Feedwater Block Valves to trip on thermal overload during opening attempts because of the high differential pressure across the valves. This event is being reported in accordance with 10CFR50.73(a)(2)(iv) as an event or condition that resulted in manual or automatic actuation of any Engineered Safety Feature.

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

On June 13, 1986 the Duane Arnold Energy Center was being returned to service following a planned maintenance outage (6-8-86 through 6-13-86) for the electrical tie-in of the newly installed auxiliary transformer. At 0322 hours with reactor power at approximately 5%, a manual scram was initiated when reactor water level was decreasing and approaching the low level scram setpoint of 170 inches above Top of Active Fuel (TAF) due to feedwater flow control problems.

At 0306 hours the 'B' Reactor Feed Pump (EIIS-System Code SJ) was started at a reactor pressure of approximately 375 psig. Operators were in the process of maintaining Reactor water level at approximately 193 inches (above TAF) when they noticed reactor level to be increasing. The Feedwater Regulating Valve (Control Components International Model #PDA9160-256BW, SJ-FCV-1621) was then throttled in the closed direction. Reactor Level continued to increase to approximately 204 inches (above TAF) so the 'B' Feedwater Block Valve (Condec Lunkenheimer Co. Model #4D12689, SJ-ISV-1636) was also throttled in the closed direction. As reactor level started to decrease, attempts by Operators to control feedwater flow by opening block valve MO-1636, and then alternate path block valve MO-1592, failed when the thermal overload breakers tripped.

At 0322 hours the Reactor was manually scrammed when the Reactor water level was at approximately 175 inches (above TAF) and continuing to drop. Following the Scram, Reactor Water Level continued to drop to approximately 155 inches (above TAF). Containment Isolation Groups II (Drywell Floor/Equipment Drain Discharge Valves), III (Containment Purge Line Valves, and Containment Analyzing Systems Sample Line Valves), IV (Residual Heat Removal (RHR) Shutdown Cooling Supply Valve), and V (Reactor Water Cleanup (RWCU) Suction and Discharge Valves) isolation signals were received at 170 inches (above TAF) and all valves isolated as expected. Per design, no Group IV valves moved because the shutdown cooling mode of Residual Heat Removal (RHR) System was not lined up at the time. All equipment responded as designed with the exception of the 'A' Control Rod Drive (CRD) Pump (EIIS System Code AA, 1P-209A) which tripped immediately following the scram from low suction pressure. This CRD pump tripping had no effect on the ability to fully insert all control rods. 1P-209A was restarted at 0326 hours. At 0327 hours the 'B' Reactor Feed Pump was secured by operators. At 0331 hours the scram was reset as Reactor Water Level was being recovered using condensate and CRD flow. At approximately 0332 hours with reactor level at 175 inches (above TAF), the Primary Containment Isolation Valves (Groups II and III) were reset. At 0336 hours Reactor Water Level had reached 194 inches (above TAF). At 0345 hours Group V (Reactor Water Cleanup) valves were returned to normal after refilling the system.

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

Investigation following the scram revealed that the Moore position controllers (Moore Products - Model 74G) on the Feedwater Control Valves CV-1621, and CV-1579 were found to cause excessive valve position hunting/ position fluctuations near the closed position. This condition was caused by desiccant in the instrument air lines, which restricted air flow through the inlet orifices in the positioners. With the Reactor at low pressure, the block valves MO-1636 and MO-1592 are unable to open against Feedwater Pump Pressure, as these valves are not designed to open against a large pressure differential. In conclusion, the fluctuating control valves were unable to close enough to sufficiently control flow. It was then necessary to close the block valve. This combination of the regulating valve being too far open and the block valve being too far closed created too high a differential pressure across the block valve and prevented it from opening. (See attached Figure 1 for feedwater flow paths.)

