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FACIL:50-331 Duane Arnold Energy Center, Iowa Electric Light & Pow 05000331

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SUBJECT: LER 86-025-01:on 861210,reactor scram from spurious high intermediate radiation monitor trip.

W/8 ltr.

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TITLE: 50.73 Licensee Event Report (LER), Incident Rpt, etc.

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	NRR/~~DRIS~~/SIB	1	1	NRR/PMAS/ILRB	1	1
	REG FILE 02	1	1	RES DEPY GI	1	1
	RES TELFORD,J	1	1	RES/DE/EIB	1	1
	RGN3 FILE 01	1	1			
EXTERNAL:	EG&G GROH,M	5	5	FORD BLDG HOY,A	1	1
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NRC Form 366 (9-83)

U.S. NUCLEAR REGULATORY COMMISSION
APPROVED OMB NO. 3150-0104
EXPIRES: 8/31/88

LICENSEE EVENT REPORT (LER)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
Duane Arnold Energy Center (DAEC)	0 5 0 0 0 3 3 1	1 OF 0 5

TITLE (4)
Reactor Scram from a Spurious High Intermediate Radiation Monitor Trip

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)
1 2	1 0	8 6	8 6	0 2 5	0 1	1 2	1 0	8 7	None	0 5 0 0 0
										0 5 0 0 0

OPERATING MODE (9): N
POWER LEVEL (10):

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

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)
Jeff Thorsteinson, Technical Support Supervisor	3 1 9 8 5 1 - 7 2 3 8

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13)

CAUSE	SYSTEM	COMPONENT	MANUFAC-TURER	REPORTABLE TO NPRDS	CAUSE	SYSTEM	COMPONENT	MANUFAC-TURER	REPORTABLE TO NPRDS

SUPPLEMENTAL 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)

At 0338 hours on December 10, 1986, a reactor shutdown was in progress. While subcritical at approximately 1-2% thermal power, an automatic scram was received when the B, C and D Intermediate Range Monitors (IRMs) tripped upscale. The cause of the upscale trips is believed to be noise generated from Source Range Monitor (SRM) insertion. The most probable root cause of the scram is this spurious signal. A positive reactivity addition due to a feedwater transient has been investigated, but is not considered as the likely root cause.

Two SRMs were inserted simultaneously which conflicts with some procedural guidelines. Therefore, the root cause of the scram could be attributed to personnel error in not explicitly following procedures, but a procedural inconsistency must be considered as strongly contributing to the error.

As a short term corrective action, the Operator was counseled on the necessity of explicitly following procedures or identifying and clarifying procedural requirements when inconsistencies exist. The effected procedures have also been corrected.

The reactor experienced no significant transients, nor were any emergency systems initiated or required. Therefore, there was no significant effect on the safe operation of the plant from the scram.

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

At 0338 hours on December 10, 1986, a reactor shutdown was in progress for drywell electrical connector repair (AMP connectors). The mode switch was in startup, the main turbine was secured and both recirculation pumps were at minimum flow. One feedwater and one condensate pump were in service. Three Intermediate Range Monitors (IRMs, EIIS System Code IG) were on range four, two IRMs were on range five, and one was bypassed. While subcritical at approximately 1-2% thermal power (due to decay heat and approximately 0.2% due to thermal neutron flux), an automatic scram was received when the B, C, and D IRMs tripped upscale. The cause of the upscale trips is believed to be noise generated from SRM insertion. The possibility of a positive reactivity addition due to a cold feedwater transient was analyzed and our review of the results indicate that this transient was possible, but not probable.

The Operators noted that the IRMs went quickly upscale at the time Source Range Monitors (SRMs, EIIS System Code IG) A and C were selected and driven. Investigation of the event revealed that an Operator had selected the B and D SRMs and inserted them simultaneously and without problems prior to the scram. When the Operator selected and began inserting the A and C SRMs an upscale trip was promptly received on the B, C, and D IRMs.

