U. S. NUCLEAR REGULATORY COMMISSION

REGION III

Docket No:	50-331
License No:	DPR-49
Report No:	50-331/99014(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:	November 11 through December 21, 1999
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 50-331/99014(DRP)

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

Operations

- Operators performed an error-free startup in a controlled and deliberate manner. The inspectors noted that operators responded appropriately when a maintenance activity caused a reactor pressure and water level perturbation (Section O1.1).

- The inspectors noted improved performance from the last refueling outage for the licensee's closeout of the primary containment drywell. The inspectors identified fibrous materials used in drywell penetrations that Nuclear Reactor Regulation inspectors will review and assess in their plant evaluation report on the licensee's response to Bulletin 96-03 (Section 02.1).

- The inspectors identified that licensee personnel needed additional guidance for using the Action Request system and the Work Request Card system. This was determined when inspectors identified suspect wiring to a relief valve bellows pressure sensing monitor. Confusion existed on whether to use the Action Request system or the Work Request Card system for evaluation and resolution (Section 02.1).

Maintenance

- Overall, improvement was noted in the conduct of maintenance activities from the previous refueling outage. However, several minor personnel performance issues occurred during the latter half of the refueling outage due to inattention to detail. Also, the weld overlay work for the reactor recirculation jet pump inlet nozzles caused damage to surrounding equipment that challenged plant operators during the startup. Licensee performance in planning and scheduling improved since the last outage (Section M1.1).

- The inspectors found that the licensee had followed an approved methodology for troubleshooting electrical grounds in the wiring for main steam line relief valve PSV 4402, which spuriously lifted during troubleshooting. However, the licensee's corrective actions to properly re-install wiring pulled from the junction box supplying power to relief valve PSV 4402 prior to the relief valve lifting was inadequate. Electrical maintenance technicians failed to examine wiring inside the junction box or conduit box (Section M4.1).

Engineering

- In general, the significant emergent work item of the refueling outage of planning and performing the weld overlay for the reactor recirculation risers was conducted well. However, there were problems in positioning of the weld overlay for the N2B riser due to drawings that had not received an adequate engineering review (Section E2.1).

- The inspectors determined that due to a lack of attention to detail, the engineering staff did not adequately address high pressure coolant injection operability concerns using the Action Request system (Section E2.2).

Plant Support

- Workers effectively minimized their accumulated dose and minimized the spread of contamination while performing maintenance on the crack arrest verification system (Section R1.1)

Report Details

Summary of Plant Status

The plant was in Refueling Outage 16 at the beginning of the inspection report period. On November 27, 1999, at 2:59 p.m., operators placed the mode switch in startup/hot standby. Control rod withdrawal commenced at 3:28 p.m. Criticality was reached at 5:31 p.m. Operators subsequently identified that the "A" intermediate range monitor was not responding and declared it inoperable. At 10:35 p.m., main steam line relief valve PSV 4402 opened while electrical maintenance technicians were troubleshooting a 125 volt direct current (VDC) system Division I ground. Operators halted the troubleshooting and the valve closed. This evolution caused reactor vessel level to swell to approximately 200 inches while the relief valve was open, and drop to 178 inches when the valve closed. Reactor pressure and level were quickly restored to normal. A subsequent fact-finding meeting was held and licensee management determined that it was acceptable to continue the startup.

On November 28, at 9:10 a.m., operators re-commenced control rod withdrawal. At 11:11 a.m., while attempting to cycle relief valve PSV 4402 during the relief valve surveillance testing, the valve failed to open. The relief valve was declared inoperable and the operators entered appropriate limiting conditions for operation. The remaining relief valves were tested satisfactorily. Licensee management decided to shut down the plant to make the necessary repairs to the relief valve and the "A" intermediate range monitor. The reactor was manually scrammed at 2:22 p.m.

