DMB

Iowa Electric Light and Power Company

September 15, 1986
NG-86-2485

Mr. James G. Keppler
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
Response to NRC Inspection Reports 86-06 and 86-10

Dear Mr. Keppler:

This letter is provided in response to the subject reports concerning inspections of activities at the Duane Arnold Energy Center. Attachment 1 provides our response in accordance with your request. Because of the related nature of these two inspections, an extension was requested of Mr. Duane Boyd of your organization for responding to Inspection Report 86-06.

Very truly yours,

R. W. McGaughy

Richard W. McGaughy
Manager, Nuclear Division

RWM/JCS/pl

Attachment: Response to IR 86-06 and IR 86-10

cc: L. Liu
L. Root
M. Thadani
NRC Resident Inspector
Commitment Control 860245, 860300
File A-102

SEP 18 1986

IE01

Response to IR 86-06

NRC Item of Violation (Severity Level IV)

10 CFR Part 50, Appendix B, Section VIII, requires in part that measures be established for the identification and control of materials, parts, and components, and that these measures be designed to prevent the use of incorrect or defective materials, parts and components.

Contrary to the above, the licensee's measures did not prevent use of a defective static inverter in the Reactor Core Isolation Cooling (RCIC) system on August 2, 1985. The inverter's factory set high input voltage trip setpoint was lower than required, and, on March 11, 1986, the inverter tripped on high input voltage, rendering the RCIC system inoperable.

Response to Item of Violation

1. Corrective Action Taken and the Results Achieved

Our investigation into this matter revealed that this inverter model is utilized in several applications at DAEC. We have found these devices to be very reliable. The inverters have internal voltage trip adjustments that allow adjustment for low and high supply voltage. The spare that was installed in August 1985 was procured early in the plant lifetime and tested prior to shipment by General Electric on June 16, 1971. Test data furnished with the inverter document that the high voltage setting was tested and found acceptable at 142 volts at that time. In 1974, the installed inverters were reset to raise the internal high voltage setting. The 1974 Maintenance Action Requests (MARs) indicate that the HPCI and RCIC inverters tripped during the equalizing process. Resetting was apparently necessary because the settings did not match the factory specification for high voltage trip. Vendor maintenance instructions specify an as-left setting of 147 volts +/- .5 volts. The spare apparently was not checked or adjusted at that time.

The inverter (formerly the spare) was verified to be operating properly on the normal supply voltage of approximately 130 volts following installation in August, 1985. In March 1986, following months of reliable operation, the unit tripped at 138 volts during an equalizing charge (which is performed above normal operating supply voltage levels).

The failure caused a Control Room annunciator to alarm. The RCIC system was immediately declared inoperable as reported in LER 86-05 for the period of approximately 4 hours that was required to troubleshoot, replace and test the inverter. Technical Specification action statements were complied with during this period.

The apparent drift of this inverter setting was not anticipated. This failure would not have occurred if the inverter had tripped at 142 volts as indicated by the test data furnished with the inverter. The manufacturer's service manual states that the setting is factory set and should not require adjustment. We had experienced no previous problems with drift of the high voltage setting. We therefore did not anticipate the need to provide adequate procedural guidance for preparing the inverter prior to returning it to service in August, 1985.

The high voltage trip setting was verified to be correct when the inverter was replaced in March, 1986. Current maintenance procedures require completion of an inspection procedure when these inverters are replaced. That inspection procedure includes checking of these settings. The inspection procedure was not yet developed at the time of the August 1985 installation. Inspection was not scheduled prior to the next refueling outage because the inverter is inoperable during the inspection.

We control activities concerning installed and spare components in two ways. If a component is modified, this is accomplished through the design change process. If a component deficiency is discovered, corrective actions are performed through the corrective maintenance process.

Design modifications which change installed components and required spare parts are controlled so that spare parts must be procured as part of the design change package before components and systems are placed in service. The package must also revise component drawings to reflect the change. These component drawings and spare parts lists are then utilized during future corrective maintenance.

The administrative controls by which we plan and initiate maintenance work have been strengthened since the installation of this inverter in August 1985. Improvements include:

1) More thorough review of post maintenance testing requirements. This testing is designed to assure that all known parameters essential to component function are operable. The number of people reviewing the testing requirements has also been increased.

2) Maintenance planners are now required to review the computerized maintenance history data. This review consists of identifying past maintenance and problems with specific components and systems. This process is utilized to identify necessary maintenance and post testing actions.

3) Preventive Maintenance (PM) procedures are considered when specifying corrective maintenance and post maintenance testing steps. These PM procedures are continually reviewed and revised to reflect vendor recommendations and operating experience.

4) The computerized maintenance system automatically incorporates PM procedures into future corrective maintenance work once these PM procedures are identified.

A review of design change and maintenance procedures to provide additional controls over warehouse spares will be performed. This review will concentrate on establishing mechanisms which ensure modifications to installed components are also performed on warehouse spares when necessary. This review will be completed by October 31, 1986.

2. Corrective Actions to be Taken

Full compliance has been achieved. In addition, we are continuing to improve the CHAMPS data base and maintenance history records for more accurate Corrective Maintenance planning. Additional procedure review which will be completed by October 31, 1986 will help ensure that future similar noncompliances do not occur.

3. Date When Full Compliance will be Achieved

Full compliance was achieved in March 1986. Procedures in effect at that time ensured that a defective unit would not be installed, as discussed above.

Response to Open Item

In the subject Inspection Report, the NRC noted apparent weakness in our consideration of seismic qualification related to design change package implementation. Secondly, NRC concern was expressed regarding the adequacy of the written safety evaluation in support of the temporary use of a non-certified relay in a portion of the RCIC logic where a certified relay was not available due to procurement lead times. We have initiated action to strengthen the documentation in both of these areas as discussed below.

Engineering evaluations of seismic considerations for the installation of electrical equipment have shown a weakness in both adequacy and documentation. As a corrective action for this area, we have administratively assured that an engineering review, performed by an engineer knowledgeable in seismic design, will be completed and documented for electrical modifications that involve seismic category I racks, panels etc.

Improvements have also been implemented in the procedures and practices utilized in preparing safety evaluations. See the attached Inspection Report 86-10 and our response for details.

Response to IR 86-10

NRC Item of Violation (Severity Level IV)

10 CFR 50.59(b) requires that the licensee prepare a written safety evaluation for any change to the facility as described in the FSAR, any change to procedures described in the FSAR, or tests or experiments not described in the FSAR. It further requires the safety evaluation to document the bases for the determination that the change, test, or experiment does not involve an unreviewed safety question.

Contrary to the above, the safety evaluations for the following changes did not provide the bases for concluding that no unreviewed safety question existed:

A. Design Change (DC) 1161
B. DC 908
C. DC 1222
D. DC 1008
E. DC 1276
F. DC 1177
G. DC 1095
H. DC 1057
I. Procedure Changes OI-64, Revision 3; OI-24, Revision 5; OI-16.0, Revision 5; and OI-49, Revision 7.

Licensee Response to Items A. through H.

As was acknowledged in the NRC letter transmitting Inspection Report 86-10, Iowa Electric has made recent efforts to upgrade the process of safety evaluations at the Duane Arnold Energy Center. The administrative control procedure now in use for the preparation of Safety Evaluations has been revised and strengthened since the evaluations identified in the Inspection Report were prepared. We are continuing in our efforts to improve development and review of Safety Evaluations, and increasing emphasis on this subject has been provided to the engineering staff, including formalized training.

The separate but related program implemented to improve the specific content and scope of safety evaluations provides personnel with a new approach using the General Electric developed "Nuclear Safety Operational Analysis" (NSOA) and better instructions in the development of the safety evaluations. A new procedure has been issued to implement the NSOA methodology. A safety evaluation handbook is currently near completion. These improvements will provide the necessary information to produce better safety evaluations. The program will be fully implemented by December 1986.

Appendix 1 to this Attachment provides a detailed response to Items A. through H.

Licensee Response to Item I.

The Safety Evaluations for four procedure changes were reviewed (see c. (14) of IR 86-10). Procedure changes are done via a Document Change Form (DCF). Presently, procedure change Safety Evaluations consist of a checklist format specifying the technical specification and the UFSAR sections reviewed. This was considered a strength. Yes/no answers are required to the three questions used in identifying an unreviewed safety question and whether or not a technical specification change was required.

