August 7, 2007

Mr. Gary Van Middlesworth
Vice-President
Duane Arnold Energy Center
3277 DAEC Road
Palo, IA 52324-9785

SUBJECT: DUANE ARNOLD ENERGY CENTER
NRC INTEGRATED INSPECTION REPORT 05000331/2007003

Dear Mr. Van Middlesworth:

On June 30, 2007, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at your Duane Arnold Energy Center. The enclosed integrated inspection report documents the inspection findings which were discussed on July 12, 2007, with you and other members of your staff.

The inspection examined activities conducted under your license as they relate to safety and compliance with the Commission's rules and regulations and with the conditions of your license. The inspectors reviewed selected procedures and records, observed activities, and interviewed personnel.

Based on the results of this inspection, there was one NRC-identified and three self-revealed findings of very low safety significance, of which three involved a violation of NRC requirements. In addition, one issue was viewed under the NRC traditional enforcement process and determined to be a Severity Level IV violation of NRC requirements. However, because these violations were of very low safety significance and because the issues were entered into the licensee's corrective action program, the NRC is treating these findings as Non-Cited Violations in accordance with Section VI.A.1 of the NRC's Enforcement Policy. Additionally, one licensee identified violation is listed in Section 4OA7 of this report.

On August 27, 2007, the U.S. Nuclear Regulatory Commission will begin the supplemental inspection for the White Emergency Preparedness finding you received in April 2, 2007, as documented in NRC Inspection Report 05000331/2006009(DRS). This inspection will be performed in accordance with NRC baseline inspection procedure (IP) 95001. The lead inspector for this inspection is Mr. Randal Baker. If there are any questions about the material requested, or the inspection, please call Mr. Randal Baker at (319) 851-5111 or e-mail him at RDB@nrc.gov.

If you contest the subject or severity of a Non-Cited Violation, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555-0001; with a copy to the Regional Administrator, U.S. Nuclear Regulatory Commission - Region III, 2443 Warrenville Road, Suite 210, Lisle, IL 60532-4352; the Director, Office of Enforcement,

U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001; and the Resident Inspector Office at the Duane Arnold Energy Center.

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter and its enclosure will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARS) component of NRC's document system (ADAMS). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

/RA/

Kenneth Riemer, Chief
Branch 2
Division of Reactor Projects

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

Enclosure: Inspection Report 05000331/2007003
w/Attachment: Supplemental Information

cc w/encl: J. Stall, Senior Vice President, Nuclear and Chief Nuclear Officer
R. Helfrich, Senior Attorney
M. Ross, Managing Attorney
W. Webster, Vice President, Nuclear Operations
M. Warner, Vice President, Nuclear Operations Support
R. Kundalkar, Vice President, Nuclear Engineering
J. Bjorseth, Site Director
D. Curtland, Plant Manager
S. Catron, Manager, Regulatory Affairs
Chief Radiological Emergency Preparedness Section, Dept. Of Homeland Security
D. McGhee, State Liaison Officer

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Letter to G. Van Middlesworth from K. Riemer dated August 7, 2007

SUBJECT: DUANE ARNOLD ENERGY CENTER
NRC INTEGRATED INSPECTION REPORT 05000331/2007003

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U.S. NUCLEAR REGULATORY COMMISSION

REGION III

Docket No:	50-331
License No:	DPR-49
Report No:	05000331/2007003
Licensee:	FPL Energy Duane Arnold, LLC
Facility:	Duane Arnold Energy Center
Location:	Palo, Iowa
Dates:	April 1 through June 30, 2007
Inspectors:	R. Orlikowski, Senior Resident Inspector R. Baker, Resident Inspector R. Langstaff, Senior Reactor Inspector J. McGhee, Reactor Engineer T. Go, Health Physicist N. Valos, Senior Operations Engineer N. Shah, Project Engineer C. Acosta Acevedo, Reactor Inspector C. Zoia, Operations Engineer M. Bielby, Senior Operations Engineer
Observers:	None
Approved by:	K. Riemer, Chief Branch 2 Division of Reactor Projects

Enclosure

SUMMARY OF FINDINGS

IR 05000331/2007003; 04/01/2007 - 06/30/2007; Duane Arnold Energy Center. Operability Evaluations, Event Follow-up, and Other Activities.

