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U. S. NUCLEAR REGULATORY COMMISSION

REGION III

Report No. 50-331/92022(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, Iowa

Inspection Conducted: October 22 througn December 3, 1992

Inspectors: M. Parker
C. Miller

Approved: R. D. Lanksbury, Chief
Reactor Projects
Section 3B

12/17/92
Date

Inspection Summary

Inspection on October 22 through December 3, 1992 (Report No. 50-331/92022(DRP))

Areas Inspected: Routine, unannounced inspection by the resident inspectors of followup of events, operational safety, maintenance, surveillance, design, plant trips, management meetings, and report review.

Results: An executive summary follows:

EXECUTIVE SUMMARY

Operations

The reactor was operating at about 100 percent power at the beginning of the period. On November 3, 1992, the "A" reactor recirculation pump motor generator (M/G) set spuriously ramped up in speed until it locked up. The resultant reactor power was 1681 megawatts (MW) (101.4 percent of rated) before operators reduced recirculation flow manually to return reactor power to 100 percent (Section 2).

On November 13, 1992, the reactor automatically scrammed due to a turbine trip caused by high condenser back pressure which resulted from a separation of the "A" circulating water (CW) pump casing. Following an outage to isolate the "A" CW header and to perform other maintenance, the reactor was started up on November 16, 1992 (Section 7).

Numerous instances of missed fire watches were identified in a Quality Assurance surveillance. The details and corrective actions of these discrepancies were being evaluated by the licensee and the NRC. Unresolved item 331/92022-01(DRP) will be used to track this issue (Section 3).

Maintenance/Surveillance

Maintenance and surveillance activities continued to show good performance. Troubleshooting and repair activities on the "A" reactor recirculation M/G set were timely and effective (Section 2). Incomplete followup of maintenance activities on the "A" control building chiller led to the chiller tripping out on low oil suction pressure when the "B" chiller was also out of service (Section 4).

Engineering and Technical Support

Significant engineering support was given for CW pump problems, high energy line break (HELB) issues, and containment free volume calculations. An engineer reviewing leak rate calculations discovered an error with a drywell free volume calculation made in 1971. Unresolved item 331/92022-02(DRP) will be used to track this issue (Section 6). The licensee issued a non-conformance report (NCR) and made a prompt operability determination. Engineering evaluation of the HELB issue was lacking in depth, and failed to consider safe shut down requirements and safe shut down instrumentation in the initial operability evaluation, and in a subsequent evaluation, prior to NRC involvement. Unresolved item 331/92022-03(DRP) will be used to track this issue (Section 6).

Safety Assessment/Quality Verification

The Quality Assurance organization continued to show good performance and a strong surveillance program. A Quality Assurance surveillance found non-conservative assumptions in engineering calculations used for HELB modeling.

DETAILS

1. Persons Contacted

 R. Anderson, Assistant Operations Supervisor
*R. Baldyga, Supervisor, Maintenance Engineering
*B. Bernier, Supervising Engineer, Mechanical Engineering
 P. Bessette, Supervisor, Regulatory Communications
 J. Bjorseth, Assistant Operations Supervisor
*D. Blair, Supervisor, Quality Assurance
*C. Bock, Group Leader, System Engineering
 D. Boone, Supervisor, Health Physics
*J. Brazant, Group Leader, EQ Engineering
*J. Dvorsky, Group Leader, Mechanical Engineering
*D. Engelhardt, Security Superintendent
*M. Flasch, Manager, Engineering
 J. Franz, Vice President Nuclear
 T. Gordon, Supervisor, Electrical Maintenance
*L. Heckert, Nuclear Licensing Engineer
*S. Huebsch, Project Engineer
*J. Kinsey, System Engineer
*K. Kleinheinz, System Engineer
*J. Kozman, Supervising Engineer, Configuration Engineering
*D. Lausar, Supervising Engineer, Project Engineering
 M. McDermott, Maintenance Superintendent
*R. McGee, Technical Support Specialist
 C. Mick, Operations Supervisor
*T. Page, Acting Supervisor, Nuclear Licensing
*K. Peveler, Manager, Corporate Quality Assurance
 J. Probst, Group Leader, Systems Engineering
 K. Putnam, Supervisor, Technical Support
*D. Robinson, Nuclear Licensing Specialist
*A. Roderick, Testing and Surveillance Supervisor
*C. Rushworth, Nuclear Licensing Engineer
*B. Schenkelberg, Fire Protection Coordinator
 P. Serra, Manager, Emergency Planning
 S. Shangari, Group Leader, Mechanical Engineering
*N. Sikka, Supervising Engineer, Electrical Engineering
*T. Sims, Nuclear Licensing Specialist
*M. Smith, Quality Control Engineer
*J. South, Electrical Engineer
*S. Swails, Manager, Nuclear Training
 J. Thorsteinson, Assistant Plant Superintendent, Operations Support
*G. Van Middlesworth, Assistant Plant Superintendent, Operations and Maintenance
 D. Wilson, Plant Superintendent, Nuclear
*P. Wojtkiewicz, System Engineer
 K. Young, Manager, Nuclear Licensing