The general integrated plant operating instructions for plant shutdown cautions against inserting all four SRMs simultaneously or excessively cycling the SRM drives because of electronic noise on the IRMs. The instructions then direct that the SRMs are to be inserted one at a time per the specific operating instruction. However, the specific operating instruction for SRM insertion does not prohibit the insertion of more than one SRM at a time. The specific operating instruction also contains a caution that the insertion of all four SRMs simultaneously or the excessive cycling of the drives may cause fluctuations on the IRM indication. These procedural cautions were added to address the SRM drive noise problems which occurred prior to 1978. Therefore, the root cause of the scram could be attributed to personnel error, but a procedural inconsistency must be considered as strongly contributing to the error. As a short term corrective action, the operator was counseled as to the necessity of explicitly following procedures or identifying and clarifying procedural requirements when inconsistencies exist.

The root cause of the possible noise spike is unknown. The routing and locations of related cabling were reviewed. No interconnections which could clearly explain the cause were found. However, given the unpredictable nature of the noise and the high sensitivity of the IRM equipment, all sources of spurious noise could not be adequately analyzed. Low level electronic noise seems to be inherent to the IRM System. Maintenance was performed on the IRM System following the scram, and prior to startup, which corrected an increase in this low level electronic noise which was being experienced on several IRMs. The significance of this maintenance with respect to the noise causing the scram is unknown at this time. This phenomena was studied as part of testing and investigation during the Spring 1987 Refuel Outage. This investigation was performed after much IRM cabling had been replaced. The testing did not identify any noise signals characteristic of the observed IRM response during the December event.

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Scrams as a result of noise spikes on IRMs and SRM insertion had been experienced prior to and including 1976. Following noise spike problems during SRM insertion, noise filters were added to the drive circuitry of all SRMs and IRMs in 1978.

To address the simultaneous insertion of multiple SRMs as a possible cause of the IRM upscale signals, the caution notes in the Startup, Shutdown, IRM, and SRM procedures have been modified to give more direct guidance on the number of SRMs that are to be inserted at one time.

Concerning the possibility of a cold feedwater transient, some discrepancies were found which were not characteristic of noise problems experienced in the past. Investigation of the process computer alarm log following the scram indicates that the Reactor Protection System (RPS, EIIS System Code JC) logics for channels A and B, were completed approximately 2.7 seconds apart. The upscale trip on IRM B took approximately 2.9 seconds to reset. An IRM noise induced RPS trip in the past was analyzed for similarity. The noise signals were of shorter duration (milliseconds versus seconds). The trip signals from IRM noise also initiated and reset at about the same instant. Although it is not considered impossible for noise to cause a similar process computer alarm log response, further investigation revealed a similar past computer response from a known positive reactivity transient.

It was determined that the cooldown rate alone, prior to the scram, was not of sufficient magnitude to yield the observed response. It was postulated that the most credible initiating event which could yield the observed response with the excessive addition of colder feedwater (actual feedwater temperature was approximately 100 degrees Fahrenheit). However, the Operators noted that no major changes in feedwater flow had been initiated in the minutes prior to the scram. There have been repeated problems with feedwater control in low flow conditions. The flow control valves tend to cause fluctuating flow under these low flow conditions. Some minor feedwater flow fluctuations were noted prior to the scram. This problem was previously discussed in LER 86-017 and an engineering study is in progress to address these problems. Corrective actions are scheduled to be completed during the cycle 9/10 Refuel Outage. Investigation into past scrams at the DAEC shows that feedwater control problems may have contributed to scrams on November 21, 1977 and June 3, 1983.