The Safety Evaluation for Procedure OI-64 (single loop operation) was cited for noting that related technical specification and UFSAR sections were not applicable even though single loop operation is addressed in both. None of the four Safety Evaluations examined stated the basis for the conclusion that no unreviewed safety question existed.

Licensee Response and Corrective Actions:

It was determined the UFSAR and technical specifications sections for single loop operation had in fact been examined during the revision of OI-64. This examination found the revision not to have an impact on either, thus the examiner wrote "n/a" (not applicable). A more accurate statement indicating sections on this topic had been reviewed would have been appropriate. At the present time, DCF's containing "n/a" are not granted approval by the Operations Committee.

The "Revision of Procedures and Instructions" administrative control procedure, which controls DCFs, will be modified to provide clear guidance as to when 10 CFR 50.59 safety evaluations are required and provide for additional documentation when such an evaluation is necessary. The method for listing applicable UFSAR and Technical Specifications sections will also be clarified. This will be accomplished by December 2, 1986.

Licensee Response to Unresolved Items

Temporary Modifications. (See 2.c. (11) of IR 86-10) The inspector determined that a procedure did not exist to effect and control all temporary modifications. A Jumper and Lifted Lead Control Procedure was found to be available, and it was noted this procedure could be used with one DCP to perform a modification, and another DCP to remove that modification. In addition, administrative control procedures for conducting special test procedures and authorizing, installing, and documenting minor modifications were noted as being purportedly utilized as necessary when other temporary modifications were required.

This type of control mechanism was considered as possibly not adequately addressing the installation of all temporary modifications, and in addition allowing safety evaluations to be bypassed. This was considered an Unresolved Item.

Licensee Response:

The Design Change process is meant to address permanent modifications only, and is procedurally prevented from being used in conjunction with the Jumper and Lifted Lead Control procedure.

The current Jumper and Lifted Lead Control procedure requires performance of an Engineering Evaluation before any jumpers or lifted leads are placed, other than as part of a procedure approved by the Operations Committee. The Engineering Evaluation contains a description of the purpose and effect of the clearance, and a list of referenced technical specifications and UFSAR sections. The Evaluation requires yes/no answers to the three questions used in identifying an unreviewed safety question.

The Jumper and Lifted Lead Control procedure will be revised to ensure that a jumper or lifted lead cannot be used to change the required performance of a system as described in the UFSAR and technical specifications. In addition, this procedure will be modified to require a written basis for the answers to the three questions identifying unreviewed safety questions. This will be accomplished by December 9, 1986.

As required by the Special Test Procedures administrative control procedure, all Special Test Procedures must contain a written safety evaluation.

As required by the Minor Modifications procedure, a written safety evaluation is required for all Minor Modifications.

The Duane Arnold Energy Center considers its current procedural controls, in lieu of a single all-encompassing procedure, to be adequate to control all of the above items which may be considered temporary modifications. With the exception of providing a better basis for the Jumpers and Lifted Leads Safety Evaluation as previously detailed, the safety evaluation process for these procedures appears adequate, and does not allow bypassing of Safety Evaluations.

Licensee Response to Open Items

Items numbered as in Section 2.c. of IR 86-10.

(12) The Minor Modification Procedure was reviewed for technical and administrative adequacy. It was noted the procedure did not specify in sufficient detail the limitations and applicability of using minor modifications. The definition of a minor modification in the procedure was considered not limiting enough to prevent the utilization of the procedure to perform a modification which would affect a structure, system, or component as described in the UFSAR. The inspectors noted that we indicated that we shared this concern and would address it in a future revision to the procedure. This was considered an open item.

Licensee Response and Corrective Actions:

The Manager of Design Engineering currently reviews and approves, before implementation, all uses of the Minor Modification Procedure to ensure it is not being misapplied. The Minor Modification Procedure will be revised to clarify the scope of this procedure. As presently defined within the Minor Modification Procedure, the Safety Evaluation required in all cases must address system interfaces. If a safety system is affected, the planned modifications must be revised to eliminate this effect or the modification cannot be done as a minor modification. Additional guidance will be provided within the procedure to ensure these controls are followed. These revisions will be completed by January 31, 1987.

(13) The Control of Design Document Changes (DDC) procedure was examined and appeared to offer a mechanism for effecting a design change without using the design change process. Of particular concern was the possibility of documenting a field discrepancy with design drawings without determining if the "as-built" configuration was the configuration used in the design analysis and safety evaluation. This was considered an open item.

Licensee Response and Corrective Actions:

Presently, discrepancies found on walkdowns for the Computerized History and Maintenance Planning System (CHAMPS) and Electrical Distribution Information System (EDIS) projects are being evaluated to determine if the "as-built" configuration is the same as that considered in the design analysis and safety evaluation, and to make provisions for corrective action where necessary. The Control of Design Document Changes procedure will be revised and strengthened by January 31, 1987 to ensure an evaluation is done on each DDC submitted for determination of its effect on the design analysis. These procedure revisions will be utilized when discrepancies are discovered by any means, including walkdowns.

APPENDIX 1 to Attachment 1

Detailed Response to Items A. through H.

Items are also numbered as in section c. of IR 86-10.

ITEM A (paragraph 2.C.(1) of IR 86-10 details)

(1) Design Change Request 1161, "Temperature Elements Replacement"

This Design Change Request (DCR) was originally written to replace the 36 temperature elements (TEs) in the main steam line break (MSLB) system with environmentally qualified (EQ) elements. The DCR was later amended to include the replacement of accident monitoring TEs in the drywell, RHR heat exchanger outlet, and Torus with EQ TEs. As part of this modification, five of the 36 TEs in the (MSLB) system were relocated.

The Safety Evaluation (SE) for DCR 1161 was cited for failure to document the bases for concluding that no unreviewed safety question exists. The main deficiency in the SE as noted in IR 86-10 was the failure to document the acceptability of the relocation of the five MSLB TEs. The potential for changes in system response time due to the relocation, or the basis for the original TE locations were not addressed in the written safety evaluation. The Safety Evaluation also states the new TEs are electrically interchangeable with the original TEs without identifying which parameters were considered when making this determination.

Corrective Action:

The Safety Evaluation for DCR 1161 will be revised to address the concerns stated in IR 86-10. This revision will be completed by January 31, 1987.

ITEM B (paragraph 2.C.(2) of IR 86-10 details)

(2) DCR 908, Post Accident Sampling System

This modification provided a keylock override of the containment isolation signal to the reactor coolant sample lines, which could be enabled only during isolation conditions for the purpose of post accident sampling. The keylock switches were located in the control room with appropriate annunciation provided. The original design featured automatic timers which would limit the override condition to ten minutes. This feature was later removed from the design in favor of administrative controls on the time limit. The administrative control requirement consists of a procedural requirement that the chemist notify the operator when he wished to start sampling and when he had finished. The inspectors noted that no time limit on the sampling period is mentioned.

The Safety Evaluation for DCR 908 was cited for stating without basis that the consequences of an accident are not increased as a result of this modification. A calculation of the increased dose in the exclusion area due to sampling was removed from the original Safety Evaluation when the requirement for timers was dropped.

Corrective Action:

The revised (6/12/85) Safety Evaluation for DCR 908 references the safety evaluation which accompanied the removal of the timers, dated 6-5-85. This states that following review of NUREG 0578 and 10 CFR 50, Appendix A, it was determined that the timers were not required. It further states the evaluation of the incremental dose in the original evaluation was eliminated because the dose was based on the timers limiting the time the isolation signal was overridden. Since the time was to be limited by administrative control procedures rather than by timers, the calculation portion of the safety evaluation was no longer applicable. As stated in the Safety Evaluation, use of an approved procedure constituting administrative control of valve position was equivalent to the existing design on other containment piping penetrations where a manual override switch with a keylock feature is used. Obtaining a reactor coolant system sample will enhance plant safety as indicated in NUREG 0578 Section 2.1.8.a and does not constitute a plant condition that would increase either the probability of occurrence or the consequences of the accident causing the containment isolation. The reopening of the sample lines following isolation was found not to increase the probability of an accident or malfunction above that which existed at that time due to line size and pressure rating.

The safety evaluation for DCR 908 does not follow the presently acceptable format in that it does not specifically address the questions used to identify an unreviewed safety question. It does not adequately identify if the consequences of an accident are increased by this modification nor does it address the radiological consequences of the removal of the timers on personnel performing the sampling. The timers were never installed by this modification and were replaced by administrative procedures to control the sampling. The procedure operating this equipment requires that conservative dose calculations be performed to determine stay time before entering the area. The Safety Evaluation for DCR 908 will be revised to reflect these concerns and address the current system configuration and usage. This will be completed by January 31, 1987.