Investigation into past scrams at the DAEC shows that desiccant in these type Moore positioners may have been a contributing factor in two other scrams [11-21-77 (40% power), 6-3-83 (100% power)]. In addition these Moore positioners failed on the 'A' and 'B' Standby Filter Units (SFU, EIIS - System Code VI) in July 1984 (See LER 84-026). In an attempt to reduce the amount of moisture and desiccant in the instrument air system, the DAEC changed out its Instrument Air Dryer System to a non-desiccant system in August 1984.

However, small amounts of very fine desiccant particles are still suspected to be present in the Instrument Air Lines. It was previously determined that these Moore positioners are the only instruments which are affected by this residual desiccant material, due to their small orifice size.

On 6-13-86 the Moore positioner was replaced on CV-1621, and cleaned out on CV-1579. These valves then successfully passed post-maintenance testing requirements. As a long term corrective action an Engineering Work Request (EWR) has been initiated to install in-line air filters upstream of all the Moore positioners susceptible to this desiccant problem. This design change will be implemented by the end of the Spring, 1987 refuel outage. As a continuing interim corrective action these Feedwater Regulating Valve Moore Positioners along with the Feedwater Minimum Flow Line Valve Moore Positioners will be cleaned out, adjusted, or changed out every time the unit is shutdown.

Moore positioners are also utilized on the Standby Filter Units (SFU), the Condensate System (EIIS Code KD) Demineralizer valves, and the Radwaste System (EIIS Code WB). However, these Moore positioners are not subjected to the low throttling service which is experienced with the feedwater positioners during reactor startup and shutdown. Therefore, these positioners are judged not to need accelerated preventive maintenance.

As another long term corrective action an EWR was previously generated to recommend a design change (installation of a smaller regulation valve bypassing the large Regulation Valve) to facilitate easier feedwater system operation under low reactor pressure conditions. As an interim measure, the

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startup procedure has been revised to provide additional directions to operators for this situation. This revision directs the operators to trip the Feedwater Pump, thus reducing the differential pressure across the Feedwater valves. The valves can then be opened and reactor level maintained with condensate flow.

On 6-13-86 the thermal overloads on MO-1636, and MO-1592 were checked and found to be properly tripping at their specified set points.

The CRD pump low suction pressure trip following the scram is a recurring phenomenon and is being investigated through a previously generated EWR. The CRD pump tripping had no safety implications which would have prevented the control rods to fully insert.

Prior to plant startup in accordance with Administrative Control Procedures, systems which initiated during the scram were reviewed and found to have performed properly with exception of the CRD pump trip. With the Reactor at approximately 5% of rated power, no significant transients occurred during the manually initiated scram.

Restart of the plant was initiated at 1952 hours on 6-13-86. This event is being reported in accordance with 10CFR50.73(a)(2)(iv) as an event or condition that resulted in manual or automatic actuation of any Engineered Safety Feature. At no time during the event was the health and safety of the public compromised.

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TEXT (If more space is required, use additional NRC Form 388A's) (17)