U. S. Nuclear Regulatory Commission (NRC)

*C. Miller, Resident Inspector
*M. Parker, Senior Resident Inspector

In addition, the inspectors interviewed other licensee personnel including operations shift supervisors, control room operators, engineering personnel, and contractor personnel (representing the licensee).

*Denotes those present at the exit interview on December 3, 1992.

2. Followup of Events (93702)

During the inspection period, the licensee experienced several events, some of which required prompt notification of the NRC pursuant to 10 CFR 50.72. The inspectors pursued the events onsite with licensee and/or other NRC officials. In each case, the inspectors verified that the notification was correct and timely, if appropriate, that the licensee was taking prompt and appropriate actions, that activities were conducted within regulatory requirements, and that corrective actions would prevent future recurrence. The specific events are as follows:

November 3, 1992 - Reactor Recirculation Pump Speed Increase

November 13, 1992 - Reactor Scram due to high condenser back pressure (See Section 7 for further details)

November 19, 1992 - Loss of Control Building Chillers (See Section 4 for further details)

Reactor Recirculation Pump Speed Increase

On November 3, 1992, the "A" reactor recirculation pump M/G set experienced a speed increase for unknown reasons. As a result of the speed increase, the M/G scoop tube locked up. Upon observing the lock up, reactor operators immediately reduced reactor power with the "B" reactor recirculation pump to restore reactor power to 100 percent. Operators then took action to balance out pump speed between the recirculation pumps. The "A" recirculation pump speed was manually reduced while maintaining the scoop tube in lock, and the "B" recirculation pump was then increased to restore the balance.

Upon further investigation, it was noted that the "A" M/G set speed indicator had "zero" indicated speed on the speed controller. The lock up was subsequently determined to be caused by a "High Speed Demand Versus Positioner Feedback" (runaway) signal. This runaway signal was caused by a tachometer generator speed signal failure. The resultant speed signal failure caused an increase of the reactor recirculation speed until the electrical stop was reached. Upon reaching the electrical stop, the high speed demand signal caused the scoop tube to lock up.

The resultant recirculation pump speed increase caused core flow to increase from 46.32×10^6 lbs/hr to 47.66×10^6 lbs/hr, as indicated on the process computer. The increase in core flow caused reactor power to increase from 1658 MW to 1681 MW (101.4 percent).

In troubleshooting the cause of the speed signal failure, the licensee identified a millivolt to current circuit card in the tachometer generator speed signal logic which had failed diodes. The licensee replaced the circuit card, which restored the speed signal, and returned the "A" reactor recirculation M/G to normal. The licensee has also taken action to repair the failed circuit card and to perform similar repairs to the "B" reactor recirculation M/G.

In reviewing this event, the inspectors confirmed that reactor power went from 1658 MW to 1681 MW (100.0 to 101.2 percent). The licensee is authorized to operate the Duane Arnold facility in accordance with Technical Specifications (TS) at steady state reactor core power levels not in excess of 1658 MW (thermal). In a review of the Updated Final Safety Analysis Report (UFSAR), the inspectors noted that this power increase did not exceed the design basis assumptions which included consideration for plant transients. In addition, the NRC staff has recognized brief power excursions up to 2 percent above licensed thermal power limits, provided the average power level over any 8 hour shift is maintained no greater than the 100 percent limit. The inspectors concluded that the event itself had no safety significance with regard to thermal power limits, and that it was within existing NRC guidance and requirements.