This report covers a three-month period of baseline resident inspection announced baseline inspections of radiation protection, engineering, and operator licensing. The inspections were conducted by Region III reactor inspectors, a health physics inspector, an operations engineer, and the resident inspectors. The significance of most findings is indicated by their color (Green, White, Yellow, Red) using Inspection Manual Chapter (IMC) 0609, "Significance Determination Process" (SDP). Findings for which the SDP does not apply may be "Green" or be assigned a severity level after NRC management review. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, "Reactor Oversight Process," Revision 3, dated July 2000.

A. Inspector-Identified and Self-Revealed Findings

Cornerstone: Initiating Events

- Green: A finding of very low safety significance was self-revealed regarding the failure of the control room crew to establish control of a critical parameter, Reactor Pressure Vessel (RPV) level, in a timely manner during the feedwater recovery efforts following the manual scram initiated due to the loss of the 1A2 bus and the associated loss of the 'B' Reactor Feed Pump (RFP) and 'B' Condensate Pump. This resulted in a second automatic reactor protection system (RPS) actuation on low RPV level. The inspectors determined that the failure to ensure positive control of RPV level to preclude receiving an automatic protective system actuation was a performance deficiency warranting further evaluation. The licensee subsequently restored feedwater flow, completed the reactor shutdown, and entered this issue into their corrective action program. This finding did not result in a violation of NRC requirements.

  The finding was more than minor because it adversely impacted the initiating events cornerstone attribute for human performance which limits the likelihood of events that upset plant stability and challenge critical safety functions. Although the crew's actions resulted in an automatic RPS actuation, the finding was determined to be of very low safety significance since it did not impact any mitigating systems capability. Additionally, no violations of NRC requirements occurred. (Section 4OA3.6)

Cornerstone: Mitigating Systems

- Green: A finding of very low safety significance and an associated Non-Cited Violation (NCV) of 10 CFR 50 Appendix B, Criterion 3, was self-revealed when PSV2302, the HPCI discharge pressure relief valve, stuck open during planned testing of the High Pressure Coolant Injection (HPCI) System. The inspectors determined that the failure to provide sufficient margin between the HPCI discharge relief valve setpoint and the peak discharge pressure of the HPCI system upon startup was a performance deficiency warranting further evaluation. The licensee completed a temporary modification to

remove the HPCI keep-fill modification and the HPCI system was returned to operable status.

The finding was determined to be more than minor because the engineering calculation error resulted in a condition where there was a reasonable doubt on the operability of the HPCI system. This issue screened as having a very low safety significance since the finding is a design deficiency confirmed not to result in a loss of operability per the part 9900 technical guidance for operability determination process for operability and functional assessment. This issue was also related to the decision making component of the human performance cross-cutting area, because engineering personnel failed to conduct an effective review of the safety-significant HPCI keep-fill modification and identify that the relief valve setpoint did not provide sufficient margin to prevent an unintended consequence. Specifically, the lifting of the relief valve due to the peak HPCI system discharge pressure seen during system startup. (Section 4OA3.5)

- Green: A finding of very low safety significance and an associated NCV of Technical Specification (TS) 3.8.1.b, Electrical Power Systems, AC Sources-Operating, was self-revealed when a leak was discovered coming from the lube oil filter (LOF) cover on the 'B' Emergency Diesel Generator (EDG) during surveillance testing. The leak rate was approximately 0.21 gallons per minute, and the licensee determined that the 'B' EDG would not have been capable of performing its seven-day unassisted operation design requirement. The licensee declared the 'B' EDG inoperable, entered the issue into their corrective action program, and initiated a work order to repair the oil leak. During the licensee's investigation, the apparent cause of the LOF leak was the installation of the wrong oil filter cover o-ring while performing the liner replacement maintenance during the recent refueling outage conducted by the vendor six weeks prior.

  The finding was determined to be more than minor because the 'B' EDG was returned to service with the incorrect o-ring installed and the leak that developed resulted in subsequent equipment inoperability. Additionally, based upon the licensee's past operability evaluation, the TS limiting condition for operation (LCO) allowed outage time for one EDG inoperable with the plant at power was exceeded. Since this issue was not a design or qualification deficiency, did not result in a loss of safety function, and was not considered potentially risk significant to a seismic, flooding, or severe weather initiating event, the issue screened as having a very low safety significance. This issue was also related to the work practices component of the human performance cross-cutting area, because maintenance personnel failed to ensure supervisory and management oversight of work activities, including contractors, supported nuclear safety. Specifically, the personnel performing maintenance activities for reassembly of the LOF were not supervised, an incorrect LOF cover 0-ring was installed, and the equipment was subsequently returned to service. (Section 1R15)