ITEM C (paragraph 2.C.(3) of IR 86-10 details)

(3) DCR 1222, Replace RHRSW Pump Bowl Assembly

This DCR replaced one RHR Service Water Pump and provided a method by which the remaining three pumps in this system might be replaced on an as-needed basis without an additional DCR. The existing motor and upper column pipe would be reused while the lower column section and bowl assembly would be replaced.

One deficiency was noted in the safety evaluation by the inspectors. The SE stated the original head and capacity requirements were met following installation of a stainless steel impeller and bowl. Plotting of the test data collected during performance of the post installation tests for Engineering Acceptance Requirements of the DCR revealed that the head and capacity requirements were not met for several flow rates and the values obtained varied from the vendor required pump head curve by as much as 7%.

This was considered a failure to accurately document the basis for the conclusion that no unreviewed safety question existed. This did not present an unreviewed Safety Question since a Technical Specification Amendment approved during this time allowed a 15% reduction in flow requirements.

Licensee Response and Corrective Actions

Safety Evaluations are required to be complete before a modification takes place. The replacement of parts for DCR 1222 which were like-for-like with the exception of material (stainless steel vs. bronze) was assumed to have no effect on pump performance. However, a revision to the SE or other engineering analysis should have been documented following the acceptance tests. In response to the concerns expressed in Inspection Report 86-10, the Safety Evaluation for DCR 1222 will be revised to provide such documentation by January 31, 1987.

ITEM D (paragraph 2.C.(4) of IR 86-10 details)

(4) Design Change Package 1008, "RCIC System PT 2502 Replacement"

This modification replaced the pressure transmitter monitoring the Reactor Core Isolation Cooling (RCIC) system pump suction pressure because of pressure transients in excess of the switch's rated design proof pressure when the turbine was tripped. The change replaced the pressure transmitter with a safety-related transmitter which had a higher pressure rating compatible with the pressure transients. The primary function of this pressure transmitter is to provide information regarding RCIC system operational conditions to the Control Room.

The Safety Evaluation for Design Change Package (DCP) 1008 was cited for not providing adequate justification to show that all environmental and seismic requirements had been met. In addition, it was noted the SE did not address transmitter reliability, availability or response time changes to the system as a result of this change.

Corrective Action:

The Safety Evaluation for DCP 1008 will be revised to address the concerns stated in IR 86-10 by January 31, 1987.

ITEM E (paragraph 2.C.(5) of IR 86-10 details)

(5) Design Change Package 1276, "Replace Non-Qualified Flow Transmitters"

This design change replaced the flow-rate transmitters of the primary accident monitoring instrumentation for the LPCI/RHR and Core Spray systems with Class 1E safety-related transmitters for environmental qualifications. It was stated in the SE that the flow-rate transmitters do not provide any automatic safety function, therefore failure of these transmitters will not result in a failure of an automatic system to perform its safety function.

The Safety Evaluation was cited for two deficiencies. It was stated that alternative verification of system flow for the Core Spray and LPCI/RHR systems is available in the control room, and that these system flows can indirectly be determined by reactor water level indication or system check valve position. However, no justification is made within the SE regarding the reliability or likelihood these additional instruments would be available in the event of an inoperable flow-rate transmitter. Secondly, this seismic analysis was considered insufficient to support the seismic qualification of the instrument racks following mounting of the new transmitters.

Corrective Action:

The Safety Evaluation for DCP 1276 will be revised to reflect these concerns by January 31, 1987.

ITEM F (paragraph 2.C.(6) of IR 86-10 details)

(6) DCP 1177, "Reactor Recirculation System Bonnet Vent Removal"

DCP 1177 was to help prevent excessive leakage to the drywell equipment drain sump through the bonnet vent connections from valves MO-4629 and MO-4630. The bonnet vent isolation valves had been leaking past their valve seats, resulting in increased leakage to the sump. The safety-related change performed was to remove the bonnet vent line and isolation valves, and to cap the bonnet vents and the equipment drain sump line where the bonnet vent line tied in. The function of the bonnet vent system was to prevent overpressurization of the bypass valve(s) if the valve(s) were closed and experienced an increasing temperature. As the valves are left open per a General Electric recommendation, the potential for bonnet overpressurization is eliminated. In addition to removing the vent valves, lines, and installing caps, DCP 1177 was also to remove the associated snubbers and pipe supports.

The Safety Evaluation for DCP 1177 was cited for not providing justification that all seismic requirements were satisfied, as only the effect of eliminating the bonnet vent line was addressed.

Corrective Action:

The Safety Evaluation for DCP 1177 will be revised to address this concern by January 31, 1987.

ITEM G (paragraph 2.C.(7) of IR 86-10 details)

(7) DCP 1095, "Main Steam leakage Control System Flowmeter Replacement"

This modification replaced the existing MSIV Leakage Control System flow instrumentation with instrumentation which was seismically and environmentally qualified, to meet the requirements of 10 CFR 50.49.

The text of the package stated that, "Based on the information available in our offices", there had been no previous additions to the panel and since the instrumentation to be installed was less than 5% of the total cabinet weight, the cabinet's seismic analysis was not affected. This was not addressed by the Safety Evaluation, which should have ensured the available information was correct and also documented in the SE. This was considered an inadequacy.

The Safety Evaluation stated replacement of the instruments with environmentally qualified instruments did not change the function of these instruments, and therefore the possibility of an accident or malfunction of a different type than was previously discussed in the FSAR was not created. This statement was, by itself, not considered adequate. Areas such as instrument accuracy, instrument reliability, and system response time were not addressed.

The Safety Evaluation stated that the safety functions of the system, passive flow limiting ability and high flow trip, were maintained. It was noted that a drawing referenced when the inspectors questioned the passive flow function stated the three inch maximum pressure drop through the system's original flowmeter was specified to insure the flowmeter remained calibrated over all expected flow conditions, not for a passive flow limiting safety function as described in the SE.

Corrective Action:

The Safety Evaluation for DCP 1095 will be revised to address these concerns by January 31, 1987.

ITEM H (paragraph 2.C.(9) of IR 86-10 details)

(9) DCR 1057, "Offgas System Valves MO-4151 and CV-4151 Replacement"

Gate valve MO-4151 was used to control the pressure in the Offgas System upstream of a jet compressor. As this valve was not adequately designed to control pressure, problems with pressure control, valve drift and seat wear were observed. DCR 1057 was to replace MO-4151 on a like basis and install a globe valve downstream for pressure control. The Safety Evaluation concluded the modification did not constitute an unreviewed safety question.

The Safety Evaluation for DCR 1057 was cited for stating the changes did not affect any safety-related equipment. The inspectors also noted the safety evaluation stated the safety evaluation in the UFSAR was not changed without providing a basis for this statement. Wiring and piping changes required for this modification were not addressed in the SE. Also not addressed were interfaces with safety systems or the effect of modified equipment failure, if any, upon any safety-related components. Although the SE stated the probability of any accident was reduced due to a more stable Offgas system, it did not address the likelihood of failure of the new equipment or failures due to the new equipment/wiring interfaces. The SE also states the margin of safety as defined by the technical specification basis was not reduced because the portion of the Offgas System affected was not addressed in the technical specifications. The possibility had not been evaluated that this system failing, due to the new components, could affect a system important to safety.

Corrective Action:

The Safety Evaluation for DCR 1057 will be revised to address these concerns by January 31, 1987.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION III
799 ROOSEVELT ROAD
GLEN ELLYN, ILLINOIS 60137

JUN 19 1986

Docket No. 50-331

Iowa Electric Light and Power
Company
ATTN: Mr. Lee Liu
President and Chief
Executive Officer
IE Towers
P. O. Box 351
Cedar Rapids, IA 52406

Gentlemen:

This refers to the routine safety inspection conducted by Mr. J. S. Wiebe and Ms. N. V. Gilles of this office on March 18 through May 19, 1986, of activities at the Duane Arnold Energy Center (DAEC) authorized by NRC Operating License No. DPR-49 and to the discussion of our findings with Mr. D. Mineck and others of his staff at the conclusion of the inspection.

The enclosed copy of our inspection report identifies areas examined during this inspection. Within these areas, the inspection consisted of a selective examination of procedures and representative records, observations, and interviews with personnel.