On November 29, at 4:31 p.m., operators took the mode switch to startup/hot standby. Control rod withdrawal started at 7:18 p.m. At 9:12 p.m., operators took the reactor critical. Post-maintenance testing for relief valve PSV 4402 was completed at 10:50 p.m. and the valve was declared operable. Post-maintenance testing for the "A" intermediate range monitor was completed at 10:55 p.m., and it was declared operable. On November 30, at 11:51 p.m., operators synchronized the main generator to the grid. On December 1, at 3:51 a.m., the main generator was disconnected from the grid for turbine over-speed testing. Main turbine over-speed testing was completed satisfactorily and operators synchronized the main generator back onto the grid at 4:54 a.m. The plant was restricted to approximately 60 percent power until December 6, due to the ongoing corrective maintenance to replace structural supports on the "B" cooling tower. Full power operations were resumed on December 7, at 5:58 a.m.

I. Operations

O1 Conduct of Operations

O1.1 Observations of Routine Activities and the Plant Startup From Refueling Outage 16

a. Inspection Scope (71707)

The inspectors conducted numerous reviews of operators and operations management during shift activities. These reviews included observations of control shift turn-overs and operator performance during plant evolutions. The inspectors also observed the plant startup on November 27, 1999, from Refueling Outage 16. Documents reviewed for the plant startup included the following:

- Integrated Plant Operating Instruction (IPOI) 2, "Startup," Revision 59
- "Instructions for Reactor Startup for Cycle 17," Revision 1, November 1999

b. Observations and Findings

The inspectors observed that operators were knowledgeable of plant and equipment status. The inspectors observed that operations personnel conducted effective shift turn-overs. In general, the conduct of operations was appropriately focused on safety.

During the plant startup on November 27, 1999, operators, in accordance with IPOI 2, performed control rod withdrawals in a controlled and deliberate manner. Operators were challenged when maintenance troubleshooting activities inadvertently opened a safety-relief valve causing a reactor pressure and water level perturbation. Operators appropriately responded by restoring reactor pressure and water level. The power ascension and maintenance activities were stopped until operators were confident that the startup would not be affected by additional challenges. During subsequent testing the licensee determined that the safety-relief valve had an electrical short and appropriately declared it inoperable and entered the proper limiting condition for operation. Licensee management decided to shut down the plant to make the necessary repairs to the safety-relief valve. Operators then manually scammed the reactor.

On November 29, operators began withdrawing control rods after completion of the safety relief valve repair. The inspectors noted good command and control by operations personnel. The operators performed an error-free startup.

c. Conclusions

Operators performed an error-free startup in a controlled and deliberate manner. The inspectors noted that operators responded appropriately when a maintenance activity caused a reactor pressure and water level perturbation.

O2 Operational Status of Facilities and Equipment

O2.1 Walk-downs Conducted in the Drywell Prior to Startup From Refueling Outage (RFO) 16

a. Inspection Scope (71707)

The inspectors followed the guidance of Inspection Procedure 71707 in walking down accessible portions of the drywell prior to startup from RFO 16. The drywell was chosen based on maintenance work activities and probabilistic risk significance. Discussions were held with the project engineer responsible for reviewing material used in the drywell. Additionally, the inspectors discussed observations of material in the drywell with Nuclear Reactor Regulation (NRR) inspectors responsible for reviewing the licensee's response to NRC Bulletin 96-03, "Potential Plugging of Emergency Core Cooling Suction Strainers by Debris in Boiling Water Reactors." The inspectors reviewed the following documents:

- Surveillance Test Procedure (STP), 3.10.1-02, "Non-Nuclear Heat Class 1 10-Year System Leakage Pressure Test," Revision 1
- Engineering Calculation CAL-M98-002, "Post LOCA [Loss of Coolant Accident] Debris Generation for ECCS [Emergency Core Cooling System] Strainers"
- Integrated Plant Operating Instructions 7, " Special Operations," Attachment 2, "Primary Containment Closeout," Revision 58

b. Observations and Findings

The inspectors performed two detailed walk-downs of the drywell, due to the significant maintenance activities that occurred during RFO 16. The first walk-down was conducted during performance of STP 3.10.1-02. There were no significant leakage problems observed by the inspectors or licensee personnel that performed the walk-down. However, the inspectors did note a fibrous material, similar in appearance to steel wool, in several instrument line penetrations inside the drywell. Also, there was rigid insulation with imbedded fibrous material visible located around large diameter piping running through drywell penetrations.