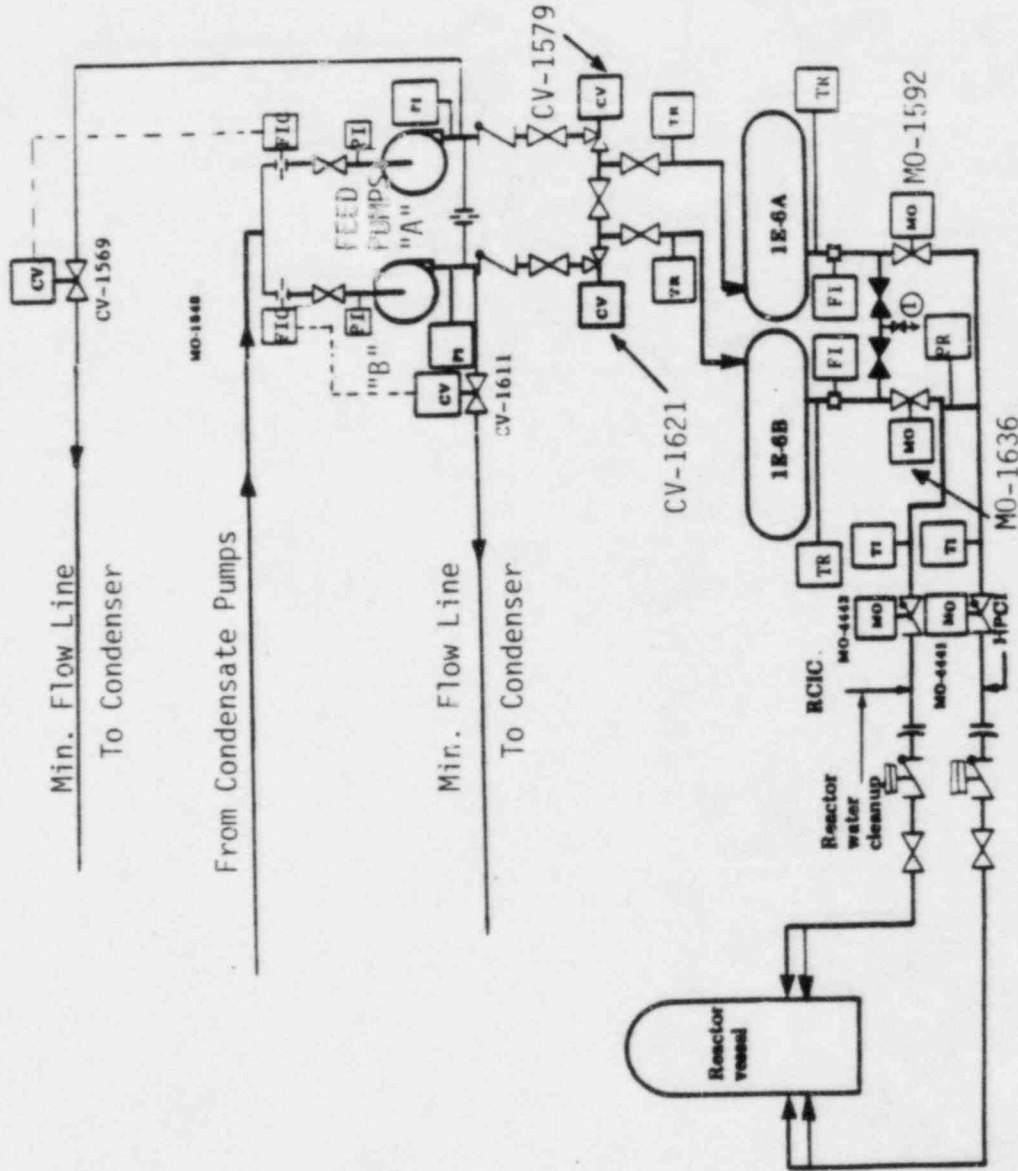


Figure 1
Feedwater Schematic

Iowa Electric Light and Power Company

July 11, 1986
DAEC-86-0506

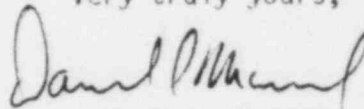
U. S. Nuclear Regulatory Commission
Document Control Desk
Washington, D. C. 20555

Subject: Duane Arnold Energy Center
Docket No. 50-331
Op. License DPR-49
Licensee Event Report No. 86-017

Gentlemen:

In accordance with 10 CFR 50.73 please find attached a copy of the
subject Licensee Event Report.

Very truly yours,



Daniel L. Mineck
Plant Superintendent - Nuclear

DLM/BNT/pl

Attachment - LER 86-017

cc: Mr. James G. Keppler
Regional Administrator
Region III
U. S. Nuclear Regulatory Commission
799 Roosevelt Road
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NRC Resident Inspector - DAEC

File A-118a

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