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RGN-002

UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION III
801 WARRENVILLE ROAD
LISLE, ILLINOIS 60532-4351

March 07, 2000

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ANO

Mr. David Wilson
Vice President, Nuclear
IES Utilities, Inc.
Alliant Tower
200 First Street SE
P. O. Box 351
Cedar Rapids, IA 52406-0351

SUBJECT: DUANE ARNOLD INSPECTION REPORT 50-331/99015(DRP)

Dear Mr. Wilson:

This refers to the inspection conducted on December 22, 1999, through February 9, 2000, at the Duane Arnold Energy Center (DAEC) facility. The NRC conducted a routine safety inspection. The enclosed report presents the results of this inspection.

During this 6-week inspection period, your conduct of activities at the DAEC facility was characterized by safety conscious operations. As an example, licensee management conservatively decided to keep the unit shut down following an automatic scram to find the source of increasing drywell leakage.

However, based on the results of this inspection, the NRC has determined that two violations of NRC requirements occurred. One violation was for a failure to follow procedures when aligning the spent fuel pool cooling system. As a result, fuel pool temperature increased approximately 50 degrees Fahrenheit over the next 2 days. The second violation was for an inadequate surveillance test procedure that resulted in an unplanned automatic scram of the reactor.

These violations are being treated as Non-Cited Violations (NCVs), consistent with Section VII.B.1.a of the Enforcement Policy. These NCVs are described in the subject inspection report. If you contest the violation or severity level of these NCVs, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington DC 20555-0001, with copies to the Regional Administrator, Region III, and the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington, DC 20555-0001.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter, its enclosure, and your response, if you choose to provide one, will be placed in the NRC Public Document Room.

Sincerely,

/s/ D. Roberts for

Melvyn N. Leach, Chief
Reactor Projects Branch 2

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

Enclosure: Inspection Report 50-331/99015(DRP)

cc w/encl: E. Protsch, Executive Vice President -
Energy Delivery, Alliant;
President, IES Utilities, Inc.
Richard L. Anderson, Plant Manager
K. Peveler, Manager, Regulatory Performance
State Liaison Officer
Chairperson, Iowa Utilities Board
The Honorable Charles W. Larson, Jr.
Iowa State Representative

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In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter, its enclosure, and your response, if you choose to provide one, will be placed in the NRC Public Document Room.

Sincerely,

Darrell Roberts for

Melvyn N. Leach, Chief
Reactor Projects Branch 2

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

Enclosure: Inspection Report 50-331/99015(DRP)

cc w/encl: E. Protsch, Executive Vice President - Energy Delivery, Alliant; President, IES Utilities, Inc.
Richard L. Anderson, Plant Manager
K. Peveler, Manager, Regulatory Performance
State Liaison Officer
Chairperson, Iowa Utilities Board
The Honorable Charles W. Larson, Jr.
Iowa State Representative

U. S. NUCLEAR REGULATORY COMMISSION

REGION III

Docket No:	50-331
License No:	DPR-49
Report No:	50-331/99015(DRP)
Licensee:	Alliant, IES Utilities, Inc. 200 First Street S.E. P. O. Box 351 Cedar Rapids, IA 52406-0351
Facility:	Duane Arnold Energy Center
Location:	Palo, Iowa
Dates:	December 22, 1999, through February 9, 2000
Inspectors:	P. Prescott, Senior Resident Inspector M. Kurth, Resident Inspector
Approved by:	M. N. Leach, Chief Reactor Projects Branch 2 Division of Reactor Projects

EXECUTIVE SUMMARY

Duane Arnold Energy Center
NRC Inspection Report 50-331/99015(DRP)

This inspection report included the resident inspectors' evaluations of aspects of licensee operations, engineering, maintenance, and plant support.

Operations

- The inspectors noted that the operators responded appropriately to an automatic reactor scram. Several equipment problems occurred during the scram and challenged the operators. These problems were properly addressed prior to startup. The inspectors found the licensee's review of the scram adequate prior to restarting the plant (Section O1.1).

- Operators performed startup activities in a controlled and deliberate manner. The inspectors noted that operators responded appropriately when a control rod double-notched and when several control rods would not withdraw using normal control rod drive pressure (Section O1.2).

- A Non-Cited Violation was identified for auxiliary operators failing to follow the procedure for placing the "A" spent fuel pool cooling train in service. This resulted in spent fuel pool temperatures approaching the safety operating limit of the spent fuel pool system. Also, the licensee did not have adequate barriers in place to alert operators when the spent fuel pool temperature increased (Section O1.3).