The rigid insulation around the large diameter piping was identified as calcium silicate/asbestos. This insulation was considered in the licensee's engineering calculation CAL-M98-002. It was evaluated not to be a concern because pieces of the insulation were considered negatively buoyant if dislodged from around the large diameter piping. The material in the instrument line penetration was identified to be lead wool. The lead wool was used in the drywell for the purpose of permanent shielding. The project engineer had determined that the lead wool was not a concern for potential plugging of ECCS strainers in the event of an accident because he considered the material to be negatively buoyant. The NRR inspectors determined to review this issue prior to issuing a final report on the adequacy of the licensee's response to Bulletin 96-03. However, the inspectors' initial review determined this was not a startup issue.

The inspectors toured the drywell after the licensee performed its final drywell close-out and noted improved material condition from RFO 15 (refer to NRC Inspection Report 50-331/99014, Section O2.2). The licensee was thorough in ensuring loose debris and trash were removed from the drywell prior to the primary containment close-out. The inspectors identified that the rubber jacket surrounding wires for the bellows pressure sensing monitor for a main steam line safety-relief valve (PSV 4402) had pulled away from its conduit connection. Operations and engineering personnel evaluated the condition and determined that relief valve PSV 4402 was environmentally qualified to perform its safety function. The licensee initiated a work request card to take corrective actions in the next refueling outage to ensure the exposed wires are covered. The inspectors questioned plant staff and managers to determine if the Action Request system would be used to document the evaluation for relief valve PSV 4402.

Several different answers were provided, which demonstrated that management needed to provide clearer expectations to plant staff for using the Action Request system and Work Request Card system. For this example, the licensee's evaluation for relief valve PSV 4402 was appropriately captured using a work request card. Action Request 18222 was generated to evaluate the need for training plant staff to ensure that the Action Request system and the Work Request Card system are properly implemented in accordance with management's expectations.

c. Conclusions

The inspectors noted improved performance from the last refueling outage for the licensee's closeout of the primary containment drywell. The inspectors identified fibrous materials used in drywell penetrations that NRR inspectors will review and assess in their plant evaluation report on the licensee's response to Bulletin 96-03. The inspectors identified that licensee personnel needed additional guidance for using the Action Request system and the Work Request Card system. This was determined when inspectors identified suspect wiring to a relief valve bellows pressure sensing monitor. Confusion existed on whether to use the Action Request system or the Work Request Card system for evaluation and resolution.

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. The inspectors also reviewed several human performance issues concerning maintenance activities that occurred during the latter portion of the refueling outage covered by this inspection period. Specific tests and work request activities observed are listed below:

Maintenance Activities

- Corrective Work Order (CWO) A46101: Balance high pressure coolant injection (HPCI) oil pressures
- CWO A49989: Add and remove noble metals chemical addition coupons on crack arrest verification system
- CWO A49990: Replace electro-chemical potential electrodes for monitoring
- Preventive Work Order (PWO) 1107559: HPCI overspeed trip test

Surveillance Test Activities

- IPOI 7, "Special Operations," Revision 58, Attachment 2, "Primary Containment Closeout"
- STP 3.1.4-01, "Scram Insertion Time Test," Revision 5
- STP 3.5.1-06, "HPCI System Low Pressure Operability Test," Revision 4
- STP 3.5.3-03, "Low Pressure RCIC [Reactor Core Isolation Cooling] System Flow Rate Test," Revision 3
- STP 3.8.1-07, "LOOP-LOCA Test "B" System," Revision 5
- STP 3.10.1-02, "Non-Nuclear Heat Class 1 10-Year System Leakage Pressure Test," Revision 1
- STP NS93003, "Main Turbine Overspeed Trip System Test," Revision 0

b. Observations and Findings

There was an overall improvement in the conduct of maintenance activities from the previous refueling outage. Work associated with these activities was effectively conducted and completed in a thorough manner. Maintenance personnel were knowledgeable of work document requirements and their assigned tasks. In general, appropriate radiological controls were in place to support the maintenance activities.

