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SUBJECT: Responds to violations noted in Insp Rept 50-331/88-17.

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Iowa Electric Light and Power Company

October 26, 1988
NG-88-3681

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 No: DPR-49
Response to Notice of Violation Transmitted with Inspection Report 88-017

File: A-102, A-103

Dear Mr. Davis:

This letter and attachment are provided in response to the subject Notice of Violation concerning certain activities at the Duane Arnold Energy Center.

If you have any questions regarding this response, please feel free to contact our office.

Very truly yours,

William C. Rothert

William C. Rothert
Manager, Nuclear Division

WCR/JSA/go

Attachment: Response to Notice of Violation Transmitted with Inspection Report 88-017

cc: U. S. NRC Document Control Desk (Original)
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IE01

NRC NOTICE OF VIOLATION ITEM 1 (SEVERITY LEVEL IV)

The Code of Federal Regulation, Title 10, Part 50, Appendix B, Section III, Design Control states, in part, that measures shall be established to assure that applicable regulatory requirements and the design basis, as specified in the license application, for those structures, systems, and components to which this appendix applies are correctly translated into specifications, drawings, procedures, and instructions. The design control measures shall provide for verifying or checking the adequacy of design.

Duane Arnold's Quality Assurance Manual, "Design Control," Chapter 3, Revision 5, dated April 8, 1988, states, in part, that the design process shall be controlled through the use of procedures to assure the applicable regulatory requirements, design bases, codes, standards, drawings, procedures or instructions. Section 3.9, "Design Verification," states, in part, that design verification is the process of reviewing, confirming, or substantiating the design by one or more methods to provide assurance that the completed design meets the design intent and when changes to previously verified designs have been made, design verification is required for the changes, including an evaluation of the effects of those changes on the overall design.

Contrary to the above, the licensee failed to adequately verify design changes associated with DCP-1261 Safety Parameter Display System Electrical Installation, dated February 02, 1984.

a. The computer point B023 (reactor temperature) was wired to the Reactor Water Cleanup Temperature selector switch output, TSS-2713, and not a specific parameter input (reactor temperature) as required. This resulted in computer point B023 defaulting to the value selected on the temperature selector switch in the control room.

b. The computer contact point B025 (reactor pressure) was installed incorrectly with an additional 6 feet of wiring which resulted in an increased resistance giving an erroneous reactor pressure input to the process computer.

RESPONSE TO NOTICE OF VIOLATION ITEM 1

In early July 1988, it was discovered that the Reactor Water Cleanup (RWCU) inlet temperature input (computer point B023) to the core thermal power (P1) calculation performed by the Plant Process Computer was directly affected by a change in position of the RWCU temperature selector switch. Further investigation determined that the computer point had been incorrectly wired to the output side of the RWCU temperature selector switch due to a design error in the Safety Parameter Display System (SPDS) installation package (Design Change Package (DCP) 1261). The error did not affect the P1 calculation until the new process computer, which receives its B023 computer point input from the SPDS data acquisition system, became operational in February, 1988.

The effects of this error had been noted in February, 1988 but when Computer Services personnel investigated the problem the selector switch had been set back to the correct position and thus no problem could be identified. The link between selector switch position and the B023 computer point indication was first detected in July 1988. It was determined that the maximum discrepancy in calculation of power because of this error was less than 0.2%. It was verified that this discrepancy did not cause thermal limits to be exceeded.

During a post event review of the July 21, 1988 scram, it was noted that reactor pressure, computer point B025, indicated approximately 20 psi higher than other reactor pressure computer points. Further investigation revealed that the computer point inputs had not been connected directly to each side of the sensing resistor but instead were connected to points approximately 6 feet upstream and downstream of it. In most electronic circuits in use at the DAEC the addition of a 6 foot piece of wire between instruments would have no effect on circuit performance. In the case of the computer point sensing circuitry however, the sensing resistor is only 3.2 ohms, thus the small resistance added by the wire is enough to cause an error of approximately 1.5% at the computer point output.

1. Corrective Actions Taken and the Results Achieved:

   The immediate corrective action for computer point B023 was to tag the selector switch, thereby informing the operators of the problem. As long as the selector switch is in the RWCU inlet temperature position when the P1 calculation is performed the error in circuit design has no effect on the calculation. To verify that similar design errors did not exist with other P1 inputs associated with the installation of DCP-1261, a review of these computer points was performed. No similar design errors were found.

   Corrective actions to eliminate the discrepancy with computer point B025 were to measure the added resistance due to the additional wire on each side of the sensing resistor and then revise the millivolt input range in the process computer. Following that process computer update the computer point was verified to be functioning satisfactorily.

   By letter dated July 19, 1988, the Manager, Design Engineering issued supplemental guidance to supervising engineers that emphasized the importance of comprehensive design verification activities.

2. Corrective Actions Which Will Be Taken to Prevent Recurrence:

   The computer point B023 will be rewired, to permanently correct the error, during the current refuel outage. In order to strengthen the design verification process on design packages generated by organizations external to Iowa Electric Nuclear Generation Division, second level reviews will be performed on those design packages interfacing with the process computer.

Iowa Electric engineering administrative procedures require that design engineers identify engineering acceptance criteria that must be met during post modification testing. Additional guidance will be provided to engineering personnel to ensure comprehensive specification of design change engineering acceptance criteria within design change packages that are safety-related or safety significant.