Maintenance

- In general, maintenance activities and surveillance tests were conducted in an acceptable manner. However, the inspectors identified mechanics prematurely working in a posted contaminated area while the responsible radiation technician exited to analyze contamination wipes taken during maintenance on the "A" standby gas treatment system. The radiation protection technician subsequently determined the area was not contaminated (Section M1.1).

- During surveillance testing of the remote shutdown panel level transmitter, I&C technicians inadvertently initiated an automatic reactor scram. A Non-Cited Violation was identified for an inadequate surveillance test procedure. Failure to adequately self-check on the part of the technicians was another contributing factor to the scram (Section M1.2).

- Mechanics identified that the low pressure coolant injection check valve's seal gasket developed a leak due to a slight high spot on the valve body's pressure seal mating surface. The high spot was removed. Mechanics properly implemented foreign material controls to prevent the intrusion of tools or parts into the normally closed low pressure coolant injection piping system (Section M1.3).

Engineering

- The licensee's Year 2000 readiness review program was shown to have been effective based on the uneventful roll-over period observed by the inspectors (Section E1.1).

- The inspectors concluded that the revised or deleted preventive maintenance activities reviewed had adequate technical basis and documentation to support the changes. No problems were identified (Section E2.1).

- The inspectors verified that the licensee effectively implemented its oil analysis program to determine component wear rates and lubricant condition. However, the inspectors were concerned with the number of oil samples that were taken from components located in the reactor building and the turbine building that were radiologically contaminated, which prevented the oil analysis from being performed. The licensee established short-term and long-term corrective actions to either prevent sample contamination or allow contaminated samples to be analyzed (Section E2.2).

Plant Support

- Radiation protection personnel performed a thorough pre-job brief in support of a maintenance activity in a locked high radiation area and contamination area. Maintenance workers made efficient use of their time, thereby minimizing their accumulated dose (Section R1.1).

Report Details

Summary of Plant Status

The licensee operated the plant at 100 percent power at the beginning of this inspection period.

On January 5, 2000, with the plant operating at 100 percent power, an automatic scram occurred when instrumentation and control technicians attempted to restore a level transmitter to service after a calibration. All control rods inserted into the core. Primary containment isolations (Groups 2, 3, and 4) occurred as expected. An unexpected partial Group 5 isolation also occurred. Reactor water level control following the scram was complicated by a positive bias on a feedwater regulating valve and a reactor recirculation pump motor generator run-back failure, but vessel level and pressure were maintained within safe operating limits. No emergency core cooling systems actuated and no safety relief valves opened.

Operators placed the mode switch in shutdown on January 5, at 11:39 p.m., and the licensee commenced a maintenance outage. On January 6, the licensee identified the source of increased drywell leakage noted prior to the scram. A body-to-bonnet leak was found on the "B" residual heat removal injection check valve, V19-149. On January 6, at 5:09 p.m., operators placed the mode switch in Mode 4, cold shutdown, to initiate repairs to V19-149. On January 8, repairs to V19-149 were completed and the drywell close-out inspection performed. On January 9, shutdown cooling was secured and Mode 3 was entered at 10:56 a.m. Operators took the plant critical at 2:29 p.m. On January 10, at 9:39 a.m., the generator was synchronized to the grid. On January 12, at 4:40 a.m., the reactor was essentially at full power. The licensee operated the plant at 100 percent power for the remainder of the inspection period.

I. Operations

O1 Conduct of Operations

O1.1 Operator Response to Reactor Scram

a. Inspection Scope (71707 and 93702)

The inspectors observed the majority of the on-shift operators' response to an automatic scram. The following documents were reviewed.

- Operator and Alarm Logs
- Integrated Plant Operating Instructions (IPOI) 5, Attachment 1, "Scram Report," Revision 24
- "Licensing Department Post Scram Review and Report," Revision 2

The inspectors also attended the scram fact-finding and startup review committee meetings.

b. <u>Observations and Findings</u>

On January 5, 2000, with the plant at 100 percent power, an automatic reactor scram occurred. Instrumentation and control technicians were conducting a calibration on the remote shutdown panel wide range (flood-up) reactor vessel level instrument LT4541. The scram signal resulted from a false low level signal caused by an inadequate instrument restoration (see Section M1.2 for details). This resulted in a level transient for the "A" and "B" reactor vessel narrow range level instruments, which share a common reference leg. The sensed level transient resulted in a reactor scram on vessel low level (170 inches). Actual reactor water level dropped to approximately 132.5 inches. Following the reactor scram, reactor water level swelled to 218 inches, which caused a turbine trip on high level (211 inches). Operators tripped one of the two feed pumps in anticipation of a high level. Reactor water level then returned to the normal level of 190 inches. The high level turbine trip was caused by the reactor feedwater regulating valve not going fully closed. The licensee determined that this was caused by a flow bias signal error associated with the feedwater regulating valve controller. Operators did not expect this response. To prevent repetition of the unexpected feedwater system response, engineering personnel revised the feedwater regulating valve controller setting.