Several minor personnel performance problems occurred during the latter half of the refueling outage. The section below describes the performance issues that occurred with the associated maintenance activity:

- Weld overlays were performed on two cracking reactor recirculation jet pump inlet nozzle welds. A HPCI instrument line and a safety-relief valve electrical connection, located in the vicinity of the reactor recirculation jet pump inlet nozzles, were damaged while performing the weld overlay work. This resulted in the need to replace a HPCI flow indication instrument line. Also, as described in Section M4.1 of this report, an electrical connection for safety-relief valve PSV 4402 was damaged during the weld overlay work causing an inadvertent engineered safety feature actuation during reactor startup. Both the HPCI instrument line and the safety-relief valve electrical connection were repaired when discovered.
- The condenser low vacuum pressure switch, PS1020B, which instrument technicians replaced during the outage, was found to have its internal wiring mis-configured. This was discovered after the condenser backpressure alarm would not reset during startup. Action Request 17828 was initiated and the internal wires were reconfigured. The switch was post-maintenance tested satisfactorily.

- On November 30, 1999, during the performance of radiation monitor sensor operability checks, the wrong radiation monitor was operated. Action Request 17834 was initiated and the correct radiation monitor was operated.

- On November 30, the licensee determined that the reactor water dissolved oxygen sampling line was not properly configured prior to startup. The oxygen sensors, which were to be valved-in prior to the reactor reaching a temperature of 284 degrees Fahrenheit, were valved-in at 300 degrees Fahrenheit. Action Request 17829 was initiated.

c. Conclusions

Overall, improvement was noted in the conduct of maintenance activities from the previous refueling outage. However, several minor personnel performance issues occurred during the latter half of the refueling outage due to inattention to detail. Also, the weld overlay work for the reactor recirculation jet pump inlet nozzles caused damage to surrounding equipment that challenged plant operators during the startup. Licensee performance in planning and scheduling improved since the last outage.

M4 Maintenance Staff Knowledge and Performance

M4.1 Main Steam Line Relief Valve Lifting

a. Inspection Scope (62703)

The inspectors reviewed licensee actions in response to main steam line relief valve PSV 4402 pilot solenoid spuriously opening. Discussions were held with the system engineer and maintenance personnel involved. The following documents were reviewed:

- Action Request (AR) 17822, "Electrical Ground Caused PSV 4402 (Main Steam Line "B" ADS [Automatic Depressurization System] Relief Valve) to Actuate"

- AR 17824, "PSV 4402 (Main Steam Line "B" ADS Relief Valve) Did Not Open During STP 3.4.3-03"

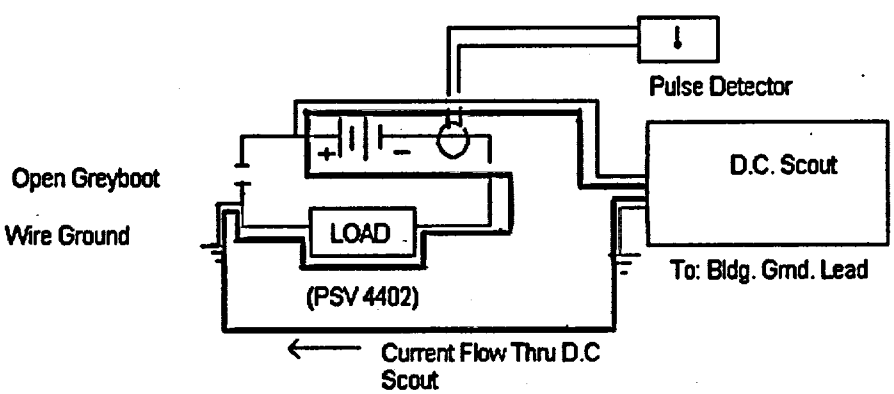
- Work Request A49950, "125 VDC Division I Ground In On Positive Bus +56VDC. Use DC Scout to Identify Grounded Circuit"

- Vendor Manual, "The DC Scout - DC Battery Distribution System Ground Locator"

b. Observations and Findings

On November 27, 1999, during plant startup from RFO 16, main steam line relief valve PSV 4402 spuriously opened for approximately 30 seconds. On-shift operators became aware of the relief valve lifting when the high temperature tail-pipe alarm annunciated and torus water level increased. Reactor water level swelled to 200 inches and reactor pressure lowered from 100 pounds per square inch gauge (psig) to 85 psig. Operators notified electrical maintenance technicians to disconnect the ground detection equipment from the 125 VDC Division I electrical system. After the equipment was disconnected, the relief closed. Indicated reactor vessel level returned to 178 inches and reactor pressure recovered.