- Severity Level IV: The inspectors identified a Level IV NCV of 10 CFR 50.9, "Completeness and Accuracy of Information." The inspectors identified that the facility licensee, on March 30, 2007, submitted to the NRC, an NRC Form 396, "Certification of Medical Examination By Facility Licensee," for a licensed operator applying for renewal of his reactor operator license, that was not complete and accurate in all material respects. Specifically, the NRC Form 396 certified that the licensed operator was not required to have a "corrective lens" restriction on his license. When the NRC

questioned the licensee on the accuracy of the most recent biennial medical examination on the submitted NRC Form 396, the licensee submitted a revised NRC Form 396 on April 19, 2007. The revised NRC Form 396 included a new date for the most recent biennial medical examination, but also showed that the licensed individual was required to have a "corrective lens" restriction added to his license. This information was material to the NRC because the NRC relies on this certification to determine whether an applicant meets the requirements to operate the controls of a nuclear power plant pursuant to 10 CFR Part 55.

The finding was determined to be more than minor because the information associated with the license renewal of the individual was provided to the NRC under a signed statement by the Site Vice President and could have impacted an NRC licensing decision. The licensed operator could have been, without NRC intervention, issued a license without a "corrective lens" restriction added to his license. The finding was determined to be of low safety significance because the license renewal application for the reactor operator was not renewed until complete and accurate information was received on a revised NRC Form 396 that showed a "corrective lens" restriction for the licensed individual. (Section 4OA5.1)

Cornerstone: Barrier Integrity

- Green: The inspectors identified a finding of very low safety significance and an associated NCV of 10 CFR 50 Appendix B, Criterion 5, when licensee staff failed to implement the appropriate controls to properly store underwater lights in the spent fuel pool. The licensee entered the issue into the corrective action program for resolution. This issue was also related to the work practices component of the human performance cross-cutting area. Specifically, the aspect related to procedural compliance, as the station procedure that described the appropriate controls for storing items in the pool, was not followed.

  The finding was determined to be more than minor because the finding could be reasonably viewed as a precursor to a more significant event. Specifically, the failure to follow the approved process for controlling the use of nylon ropes in the spent fuel pool could result in the ropes being in place for an extended period of time. This increased the potential for unplanned radiation exposure either due to wicking or from damage to the underlying fuel assemblies, if the ropes degraded causing the lights to fall. The finding was considered to be of very low safety significance since it was determined to affect only the fuel cladding function of the Barrier Cornerstone. (Section 4OA5.2)

B. Licensee-Identified Violations

A violation of very low safety significance, which was identified by the licensee, has been reviewed by the inspectors. Corrective actions taken or planned by the licensee have been entered into the licensee's corrective action program. This violation and corrective actions are listed in Section 4OA7 of this report.

REPORT DETAILS

Summary of Plant Status

Duane Arnold Energy Center operated at full power for the entire assessment period except for brief down-power maneuvers to accomplish rod pattern adjustments and to conduct planned surveillance testing activities with the following exceptions:

- On April 2, 2007, the reactor was manually scrammed for a forced outage due to a loss of the 1A2 non-essential bus, which resulted in a loss of the 'B' Reactor Feed Pump and the 'B' Recirculation Pump. The reactor was restarted on April 4, and the generator connected to the grid on April 5. Full power was achieved on April 7.

- On May 31, 2007, a rapid power reduction to 58 percent reactor power was performed to permit securing and repair of a steam leak from the pump casing on the motor driven 'A' Reactor Feedwater Pump (RFP). The plant was returned to full power on June 2, following the repair, post maintenance testing, and restoration of the 'A' RFP.

1. REACTOR SAFETY

Cornerstone: Initiating Events, Mitigating Systems, Barrier Integrity, and Emergency Preparedness

1R01 Adverse Weather (71111.01)

.1 Summer Weather Preparations

a. Inspection Scope

The inspectors performed a detailed review of the licensee's procedures and a walkdown of three systems to observe the licensee's preparations for summer conditions for a total of one sample. The documents listed in the Attachment were used by the inspectors to accomplish the objectives of the inspection procedure. During the inspection, the inspectors focused on plant specific system design features and implementation of procedures for responding to or mitigating the effects of adverse weather. Inspection activities included, but were not limited to, a review of the licensee's adverse weather procedures, preparations for the summer season, and a review of analysis and requirements identified in the Updated Final Safety Analysis Report (UFSAR).