The "A" reactor recirculation pump did not run to the 20 percent minimum speed setpoint. This should have occurred due to feedwater flow less than 20 percent. Operators manually secured the "A" reactor recirculation pump. Subsequent troubleshooting found that a timer relay had failed.

Primary containment isolations for Group 2 (radwaste and drywell drain systems), Group 3 (containment atmosphere control system), and Group 4 (shutdown cooling) occurred as expected. An unexpected partial outboard only Group 5 (reactor water cleanup) isolation, which normally occurs at 119.5 inches, was also initiated. Control room operators took the reactor water cleanup system isolation to completion. Operations personnel responsible for reviewing the scram had checked the latest level switch calibration data. The two switches that actuated the outboard isolation had as-left trip setpoints of 122.56 and 126 inches. The as-left trip setpoints were within the Technical Specification requirements. The operations personnel concluded this was close to the lowest recorded level of 132.5 inches. The inspectors questioned if instrument inaccuracies had been considered to support this conclusion. The licensee subsequently reviewed this information and noted that when instrumentation tolerances were considered, potential overlap did exist.

Other minor equipment distractions occurred while the operators responded to the scram. During the post trip panel walk-downs, operators noted annunciators A4 and B4, "Drywell High Pressure Loop "A" ("B") Fans to Slow Speed," were flashing. The fans did not change speed. The pressure switches that directly actuate the annunciators are set to trip at 2 pounds per square inch gauge drywell pressure. There is no logic tie between the level instrumentation that initiated the scram and the drywell pressure switches. The licensee determined this was probably a "sympathy" annunciator due to induced current over signal cabling. Annunciator D2, "Fuel Pool Cooling Panel 1C-65/1C-66 Trouble," on panel 1C04B alarmed during the scram. An auxiliary operator found the "A" fuel pool demineralizer isolated on high differential pressure, which deadheaded the running "A" fuel pool cooling pump. The operator secured the pump. A problem was later found with a relay that failed on the "B" fuel pool demineralizer pre-coat logic, which caused the isolation. The licensee determined this was not related to the scram. The "B" intermediate range monitor was erratic when inserted into the core following the scram. The detector was replaced prior to startup.

c. Conclusions

The inspectors noted that the operators responded appropriately to an automatic reactor scram. Several equipment problems occurred during the scram and challenged the operators. These problems were properly addressed prior to startup. The inspectors found the licensee's review of the scram adequate prior to restarting the plant.

O1.2 Reactor Startup and Return to Full Power Operation

a. Inspection Scope (71707)

From January 9 through 12, 2000, the inspectors observed portions of the reactor startup, approach to criticality, and return to full power operation. Documents reviewed included the following:

- Integrated Plant Operating Instruction (IPOI) 2, "Startup," Revision 59
- Abnormal Operating Procedure (AOP) 255.1, "Control Rod Movement/Indication Abnormal," Revision 19

b. Observations and Findings

In accordance with IPOI 2, on January 9, 2000, operators commenced control rod withdrawals in a controlled and deliberate manner. Operators appropriately responded when control rod 18-31 double-notched from position 36 to 40. Control rod movement activities were stopped and the occurrence was evaluated by shift management and the reactor engineer. Control rod movement activities were allowed to continue. Also, several control rods did not initially withdraw using the normal control rod drive pressure of 250 pounds per square inch differential (psid). The operators followed the instructions in AOP 255.1 and raised drive pressure approximately 50 psid, which allowed the control rods to move. Control rod drive pressure was returned to 250 psid per the procedure.

c. Conclusions

Operators performed startup activities in a controlled and deliberate manner. The inspectors noted that operators responded appropriately when a control rod double-notched and when several control rods would not withdraw using normal control rod drive pressure.

O1.3 Fuel Pool Temperature Rise

a. Inspection Scope (71707)

The inspectors reviewed the events that led to an unexpected increase of the spent fuel pool cooling water temperature. Discussions were held with the operators involved in the event and the system engineer. The licensee's fact-finding meeting was attended. The following documents were reviewed:

- Operations Instruction (OI) 435, "Fuel Pool Cooling System," Revision 27
- Updated Final Safety Analysis Report (UFSAR)
- General Electric Design Document Number 22A1423, "Fuel Pool Cooling and Cleanup System"

b. Observations and Findings

On January 13, 2000, a control room operator noted that the control room recorder for spent fuel pool cooling water temperature indicated 141 degrees Fahrenheit (°F), vice the normal temperature band of approximately 90°F to 100°F. Subsequently, an auxiliary operator found the reactor building closed cooling water (RBCCW) outlet valve V-12-33 closed. The valve was not throttled, as required, for allowing cooling water to the "A" fuel pool cooling heat exchanger. The auxiliary operator opened the RBCCW valve and within 24 hours spent fuel pool temperature returned to normal.

