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)
Duane Arnold Energy Center	0 5 0 0 0 3 3 1	1 OF 0 6

TITLE (4)
Degraded Offsite Voltage

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)
									None	0 5 0 0 0
0 7	1 4	8 4	8 4	— 0 2 8 —	0 1	1 1	1 6	8 4	None	0 5 0 0 0

OPERATING MODE (9): N
POWER LEVEL (10): 0 0 0

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)	X 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)
James R. Probst, Technical Support Engineer	3 1 9 8 5 1 - 7 3 0 7

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
C	F K			No					
B	J E	T S	P 0 5 5	Yes					

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)

On 7/14/84 at 1524, with the reactor subcritical in the startup mode at 0% thermal power, a degraded voltage condition on the offsite power grid resulted in the automatic switching of the essential buses from offsite to onsite power. As per design, the RPS logic de-energized, initiating a scram. All required systems operated as designed, including the two Diesel Generators assuming the essential loads. However, the HPCI Inboard Steam Supply Valve also closed. The reason for this response is unknown and under investigation. After consultation with the load dispatcher confirming a return to grid normality, the diesels were secured. A post-event review confirmed operability of all systems, which were then returned to service. Reactor startup commenced three and one half hours following the scram.

On 10/15/84, a special test procedure demonstrated that the cause of the HPCI Inboard Steam Supply Containment Isolation Valve closure was a spurious isolation signal upon re-energizing from either or both of two temperature switches in the HPCI Steam Leak Detection System. A minor design change within the PCI Steam Leak Detection System has resolved this problem. See the attached LER update text for details.

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

On 7/14/84 at 1524, with the reactor at 0% thermal power, subcritical in the startup mode with rod groups one and two withdrawn and pressure at 176 psig, the essential buses were automatically transferred from offsite to onsite power due to degraded offsite grid voltage. As per design, the Reactor Protection System (JC) was de-energized during the transfer, resulting in a scram. The two essential and two non-essential 4160 kV buses (EA) were receiving offsite power via the Startup Transformer (EA) when the essential bus' undervoltage protection instrumentation sensed grid voltage less than 92.3% of nominal for 8+ seconds. Consequently the essential bus breakers tripped, as per design. The non-essential buses remained energized and no load shedding occurred as grid voltage did not drop below 65% of nominal. Both Diesel Generators (EK) auto-started upon loss of offsite power to the essential buses, and within ten seconds had assumed the essential bus' loads. The degraded voltage was caused by the accidental motoring of a 650 MWe generator for over twenty seconds at the Louisa Generating Station (fossil fuel) of Iowa Illinois Gas and Electric in Muscatine, Iowa.

Upon switching of the essential buses from offsite to onsite power, the RPS motor-generator set (JC) was de-energized, resulting in the Electrical Protection Assembly breakers (EC) tripping. This, in turn, de-energized the RPS logic, causing the scram. As per design, Group 1, 2, 3, 4, and 5 isolations and Reactor Building isolation occurred. The Recirculation Pumps (AD) tripped, the Circulation Water pumps (KE) tripped, the Control Rod Drive pump (AA) in use, pump B, tripped, the Recirculation pump trip breakers (EA) tripped, the Uninterruptable AC Motor-Generator set (EE) was transferred to DC, the Control Building habitability systems (VI) isolated, and several other systems powered off the essential buses also responded, all per plant design. The Process Computer (IQ) failed at the time of the scram due to a momentary electrical disturbance as its power source, the Uninterruptable AC Motor-Generator set, was transferred to DC. In addition, its "B" Drum Cabinet (for memory storage) and the Virtual Address Extension disk drives and CPU were without power during the switchover from offsite to onsite power, as they are powered off an essential bus.

The HPCI Inboard Steam Supply Valve (BJ-ISV-2238) closed following the scram. Although this was not per design, if necessary the valve could have been opened within a short time by manual operation after resetting the HPCI logic, which was later done at 1810 after Operations personnel had returned other systems to normal. The isolation signal was received from the Steam Leak Detection System (JM). Extensive review indicates the signal origin may have been either or both of two temperature differential switches within the Steam Leak Detection System logic. These switches contain elements powered off essential buses, and therefore would have lost and regained power during the switchover to onsite power, possibly resulting in a spurious signal(s). Investigation is continuing.

The reactor scram was reset at 1556, following the restarting of the RPS m-g sets. All required systems were returned to normal by 1600. At 1620, after the Iowa Electric system control center had advised the plant that recurrence of the grid voltage excursion was unlikely the essential buses were transferred back to the start-up transformer, and both diesels were unloaded. The diesels were secured at 1642. After a post-event review had determined the operability of all safety-related systems and that there had been no effect upon safe plant operation, startup of the reactor was commenced at 1901.