During this inspection, certain of your activities appeared to be in violation of NRC requirements, as specified in the enclosed Notice. A written response is required for Violation No. 1. No response is required for Violation No. 2.

Certain other activities, set forth in Paragraph 8 of this inspection report, appear to indicate weaknesses in your engineering evaluations and documentation associated with modifications to the plant. While no specific violations have yet been identified, the NRC is concerned that these weaknesses could lead to inadequate consideration of the effect of modifications on safety related systems, with a resulting degradation of a safety-related system. Please advise us in writing, within 30 days of the date of this letter, of the action you have taken or plan to take, to correct these weaknesses.

In accordance with 10 CFR 2.790 of the Commission's regulations, a copy of this letter, the enclosures, and your response to this letter will be placed in the NRC Public Document Room.

The responses directed by this letter and the accompanying Notice are not subject to the clearance procedures of the Office of Management and Budget as required by the Paperwork Reduction Act of 1980, PL 96-511.

~~8606240059~~

We will gladly discuss any questions you have concerning this inspection.

Sincerely,

Charles E. Norelius

Charles E. Norelius, Director
Division of Reactor Projects

Enclosures:
1. Notice of Violation
2. Inspection Report
No. 50-331/86006(DRP)

cc w/enclosures:
D. Mineck, Plant Superintendent
Nuclear
W. Miller, Assistant Plant
Superintendent Technical
Support
DCS/RSB (RIDS)
Licensing Fee Management Branch
Resident Inspector, RIII
Thomas Houvenagle, Iowa State
Commerce Commission

NOTICE OF VIOLATION

Duane Arnold Energy Center | Docket No. 50-331

As a result of the inspection conducted on March 18 through May 19, 1986, and in accordance with the "General Policy and Procedure for NRC Enforcement Actions," 10 CFR Part 2, Appendix C (1985), the following violations were identified:

1. 10 CFR Part 50, Appendix B, Section VIII, requires in part that measures be established for the identification and control of materials, parts, and components, and that these measures be designed to prevent the use of incorrect or defective materials, parts, and components.

   Contrary to the above, the licensee's measures did not prevent use of a defective static inverter in the Reactor Core Isolation Cooling (RCIC) system on August 2, 1985. The inverter's factory set high input voltage trip setpoint was lower than required, and, on March 11, 1986, the inverter tripped on high input voltage, rendering the RCIC system inoperable.

   This is a Severity Level IV violation (Supplement I).

2. 10 CFR Part 50, Appendix B, Section XVI, requires in part that measures be established to assure that conditions adverse to quality . . . are promptly identified and corrected . . . and that the measures taken shall assure that the cause of the condition is determined and corrective actions taken to preclude repetition.

   The Iowa Electric Light and Power Company, Quality Assurance Manual, Sections 14.3.2 and 14.3.3 identify Licensee Event Reports (LERs) as a system which has mechanisms for identifying and correcting conditions adverse to quality.

   Contrary to the above, corrective action taken in response to LER 85-012 which occurred on April 25, 1985, did not correct the condition adverse to quality identified in that report, namely, failure to identify proper power supply circuitry for reactor building radiation monitors. In the subject LER, this condition led to a Group III isolation and Standby Gas Treatment System initiation. Corrective action taken was not adequate to prevent recurrence of this condition, and on March 29, 1986, similar circumstances led to a one half Group III isolation of the "A" Standby Gas Treatment System.

   This is a Severity Level IV violation (Supplement I).

Pursuant to the provisions of 10 CFR 2.201, you are required to submit to this office within 30 days of the date of this Notice a written statement or explanation in reply, including for Violation No. 1: (1) corrective action taken and the results achieved; (2) corrective action to be taken to avoid further violations; and (3) the date when full compliance will be achieved. No written response is required for Violation No. 2. Consideration may be given to extending your response time for good cause shown.

JUN 19 1986

Date

Charles E. Norelius

Charles E. Norelius, Director
Division of Reactor Projects

U.S. NUCLEAR REGULATORY COMMISSION

REGION III

Report No. 50-331/86006(DRP)

Docket No. 50-331 License No. DPR-49

Licensee: Iowa Electric Light and Power Company
IE Towers, P. O. Box 351
Cedar Rapids, IA 52406

Facility Name: Duane Arnold Energy Center

Inspection At: Palo, IA

Inspection Conducted: March 18 through May 19, 1986

Inspectors: J. S. Wiebe
N. V. Gilles

Approved By: D. C. Boyd, Chief
Reactor Projects Section 2D

6-19-86
Date

Inspection Summary

Inspection on March 18 through May 19, 1986 (Report No. 50-331/86006(DRP))
Areas Inspected: Routine unannounced inspection by the resident inspectors of licensee action on previous inspection findings, operational safety, maintenance, surveillance, Licensee Event Reports, outage activities, and modifications and facility changes.
Results: Of the seven areas inspected, two violations were identified in one area (Paragraph 6 - failure to perform post maintenance testing on the Reactor Core Isolation Cooling System static inverter - and failure to take adequate corrective action to preclude repetition).

~~8606240471~~

DETAILS

1. Persons Contacted

   R. Hannen, Assistant Plant Superintendent, Operations
   *B. Lacy, Maintenance Superintendent (Acting)
   *R. Lessly, Design Engineering Manager
   *E. Matthews, Quality Assurance Manager
   *C. Mick, Operations Supervisor
   *W. Miller, Assistant Plant Superintendent, Technical Support
   *D. Mineck, Plant Superintendent, Nuclear
   *J. Probst, Technical Support Engineer
   *J. Smith, Technical Support Supervisor
   *K. Young, Assistant Plant Superintendent, Radiation Protection/Security

   In addition, the inspectors interviewed several other licensee personnel including Operations Shift Supervisors, Control Room Operators, engineering personnel, and contractor personnel (representing the licensee).

   *Denotes those personnel present at the exit interviews.

2. Licensee Action on Previous Inspection Findings

   a. (Closed) Open Item (331/85015-01(DRP)): Personnel Errors. A scheduled outage was conducted during the last half of the month of March. During the outage and during the outage recovery effort, no significant personnel errors occurred even though significant numbers of contractors were onsite to support the outage. Based on the above, the inspectors consider the licensee's efforts in this area to be effective in reducing personnel errors. This item is considered closed.

   b. (Closed) Open Item (331/85021-10(DRP)): Long Term Corrective Action for Sequencing Diesel Generator Loads. A permanent modification was installed and tested to sequence the required Loss of Coolant Accident loads onto the emergency busses even if power is being received from a transformer instead of the diesel generator. This item is considered closed.

   c. (Open) Violation Severity Level IV (331/85029-04(DRP)): Surveillance Test on the Equipment Drain Sump Flow Timers Did Not Verify Proper Alarm and Initiating Action. A study of the Surveillance Test Program to ensure 10 CFR Part 50.36(c)(3) is met is scheduled for completion by December 31, 1986. This item remains open pending NRC review of the licensee's study.

   d. (Closed) Unresolved Item (331/85034-03(DRP)): Vibration Readings on Core Spray Pump. The licensee has agreed that if data taken for Section XI of the ASME Boiler and Pressure Vessel Core is in the required action range, the affected component will be declared inoperable unless a known instrument problem has caused the data to fall in the required action range, in which case the instrument problem will be corrected and the data will be retaken.

With respect to the vibration readings on the Core Spray Pump, the licensee's consultant has determined that (1) the pump and motor are operating within acceptable vibration limits, and (2) the cause of the vibration readings above 30 mils is due to one or more of the following effects:

- A mechanical resonance occurring in the magnetically mounted transducer during the vibration measurement.
- An inherent problem that occurs when low frequency accelerometer signals are integrated to produce velocity and displacement outputs.

The licensee has relocated the measurement points to make the point more accessible and reduce the above effects. This item is considered resolved.

e. (Open) Open Item (331/86002-01(DRP)): Limitorque Operators With Nonqualified Wire. During the Spring 1986 outage the licensee replaced the wiring in 34 Limitorque Valve Operators that perform a safety function in a High Energy Line Break (HELB) environment with qualified wiring. Eleven other Limitorque Valve Operators were previously replaced, and according to the licensee, have qualified wiring. The remaining 51 Limitorque Valve Operators in the EQ program are in harsh (radiation only) environments. The licensee has inspected 11 of these valves to determine the type of wire in the valve operators. Based on the type of wire (in some cases engineering judgement was used to determine the type of insulation) and insulation material, the licensee evaluated the insulation material and determined that the insulation would not fail under accident conditions. This type of evaluation, however, is not consistent with NRC requirements for having traceable documentation for the wires and qualification testing of a similar or actual wire sample of the type installed in the valve operators. The licensee has committed to replacing the wiring in the affected valve operators with wiring that is consistent with NRC requirements. The wire replacements will be accomplished on a noncontrolling basis during outages until the next refueling outage when all the wiring will be replaced prior to startup.