Electrical maintenance technicians were troubleshooting a ground on the 125 VDC Division I electrical system at the time of the relief valve lifting. The system's in-line ground detection system indicated a ground on the negative portion of the circuit. Electrical maintenance technicians were using a "non-intrusive" ground detection device (DC Scout) to locate the source of the ground. The ground detection device had narrowed down the ground to a DC power sub-panel. As directed by the vendor manual, electrical maintenance technicians had tied the DC Scout input lead into the positive side of the 125 VDC system. The DC Scout output lead was tied into the building ground. When amperage allowed through the DC Scout was increased to 50 milliamps, the pilot solenoid valve for relief valve PSV 4402 actuated. The reason the pilot valve actuated was because there was also an open environmentally qualified wire splice, known as a greyboot connection, at the pilot valve in addition to the ground. The greyboot connection was open on the positive side of the pilot solenoid. This masked the fact that the ground was also on the positive side of the pilot valve circuitry. This resulted in the electrical maintenance technicians erroneously connecting the DC Scout ground detection device to the positive bus.



The simplified diagram above shows the 125 volt system with the actual system faults and the DC Scout ground detection device as it was installed. The heavy line represents the current and flow-path which caused actuation of the pilot solenoid valve. The inspector's review of the vendor manual found that the electrical maintenance technicians connected the DC Scout in what would be considered the proper manner to troubleshoot the ground as indicated by the 125 VDC system's ground detection system.

On November 28, while performing STP 3.4.3-03, relief valve PSV 4402 did not open, as expected, when operators placed the control room hand-switch to the open position.

A review of work orders completed during Refueling Outage 16, found that the sealtite (insulation jacket around the wiring) had been pulled from a junction box that supplied power to relief valve PSV 4402. The sealtite was re-installed back into the junction box and re-tightened; however, the wiring internal to the junction box and the conduit box for the solenoid was not inspected for potential damage caused when the wiring was pulled out of the junction box. The junction box and the conduit box are approximately three feet apart. A work order to examine the wiring in the conduit box, following failure of relief valve PSV 4402 to open during the surveillance test, found that the insulation around a lead had been stretched and exposed the wiring. Also, an examination of the greyboot connection on the same wire found that the connection had been pulled apart. The necessary repairs were made to the wiring and greyboot connection. The post-maintenance testing was performed satisfactorily.

c. Conclusions

The inspectors found that the licensee had followed an approved methodology for troubleshooting electrical grounds in the wiring for main steam line relief valve PSV 4402, which spuriously lifted during troubleshooting. However, the licensee's corrective actions to properly re-install wiring pulled from the junction box supplying power to relief valve PSV 4402 prior to the relief valve lifting was inadequate. Electrical maintenance technicians failed to examine wiring inside the junction box or conduit box.

III. Engineering

E2 Engineering Support of Facilities and Equipment

E2.1 Emergent Work for the Weld Overlay Repairs to the Reactor Recirculation Riser Nozzles

a. Inspection Scope (37551)

The inspectors reviewed licensee actions for repairs to the reactor recirculation risers N2B and N2D, which were found to have indications in the safe-end to nozzle welds. The inspectors observed licensee meetings discussing the problem and the subsequent weld repair technique to be used. The inspectors observed pre-job welding on the mock-up by the vendor and in-field observations of portions of the repair. A review of the work order package was also conducted. Discussions of the problem and repair were held with the responsible engineers and vendor personnel performing the repair. The following documents were reviewed:

- AR 13836, "Weld Measurements on Recirc Nozzle N2B - ALARA [As Low As Reasonably Achievable] Awareness"
- AR 17604, "Two Linear Indications Found in Recirc Riser Nozzle N2B, Weld #RRB-F002 (Nozzle to Safe-End)"

- AR 17481, "Linear Indication Found in Recirc Riser Nozzle N2D, Weld #RRD-F002(Nozzle to Safe-End)"