The inspectors evaluated summer readiness of the following three systems for a total of one sample:

- Pumphouse Heating and Ventilation System;
- Main Turbine Electro-Hydraulic Control System; and
- Residual Heat Removal Service Water (RHRSW) System.

b. Findings

No findings of significance were identified.

1R04 Equipment Alignment (71111.04)

.1 Partial System Walkdown

a. Inspection Scope

The inspectors performed partial walkdowns of accessible portions of trains of risk-significant mitigating systems equipment. The documents listed in the Attachment were used by the inspectors to accomplish the objectives of the inspection procedure. Equipment alignment was reviewed to identify any discrepancies that could impact the function of the system and potentially increase risk. Redundant or backup systems were selected by the inspectors during times when the trains were of increased importance due to the redundant trains of other related equipment being unavailable. Inspection activities included, but were not limited to, a review of the licensee's procedures, verification of equipment alignment, and an observation of material condition, including operating parameters of in-service equipment. Identified equipment alignment problems were verified by the inspectors to be properly resolved.

The inspectors selected the following equipment trains to verify operability and proper equipment line-up for a total of four samples:

- Reactor Core Isolation Cooling (RCIC) system with High Pressure Coolant Injection (HPCI) system Out of Service (OOS);
- 'B' RHRSW system with 'A' RHRSW system OOS during strainer modifications;
- 'B' Emergency Service Water system with the Standby Liquid Control system OOS for planned testing; and
- 'A' River Water Supply (RWS) system with 'B' RWS system OOS.

b. Findings

No findings of significance were identified.

.2 Complete System Walkdown

a. Inspection Scope

The inspectors performed a complete system alignment inspection of the Core Spray (CS) system. This system was selected because it was considered both safety-significant and risk-significant in the licensee's probabilistic risk assessment. The inspection consisted of a review of plant procedures (including selected abnormal and emergency procedures), drawings, and the UFSAR to identify proper system alignment. The inspectors also reviewed selected issues documented in Corrective Action Processes (CAPs), to determine if they had been properly addressed in the licensee's corrective actions program. As part of this inspection, the documents in the Attachment were utilized to evaluate the potential for an inspection finding.

This review represented one sample.

b. Findings

No findings of significance were identified.

1R05 Fire Protection (71111.05)

.1 Quarterly Fire Zone Walkdowns (71111.05Q)

a. Inspection Scope

The inspectors walked down risk-significant fire areas to assess fire protection requirements. The documents listed in the Attachment were used by the inspectors to accomplish the objectives of the inspection procedure. Various fire areas were reviewed to assess if the licensee had implemented a fire protection program that adequately controlled combustibles and ignition sources within the plant, effectively maintained fire detection and suppression capability, maintained passive fire protection features in good material condition, and had implemented adequate compensatory measures for OOS, degraded or inoperable fire protection equipment, systems or features. Fire areas were selected based on their overall contribution to internal fire risk as documented in the plant's Individual Plant Examination of External Events, their potential to adversely impact equipment which is used to mitigate a plant transient, or their impact on the plant's ability to respond to a security event. Inspection activities included, but were not limited to, the control of transient combustibles and ignition sources, fire detection equipment, manual suppression capabilities, passive suppression capabilities, automatic suppression capabilities, compensatory measures, and barriers to fire propagation.

The inspectors selected the following areas for review for a total of 12 samples:

- Area Fire Plan (AFP) 1; Reactor Building Torus Area and North Corner Rooms, Elevations 716'9" and 735'7.5";
- AFP 3; Reactor Building HPCI, RCIC and Radwaste Tank Rooms, Elevations 716'9" and 747'0";
- AFP 4; Reactor Building North Control Rod Drive Module Area and Control Rod Drive Repair Room, Elevation 757'6";
- AFP 9; Reactor Building Reactor Building Closed Loop Cooling Water Heat Exchanger Area, Equipment Hatch Area, and Jungle Room, Elevation 812'0";
- AFP 14; Turbine Building Reactor Feed Pump Area, Turbine Lube Oil Tank Area, and 1A2 Switchgear Room, Elevation 734'0";
- AFP 19; Turbine Building South Turbine Building Ground Floor, Elevation 757'6";
- AFP 20; Turbine Building Aux Boiler Room, Emergency Diesel Generator Rooms, and Generator Day Tank Rooms, Elevation 757'6";
- AFP 31; Intake Structure Pump Rooms, Elevation 767'0";
- AFP 32; Intake Structure Traveling Screen Areas, Elevation 754'0";
- AFP 34; Radwaste Building Drum Filling, Storage, and Shipping Area, Elevation 757'6";
- AFP 35; Radwaste Building Radwaste Treatment And Access Area, Elevation 773'6"; and

- AFP 36; Radwaste Building Precoat and Access Area, Control Room, and Heating, Ventilation, Air-Conditioning (HVAC) Equipment Room, Elevation 786'0".

b. Findings

No findings of significance were identified.