On January 11, two auxiliary operators had been tasked with swapping fuel pool cooling pumps, in accordance with OI 435, in support of maintenance activities. The auxiliary operators had cleared the tag-out on the "A" fuel pool cooling pump and were in the process of lining up the "A" fuel pool cooling train. One of the auxiliary operators read the procedure steps as the other operator performed the task. The senior auxiliary operator reading the procedure assumed that RBCCW was lined up to the "A" fuel pool cooling heat exchanger from flushing activities that had occurred earlier. Therefore, the position of V-12-33 was never visually checked. The auxiliary operator performing the valve manipulations assumed the senior auxiliary operator knew from experience the correct position of V-12-33. The OI 435 stated to close the RBCCW isolation on the idle heat exchanger.

The 10 CFR Part 50, Appendix B, Criterion V, requires, in part, that activities affecting quality shall be prescribed by documented procedures and shall be accomplished in accordance with these procedures. Operating Instruction 435, "Fuel Pool Cooling System," Step 4.2.(4)(a), stated, "Line up RBCCW to the fuel pool heat exchangers as follows: On the operating heat exchanger, throttle open fuel pool heat exchanger 1E-211A RBCCW outlet isolation V-12-33 one-quarter turn." Contrary to the above, on January 11, an auxiliary operator failed to perform Step 4.2.(4)(a) when placing the "A" spent fuel pool train into service. The Severity Level IV violation (50-331/99015-01 (DRP)) is being treated as a Non-Cited Violation consistent with Section VII.B.1.a of the NRC Enforcement Policy. This violation is entered in the licensee's corrective action program as Action Request 18515.

The inspectors noted the high temperatures did not exceed the UFSAR safety operating limit of 150°F for the fuel pool cooling system. However, several problems were evident from this event. No alarm existed for high temperature of fuel pool cooling water. As part of the corrective actions, the licensee subsequently started to log fuel pool cooling water temperature on a daily basis and changed the computer points of the two outlet and inlet heat exchanger probes to alarm on high temperature. Operations management was tasked to review the event for lessons learned. This included the failure to adequately peer check and to get supervisory approval prior to skipping a procedure step.

c. Conclusions

A Non-Cited Violation was identified for auxiliary operators failing to follow the procedure for placing the "A" spent fuel pool cooling train in service. This resulted in spent fuel pool temperatures approaching the safety operating limit of the spent fuel pool system. Also, the licensee did not have adequate barriers in place to alert operators when the spent fuel pool temperature increased.

O8 Miscellaneous Operations Issues (92901)

O8.1 (Closed) Licensee Event Report (LER) 50-331/99-004-00: Entry into Technical Specification (TS) 3.0.3 Due to Both Trains of Standby Gas Treatment (SBGT) Inoperable Caused by a Surveillance Test Coincident with a Component Failure. During monthly surveillance testing of the "B" SBGT train, maintenance technicians identified that the "A" SBGT train flow indicating controller, FIC 5828, was not indicating flow properly. The cause of the failure was found to be a capacitor in the low voltage power supply. This capacitor, last replaced in January 1996, had a 10-year replacement preventive maintenance task. This was in agreement with the vendor's recommendation. The licensee concluded that the capacitor failure was a random occurrence.

The monthly "B" train SBGT surveillance test required the operator to move the SBGT mode selector switch from manual to the auto position for normal standby mode. Manipulating the mode switch to manual at the completion of the surveillance test led to the "B" SBGT train being inoperable for approximately 15 seconds, which when coupled with the capacitor failure on the "A" SBGT train, created a momentary loss of safety function and entry into the limiting condition of operation TS 3.0.3.

Failure of the "A" SBGT train flow indicator appeared random. The inspectors did not identify any other recent similar failures. Operators entered the appropriate TS requirements. The indicator was subsequently replaced. This item is closed.