Further, Iowa Electric will develop electrical loop diagrams in future design packages that involve safety-related or safety significant instrumentation interfacing with the plant process computer. These corrective actions to prevent recurrence will be in place by January 13, 1989.

Field walkdowns to detect wiring problems similar to that associated with the 3.2 ohm sensing resistor for computer point B025 will be performed for all critical P1 computer inputs. This walkdown, and disposition of discrepancies, will be completed prior to startup from the current refueling outage.

3. Date When Full Compliance Will Be Achieved:

Full compliance will be achieved by December 2, 1988 (prior to startup from the present refuel outage) with the correction of the design error associated with computer point B023.

NRC NOTICE OF VIOLATION ITEM 2 (SEVERITY LEVEL IV)

The Code of Federal Regulations, Title 10, Part 50, Appendix B, Section II, Instructions, Procedures, and Drawings states, "Activities affecting quality shall be prescribed by documented instructions, procedures, or drawings, of a type appropriate to the circumstances and shall be accomplished in accordance with these instructions, procedures, or drawings."

Duane Arnold Energy Center Administrative Control Procedure 1411.5, Use of Temporary Shielding on Safety Related Piping Systems, Revision 0, dated April 3, 1984, states in Paragraph 4.2, "The Design Engineering Group has the responsibility to evaluate the impact of temporary shielding on safety related systems," and Paragraph 4.4, "The ALARA Coordinator has the responsibility to assure that the installation of temporary shielding on safety related systems has been analyzed by the Design Engineering Group."

Contrary to the above, the inspector identified a masonry shield wall in the reactor building surrounding the northeast drywell equipment hatch on August 9, 1988, which was not evaluated by Design Engineering for its impact on safety related equipment, specifically seismic evaluation, nor was it reviewed by the ALARA Coordinator to assure that the shield wall had been analyzed prior to installation.

RESPONSE TO NOTICE OF VIOLATION ITEM 2

1. Corrective Actions Taken and the Results Achieved:

Immediate corrective actions were to determine if the shielding had been seismically qualified. A review of plant records did not indicate that this had been done. Following this determination, seismic calculations were performed but the shielding, by itself, could not be qualified. Therefore three steel I-beams were installed horizontally in front of the shielding to ensure that a seismic event would not cause the concrete blocks to fall on the nearby instrument rack. In addition, a review of similar radiation shielding in the plant was performed to verify that the appropriate evaluations had been performed and documented. No problems were noted. This event was reported for information in LER 88-012.

2. Corrective Actions Which Will Be Taken to Prevent Recurrence:

Plant documentation indicated that the request for concrete block shielding was reviewed by the ALARA Coordinator, but the review was completely inadequate in providing an engineering evaluation. The corrective actions stated below are intended to clarify that the ALARA Coordinator must assure an engineering evaluation is performed when required.

Long term corrective actions, to ensure that both temporary and permanent radiation shielding is properly qualified and installed, will involve revision of the current procedure which governs the use of radiation shielding. Specific changes to this procedure will include:

A method to ensure that radiation shielding qualification is periodically reviewed to verify that qualification of shielding has not been changed by a change in plant conditions/systems.

Specific guidelines for determining whether a seismic analysis of radiation shielding is required.

Clarification of the difference between temporary and permanent radiation shielding and the method for processing each type.

Clarification of the applicability of this procedure to various types of radiation shielding.

The revision of this procedure will be completed by December 31, 1988.

3. Date When Full Compliance Will Be Achieved:

Iowa Electric achieved full compliance when the I-beam retaining structure was installed in front of the shielding on August 17, 1988.

NRC NOTICE OF VIOLATION ITEM 3 (SEVERITY LEVEL IV)

Technical Specifications 3.13.F.1 and 4 states, in part, that all fire barrier penetration seals protecting safety-related areas shall be intact. If Specification 3.13.F.1 cannot be met then a continuous fire watch shall be established within one hour on at least one side of the affected area, or verify the operability of fire detectors on at least one side of the non-functional fire barrier and establish an hourly fire watch patrol.

Contrary to the above, with a fire barrier penetration seal in the northwest corner room not intact on August 3, 1988 at 10:00 a.m., the licensee failed to establish a fire watch patrol until August 4, 1988, at 9:30 a.m., approximately 24 hours after the fire watch patrol was required.

RESPONSE TO NOTICE OF VIOLATION ITEM 3:

1. Corrective Actions Taken and The Results Achieved:

   Upon discovery that the required firewatch had not been established, the Control Room was notified and the appropriate fire watch was established This event was reported in LER 88-009.

2. Corrective Actions Which Will Be Taken to Prevent Recurrence:

   Corrective action to ensure that firewatches are established when required consists of improved guidance through the use of an Administrative Control Procedure (ACP). This procedure will require the person who identifies the need for a firewatch to notify those responsible for performing the firewatch. (At the time of the missed firewatch it was appropriate for either Operations or the person requesting the firewatch to notify firewatch personnel). By requiring the person who identifies the need for a firewatch to contact the firewatch personnel directly, the possiblity of failing to initiate a firewatch due to a miscommunication is minimized. The ACP will be in effect by December 15, 1988. In the interim, administrative guidance to implement the concepts of the ACP has been issued.

3. Date When Full Compliance Will Be Achieved:

   Iowa Electric was in full compliance on August 4, 1988 when the required firewatch was established.