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

The following is being added to LER 84-028 as Revision One, and deals with the HPCI Inboard Steam Supply Valve Isolation.

On 10/15/84, a special test procedure was run on the "A" logic of the HPCI Steam Leak Detection System (HPCI SLDS). The reactor was in the shutdown mode for a planned maintenance outage, with reactor vessel pressure below that at which HPCI is required to be operable by Technical Specifications. A review of the isolation logic for the HPCI Inboard Steam Supply Valve indicated that the spurious isolation of this valve following the loss of offsite power on 7/14/84 most likely originated in the HPCI SLDS isolation logic. Within the HPCI SLDS logic (see Figures 1 and 2), the "A" logic controls the AC Inboard Steam Supply Containment Isolation Valve (BJ-ISV-2238) and the "B" logic controls the DC Outboard Steam Supply Containment Isolation Valve (BJ-ISV-2239). Power for each HPCI SLDS train is provided by a separate AC essential bus.

By design, each logic train for the HPCI SLDS consists of two temperature switches and two differential temperature switches, each a permissive for energizing a downstream relay (JM-94-K4A or B). This relay closes contacts in the HPCI isolation logic, which energize another relay that in turn results in the isolation signal for the associated HPCI Steam Supply Containment Isolation Valve being sealed in (see Figure 1). Half of each logic side consists of a HPCI Equipment Room High Differential Temperature Switch (JM-TDS-2260A or B) in parallel with a HPCI Equipment Room High Temperature Switch (JM-TS-2261A or B). The momentary closure of either switch in a logic train energizes relay K4 (A or B), resulting in isolation of the associated HPCI Steam Supply Containment Isolation Valve. The other half of each logic side consists of a Suppression Pool Area High Temperature Switch (JM-TS-2526C and D) in parallel with a Suppression Pool Area Differential Temperature Switch (JM-TDS-2521C or D). As there is a 15 minute timer immediately downstream of these switches, either switch must remain closed for a minimum of fifteen minutes to energize relay K4. Due to this significant time delay, these instruments were eliminated as the cause of the spurious HPCI isolation on 7/14/84. The test procedure run on 10/15/84 de-energized the instruments in the "A" SLDS logic, which controls the inboard HPCI steamline isolation valve that isolated, and restored power approximately ten seconds later, simulating the conditions of 7/14/84.

Three trials of the test procedure were performed. On the first and third trials the suspected switches (TS-2261A and TDS-2260A) closed and quickly reopened upon restoration of power. The result was an isolation signal of short duration which was sealed into the HPCI isolation logic. The second test produced no isolation signal. The test results proved a random signal could be expected from the two suspect instruments in the HPCI "A" SLDS logic upon re-energizing after loss of offsite power. The instruments are both Riley Pan Alarm Model 86, and appear in the identical configuration in the HPCI "B" SLDS logic, which signals for closure of the outboard HPCI steamline isolation valve. It is believed these "B" instruments did not trip during the 7/14/84 event due to the random nature of the spurious signal, as demonstrated by the test. Discussions with the manufacturer have also indicated that a short isolation signal upon re-energizing is not unexpected for these instruments due to their internal design.

After review, a minor design change was instituted to eliminate the random HPCI AC power dependency that results from powering up the HPCI SLDS logic. The design change implemented in both the "A" and "B" logic trains was to introduce a short time delay of approximately one second on restoration of AC power. (See Figures 1 and 2.) Prior to the design change, a power monitor relay (JM-30-K2A or B) for each side, connected in parallel to the HPCI SLDS instruments, activated a Control Room annunciator on loss of logic power. The design change involved replacing this relay with an Agastat (Control Products Division - Amerace Corporation) relay with time pickup controls (JM-2-K2), retaining the annunciation contacts and adding an additional set of closed when energized contacts in series with the two instruments in question. This design results in the following; a) with no loss of offsite power there is no time delay introduced for HPCI isolation from the SLDS; b) with a loss of offsite power the logic and instruments themselves are de-energized. On restoration of power the spurious isolation signals are prevented from energizing the downstream relay (K4) by the open contact controlled by the power monitor relay (K2). When this contact closes approximately one second later, the temperature switches have stabilized internally and the isolation signal is no longer present unless it is a genuine signal.