In order to test the feasibility of a feedwater transient causing this scram, attempts were made to simulate the feedwater transient necessary to cause the observed response. These tests were performed at a simulator which approximates the Duane Arnold Energy Center's configuration. Preliminary analysis of the results indicates that a feedwater addition could have caused the response observed by the Operators and documented in the computer alarm logs. However, the magnitude of the simulated feedwater addition appears to be greater than the actual addition observed. Further analysis was performed when the complete data was received from the simulator. The analysis concluded that a feedwater transient could not be completey ruled out but was unlikely. As an additional corrective action, reactor engineers will

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evaluate the need to augment reactor operator training in this area (operation at low feedwater flows).

Analysis of this event was hampered by the IRMs being on slow speed. One IRM on each side of the RPS will be placed on fast speed during startup and shutdown when practical. Also, because the intermediate ranges of reactor power show an increased sensitivity to noise and feedwater control difficulties, the possibility of manually scramming the reactor at an appropriate power level during a controlled shutdown is still being evaluated.

Following the scram signal, all control rods fully inserted. No Emergency Core Cooling Systems actuated, or were required, and no isolations occurred. All safety related equipment responded per design. Problems were noted with the Control Rod Drive (CRD, EIIS System Code AA) pumps. The B CRD pump tripped on low suction pressure following the scram. When the B CRD pump could not be restarted, the Operators attempted to start the A CRD pump. When the A CRD pump would not start, the Operators temporarily manually overrode the low suction pressure trip function to start the A pump. A review of CRD System operation identified excessive charging water flow, post-scram, to be causing CRD pump runout. Following review of design data, operating procedures were revised to limit excessive charging water header flow following a scram. On Friday, December 19, 1986, with the plant shutdown, the CRD Charging Water Header Throttle Valve (EIIS Code V-17-24) was throttled to approximately one-third open. The reactor was then manually scrammed. The CRD pump did not trip on low suction pressure and CRD pump performance was verified to be within acceptable limits.

During attempts to align shutdown cooling, the inboard shutdown cooling isolation valve (EIIS Code JM-ISV-1908) MO-1908 would not open on a signal from the Control Room. Following unsuccessful attempts to remotely open the valve, the operators manually opened MO-1908. The motor was later inspected and found to have failed. The failed motor was replaced and the valve was returned to service. This valve also failed to open during the shutdown on October 16, 1986 (see LER 86-020). The results of the failure mode investigation were reported in the update to LER 86-020.

An automatic scram resulted from an IRM upscale trip in 1984 (LER 84-037). This event was believed to have been caused by a spurious upscale spike on IRMs E and F (no other channels alarmed or tripped). The SRMs were still withdrawn from the core. The root source of the noise could not be determined.

The purpose of the Intermediate Range Monitoring System is to provide neutron flux information to the Operators during startup, heatup, power ascension, power descension, cooldown, and shutdown. The IRM system will also initiate appropriate trip signals to prevent damage to the fuel from an abnormal operational transient while operating within the intermediate range of reactor power. The purpose of the Source Range Monitoring System is to provide thermal neuton flux information from various points within the core,

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NRC Form 366A
(9-83)

LICENSEE EVENT REPORT (LER) TEXT CONTINUATION

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during reactor startup, shutdown, and low flux level operations. The SRM System is also used to monitor neutron flux level during refuel operations.

All safety systems responsed per design to the scram signal. With the reactor in the IRM range of power, no significant transients occurred following the automatic scram. There was no effect on the safe operation of the plant, nor was there any effect on public health and safety. This event is being reported pursuant to 10CFR50.73(a)(2)(iv) as an "automatic actuation of any Engineered Safety Feature (ESF), including the Reactor Protection System (RPS)."

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Iowa Electric Light and Power Company

December 10, 1987
DAEC-87-1189

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

Subject: Duane Arnold Energy Center
Docket No: 50-331
Op. License DPR-49
Licensee Event Report #86-025, Rev 1

Gentlemen:

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

Very truly yours,

Rick Hannen

Rick L. Hannen
Plant Superintendent - Nuclear

RLH/JCT/go

cc: Mr. A. Bert Davis
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NRC Resident Inspector - DAEC

File A-118a

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