- NG-99-1613, "Authorization for Use of Code Cases for Nozzle-to Safe End Repairs"

- Engineering Change Procedure (ECP)-1627, "Design of Weld Overlay for N2 Recirc Inlet Nozzles"

b. Observations and Findings

There was one significant emergent work issue that occurred during the recently completed RFO 16. During performance of ultrasonic testing for in-service inspection of the reactor recirculation risers, indications in the nozzle to safe-end welds were identified. The original scope of examinations included inspection of three recirculation risers and one core spray nozzle to safe-end welds. The original inspections identified two circumferential cracks on the N2B riser. The deeper of the two cracks was determined to be 65 percent through wall. The licensee subsequently expanded the scope of the recirculation riser and core spray nozzle welds to include all eight risers and both core spray nozzles. The expanded scope identified a 65 percent through wall crack on the N2D riser. Also, an indication was found on the N2F riser that was further evaluated by grinding the weld crown flush and re-inspecting the suspect area. The re-inspection was able to show that the flaw was internal with no connection to the inside or outside surface of the piping.

The licensee retrieved the radiographs taken when the recirculation risers were replaced in 1978. The radiographs were then digitized for enhancement. This review identified a small area of internal incomplete fusion between welds. The reason this was identified during the current inspections was due to improvements in ultrasonic testing equipment. The welds were susceptible to inter-granular stress corrosion cracking due to the creviced areas in the welds. The licensee concluded that the 1978 safe-end replacement created some conditions in the weld root area that made it vulnerable for initiation of inter-granular stress corrosion cracking.

The licensee wanted to perform the repairs with the reactor vessel flooded up to reduce radiation dose to personnel performing the weld repairs and avoid impacting the schedule that would occur if the vessel had to be drained. The method of performing the weld repairs with the vessel flooded-up had was not yet been approved by the American Society of Mechanical Engineers (ASME). A conference call was conducted between the licensee and NRC on November 9, 1999. An ASME Code case was under review by NRC to allow use of gas tungsten arc welding without preheat or post-weld heat treatment. After reviewing the licensee's proposed repair methodology, NRC gave initial approval for the licensee to proceed with the weld repairs.

There was good management oversight of the planning for the weld repairs. The licensee considered ongoing and upcoming refueling outage activities that could be impacted by this emergent work. However, there were problems on the first riser to be repaired, which was riser N2B. Weld measurements on the N2B were required for planning the weld overlay repair work. The first attempt to measure the exact weld overlay location was taken on the wrong weld. A second attempt also failed to attain the appropriate weld measurements. Personnel were not given adequately detailed drawings to perform the measurements. The initial measurements were taken with original drawings that were not updated for changes to the piping when the 1978 repairs were done. Engineering personnel subsequently revised the drawings to reflect the proper piping configurations. Coverage of this issue from the health physics perspective was detailed in Inspection Report 50-331/99013.

c. Conclusions

In general, the significant emergent work item of the refueling outage of planning and performing the weld overlay for the reactor recirculation risers was conducted well. However, there were problems in positioning of the weld overlay for the N2B riser due to drawings that had not received an adequate engineering review.

E2.2 Engineering Support for High Pressure Coolant Injection (HPCI) System

a. Inspection Scope (37551)

The inspectors evaluated the engineering support provided for the HPCI system. The following documents were reviewed:

Action Request (AR) 17823, "Surveillance Test Procedure (STP) 3.5.1-06 HPCI System Low Pressure Operability Found System Response Time too Long"

Preventive Work Order (PWO) 1107559, "HPCI Overspeed Trip Test"

b. Observations and Findings

On November 13 through 15, 1999, the licensee performed PWO 1107559. The work was completed satisfactorily and on November 28, the plant startup activities had commenced and STP 3.5.1-06 surveillance testing was conducted. Operations personnel initiated AR 17823 when the corrected response time for the turbine stop valve operation was longer than desired.

The system engineer reviewed the test data and discussed the results with previous HPCI system engineers. Based on the cursory review, the engineer determined that the response time was an expected value with the reactor at low pressure and AR 17823 was administratively closed. However, the inspectors reviewed previous results from HPCI low pressure operability testing and noted that corrected response times were within the desired range; therefore, the longer response time test result was not expected and the previous testing results contradicted the engineer's resolution to AR 17823. The licensee agreed and re-opened AR 17823 and provided additional data to demonstrate that HPCI was considered operable.