   This item remains open pending replacement of the wiring in all the affected valve operators.

f. (Closed) Open Item (331/85021-01(DRP)): Incorporation of Ambient Vaporizer Temperature Control Valve and Pressure Control Valve Into the Preventive Maintenance Tracking Program. The inspectors verified that the above equipment was added to the plant preventive maintenance program. This item is considered closed.

3. Operational Safety Verification

The inspectors observed control room operations, reviewed applicable logs and conducted discussions with control room operators during the inspection. The inspectors verified the operability of selected emergency systems, reviewed tagout records and verified proper return to service of affected components. Tours of the reactor building and turbine building were conducted to observe plant equipment conditions, including potential fire hazards, fluid leaks, and excessive vibrations and to verify that maintenance requests had been initiated for equipment in need of maintenance. The inspectors, by observation and direct interview, verified that the physical security plan was being implemented in accordance with the station security plan.

The inspectors observed plant housekeeping/cleanliness conditions and verified implementation of radiation protection controls. During the inspection, the inspectors walked down the accessible portions of the Residual Heat Removal, Emergency Service Water, Core Spray, Diesel Generator, and Standby Liquid Control Systems to verify operability. The inspectors also witnessed portions of the Radioactive Waste System controls associated with radwaste shipments and barreling.

These reviews and observations were conducted to verify that facility operations were in conformance with the requirements established under technical specifications, 10 CFR, and administrative procedures.

No problems or concerns were identified.

4. Monthly Maintenance Observation

Station maintenance activities of safety-related systems and components listed below were observed/reviewed to ascertain that they were conducted in accordance with approved procedures, regulatory guides and industry codes or standards, and in conformance with technical specifications.

The following items were considered during this review: the limiting conditions for operation were met while components or systems were removed from service; approvals were obtained prior to initiating the work; activities were accomplished using approved procedures and were inspected as applicable; functional testing and/or calibrations were performed prior to returning components or systems to service; quality control records were maintained; activities were accomplished by qualified personnel; parts and materials used were properly certified; radiological controls were implemented; and fire prevention controls were implemented.

Work requests were reviewed to determine status of outstanding jobs and to assure that priority is assigned to safety-related equipment maintenance which may affect system performance.

The following maintenance activities were observed/reviewed:

- HPCI Throttle Valve Repair
- Residual Heat Removal Discharge Check Valve Repair
- Traversing Incore Probe Ball Valve Repair
- Safety Relief Valve Power Cable Replacement
- Intermediate Range Monitor Overlap Adjustment
- Limitorque Valve Operator Wire Replacement

No problems or concerns were identified.

5. <u>Monthly Surveillance Observation</u>

The inspectors observed technical specifications required surveillance testing on the Emergency Diesel Generators, Drywell Pressure Instruments, and Operator Daily and Shiftly Checks (selected portions) and verified that testing was performed in accordance with adequate procedures, that test instrumentation was calibrated, that limiting conditions for operation were met, that removal and restoration of the affected components were accomplished, that test results conformed with technical specifications and procedure requirements and were reviewed by personnel other than the individual directing the test, and that any deficiencies identified during the testing were properly reviewed and resolved by appropriate management personnel.

No problems or concerns were identified.

6. <u>Licensee Event Reports Followup</u>

Through direct observations, discussions with licensee personnel, and review of records, the following event reports were reviewed to determine that reportability requirements were fulfilled, immediate corrective action was accomplished, and corrective action to prevent recurrence had been accomplished in accordance with technical specifications.

a. (Closed) Licensee Event Report (LER) 86-005 (331/86-005-LL): Reactor Core Isolation Cooling (RCIC) Inverter Trip on High Input Voltage. The inverter tripped on high input voltage due to battery equalization in progress at the time. The licensee experienced similar problems with inverters in the RCIC and High Pressure Coolant Injection (HPCI) systems in 1974, and raised the high input voltage sensor trip points to the manufacturer's recommended value on these inverters. However, the licensee did not change the setpoint on a spare inverter in storage, which was eventually installed in the RCIC system on August 2, 1985. The result was the installation of a defective inverter into the RCIC system and the subsequent inverter trip on March 11, 1986. This is an apparent violation (331/86006-01(DRP)) of 10 CFR 50, Appendix B, Section VIII which requires in part that measures established for the identification and control of materials, parts, and components be designed to prevent the use of incorrect or defective materials, parts, and components. A replacement inverter was installed in the RCIC system, and its trip setpoints checked prior to installation. The licensee is developing a procedure to adequately test this and similar inverters prior to installation. The licensee is also reviewing controls in place to ensure spares installed from stock reflect operating experience. This LER is considered closed.

b. (Closed) LER 86-006 (331/86-006-LL): Reactor Water Cleanup (RWCU) System Isolation Due to Failed Tank Head Gasket. The isolation was caused by a high differential flow condition between the inlet and outlet of the "B" filter demineralizer as a result of gross leakage in the area of the main head flange. A suspected cause of the failure was lower than expected torque values for the tank head bolts. The main head flange gasket was replaced and the bolts were torqued to the required value. The licensee checked bolt torque values on the "A" filter demineralizer, which were also lower than expected. The bolts were torqued to the required value. The licensee plans to recheck these torque values prior to restart from the next refuel outage and to inspect and replace the gaskets on the "A" filter demineralizer. This LER is considered closed.

c. (Closed) LER (331/86-007-LL): RCIC Isolation Due to Temperature Switch Design Problem. The cause of the isolation was an internal design problem with a temperature differential switch in the Steam Leak Detection System (SLDS). The isolation was received when the temperature switch was taken to the READ position as part of a daily surveillance test. The manufacturer of the temperature switch is aware of a design problem with this particular model producing spurious signals when switched to the READ position. To correct this problem the licensee has installed a short time delay within the RCIC and HPCI SLDS circuitry to eliminate isolations from spurious signals, without affecting system response to a real event. This LER is considered closed.

d. (Closed) LER 86-008 (331/86-008-LL): Reactor Protection System (RPS) Trip from Spurious Signal While in Cold Shutdown. The trip was caused by a spurious signal in the "A" RPS logic. The "B" RPS channel was already tripped due to a 24 VDC battery discharge test in progress. The cause of the spurious signal is unknown since no annunciators were received with the trip, and the process computer was de-energized for maintenance at the time. The licensee believes the trip was caused by a spurious electrical spike generated from outage activities. The trip signal was reset immediately. The licensee has reviewed recent functional tests performed on each trip sensor, and no deviations were found. This LER is considered closed.

e. (Closed) LER 86-009 (331/86-009-LL): Standby Gas Treatment System (SGTS) Actuation During Outage Maintenance. Actuation of the SGTS was caused by a downscale trip of the "A" Reactor Building Exhaust

Ventilation Radiation Monitor. The radiation monitor tripped because of maintenance work being performed on a drywell cooling valve which shares common instrument AC power with the monitor. The operator who prepared the equipment tags for the maintenance did not realize the connection between the two components, since he did not consult the Electrical Distribution List. This document was revised to reflect the common power supply when a similar event occurred in 1985, documented in LER 85-012. This is an apparent violation (331/86006-05(DRP)) of 10 CFR 50, Appendix B, Section XVI which requires in part that measures be established and taken to assure that corrective actions are taken to preclude repetition. Furthermore, the Iowa Electric Light and Power Company QA Manual Sections 14.3.2 and 14.3.3 identify LERs as a system to correct conditions adverse to quality. The licensee has placed temporary warning tags on the breakers involved and has instructed operators to consult the Electrical Distribution List or the Electrical Distribution Information System (EDIS) in the future. A work request was initiated to modify the power supply circuits to separate dissimilar equipment. The licensee is continuing to develop its EDIS, a computerized system which will eventually replace all documents presently used by operators. This LER is considered closed.

f. (Closed) LER 86-011 (331/86-011-LL): Main Steam Line Isolation (MSIV) Failure to Open Due to Stem Binding. During observation of MSIV movement prior to running a surveillance test, the licensee observed the valve to stick in the partially open position. Removal of the packing gland and packing material revealed surface galling on both the valve stem and junk ring. The licensee discovered that a small piece of stellite material had broken off the junk ring and caused the valve stem to bind. The licensee suspects that a metallurgical flaw caused the stellite to break away and believes this to be an isolated occurrence. The galling was removed from the valve stem and junk ring, and the valve was reassembled and stroke tested satisfactorily. The seven other MSIV stems were inspected and no evidence of surface galling found. This LER is considered closed.