II. Maintenance

M1 Conduct of Maintenance

M1.1 General Comments

a. Inspection Scope (62707 and 61726)

The inspectors observed all or portions of the surveillance test activities and work request activities listed below. The applicable surveillance test or work package documentation was reviewed. Specific tests and work request activities observed are listed below:

Maintenance Activities

- Corrective Work Order (CWO) A45941: Inspect and replace body-to-bonnet gasket on residual heat removal low pressure coolant injection (LPCI) inject loop "B" check valve, V19-0149
- CWO A46203: Replace IRM detector "B"
- Preventive Work Order (PWO) 1110969: Inspect "A" accumulator on the standby liquid control (SBLC) system
- PWO 1110970: Inspect "B" accumulator on the SBLC system
- PWO 1111005: Inspect/replace filters on the "A" SBGT system
- PWO 1111097: Calibrate electric coil differential temperature transmitter, TDT5805A, on the "A" SBGT system

Surveillance Test Activities

- Surveillance Test Procedure (STP) 3.5.1-07, "HPCI [High Pressure Coolant Injection] System Simulated Automatic Actuation," Revision 4
- STP 3.5.3-04, "RCIC [Reactor Core Isolation Cooling] Simulated Auto Actuation Test," Revision 4
- STP NS 340004, "Safeguards Systems Area Cooling Logic System Functional Test (HPCI/RCIC)," Revision 0

b. Observations and Findings

In general, the work associated with these activities was conducted in a professional and thorough manner. Technicians were knowledgeable of their assigned tasks and work document requirements. However, instrumentation and control technicians initiated an unexpected automatic reactor scram while performing a reactor water level instrumentation calibration surveillance (see Section M1.2 for details). Also, the inspectors identified mechanics prematurely working in a posted potentially contaminated area while the responsible radiation protection technician exited to analyze contamination wipes taken during maintenance on the "A" SBGT system. The radiation protection technician subsequently determined that the area was not contaminated.

c. Conclusions

In general, maintenance activities and surveillance tests were conducted in an acceptable manner. However, the inspectors identified mechanics prematurely working in a posted potential contaminated area while the responsible radiation technician exited to analyze contamination wipes taken during maintenance on the "A" SBGT system. The radiation protection technician determined the area was not contaminated.

M1.2 Calibration of Reactor Water Level Instrumentation

a. Inspection Scope (62707)

The inspectors reviewed the maintenance activities that resulted in an automatic reactor scram and subsequent licensee management response. The following documents and procedures were reviewed.

- STP 3.3.3.2-01, "Remote Shutdown System Instrument Calibration," Revision 3
- Equipment Maintenance Procedure I.PDT-G080-1, "GE [General Electric] Differential Pressure Transmitters," Revision 9
- "Licensing Department Post Scram Review and Report," Revision 2

The inspectors also attended the scram fact-finding and startup review committee meetings.

b. Observations and Findings

Instrumentation and control (I&C) technicians were performing STP 3.3.3.2-01. The purpose of the STP was to calibrate remote shutdown panel instrumentation and to verify control circuits associated with reactor pressure and level could perform their intended function. Specifically, I&C technicians were restoring the wide range (flood-up) reactor vessel level instrument LT4541 following calibration of the transmitter. Although the level transmitter was valved in slowly, the procedure did not require it to be back-filled, vented, or pressurized first. This caused a pressure pulse in the sensing line that resulted in a sensed low level in reactor water level instrumentation. The sensed level was less than 170 inches, which resulted in the low reactor water level scram (see Section O1.1 for details). The level transmitter LT4541 shared a common instrumentation line with other transmitters that sensed a low level and actuated the scram.

There were several missed opportunities to identify the procedure weaknesses. During activities to implement Improved Technical Specifications in June 1998, the flood-up reactor level transmitter calibration was added to STP 3.3.3.2-01. The STP had insufficient detail to return the instrument to service properly. The decision was also made to calibrate the instrument on-line. During the STP review process, the validation and walk-down checks did not identify the procedure deficiency. The revised test was first performed in July 1998, with the plant on-line. However, the I&C technicians had back-filled the instrument line prior to valving in the level transmitter. No request to incorporate back-filling into the procedure was done. A preventive work order was performed on LT4541 during the last refueling outage. The licensee's preventive maintenance program missed the fact that a preventive work order and a surveillance test calibrated the same instrument. The pre-job brief prior to performing STP 3.3.3.2-01 did not address that this instrument shared a common line with other level transmitters that could actuate a scram, the sensitivity of the instrument rack, or if back-filling was needed.

Procedure 3.3.3.2-01 was subsequently revised. Criterion V of 10 CFR Part 50, Appendix B, requires, in part, that activities affecting quality shall be prescribed by documented instructions or procedures of a type appropriate to the circumstances. Contrary to the above, STP 3.3.3.2-01, "Remote Shutdown System Instrument Calibration," Revision 3, did not incorporate instructions to back-fill, vent, and properly valve in the flood-up reactor water level transmitter LT4541. This Severity Level IV violation (50-331/99015-02 (DRP)) is being treated as a Non-Cited Violation consistent with Section VII.B.1.a of the NRC Enforcement Policy. This violation is entered in the licensee's corrective action program as Action Request 18262.

