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SUBJECT: LER 90-013-00:on 900910,RWCU isolation due to high area differential temp.W/901005 ltr.

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Iowa Electric Light and Power Company

October 5, 1990
DAEC-90-0848

Mr. A. Bert Davis
Regional Administrator
Region III
U. S. Nuclear Regulatory Commission
799 Roosevelt Road
Glen Ellyn, IL 60137

Subject: Duane Arnold Energy Center
Docket No: 50-331
Op. License DPR-49
Licensee Event Report #90-013

Gentlemen:

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

Very truly yours,

Gary Van Middlesworth for RLH

Rick L. Hannen
Plant Superintendent - Nuclear

RLH/LLS/sjo

cc: Director of Nuclear Reactor Regulation
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File A-118a

OCT 11 1990 IE22
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NRC Form 366A (6-89) U S NUCLEAR REGULATORY COMMISSION

LICENSEE EVENT REPORT (LER)

APPROVED OMB NO 3150-0104

EXPIRES 4/30/92

ESTIMATED BURDEN PER RESPONSE TO COMPLY WITH THIS INFORMATION COLLECTION REQUEST 50 0 HRS FORWARD COMMENTS REGARDING BURDEN ESTIMATE TO THE RECORDS AND REPORTS MANAGEMENT BRANCH (P-530) U S NUCLEAR REGULATORY COMMISSION, WASHINGTON DC 20555, AND TO THE PAPERWORK REDUCTION PROJECT (3150-0104) OFFICE OF MANAGEMENT AND BUDGET, WASHINGTON DC 20503

FACILITY NAME (1): Duane Arnold Energy Center

DOCKET NUMBER (2): 0 5 0 0 0 3 3 1

PAGE (3): 1 OF 0 4

TITLE (4): Reactor Water Cleanup Isolation Due to High Area Differential Temperature

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 9	1 0	9 0	9 0	0 1 3	0 0	1 0	0 2	9 0	None	0 5 0 0 0
										0 5 0 0 0

OPERATING MODE (9): N

POWER LEVEL (10): 0 0 3

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
Leonard L. Sueper, Technical Support Engineer	3 1 9 8 5 1 - 7 3 6 5

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	

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 September 10, 1990 at 0016 hours, during reactor startup, an automatic initiation of the Group V Primary Containment Isolation System isolation logic occurred. As a result, the outboard Reactor Water Cleanup suction and discharge valves closed as designed. The initiating event was a high differential area temperature as sensed by temperature switch TDS2743F in the Steam Leak Detection logic.

The intermediate cause of the system isolation was the failure to establish a new trip point for temperature differential switch TDS2743F after one of its inputs was relocated by a design change. The root cause of the event was inadequate procedural controls in the design change process. As a corrective action, an updated setpoint for TDS2743F has been established. In addition, the design engineering department has initiated a self-evaluation of overall department performance and design verification in particular that will result in additional verification criteria for design changes.

This event is being reported pursuant to 10CFR50.73 (a)(2)(iv) as an Engineered Safety Feature actuation.

I. DESCRIPTION OF EVENT:

On September 10, 1990 at 0016 hours, during reactor startup, an automatic initiation of the Group V Primary Containment Isolation System (PCIS, EIIS System Code JM) isolation logic occurred. As a result, the outboard Reactor Water Cleanup (RWCU, EIIS System Code CE) suction and discharge valves closed as designed. The initiating event was an area high differential temperature as sensed by temperature switch TDS2743F (CE-TDS) in the Steam Leak Detection (SLD, EIIS System Code IJ) logic.

SYSTEM DESCRIPTION

The Reactor Water Cleanup system is used to remove impurities from the reactor coolant which would otherwise plate out on heat transfer surfaces or become activated and contribute to general area radiation levels. The RWCU system is also used to drain excess water from the reactor during certain operational modes.