1R06 Flood Protection Measures (71111.06)

a. Inspection Scope

The inspectors performed an annual review of flood protection barriers and procedures for coping with external flooding for a total of one sample. The document listed in the Attachment was used by the inspectors to accomplish the objectives of the inspection procedure. Inspection activities focused on verifying that flood mitigation plans and equipment were consistent with design requirements and risk analysis assumptions. Inspection activities included, but were not limited to, a review and/or walkdown to assess design measures, seals, drain systems, contingency equipment condition and availability of temporary equipment and barriers, performance and surveillance tests, procedural adequacy, and compensatory measures.

b. Findings

No findings of significance were identified.

1R07 Heat Sink Performance (71111.07B)

.1 Biennial Review of Heat Sink Performance

a. Inspection Scope

The inspectors reviewed the performance of the ‘B’ Emergency Diesel Generator (EDG) coolers and the ‘B’ Residual Heat Removal (RHR) heat exchanger. These heat exchangers were chosen for review based on their high risk assessment worth in the licensee’s probabilistic safety analysis. This review resulted in the completion of two inspection samples. While on-site, the inspectors verified that the inspection, maintenance and testing conducted on these heat exchangers were adequate to ensure proper heat transfer. This was done by conducting independent heat transfer capability calculations, reviewing the methods and calculations used to inspect and test the heat exchangers, and verifying that the as-found results were appropriately dispositioned, such that the final condition was acceptable. The inspectors also verified, by review of procedures and test results, that chemical treatments, ultrasonic tests, eddy current tests and methods used to control biotic fouling corrosion and macrofouling were sufficient to ensure required heat exchanger performance.

The inspectors verified that the condition and operation of these heat exchangers were consistent with design assumptions in heat transfer calculations by conducting a walk-down of the intake bay, the selected heat exchangers and the pumps that supply

these heat exchangers and by reviewing related procedures and surveillance. The inspectors also verified that redundant and infrequently used heat exchangers were flow tested periodically at maximum design flow. This was performed by reviewing related procedures and surveillance.

The inspectors verified the performance of the ultimate heat sink and its sub-components, such as piping, intake screens, intake bays, pumps, valves, etc. by reviewing procedures, surveillance, and visual inspections conducted on the system.

The inspectors verified that the licensee had entered significant heat exchanger and heat sink problems into their corrective action program. The inspectors reviewed issues to verify that the corrective actions taken were appropriate.

The documents that were reviewed are listed in the Attachment.

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification Program (71111.11)

a. Inspection Scope

The inspectors observed performance of two training crews on Simulator Exercise Guide 2007B-015 PM, Revision 0. The scenario included a lockout of the Startup Transformer due to an internal fault, a subsequent loss of both essential electrical busses, complicated by a small break loss of coolant accident inside the drywell with a failure of the HPCI. The documents listed in the Attachment were used by the inspectors to accomplish the objectives of the inspection procedure. The inspection activities assessed the licensee's effectiveness in evaluating the requalification program, ensuring that licensed individuals operated the facility safely and within the conditions of their license, and evaluated licensed operators' mastery of high-risk operator actions. Inspection activities included, but were not limited to, a review of high risk activities, emergency plan performance, incorporation of lessons learned, clarity and formality of communications, task prioritization, timeliness of actions, alarm response actions, control board operations, procedural adequacy and implementation, supervisory oversight, group dynamics, interpretations of Technical Specifications (TSs), simulator fidelity, and the licensee critique of performance.

This review represented one sample.

The crew performance was compared to licensee management expectations and guidelines as presented in the following documents:

- Administrative Control Procedure (ACP) 110.1, "Conduct of Operations," Revision 7;
- ACP 101.01, "Procedure Use and Adherence," Revision 41; and

- ACP 101.2, “Verification Process and SELF/PEER Checking Practices,” Revision 5.

b. Findings

No findings of significance were identified.