Project engineers performed an evaluation to verify that the I&C technicians valving in LT4541 actually caused the scram on low reactor water level. The inspectors found the project engineers' assumptions appropriate that void space in the line and the immediate volume increase (by opening the valve) could have resulted in a pressure drop that corresponded to the approximate water level decrease sensed by the other level transmitters on the same instrumentation line.

c. Conclusions

During surveillance testing of the remote shutdown panel level transmitter, I&C technicians inadvertently initiated an automatic reactor scram. A Non-Cited Violation was identified for an inadequate surveillance test procedure. Failure to adequately self-check on the part of the technicians was another contributing factor to the scram.

M1.3 Identifying and Repairing Drywell Unidentified Leakage Source

a. Inspection Scope (62707)

The inspectors reviewed the licensee actions to identify and repair the source of increasing drywell unidentified leakage. Discussions were held with operations, engineering, and maintenance personnel involved. The following documents were reviewed:

- Action Request 18251, "Increase in Drywell Unidentified Leakage"
- Work Request A45941, "Inspect and Replace Body-to-Bonnet Gasket"
- Administrative Control Procedure 1408.20, "Foreign Material Control," Revision 0

b. Observations and Findings

On January 3, 2000, the licensee initiated Action Request 18251 to address an increase in drywell unidentified leakage. Normal leakage was 0.05 gallons per minute (gpm); however, the leakage had gradually increased to 0.24 gpm since mid-December and continued to trend slowly upward. The licensee toured the drywell during the recent forced outage and identified a body-to-bonnet pressure seal gasket leak on the low pressure coolant injection (LPCI) Loop B Injection Check Valve V19-0149. Work Order A45941 was initiated and the maintenance shop planned the pressure seal gasket replacement.

During valve disassembly, the licensee identified that the valve body's pressure seal mating surface was slightly uneven and a steam cut had formed in the pressure seal gasket at that location. The mating surface had a high spot of 0.003 inches. The high spot was sanded down to within 0.001 inches.

The inspectors observed mechanics implementing foreign material control practices to prevent the intrusion of tools or parts into the normally closed LPCI piping system. Quality control technicians inspected the system opening prior to closure to ensure no foreign materials were present. Also, direct radiation protection coverage was provided during the system breech to ensure unknown radiological conditions were evaluated prior to mechanics continuing their work. The valve was reassembled and hydrostatically post-maintenance tested satisfactorily.

c. Conclusions

Mechanics identified that the low pressure coolant injection check valve's seal gasket developed a leak due to a slight high spot on the valve body's pressure seal mating surface. The high spot was removed. Mechanics properly implemented foreign material controls to prevent the intrusion of tools or parts into the normally closed LPCI piping system.

III. Engineering

E1 Conduct of Engineering

E1.1 Licensee's Effectiveness of Year 2000 (Y2K) Preparations during Y2K Rollover

a. Inspection Scope (Contingency Plan Implementing Procedure (CPIP) 500 and 71707)

The inspectors followed the guidance in CPIP 500 for observing licensee activities during the Y2K rollover. Communication with the Regional incident response center was verified on a routine basis and the status of various plant systems was also relayed at pre-arranged times during the night. The inspectors observed control room on-shift operations activities. Also, Technical Support Center activities were observed, which included aspects of Y2K management planning, implementation planning, and initial assessment of information received from multiple sources of problems encountered at other plants earlier in the evening due to time zone differences.

b. Observations and Findings

The licensee's program for Y2K plant readiness was effective. There were no unanticipated plant problems noted due to the effects of Y2K software-induced problems.

c. Conclusions

The licensee's Y2K readiness review program was shown to have been effective based on the uneventful roll-over period observed by the inspectors.

E2 Engineering Support of Facilities and Equipment

E2.1 Preventive Maintenance (PM) Optimization Program Review

a. Inspection Scope (37551 and 62707)

The inspectors reviewed a sample of PM activities that were deleted (12), added (6), or revised (5), by the responsible system engineers for plant equipment that was safety-related or important to safety. The review focused on the technical justification for requesting the changes documented on the "PM Input Request Form." The inspectors also reviewed Administrative Control Procedure 1208.3, "Preventive Maintenance Program," Revision 5. Discussions were held with the project engineers responsible for the PM optimization program.

b. Observations and Findings

The inspectors conducted a review of the PM optimization program based on findings of a radiation protection inspection (see Inspection Report 50-331/99008(DRS)). In that inspection report, the inspectors questioned the practice of discontinuing the calibration of area radiation monitors referenced in the UFSAR without initiating a UFSAR change. Also, there was insufficient technical justification for discontinuing the practice of calibrating the area radiation monitors. The licensee subsequently reviewed its practices and made programmatic changes to address this problem.