The Steam Leak Detection system is designed to detect steam and water leakage both inside and outside of primary containment and, in certain instances, isolates the leaking system. The portion of the SLD system relating to the Reactor Water Cleanup system detects leaks by measuring discrepancies between the measured flows entering and exiting RWCU, equipment area ambient temperatures, and the differential temperature between equipment area ventilation inlet and exhaust. There are six (6) differential temperature switches, any of which may initiate an RWCU (Group V) isolation upon detecting a condition which is indicative of an equipment area high differential temperature.

Plant Technical Specification Table 3.2-A requires that the setpoint for the Reactor Water Cleanup Area High Differential Temperature isolation signal be set at 14 degrees Farenheit above the 100% operation ambient differential temperature conditions as determined by plant test procedures. Plant Technical Specifications require at least one of the three instruments in each trip system be operable.

II. CAUSE OF EVENT

The intermediate cause of the system isolation was the failure to establish a new area differential temperature trip point for temperature switch TDS-2743F after one of its inputs, temperature element TE2743F, was relocated to make it more effective in detecting steam leaks in the RWCU Heat Exchanger Room.

TE2743F was relocated closer to the area ventilation inlet by a design change completed during the recent refueling outage. This was done to increase the effectiveness of the temperature element to detect steam leaks. Together with its companion temperature element which senses the RWCU Heat Exchanger Room outlet temperature, TE2743F inputs into temperature differential switch TDS2743F to determine the area differential temperature. TE2743F had originally been located near a ventilation duct that was intended to supply inlet air to an area within the RWCU Heat Exchanger Room. However, the actual ventilation flows through the area bypassed this duct such that TE2743F did not sense the true area inlet temperature. For the recent design change, the design engineer and the design verifying engineer failed to fully evaluate the effect of relocating TE2743F on the response characteristics of TDS2743F beyond the fact that relocating the temperature element would improve the ability of TDS2743F to detect a steam leak.

During plant startup, as the RWCU room warmed up, the temperature of TE2743F rose at a lower rate than it would have at its former location because of its closer proximity to the colder air inlet. As a result, the differential temperature between it and the ventilation outlet temperature increased above the former full power differential temperature that had been established originally for TDS2743F. Eventually the differential temperature exceeded 14 degrees above the former full power area differential temperature and the Group V isolation occurred as designed.

The Group V isolation occurred because the new full power area differential temperature was not determined for TDS2743F and the Technical Specification trip setpoint was not adjusted accordingly.

The root cause of the event was the lack of adequate technical review guidelines to support the design change verification. For the modification in question, the design engineer believed that the technical specification setpoint for the RWCU area high differential temperature trip was determined analytically and that the setpoint would only be exceeded during an actual steam leak. The engineer lacked the guidance that would have lead him to discover the existence of temperature gradients within the Reactor Water Cleanup room during power operation and to evaluate the operation of the design change in such an environment. As such, the design and verifying engineers had only considered the response of the temperature element during accident conditions for the purposes of performing the safety evaluation.

III. ANALYSIS OF EVENT

This event had no effect on the safe operation of the plant. The effective setpoint for TDS2743F following the design change was very conservative relative to the required setpoint and as such would only have made RWCU isolation more timely. As a result, the isolation occurred at low reactor power with no significant steam leaks present. Reactor Water Cleanup is not an ECCS system and has no safety-related function other than isolation. The isolation would not have prevented an ECCS system from providing adequate core cooling in the event of an accident.

IV. CORRECTIVE ACTIONS

A Special Test was initiated to determine the proper settings for TDS2743F. The information from the special test was used as the bases for a new trip setpoint for TDS2743F. All applicable surveillance tests and procedures have been updated accordingly.

Prior to this event the design engineering department had already initiated a self-evaluation of overall department performance and design verification effectiveness in particular. As a result of the self-evaluation, the engineering staff is being retrained on design change verification. In addition to other enhancements that arise from the self-assessment process, the use of the guidelines contained in INPO Good Practice TS-415, 'Technical Reviews of Design Changes', will be integrated into the design review method of the design verification process along with supplemental operational guidelines as recommended by the operations department. These changes to the procedures shall be implemented by January 31, 1991.

V. ADDITIONAL INFORMATION

A review of the LER database did not indicate any similar ESF actuation involving an inadequate review of a design change.