Surveillance tests on the HPCI Steam Supply Containment Isolation Valves over the past year have timed all valve closures at under 11 seconds for the Inboard Valve and under 8 seconds for the Outboard Valve compared to the Tech Spec value of 13 seconds. Channel response time for the Primary Containment Isolation System is not specified in Technical Specifications. The addition of a one second time delay in logic response time on loss and regain of essential AC bus power is judged not to be a reduction of the margin of safety. The increase in HPCI reliability, however, results in an increase in the overall margin of safety at DAEC.

General Electric has stated that it is performing a 10 CFR Part 21 evaluation on a similar problem for a requisition plant, and that this modification at DAEC is similar to that implemented at other GE plants with a similar configuration.

On 10/21/84, a revised version of the special test procedure run on 10/15/84 was performed to determine the effectiveness of the design change to the "A" and "B" HPCI SLDS logic. The reactor remained in a shutdown mode for the planned maintenance outage, with reactor vessel pressure below that at which HPCI operability is required. The HPCI SLDS logic and instrumentation power supplies (see Figure 2) were removed and restored approximately ten seconds later for each logic train. In twenty-four trials for each side, Control Room annunciation indicated spurious isolation signals from either or both of the suspected temperature switches. However, none of the spurious signals sealed in or would have caused an isolation due to the design modification. This indicated the additional power monitor relay contact is successfully preventing unnecessary HPCI isolation due to the re-energizing of the instruments in the HPCI Steam Leak Detection System. Further, post-installation/maintenance testing and routine operability testing during subsequent startup successfully demonstrated HPCI operability.

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ADDITIONAL INFORMATION

The Reactor Core Isolation Cooling Steam Leak Detection System (JM) has logic similar to the HPCI SLDS. In the RCIC SLDS logic, however, the two logic trains are powered off electrically separate 125 DC batteries through an inverter. The instruments (TS and TDS) are the same model and manufacturer (Riley Pan Alarm Model 86). De-energizing and restoring the power to the RCIC SLDS inverters could therefore result in a spurious isolation signal from the RCIC SLDS. As RCIC isolation will not occur as a direct result of loss and restoration of AC power, immediate incorporation of a similar design change in RCIC was judged unnecessary. However, it is our current intent to incorporate the time delay in these circuits in the near future. Consideration is also being given to a design change to power the HPCI as well as the RCIC Outboard Steam Supply Isolation Valve (DC) Steam Leak Detection logic from inverted DC (which has proven highly reliable). In this manner, the AC dependency of the HPCI SLDS logic for the Outboard (DC) Isolation Valve will be eliminated.

Our discussions with GE indicate that the tendency for these devices to generate spurious isolation signals during a power-up situation will be the subject of GE notifications to BWR operators. In parallel, Iowa Electric is notifying other utilities of our experience via the INPO "Network" system.

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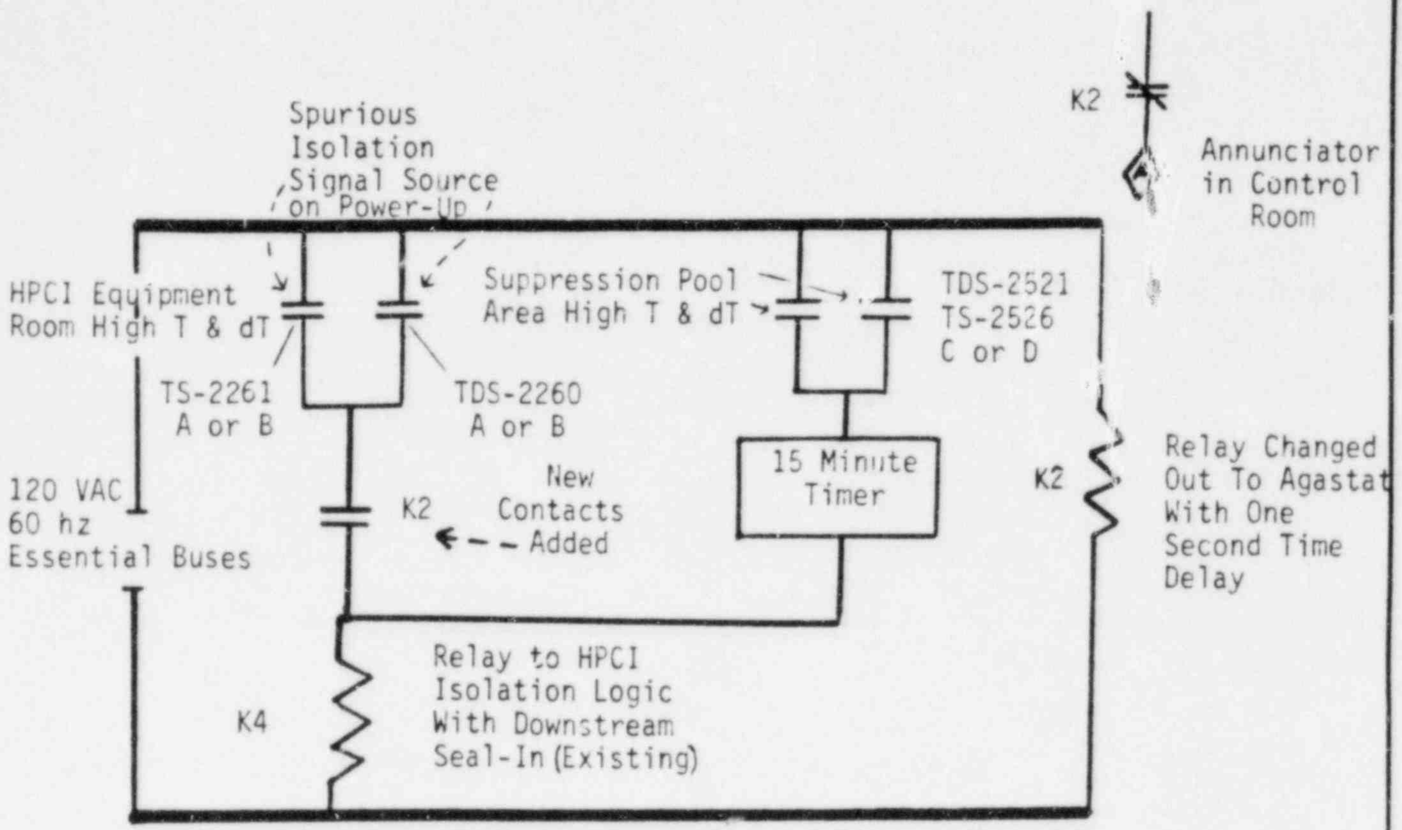