No violations or deviations were identified in this area.

3. Operational Safety Verification (71707) (71710)

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. It was observed that the Plant Superintendent, Assistant Plant Superintendent of Operations, and the Operations Supervisor were well informed of the overall status of the plant and that they made frequent visits to the control room and regularly toured the plant. 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 and cleanliness conditions and verified implementation of radiation protection controls. During the inspection, the inspectors walked down the accessible portions of the high pressure coolant injection (HPCI) system to verify operability by comparing system lineup with plant drawings, as-built configuration or present valve lineup lists; observing equipment conditions that could degrade performance; and verifying that instrumentation was properly valved, functioning, and calibrated.

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

Missed Fire Watches

As a result of previous problems with missed fire watches (inspection report 331/92017), Quality Assurance performed a followup surveillance to determine how well fire watch surveillances were being performed. This was accomplished by comparing fire watch requirements and logs to security computer records for the areas with card readers, during the period October 3 to 10, 1992. The surveillance identified approximately 15 instances in which areas were required to be checked by the program but for which the security computer had no record of personnel entry. Of those instances, several were required to have fire watches by TS. Other areas were still being evaluated to determine whether TS requirements were met.

The inspectors attended the licensee's problem solving meeting and discussed the details of the events with the Quality Assurance auditor and licensee fire protection and technical support personnel. Five instances involved a particular individual who consistently missed the pump house basement where thermo-lag material was installed. At the entrance to the basement area was a wire mesh door. The individual assumed that just observing the stairway area through the door was sufficient to meet the intent of the fire watch, even though the majority of the thermo-lag installation could not be observed from that position, nor could the majority of the room on three elevations be viewed for combustibles, fire hazards, or degraded fire protection equipment. The licensee initially agreed with the fire watch. The inspectors did not believe that this activity met the intent of a fire watch; and after confirming that position with Region III asked the licensee to review this decision.

Three additional instances of missed fire watches occurred when the individual responsible could not access the areas due to radiography barricades. The individual notified his supervisor and the central alarm station that he could not enter the area, in addition to noting this in his fire watch check sheet. The individual's supervisor did not take further action to ensure that the requirements of the fire watch could be met. The licensee initially indicated that two of the three missed areas were still covered by fire watches who fortuitously happened to be in the torus room, on the other side of the penetrations from the reactor core isolation cooling (RCIC) room and the southeast corner room (SECR). The inspectors informed the licensee that these may not be acceptable fire watches since the individuals did not know that they were supposed to be watching those particular penetrations and were probably not observing conditions in the immediate vicinity of the penetration. The other area for which coverage was missed was a HPCI room to offgas recombiner room penetration. The HPCI penetration was later found to be acceptable based on a further review of its condition. The licensee also later determined that the RCIC penetration was acceptable because it had been repaired previously, but had not been signed off on the fire protection impairment request.

Seven additional areas were missed with no identifiable reason given, since the fire watch checklists indicated that these areas were checked. These instances included an additional instance in the RCIC room which had been previously repaired, two in the northwest corner room (NWCR), and one in the SECR for which the licensee credited fortuitous fire watches in adjacent rooms as described above. The other instances included two failures to check the residual heat removal (RHR) valve room thermo-lag installations, and an additional failure to check a HPCI room to offgas recombiner room penetration which was later determined acceptable as described above.

After discussion with the personnel involved, several problems were noted. Hourly fire watches were given very little, if any, information as to what items must be checked in the rooms to which they were assigned. The licensee's fire protection program did not give details on hourly fire watch responsibilities, nor did the surveillance test procedure which initiates fire watches, or the licensee's training program for hourly fire watches. The checksheets, which had been developed as a result of previously missed fire watches, were not being used properly, in that personnel would observe several areas of a 1 hour tour before recording these on the checksheet. The number of areas required to be checked in a 1 hour period (about 75) also appeared excessive. Additionally, no clear guidance was given on how to deal with fire watches which could not be performed. In response to these concerns the licensee reduced the number of areas which a fire watch must cover in an hour, made provisions for contacting Operations Shift Supervisors in the event that a fire watch cannot be performed, and was working on improved guidance for fire watches.