1R12 Maintenance Effectiveness (71111.12)

a. Inspection Scope

The inspectors reviewed plant systems to assess maintenance effectiveness. The documents listed in the Attachment were used by the inspectors to accomplish the objectives of the inspection procedure. Maintenance activities were reviewed to assess maintenance effectiveness, including maintenance rule activities, work practices, and common cause issues. Inspection activities included, but were not limited to, the licensee's categorization of specific issues including evaluation of maintenance performance criteria, appropriate work practices, identification of common cause errors, extent of condition, and trending of key parameters. Additionally, the inspectors reviewed implementation of the Maintenance Rule (10 Code of Federal Regulations (CFR) 50.65) requirements, including a review of scoping, goal-setting, performance monitoring, short-term and long-term corrective actions, functional failure determinations associated with reviewed condition reports, and current equipment performance status.

The inspectors performed the following maintenance effectiveness reviews for a total of two samples:

- An issue/problem-oriented review of the Reactor Recirculation System because the system had experienced unplanned changes in pump speed; and
- A function-oriented review of the Standby Gas Treatment (SBGT) System because it was designated as risk-significant under the Maintenance Rule.

b. Findings

No findings of significance were identified.

1R13 Maintenance Risk Assessments and Emergent Work Control (71111.13)

a. Inspection Scope

The inspectors reviewed the licensee’s evaluation of plant risk, scheduling, and configuration control. An evaluation of the performance of maintenance associated with planned and emergent work activities was completed by the inspectors to determine if they were adequately managed. In particular, the inspectors reviewed the program for conducting maintenance risk safety assessments and to ensure that the planning, assessment and management of on-line risk was adequate. The documents listed in the Attachment were used by the inspectors to accomplish the objectives of the inspection procedure. Licensee actions taken in response to increased on-line risk were reviewed including the establishment of compensatory actions, minimizing activity

duration, obtaining appropriate management approval, and informing appropriate plant staff. These activities were accomplished when on-line risk was increased due to maintenance on risk-significant structures, systems, and components (SSCs).

The following activities were reviewed for a total of four samples:

- The inspectors reviewed the maintenance risk assessment for work planned during the weeks ending April 20, May 4 and 11, and June 1, 2007.

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations (71111.15)

a. Inspection Scope

The inspectors reviewed the licensee's operability evaluations of degraded or non-conforming systems. The documents listed in the Attachment were used by the inspectors to accomplish the objectives of the inspection procedure. Operability evaluations were reviewed that affected mitigating systems or barrier integrity cornerstones to ensure adequate justification for declaration of operability and that the component or system remained available. Inspection activities included, but were not limited to, a review of the technical adequacy of the evaluation against the TSs, UFSAR, and other design information; validation that appropriate compensatory measures, if needed, were taken; and comparison of each operability evaluation for consistency with the requirements of ACP 114.5, "Action Request System" and ACP 110.3, "Operability Determination."

The inspectors reviewed the following operability evaluations for a total of five samples:

- 5/16 Inch Linear Indication on HPCI Turbine Steam Supply Valve, MO2202, Disk Face Identified During Final PT;
- HPCI System Discharge Check Valve Leakage;
- 'B' EDG with the Speed Sensing Switch Out of Calibration;
- 1VAC012 Room Cooler (Southeast Corner Room) Work Order Documentation Did Not Identify Weld Filler Material Used During Weld Repair Performed; and
- 'B' EDG with a Self-Revealed Oil Leak on the Lube Oil Filter (LOF).

b. Findings

Introduction: A finding of very low safety significance (Green) and an associated NCV of Technical Specification (TS) 3.8.1.b, Electrical Power Systems, AC Sources-Operating, was self-revealed when a lubricating oil leak was discovered coming from the LOF cover on the 'B' EDG during surveillance testing.

Description: On April 11, 2007, while operating at 98 percent reactor power, a 0.21 gallon per minute lube oil leak was observed coming from the 'B' EDG LOF cover during performance of STP 3.8.1-04. The STP was aborted and the EDG was shutdown. The

licensee performed an apparent cause evaluation and determined that an incorrect LOF cover o-ring had been installed on February 12, 2007, during the cylinder liner replacement maintenance. The licensee also performed a past operability evaluation and determined that the 'B' EDG was inoperable from February 12, 2007, until the leak was repaired and the EDG tested on April 12, 2007 (about 13.7 days). Licensee TS require two operable EDGs when in Modes 1, 2, and 3. During the period when the 'B' EDG was inoperable, the associated TS limiting condition for operation (LCO) was not entered, and subsequently the required completion time for the LCO was exceeded.