System engineers documented on a "PM Input Request Form," the reason for changing or deleting a preventive maintenance task. The inspectors found that the technical justifications to delete, add, or revise a preventive maintenance task, as documented on the "PM Input Request Forms" reviewed, had sufficient justification to warrant the changes. The questions on the form covered a wide spectrum of possible reasons for the change. Examples of form questions for the system engineer to consider prior to initiating a change to the PM included: (1) were vendor recommendations reviewed; (2) was performance trending of the equipment or component, grouping of tasks to optimize equipment out-of-service time reviewed, and, (3) was the impact on other programs (e.g., vibration, oil, environmental qualification), and industry experience considered? System engineers were required to reference the source for the change or provide technical justification on the form.

Generally, the system engineer initiated the PM change. By procedure, the PM change was also reviewed by the respective component engineer and the PM coordinator. There appeared to be sufficient checks prior to instituting the PM change.

c. Conclusions

The inspectors concluded that the revised or deleted PM activities reviewed had adequate technical basis and documentation to support the changes. No problems were identified.

E2.2 Predictive Maintenance Program - Oil Analysis

a. Inspection Scope (37551)

The inspectors evaluated the implementation of the licensee's oil analysis predictive maintenance program. Interviews were held with maintenance, engineering, and chemistry personnel. The following documents were reviewed:

- General Maintenance Procedure GMP-Test-56, "Oil Samples - General," Revision 1
- Plant Chemistry Procedures PCP 4.33, "Oil Analysis," Revision 12
- Maintenance Directive MD-045, "Rotating Equipment Master Lube List," Revision 6

b. Observations and Findings

The oil analysis program was overseen by a responsible program owner and was managed by the program engineering group. Periodic sampling frequencies were established based on the significance and historical condition of plant equipment and components. Samples were analyzed onsite by chemistry technicians. Also, samples were sent offsite for analysis by an independent laboratory on an annual basis. Samples were analyzed for particulate size and amount, ferrous and nonferrous counts, viscosity, and analyzed (by the independent laboratory) for wear metal composition and amount. Sample analysis information was available to site personnel using an equipment monitoring database. Suspect results were identified and trended using an easy-to-read color-coded matrix.

The inspectors identified that, within the last year, a number of samples taken from components in the reactor building and the turbine building were not analyzed due to positive indications of radioisotope contamination. The onsite laboratory analysis equipment was located outside the reactor building and the turbine building; therefore, the radiologically contaminated samples were not analyzed. The inspectors were concerned that the oil analysis program was not being fully implemented due to samples that were radiologically contaminated. The inspectors noted that sampling instructions did not provide sufficient precautions to prevent contamination for samples that were traditionally analyzed to environmental levels; therefore, trace amounts of radioisotopes contaminated the samples. The inspectors observed the collection of oil samples in the reactor building and noted that used rags, gloves, and oil samples were placed in the same bag, thereby increasing the probability of low-level (environmental) sample contamination.

Action Request 18412 was initiated to evaluate the adequacy of the guidance provided in GMP-Test-056 to prevent the radiological contamination of samples. Also, the licensee has implemented long-term corrective actions to locate oil analysis equipment in the chemistry laboratory that will allow the analysis of radiologically contaminated oil samples.

c. Conclusion

The inspectors verified that the licensee effectively implemented its oil analysis program to determine component wear rates and lubricant condition. However, the inspectors were concerned with the number of oil samples that were taken from components located in the reactor building and the turbine building that were radiologically contaminated, which prevented the oil analysis from being performed. The licensee established short-term and long-term corrective actions to either prevent sample contamination or allow contaminated samples to be analyzed.

E8 Miscellaneous Engineering Issues (92903)

E8.1 (Closed) LER 50-331/99-005-00: Actuation of Engineered Safety Feature, the Standby Diesel Generator (SBDG), Due to a Lightning Strike. On October 29, 1999, with the plant in a refueling outage, the "A" SBDG automatically started but was not required to load. The cause of the automatic start was a momentary under-voltage condition sensed by the 1A3 essential bus under-voltage relay that initiated the "A" SBDG start logic. The momentary under-voltage condition was caused by a lightning strike that induced a voltage transient on the 161KV switchyard. After performing an operating checklist, the "A" SBDG was secured and returned to standby readiness mode.