Two violations were identified.

7. <u>Outage Activities</u>

The licensee entered a scheduled outage on March 15, 1986, to perform maintenance activities and technical specification required surveillance tests. The outage was scheduled to last two weeks, but was extended a week when problems were discovered late in the outage with one of the MSIV and an instrument air compressor. In the process of shutting down for the outage, the licensee discovered a leaking check valve in the injection line of one train of the Residual Heat Removal (RHR) system, causing pressurization of that system. During the outage, the licensee disassembled the valve when attempts to seat the valve with hydrostatic pressure failed. A small shoulder on a hinge pin in the valve was found which was causing the valve disc to bind on the seat. The shoulder was ground down and the valve reassembled. The valve was again hydro tested and found to be leak tight. This same valve was inspected during the last outage and the problem with the hinge pin was not discovered. The inspectors are concerned that inspection procedures may be inadequate in addressing problems of this type, and intend to review the inspection procedures for the RHR check valve (Open Item No. 331/86006-02(DRP)). During startup after the outage, the check valve again began to leak, but the licensee had procedures in place to prevent pressurization of the RHR system through operational means. Once reactor pressure reached a higher level, the check valve seated and no further problems occurred.

One item of concern was identified.

8. Modifications and Facility Changes

During review of a modification of the HPCI and RCIC steam leak detection circuitry, the inspectors noted that the HPCI relays (Quality Level 1, Seismic) were being installed using Quality Level 4 hardware. No engineering evaluation could be found which showed that the installed configuration was able to withstand a seismic event. It is the inspectors' judgement that because the weight of the relays is small no safety concern exists with the installed configuration. The inspectors are concerned, however, that the lack of an evaluation may indicate that engineering is not adequately considering the ability of installed equipment to withstand a seismic event. Similar instances have occurred for modifications such as: (1) acid pipe routed through the diesel generator room, and (2) replacement battery did not fit battery rack. Although these examples were shown to not affect the operability of the equipment, the initial engineering design appeared not to consider seismic aspects adequately. The inspectors are concerned that this apparent lack of consideration could ultimately lead to a condition where safety-related equipment is degraded. This item is open pending NRC review of the licensee's actions to correct this weakness (331/86006-03(DRP)).

During the above inspection, the licensee informed the inspectors that the RCIC relays in the steam leak detection circuitry were being replaced with Quality Level 4 relays. The original relays were being replaced to prevent spurious isolations of the RCIC system and Quality Level 1 relays would not be available for five to seven weeks. It was the licensee's position that the Quality Level 4 relays were better than the relays which were allowing the spurious isolations to occur. The Quality Level 4 relays are identical to the Quality Level 1 relays except for the required documentation. The relays are manufactured at the same facility using the same specifications. The Quality Level 4 relays were inspected for physical damage and correct nameplate data and were functionally tested. The licensee conditionally released the relays for installation and operation in accordance with their administrative procedures. The documentation included a 10 CFR 50.59 review that was marginally adequate. The conditional release expires during the next refueling outage; however, the licensee agreed to replace the relays when they become available and the RCIC system is out of service for other reasons. This item is open pending installation of the Quality Level 1 relays (331/86006-04(DRP)).

Two items of concern were identified.

9. Open Items

Open items are matters which have been discussed with the licensee, which will be reviewed further by the inspectors, and which involve some action on the part of the NRC or licensee or both. Open items disclosed during the inspection are discussed in Paragraphs 7 and 8.

10. Exit Interview

The inspectors met with licensee representatives (denoted in Paragraph 1) throughout the inspection period and at the conclusion of the inspection on May 20, 1986, and summarized the scope and findings of the inspection activities. The inspectors also discussed the likely informational content of the inspection report with regard to documents or processes reviewed by the inspectors. The licensee did not identify any such documents or processes as proprietary.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION III
799 ROOSEVELT ROAD
GLEN ELLYN, ILLINOIS 60137

RECEIVED
AUG 18 1986
I. E. L. P. Co.
By ________

AUG 14 1986

Docket No. 50-331

RECEIVED
AUG 18 1986
NUCLEAR LICENSING and FUELS

Iowa Electric Light and Power
Company
ATTN: Mr. Lee Liu
President and Chief
Executive Officer
IE Towers
P. O. Box 351
Cedar Rapids, IA 52406

Gentlemen:

This refers to the special safety inspection conducted by Messrs. R. Hasse and S. Hare of this office and R. Pierson of The Office of Inspection and Enforcement on July 7-18, 1986, of activities at the Duane Arnold Energy Center authorized by NRC Operating License No. DPR-49 and to the discussion of our findings with Mr. D. Mineck and others of your staff at the conclusion of the inspection.

The enclosed copy of our inspection report identifies areas examined during the inspection. Within these areas, the inspection consisted of a selective examination of procedures and representative records, observations, and interviews with personnel.

The focus of this inspection was on the adequacy of safety reviews performed pursuant to 10 CFR 50.59. As noted in the attached Notice and Inspection Report, your documentation of these reviews was inadequate and in violation of the requirements of 10 CFR 50.59. We do wish to acknowledge your recent efforts to upgrade the process of safety evaluations at your facility and encourage your continued effort. A written response is required to the enclosed Notice of Violation and should include your schedule for revising those safety evaluations noted as deficient in the enclosed Inspection Report.

In accordance with 10 CFR 2.790 of the Commission's regulations, a copy of this letter, the enclosures, and your response to this letter will be placed in the NRC Public Document Room.

The responses directed by this letter and the accompanying Notice are not subject to the clearance procedures of the Office of Management and Budget as required by the Paperwork Reduction Act of 1980, PL 96-511.

~~8608190685~~

We will gladly discuss any questions you have concerning this inspection.

Sincerely,

Carl J. Paperiello, Director
Division of Reactor Safety

Enclosures:
1. Notice of Violation
2. Inspection Report
No. 50-331/86010(DRS)

cc w/enclosures:
D. Mineck, Plant Superintendent
Nuclear
W. Miller, Assistant Plant
Superintendent Technical
Support
DCS/RSB (RIDS)
Licensing Fee Management Branch
Resident Inspector, RIII
Thomas Houvenagle, Iowa State
Commerce Commission

NRC / INPO DISTRIBUTION
Response Due: 9/15/86
SOI ? no

- McGaughy
- Lessly
- Matthews
- Mineck
- Rothert
- Wilson
- Borton
- Browning
- Burns
- Gucciardo
- Hannen
- Hingtgen
- Hughel
- Kozman
- Loghlein
- McDermott
- Miller
- Murphy
- Root, E.
- Salmon
- Shea (N&H)
- [redacted]
- Swails
- VanMiddlesworth
- Vavra
- Young

Codes: A-102
Comm. Cont: 860300
References:
Records Control
NRC IR 86-0010

NOTICE OF VIOLATION

Iowa Electric Light and Power Company

Docket No. 50-331

As a result of the inspection conducted on July 7-18, 1986, and in accordance with 10 CFR Part 2, Appendix C - General Statement of Policy and Procedure for NRC Enforcement Actions (1985), the following violation was identified:

10 CFR 50.59(b) requires that the licensee prepare a written safety evaluation for any change to the facility as described in the FSAR, any change to procedures described in the FSAR, or tests or experiments not described in the FSAR. It further requires the safety evaluation to document the bases for the determination that the change, test, or experiment does not involve an unreviewed safety question.

Contrary to the above, the safety evaluations for the following changes did not provide the bases for concluding that no unreviewed safety question existed:

A. Design Change (DC) 1161,
B. DC 908,
C. DC 1222,
D. DC 1008,
E. DC 1276,
F. DC 1177,
G. DC 1095,
H. DC 1057.
I. Procedure Changes OI-64, Revision 3; OI-24, Revision 5; OI-16.0, Revision 5; and OI-49, Revision 7.

This is a Severity Level IV violation (Supplement I).