The missed fire watches and log discrepancies will be further reviewed by the NRC in conjunction with items previously discussed in Temporary Instruction 2515/115, "Verification of Plant Records". These issues will be followed as an unresolved item (331/92022-01(DRP)).

No violations or deviations were identified in this area.

4. <u>Monthly Maintenance Observation (62703)</u>

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

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 was assigned to safety-related equipment maintenance which might affect system performance.

Portions of the following maintenance activities were observed and/or reviewed:

- RHR Minimum Flow Bypass Valve, MO-1935, spring pack replacement and operator diagnostic testing

- Reactor Recirculation M/G "A" speed controller repairs

- Control Building Chiller "A" and "B" troubleshooting

- Circulating Water Pump, 1P-4A, troubleshooting and repair

- Intermediate Range Neutron Monitor "D" troubleshooting

- Torus Hard Pipe Vent installation and testing

- Moisture Separator Reheater drain line, GBD-56, replacement

- Control Rod Drive Hydraulic Pump "B" seal repair

Control Building Chillers

On November 19, 1992, the licensee entered an administrative 72 hour limiting condition for operation based on the unavailability of both control building chillers. The "B" chiller was out of service for preventive maintenance when the "A" chiller tripped on low oil suction pressure, and could not be restarted. Technicians added approximately 5 to 10 gallons of oil after observing a low oil level, and were able to restart the chiller. The low oil level was the result of corrective maintenance performed earlier in November 1992 to repair a leaking fitting. During this maintenance, mechanics did not add an adequate amount of oil to the chiller to ensure that it would operate properly under all load conditions. In addition, operator logs had not previously indicated the need to check chiller oil levels, and no followup action was initiated to periodically check the chiller oil level subsequent to performing the maintenance activity.

The licensee intends to improve the auxiliary operator logs by including readings on oil level and pressure. The inspectors also discussed with the licensee the need to look for similar problems on log readings taken for other chillers, especially the offgas glycol chillers. The inspectors will follow the licensee's corrective actions during future routine inspections.

Following completion of maintenance on the RHR and Control Building Ventilation systems, the inspectors verified that these systems had been returned to service properly.

No violations or deviations were identified in this area.

5. Monthly Surveillance Observation (61726)

The inspectors observed TS required surveillance testing 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 TS 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.

The inspectors also witnessed portions of the following test activities:

SpTP-181 - MO-1935, RHR Minimum Flow Bypass Valve, Performance Testing in Response to Generic Letter 89-10.

SpTP-183 - MO-2010, RHR Cross Tie Valve, Performance Testing in Response to Generic Letter 89-10.

STP-41A006-Q - Scram Discharge Volume Level Switch Calibrations

No violations or deviations were identified in this area.

6. Design, Design Changes and Modifications (37700)

a. Containment Design Calculation

During an engineering review of calculations used for containment integrated leak rate testing, licensee engineers discovered an error in a 1971 calculation used to compute primary containment free volume. The calculation, performed by the architect engineer, failed to account for ten vertical feet of the containment's cylindrical portion. This resulted in a value of 8042 cubic feet (ft^3) less than the actual volume of about 117,482 ft^3.

Upon discovery of this error, the licensee wrote NCR 92-134 and reviewed possible design calculations and accident analyses which would be in question if the incorrect containment volumes were used. It was not immediately clear where this incorrect calculation had been used. The initial operability determination looked at containment spray, loss of coolant accidents, containment atmosphere dilution, leak rates, and other areas, and found that all areas reviewed were more conservative due to the larger containment size. Licensee and General Electric engineers are continuing to evaluate possible effects of this larger containment volume on pertinent design basis information. This issue will be tracked as unresolved item 331/92022-02(DRP) pending further licensee and NRC review.

b. <u>High Energy Line Break (HELB) Modeling</u>

A Quality Assurance surveillance reviewed engineering assumptions for HELB modeling of the HPCI and RCIC systems and found some discrepancies. In particular, an analysis of a steam line break in the HPCI and RCIC rooms did not evaluate conditions in the reactor building which could occur if the doors in either room, which open outward, were blown open. The doors had been assumed to hold shut, but no calculations had been performed to validate that assumption. The licensee wrote NCR 92-133 to document the conditions, then began evaluating the conditions in the reactor building, assuming a failure of the HPCI or RCIC doors during a steam line break.