The under-voltage relays are normally energized and trip on reduced voltage. Since the transient was a voltage reduction, no equipment limits were exceeded, and the under-voltage relays operated as designed. Switchyard equipment functioned reliably as confirmed by instrument data. The inspectors questioned if the licensee had performed a walk-down of potentially affected equipment. Engineering personnel subsequently inspected switchyard and plant equipment and verified there was no damage and that equipment was operating properly. This item is closed.

IV. Plant Support

R1 Radiological Protection and Chemistry Controls

R1.1 Radiation Protection Support for Maintenance Work in a Locked High Radiation Area and Contaminated Area

a. Inspection Scope (71750)

The inspectors assessed the adequacy of radiation protection work practices during maintenance work on the radioactive waste sludge tank 1T62A, which is located in a locked high radiation and contaminated area. The inspectors attended the pre-job brief and observed portions of the maintenance work. Radiation Work Permit 29, Job Step 7, was used to provide instructions to workers performing the maintenance.

b. Observations and Findings

A radiation protection technician performed a thorough pre-job brief in support of maintenance activities on 1T62A. The pre-job brief included current dose rate and contamination level information. Also, hold points were established if problems were encountered during the work. Maintenance workers were attentive and clearly explained their job task, which allowed radiation workers to develop potential hold points.

The inspectors observed maintenance workers wearing the proper contamination clothing to minimize personnel contamination. Workers made use of low dose areas to minimize their dose. A radiation worker periodically monitored dose rates and individuals' accumulated dose to minimize personnel dose. The workers made efficient use of their time thereby minimizing their accumulated dose.

c. Conclusions

Radiation protection personnel performed a thorough pre-job brief in support of a maintenance activity in a locked high radiation area and contamination area. Maintenance workers made efficient use of their time, thereby minimizing their accumulated dose.

V. Management Meetings

X1 Exit Meeting Summary

The inspectors presented the inspection results to members of licensee management at the conclusion of the inspection on February 9, 2000. The licensee acknowledged the findings presented. The inspectors asked the licensee whether any materials examined during the inspection should be considered proprietary. No proprietary information was identified.

PARTIAL LIST OF PERSONS CONTACTED

Licensee

R. Anderson, Plant Manager
J. Bjorseth, Maintenance Superintendent
D. Curtland, Operations Manager
R. Hite, Manager, Radiation Protection
M. McDermott, Manager, Engineering
K. Peveler, Manager, Regulatory Performance
G. Van Middlesworth, Site General Manager
D. Wilson, Vice President Nuclear

INSPECTION PROCEDURES USED

IP 37551:	Onsite Engineering
IP 61726:	Surveillance Observation
IP 62707:	Maintenance Observation
IP 71707:	Plant Operations
IP 71750:	Plant Support
IP 92901:	Followup - Operations
IP 92903:	Followup - Engineering
IP 93702:	Onsite Response to Events
CPIP500:	Y2K Contingency Plan Implementing Procedure

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

50-331/99015-01	NCV	Failure to follow procedure when placing spent fuel pool train into service
50-331/99015-02	NCV	Inadequate procedure for valving in reactor vessel level transmitter

Closed

50-331/99004-00	LER	Entry into TS 3.0.3 due to both trains of SBGT being inoperable
50-331/99005-00	LER	Actuation of engineered safety feature, the SBDG, due to a lightning strike
50-331/99015-01	NCV	Failure to follow procedure when placing spent fuel pool train into service
50-331/99015-02	NCV	Inadequate procedure for valving in reactor vessel level transmitter

Discussed

None

LIST OF ACRONYMS USED

AOP	Abnormal Operating Procedure
AR	Action Request
CFR	Code of Federal Regulations
CPIP	Contingency Plan Implementing Procedure
CWO	Corrective Work Order
DAEC	Duane Arnold Energy Center
DRP	Division of Reactor Projects
gpm	Gallons per minute
HPCI	High pressure coolant injection
I&C	Instrument and calibration
IP	Inspection procedure
IPOI	Integrated Plant Operating Instructions
LER	Licensee Event Report
LPCI	Low pressure coolant injection
NCV	Non-cited violation
NRC	Nuclear Regulatory Commission
OI	Operating Instruction
PM	Preventive maintenance
psid	Pounds per square inch differential
PWO	Preventive Work Order
RBCCW	Reactor building closed cooling water
RCIC	Reactor core isolation cooling
SBDG	Standby diesel generator
SBGT	Standby gas treatment system
SBLC	Standby liquid control
STP	Surveillance Test Procedure
TS	Technical Specification
UFSAR	Updated Final Safety Analysis Report
Y2K	Year 2000