Pursuant to the provisions of 10 CFR 2.201, you are required to submit to this office within thirty days of the date of this Notice a written statement or explanation in reply, including for each violation: (1) corrective action taken and the results achieved; (2) corrective action to be taken to avoid further violations; and (3) the date when full compliance will be achieved. Consideration may be given to extending your response time for good cause shown.

8-14-86
Dated

Carl J. Paperiello, Director
Division of Reactor Safety

~~8608190690~~

U.S. NUCLEAR REGULATORY COMMISSION

REGION III

Report No. 50-331/86010(DRS)

Docket No. 50-331 License No. DPR-49

Licensee: Iowa Electric Light and
Power Company
Post Office Box 351
Cedar Rapids, IA 52406

Facility Name: Duane Arnold Energy Center

Inspection At: Palo, IA and Cedar Rapids, IA

Inspection Conducted: July 7-18, 1986

Inspectors: R. Hasse 8-13-86
Team Leader Date

S. Hare 8-14-86
Date

R. Pierson 8-13-86
IE Headquarters Date

Approved By: M. Phillips, Chief 8/13/86
Operational Programs Section Date

Inspection Summary

Inspection on July 7-18, 1986 (Report No. 50-331/86010(DRS))
Areas Inspected: Special safety inspection of the adequacy of safety evaluations performed pursuant to 10 CFR 50.59.
Results: One violation with nine examples was identified (failure to document bases for conclusion that no unreviewed safety question exists - Paragraph 2.c).

~~8608190694~~

DETAILS

1. Persons Contacted

Iowa Electric Light and Power Company*

D. Mineck, Plant Superintendent-Nuclear
R. Lessly, Manager, Design Engineering
W. Rothert, Manager, Nuclear Projects
W. Miller, Assistant Plant Superintendent, Technical Support
R. Salmon, Technical Support
L. Vavra, Supervising Engineer
J. Loehrlein, Supervisory Engineer-Nuclear Projects
O. Olson, Systems Engineering
D. Wilson, Manager, Licensing and Emergency Planning

US NRC*

N. Gilles, Resident Inspector, DAEC

Other personnel were contacted as a matter of routine during the inspection.

*All personnel listed attended the exit interview on July 18, 1986.

2. 10 CFR 50.59 Safety Evaluations

The purpose of this special inspection was to determine the adequacy of. the safety evaluations performed by the licensee pursuant to the requirements of 10 CFR 50.59. Secondary objectives were to determine if the evaluations were performed when required and properly reported.

The inspection consisted of an in depth review by three inspectors of the safety evaluations for ten modifications. The safety evaluations for several procedure changes and special tests were also reviewed. The inspectors also reviewed the procedures controlling activities subject to the provisions of 10 CFR 50.59 to determine their adequacy.

a. Requirements

The intent of 10 CFR 50.59 is to assure that changes to the facility or its mode of operation as described in the FSAR do not create an unreviewed safety question or require a change in the technical specifications without an appropriate license amendment. Further, the regulation explicitly requires that a written safety evaluation be generated providing the bases for the conclusion that a given change, test, or experiment does not involve an unreviewed safety question.

The basis for the general conclusion that no unreviewed safety question exists logically consists of two elements: (1) the identification of all changes or issues effected by the modification, procedure change, test, or experiment; and (2) the basis for concluding that these changes or issues do not involve an unreviewed safety question.

b. Summary of Results

The following is a summary of the results of the inspection. Details on the review of individual items are presented in Paragraph 2.c.

(1) The documentation of the safety evaluations was found to be generally inadequate and in violation of the requirements of 10 CFR 50.59. Paragraph (b) of 10 CFR 50.59 requires that the bases for the conclusion that no unreviewed safety question exists be documented in the safety evaluation. The evaluations frequently stated only the conclusions with no bases for the conclusions nor what specific issues were addressed. Thus, it was necessary for the reviewer to determine what needed to be addressed, then search the design package to determine if it had been addressed in documented form, or interview the design engineer to determine if it had been addressed at all.

(2) The inspectors identified no apparent unreviewed safety questions or required technical specification changes. However, the licensee must revise the safety evaluations to document issues addressed and the bases for the conclusion that no unreviewed safety question exists.

(3) Several programmatic concerns were identified:

a. The licensee had no procedure addressing temporary modifications generically.

b. The procedure for minor modifications did not delineate in sufficient detail the limitations and applicability of the procedure.

c. The procedure for controlling design document changes appeared to provide a potential vehicle for effecting a design change without using the design change system.

c. Inspection Details

The detailed inspection results are presented in the following paragraphs.

(1) Design Change Request 1161, "Temperature Elements Replacement"

This DCR was originally written to replace the 36 temperature elements (TEs) in the main steam line break (MSLB) system with environmentally qualified (EQ) elements. These TEs were part of the containment isolation system. The DCR was later amended by Engineering Work Request (EWR) 83-368 to include the replacement of accident monitoring TEs in the drywell, RHR heat exchanger outlet, and Torus with EQ TEs. As part of this modification, five of the 36 TEs in the MSLB system were relocated.

The main deficiency in the Safety Evaluation for this modification was the failure to document the acceptability of the relocation of the five MSLB TEs. Potential changes in system response time due to the relocation or the basis for original TE locations were not addressed in the written safety evaluation. Also, the Safety Evaluation states that the new TEs are electrically interchangeable with the original TE's without identifying those parameters considered (eg. response characteristics). This failure to document the bases for concluding that no unreviewed safety question exists is considered a violation of 10 CFR 50.59 (331/86010-1A).

(2) Design Change Request 908, "Post Accident Sampling System"

This modification provided a keylock override of the containment isolation signal to the reactor coolant sample lines. The override capability could be enabled only during isolation conditions. Its purpose was to provide post accident sampling capability. The keylock switches were located in the control room and appropriate annunciation was provided. The original design had automatic timers on the bypass to limit this condition to ten minutes. This feature was later removed from the design in favor of administrative control. The administrative control consisted of a procedural requirement that the chemist notify the operator when he wished to start sampling and when he had finished. The inspector could find no time limit on the sampling period.

The Safety Evaluation for this modification states without basis that the consequences of an accident are not increased as a result of this modification. A calculation of the increased dose in the exclusion area due to sampling had been included in the original Safety Evaluation when the timers were part of the design. However, this calculation was removed from the Safety Evaluation when the timers were removed. The failure to document a basis for this conclusion is considered a violation of 10 CFR 50.59 (331/86010-1B).

(3) Design Change Request 1222, Replace RHRSW Pump Bowl Assembly"

This DCR replaced one (1) RHR Service Water Pump and provided a method by which the remaining three (3) RHRSW pumps might be replaced (without additional DCR) on an as-needed basis. The existing motor and upper column pipe would be reused while the lower column section and bowl assembly would be replaced. A seismic support would be added for each pump. This change was safety-related. The 10 CFR 50.59 Safety Evaluation concluded that the modification did not constitute an unreviewed safety question.

One deficiency was noted in the Safety Evaluation. The Safety Evaluation stated that original head and capacity requirements were met following replacement of the impeller and bowl with stainless steel instead of bronze. Plotting of the test data collected during performance of the Engineering Acceptance Requirements for DCR 1222 revealed that in fact the original head and capacity requirements were not met at flows of 2000, 2200, 2400, and 2600 gallons per minute (gpm). The pump head values varied from the vendor provided pump head curve by as much as 7%. This does not present an unreviewed Safety Question since the licensee (in an independent submittal) had submitted, and the NRC had approved, Technical Specification Amendment No. 108 which allows a 15% reduction in flow requirements. Consequently, although the current flow requirements are met, the Safety Evaluation was incorrect in stating that there is an additional 15% added to the current operating margin for flow requirements. The failure to accurately document the basis for the conclusion that no unreviewed safety question existed is considered a violation of 10 CFR 50.59 (331/86010-1C).

(4) Design Change Package 1008, "RCIC System PT 2502 Replacement"

The pressure transmitter (PT 2512) for the Reactor Core Isolation Cooling (RCIC) system pump suction pressure was replaced because the RCIC suction line experienced pressure transients in excess of the rated design proof pressure rating of the installed transmitter whenever the turbine was tripped. This change was performed to replace the pressure transmitter with a safety-related transmitter which had a higher pressure rating compatible with these pressure transients. This change was safety-related. The 10 CFR 50.59 Safety Evaluation concluded that the modification did not constitute an unreviewed safety question. The primary function of this pressure transmitter is to provide information regarding RCIC system operational conditions to the control room operators.