In addition, from February 12, 2007, through April 12, 2007, the 'A' EDG was also inoperable for planned maintenance during the following periods:

- Modes 4 and 5 (following completion of irradiated fuel movements in the afternoon) from approximately 2000 on March 1, 2007, until approximately 0100 on March 8, 2007; and

- Modes 1, 2, and 3 from approximately 1220 on April 1, 2007, until approximately 1725 on April 1, 2007.

As stated, TS require two operable EDGs when in Modes 1, 2, and 3. The TS also require one operable EDG when the plant is in Modes 4 and 5, or when movement of irradiated fuel is performed within secondary containment. Based on the above information, both EDGs were inoperable for approximately 6.2 days while the plant was in Mode 4 or 5. Both EDGs were also inoperable for about 5.1 hours while in Modes 1, 2, or 3. In both cases, the associated TS LCOs were not entered, and subsequently the required completion times for the LCOs were exceeded. The licensee evaluated these conditions to be of very low safety significance since the 'B' EDG would have operated for greater than 24 hours without operator action prior to failure, and the licensee's station blackout coping time is 6 hours. The licensee determined that this event did not result in a loss of safety system function.

<u>Analysis</u>: The inspectors determined that failing to ensure that correct, qualified replacement components were installed during maintenance activities associated with the EDG was an example of not complying with a standard that was reasonably within the licensee's ability to foresee, correct, and prevent, and was therefore a performance deficiency.

The inspectors reviewed this issue against the guidance contained in Appendix B, "Issue Screening," of Inspection Manual Chapter (IMC) 0612, "Power Reactor Inspection Reports." In particular, the inspectors compared this finding to the findings identified in Appendix E, "Examples of Minor Issues," of IMC 0612 to determine whether the finding was minor. Example f, of Section 4 for Insignificant Procedural Errors, was germane. The inspectors determined that the finding was more than minor because the 'B' EDG was returned to service with the incorrect o-ring installed and the leak that developed resulted in subsequent equipment inoperability. Additionally, the TS LCO allowed outage times for EDGs being inoperable with the plant at power were exceeded.

The inspectors evaluated the finding using the Phase 1 mitigating systems cornerstone worksheet from IMC 0609, Appendix A, "Significance Determination of Reactor

Inspection Findings for At-Power Situations." Since this issue was not a design or qualification deficiency, did not result in a loss of safety function, and was not considered potentially risk significant to a seismic, flooding, or severe weather initiating event, the issue screened as having a very low safety significance (Green).

The inspectors also determined that this finding was also related to the work practices component of the human performance cross-cutting area, because maintenance personnel failed to ensure that supervisory and management oversight of work activities, including contractors, supported nuclear safety. Specifically, the personnel performing maintenance activities for reassembly of the LOF were not supervised, an incorrect LOF cover 0-ring was installed, and the equipment was subsequently returned to service.[H.4(c)]

Enforcement: TS 3.8.1.b, Electrical Power Systems, AC Sources-Operating, Condition B, requires in part that, when one EDG is inoperable while in Modes 1, 2, and 3, the EDG be restored to an operable status within 7 days. Condition D requires that, when both EDGs are inoperable while in Modes 1, 2, and 3, that one EDG be restored to an operable status with 2 hours. Condition E requires that if the required actions and associated completion times of Condition B or D are not met, the plant shall be placed in Mode 3 within 12 hours AND be placed in Mode 4 within 36 hours. Contrary to these requirements, the licensee's past operability evaluation demonstrated that, while in Modes 1, 2, or 3, the 'B EDG had been inoperable for greater than 13 days, that both EDGs had been inoperable for approximately 5 hours, and that the required completion times for both Condition B and Condition D had been exceeded. The failure to comply with applicable LCO completion times is a violation of the licensee's TS. However, because of the low safety significance and because it was entered into the licensee's corrective action program, the NRC is treating this issue as an NCV in accordance with Section VI.A.1 of the NRC's Enforcement Policy (NCV 05000331/2007003-04). This issue was entered into the licensee's corrective action program as CAP 049012.

1R19 Post-Maintenance Testing (71111.19)

a. Inspection Scope

The inspectors reviewed post-maintenance testing (PMT) activities. The documents listed in the Attachment were used to accomplish the objectives of the inspection procedure. PMT procedures and activities were verified to be adequate to ensure system operability and functional capability. Inspection activities were selected based upon the SSCs ability to impact risk. Inspection activities included, but were not limited to, witnessing or reviewing the integration of testing activities, applicability of acceptance criteria, test equipment calibration and control, procedural use and compliance, control of temporary modifications or jumpers required for test performance, documentation of test data, system restoration, and evaluation of test data. Also, the inspectors verified that maintenance and PMT activities adequately ensured that the equipment met the licensing basis, TS, and UFSAR design requirements.