On December 5, STP NS52002, "HPCI Response Time Correction Factor Verification," surveillance testing was completed and operators determined that the response time result was in excess of the required 5.4 seconds. Also, operators noted that the timing verification was performed to support the quarterly HPCI operational surveillance test. If the test results were greater than 5.4 seconds, then the quarterly surveillance test would be affected. Operators initiated AR 17844 to determine the operability of HPCI and to resolve the response time test prior to the next HPCI quarterly operability test.

Engineering personnel determined that HPCI was considered operable due to the large margin of safety that was applied to the surveillance testing and HPCI response time correction factor. The engineer attributed the increased response time to the HCPI overspeed trip testing. Surveillance test NS52002 was re-performed and the response time was less than 5.4 seconds. The engineer considered the previous test result to be an anomaly. The engineer suggested to verify and adjust HPCI system oil pressures accordingly for the proper response time for the turbine stop valve prior to performing the HPCI quarterly operability surveillance test.

The inspectors reviewed the engineering evaluation and acknowledged the resolution of AR 17844. However, the inspectors identified that due to a lack of attention to detail, the engineering staff did not recognize that the increased response time result also effected the HPCI low pressure operability test that was performed November 28. The engineering staff agreed and determined that although this was an oversight, HPCI was operable due to improved re-test results and the large margin of safety.

c. Conclusions

The inspectors determined that due to a lack of attention to detail, the engineering staff did not adequately address HPCI operability concerns using the Action Request system.

IV. Plant Support

R1 Radiation Protection and Chemistry Controls

R1.1 Radiation Protection Support for Maintenance Work in High Radiation Area

a. Inspection Scope (71750)

The inspectors assessed the adequacy of radiation protection work practices during maintenance work on the crack arrest verification (CAV) system which is located in a high radiation area. The inspectors observed portions of the CAV maintenance work. Radiation Work Permit (RWP) 106, Job Step 12, was used to provide instructions to workers performing the maintenance.

b. Observations and Findings

Radiation Work Permit 106, Job Step 6, provided sufficient radiological protection instructions to workers to support the CAV system maintenance activities. In accordance with RWP 106, Job Step 12, workers wore the proper protective clothing and rubber gloves to prevent personnel contamination. Workers used absorbent rags to control the spread of liquid contamination during system disassembly and reassembly. The inspectors noted that the workers effectively minimized their accumulated dose during the maintenance by moving to low dose areas to complete certain job tasks. Radiation protection personnel performed numerous dose rate and contamination surveys to ensure accumulated dose and personnel contamination were kept to a minimum.

c. Conclusions

Workers effectively minimized their accumulated dose and minimized the spread of contamination while performing maintenance on the crack arrest verification system.

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 December 21, 1999. 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

<u>Licensee</u>

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 62703:	Maintenance Observation
IP 62707:	Maintenance Observation
IP 71707:	Plant Operations
IP 71750:	Plant Support

ITEMS OPENED, CLOSED, AND DISCUSSED

<u>Opened</u>

None

<u>Closed</u>

None

<u>Discussed</u>

None

LIST OF ACRONYMS USED

ADS	Automatic Depressurization System
AR	Action Request
ASME	American Society of Mechanical Engineers
CAV	Crack Arrest Verification
CFR	Code of Federal Regulations
CWO	Corrective Work Order
DAEC	Duane Arnold Energy Center
DRP	Division of Reactor Projects
ECCS	Emergency Core Cooling System
HPCI	High Pressure Coolant Injection
IP	Inspection Procedure
IPOI	Integrated Plant Operating Instructions
LOOP-LOCA	Loss of Offsite Power - Loss of Coolant Accident
NRC	Nuclear Regulatory Commission
NRR	Nuclear Reactor Regulation
psig	Pounds Per Square Inch Gauge
PWO	Preventive Work Order
RFO	Refuel Outage
RWP	Radiation Work Permit
STP	Surveillance Test Procedure
VDC	Volts Direct Current