The 50.59 Safety Evaluation did not provide adequate justification to show that all environmental and seismic requirements had been met. Furthermore, the Safety Evaluation did not address transmitter reliability or availability, or response time changes. If a particular item is not applicable this should be stated with enough amplifying information as to why the evaluator feels this is the case.

As presently written the 50.59 Safety Evaluation for DCR No. 1008 provides inadequate engineering bases for concluding there were no unreviewed safety questions or that the new transmitter meets all the environmental and seismic requirements of PT 2502. This is considered a violation of 10 CFR 50.59 (331/86010-1D).

(5) Design Change Package 1276, "Replace Non-Qualified Flow Transmitters"

This change was performed to meet the requirements of 10 CFR 50.49 on environmental qualification of electrical equipment. The flow-rate transmitters of the primary accident monitoring instrumentation for LPCI/RHR and Core Spray System were replaced with Class IE safety-related transmitters. This change was safety-related. Although the 10 CFR 50.59 Safety Evaluation does not state explicitly that the modification does not constitute an unreviewed safety question, it is stated that these flow-rate transmitters do not provide any automatic safety function and therefore failure of these transmitters will not result in a failure of an automatic system to perform its safety function.

There were two deficiencies in the Safety Evaluation. First, the Safety Evaluation states that alternative verification of system flow for the Core Spray and LPCI/RHR systems is available in the control room, and further states that these system flows can be indirectly determined by reactor water level indication or system check valve position switches. No justification is made as to the reliability or likelihood that these additional instruments will be available in the event of an inoperable transmitter. Secondly, the seismic analysis is insufficient as written to support the seismic qualification of the instrument racks 1C-123 and 1C-124 following mounting of the new transmitters.

These deficiencies constitute a violation of 10 CFR 50.59 (331/86010-1E).

(6) Design Change Package 1177, "Reactor Recirculation System Bonnet Vent Removal"

The basis for this modification was excessive leakage to the drywell equipment drain sump through the bonnet vent connections from valves MO-4629 and MO-4630. The bonnet vent isolation valves had been leaking past their valve seats increasing the leakage to the sump which was approaching the technical specification limit on drywell leakage. The change was performed to remove the bonnet vent line and isolation valves, cap the bonnet vents and cap the equipment drain sump line where the bonnet vent line tied in. This change was safety-related.

The 10 CFR 50.59 Safety Evaluation concluded that the modification did not constitute an unreviewed safety question. The function of the bonnet vent system was to prevent overpressurization of the bypass valve(s) which could occur if the valve(s) were closed and experienced an increasing temperature. Since the valves are left in the open position per a recommendation in General Electric SIL No. 104, and as implemented in DAEC Operating Instruction 64, the potential for bonnet overpressurization is eliminated. The Design Change Package specified that in addition to removing the vent valves, lines, and installing caps, the associated snubbers and pipe supports were to be removed. The 50.59 Safety Evaluation as written, only addresses what effect eliminating the bonnet vent feature had on the operability of the system. It did not provide justification to show that all seismic requirements were satisfied. When a change to a system is made, the Safety Evaluation should address all aspects of the modification, not just the primary functional change of the modification. This is considered a violation of 10 CFR 50.59 (331/86010-1F).

(7) Design Change Package 1095, "Main Steam Leakage Control System Flowmeter Replacement"

This change was performed to meet the requirements of 10 CFR 50.49 on environmental qualification of electrical equipment. The modification replaced the existing flow instrumentation for the MSIV Leakage Control System with flow instrumentation which was seismically and environmentally qualified for the expected service conditions. This change was safety-related. Three deficiencies were noted in the Design Change Package and the 50.59 Evaluation.

In the text of the package a statement was made that "Based on the information available in our offices, there are no previous additions to the panel" (1C-14) and since the weight of the new instrumentation to be installed was less than 5% of the total cabinet weight, the cabinet's seismic analysis was not affected. The Safety Evaluation did not address this statement. Specifically the SE should have ensured that the "information based in our offices" was correct and documented in the SE.

Further, the SE states that the replacement of the instruments with environmentally qualified instruments did not change the function of these instruments and therefore the possibility of an accident or malfunction of a different type then was previously discussed in the FSAR was not created. This statement by itself is not adequate. Examples of areas that should have been addressed but weren't are instrument accuracy, instrument reliability, and system response time.

The SE also stated that the safety functions of the system, passive flow limiting ability, and high flow trip were maintained. The inspectors questioned the passive flow function and the licensee representative responded by referencing a drawing, 7884-APED-B21-43-1, that specified the maximum pressure drop through the systems original flowmeter as three inches. The inspector noted that this pressure drop was specified to insure the flowmeter remained calibrated over all expected flow conditions, not a passive flow limiting safety function as described in the SE. These inadequacies are considered a violation of 10 CFR 50.59 (331/86010-1G).

(8) Design Change Request 1236, "Modify RHR Minimum Flow Requirements/RHRSW Pumps"

This DCR added additional information to two Residual Heat Removal Data Sheets that changed the minimum RHR Service Water pump flow rate from 2400 gpm to 2040 gpm per pump. This DCR did not require any physical change to the pump. This change was safety-related. The 10 CFR 50.59 Safety Evaluation concluded that the modification did not constitute an unreviewed safety question. The purpose of this change was to lower the technical specification required RHRSW pump flow rate from the design flow rate (2400) to a more easily achievable flow rate. The change in flow rate was justified in two General Electric Reports which analyzed the change (NEDC-22082-P and NEDE 30051) and found the reduction in flow to not reduce the capability of the system to mitigate accident conditions.

The inspector did not identify any deficiencies in the Safety Evaluation.

(9) Design Change Request 1057, "Offgas System Valves MO-4151 and CV-4151 Replacement."

Valve MO-4151 was used to control the pressure in the Offgas System upstream of a jet compressor. Since MO-4151 was a gate valve not designed to control pressure, problems with pressure control, valve drift and seat wear were observed. This modification replaced the worn valve on a like basis and installed a globe valve downstream of MO-4151 to control the pressure to the suction of the jet compressor. The 10 CFR 50.59 Safety Evaluation concluded that the modification did not constitute an unreviewed safety question.

This modification replaced a worn valve and installed a new globe valve to throttle the flow and prevent overpressurization of loop seals which could have resulted in an unplanned radioactive release to the plant environment.

The Safety Evaluation stated that the changes did not affect any safety-related equipment and the safety evaluation in the UFSAR was not changed. The evaluation did not contain any basis for that statement. Wiring and piping changes were required for this modification yet nothing in the Safety Evaluation addressed if the wiring or piping interfaced with essential components or if their failure could affect any safety-related component. Further, the evaluation stated the probability of any accident was reduced due to a more stable Offgas System. It did not address new accidents that could be caused by the failure of the new equipment nor does it address the likelihood of failure due to the new equipment/wiring interfaces. The Safety Evaluation also stated the margin of safety as defined by technical specification basis was not reduced because the portion of the Offgas system affected was not addressed in the technical specifications. Even though the system was not mentioned in technical specifications, the possibility that this system failing, due to the new components, could affect a system important to safety had not been evaluated. This is considered a violation of 10 CFR 50.59 (331/86010-1H).

(10) Design Change Package 1335, "Unit Auxiliary Transformer Installation"

As a result of the Unit Auxiliary Transformer failure in 1984, this DCP was generated to cover the replacement of the failed transformer with a new transformer of an updated design which required changes to existing plant documents. This change was not safety-related. The 10 CFR 50.59 Safety Evaluation concluded that the modification did not constitute an unreviewed safety question.

The inspector did not identify any deficiencies in the Safety Evaluation.

(11) Temporary Modifications

During the course of the inspector's review, it was determined that a procedure did not exist to effect and control all temporary modifications. The licensee indicated that jumpers and lifted leads were controlled through the use of a Jumper and Lifted Lead Control Procedure, No. 1410.6, Revision 3, effective October 10, 1985. In conjunction with this procedure the licensee could utilize one Design Change Package to perform the modification and another Design Change Package to remove the modification.

In addition, Special Test Procedure, No. 1407.4, Revision 1, effective September 19, 1984 which identified requirements and responsibilities for conducting special test procedures and minor modifications Procedure, No. 109.1, Revision 0, effective March 7, 1986 which establishes the requirements for the authorization, installation and documentation of minor modifications were purportedly utilized as necessary when other temporary modifications were required.