An operability determination was issued on November 13, 1992, stating that although certain areas of the reactor building would experience a harsh environment, an isolation of the HPCI or RCIC steam line break events would still occur. This determination recognized the fact that some of the equipment relied on for isolation was not qualified for the harsh environment it would experience, but considered that the isolation function of the equipment would occur prior to degradation or failure.

The inspectors discussed the operability determination with licensee engineers and Region III equipment qualification inspectors and management. The inspectors determined that the licensee had not adequately determined the availability of equipment in the reactor building following the isolation of the break which might be necessary to maintain the plant in a safe shutdown condition.

The licensee further evaluated the postulated condition of equipment required to shut down the plant following a HPCI or RCIC steam line break, and determined that safe shutdown could be accomplished. This was documented in a supplement to the original operability determination on November 23, 1992. The inspectors requested further details regarding the status of instrumentation needed to bring the plant to safe shutdown and maintain it there.

The licensee and the inspectors are continuing to evaluate the operability considerations and hardware modifications needed to resolve this issue. The apparent lack of depth of the engineering evaluation and the apparent lack of consideration of safe shut down requirements and safe shut down instrumentation will be tracked as unresolved item 331/92022-03(DRP).

No violations or deviations were identified in this area.

7. Plant Trips (93702)

Following the plant trip on November 13, 1992, the inspectors ascertained the status of the reactor and safety systems by observation of control room indicators and discussions with licensee personnel concerning plant parameters, emergency system status, and reactor coolant chemistry. The inspectors verified the establishment of proper communications and reviewed the corrective actions taken by the licensee.

On November 13, 1992, the reactor automatically scrammed from a turbine trip at greater than 30 percent power. The turbine trip was caused by high condenser back pressure. The reactor was at approximately 100 percent power prior to the event. All engineered safety features responded as expected following the reactor scram, including Primary Containment Isolation Signal (PCIS) Groups II through V Isolations, resulting from a reactor vessel level shrink.

The high condenser back pressure was subsequently determined to be attributable to a separation of the circulating water pump bell housing from the casing assembly. Inspection of the pump identified that 28 of the 40 flange bolts on the pump bell housing had failed. These bolts had been replaced during the last refueling outage, which ended on April 24, 1992. The licensee has sent the failed bolts offsite for analysis to determine the cause of the failure.

The licensee conducted a short forced outage to isolate the "A" CW header, replace moisture separator reheater drain piping, install electromagnetic interference reducing chokes on reactor recirculation flow instruments, and other work. Operators commenced a reactor startup on November 16, 1992; and the reactor was declared critical with a 64 second period. The generator was paralleled to the grid later the same day. Initially, power ascension was limited by condenser back pressure and condensate demineralizer inlet temperatures due to only one CW pump being in service. Following a change in cooling tower lineups and an analysis allowing a reduction in CW discharge pressure, the licensee was able to increase reactor power up to 100 percent. The "A" CW pump and casing were sent to the manufacturer for repairs, and are expected to be returned in mid December 1992. Following the return of the pump, the licensee intends to perform a short outage to return the pump to service.

No violations or deviations were identified in this area.

8. Management Meetings (30702)

On November 12, 1992, the representatives of Iowa Electric Light and Power Company and the NRC held a public meeting to discuss the results of the Systematic Assessment of Licensee Performance (SALP-10) report. The meeting was held at the plant site with the Regional Administrator chairing the meeting.

9. Report Review (90713)

During the inspection period, the inspectors reviewed the licensee's Monthly Operating Report for October 1992. The inspectors confirmed that the information provided met the requirements of TS 6.11.1.C and Regulatory Guide 1.16.

No violations or deviations were identified in this area.

10. Unresolved Items

Unresolved items are matters about which more information is required in order to ascertain whether they are acceptable items, violations or deviations. Unresolved items disclosed during the inspection are discussed in Sections 3 and 6.

11. Exit Interview (30703)

The inspectors met with licensee representatives (denoted in Section 1) on December 3, 1992, and informally throughout the inspection period and summarized the scope and findings of the inspection activities. The inspectors also discussed the likely information 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. The licensee acknowledged the findings of the inspection.