FIGURE 1: HPCI STEAM LEAK DETECTION SYSTEM - Following Design Change (A and B Sides are Identical)

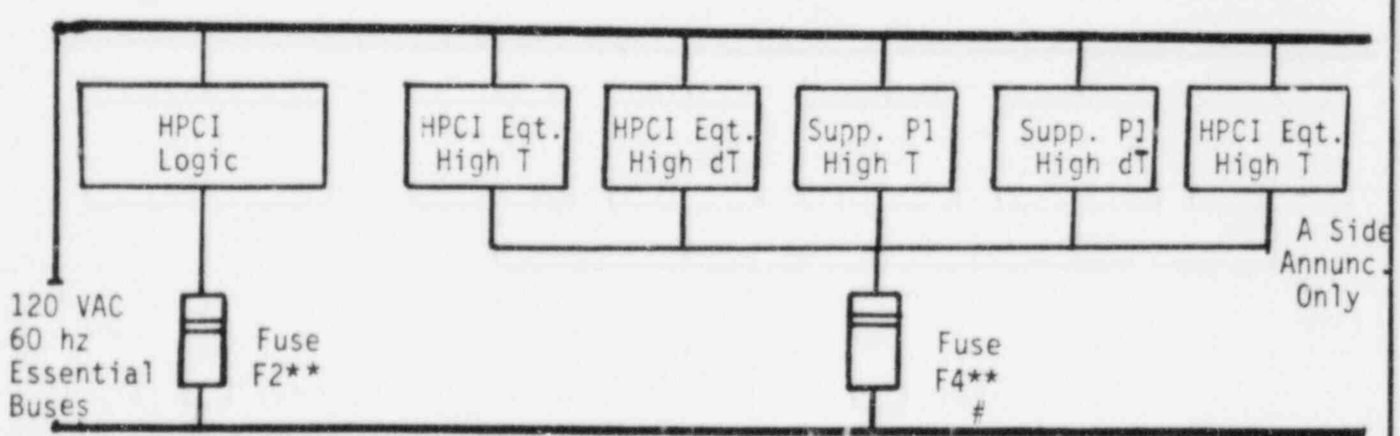

FIGURE 2: HPCI STEAM LEAK DETECTION SYSTEM - Power Supplies (No Design Change) A or B Sides.

#Pulled During First Test - 10/15/84
**Pulled During Second Test - 10/21/84

Iowa Electric Light and Power Company
November 16, 1984
DAEC-84-732

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. 84-028, Rev. 1

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
Duane Arnold Energy Center

DLM/JRP/kp

attachment

cc: Mr. James G. Keppler
Regional Administrator
Region III
U. S. Nuclear Regulatory Commission
799 Roosevelt Road
Glen Ellyn, IL 60137

NRC Resident Inspector - DAEC

File A-118a

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