The inspectors selected the following PMT activities for review for a total of six samples:

- Preventative Work Order (PWO) 1133610, SV2259 - HPCI Turbine Remote Trip Valve - Replace the Solenoid Valve;
- PWO 1136111, Replace Scoop Tube Deviation Relay;
- Corrective Work Order (CWO) A77477, Steam Leaks by MO2202 - HPCI Turbine Steam Supply Valve;
- PWO 1139506, Re-calibration of the ‘B’ EDG Speed Sensing Switch;
- CWO A73696, Remove and Replace V13-0059 - ‘A’ Core Spray Pump Motor Cooler ESW Inlet Valve; and
- Surveillance Test Procedure (STP) NS640101, Core Flow Instrumentation Calibration.

b. Findings

No findings of significance were identified.

1R20 Outage Activities (71111.20)

.1 Forced Outage Due to Loss of the 1A2 Electrical Bus

a. Inspection Scope

On April 2, the licensee inserted a manual reactor scram to shutdown the reactor for a forced outage, due to a loss of the 1A2 electrical bus, that resulted in a loss of the ‘B’ Reactor Feed Pump and the ‘B’ Recirculation Pump. Activities monitored by the inspectors included the Control Room Operator’s post scram actions and plant restoration. Troubleshooting activities associated with the loss of the 1A2 electrical bus were observed by the inspectors and the restoration of power to the 1A2 bus was observed. Outage configuration management was also monitored on a daily basis by verifying that the licensee maintained appropriate defense in depth to address all shutdown safety functions and satisfy TS requirements. The inspectors observed the reactor plant startup and power ascension to full power. This counts as one inspection sample.

The reactor was restarted on April 4, following restoration of the 1A2 bus, and the generator connected to the grid on April 5. The documents listed in the Attachment were used to accomplish the objectives of the inspection procedure.

b. Findings

No findings of significance were identified.

1R22 Surveillance Testing (71111.22)

a. Inspection Scope

The inspectors reviewed surveillance test activities. Inspection procedure objectives were accomplished as indicated by the documents listed in the Attachment to this inspection report. Surveillance testing activities were reviewed to assess operational readiness and ensure that risk-significant SSCs were capable of performing their

intended safety function. Surveillance activities were selected based upon risk significance and the potential risk impact from an unidentified deficiency or performance degradation that a SSC could impose on the unit if the condition were left unresolved. Inspection activities included, but were not limited to, a review for preconditioning, integration of testing activities, applicability of acceptance criteria, test equipment calibration and control, procedural use, control of temporary modifications or jumpers required for test performance, documentation of test data, TS applicability, impact of testing relative to Performance Indicator (PI) reporting, and evaluation of test data.

The inspectors selected the following surveillance testing activities for review for a total of six samples:

- STP NS930001, Main Turbine Operational Tests (routine);
- STP 3.5.1-13, HPCI System Water Fill Test (routine);
- STP 3.5.1-10, HPCI System Operability Test and Comprehensive Pump Test (inservice testing);
- STP 3.1.7-01, Standby Liquid Control Operability Test (inservice testing);
- STP 3.3.3.2-04, Remote Shutdown Panel Functional test for RHR (routine); and
- STP 3.3.1.1-34, Recirculation Flow Unit Calibration (routine).

b. Findings

No findings of significance were identified.

1R23 Temporary Plant Modifications (71111.23)

a. Inspection Scope

Temporary modifications were reviewed to assess the modification’s impact on the safety function of the associated systems. The documents listed in the Attachment were used to accomplish the objectives of the inspection procedure. Inspection activities included, but were not limited to, a review of design documents, safety screening documents, UFSAR, and applicable TSs to determine that the temporary modification was consistent with modification documents, drawings and procedures. Inspectors also reviewed the post-installation test results to confirm that tests were satisfactory and the actual impact of the temporary modification on the permanent system and interfacing systems were adequately verified.

The inspectors selected the following temporary modifications for review for a total of three samples:

- Manual Calculation of Feedwater Correction Factor;
- Raised Setpoint For Main Turbine Hi Vibration Alarm; and
- Leak Repair for the Number Seven Stud of the ‘A’ Reactor Feed Pump.

b. Findings

No findings of significance were identified.

1EP6 Drill Evaluation (71114.06)

.1 Simulator Based Training Evolution

a. Inspection Scope

The inspectors observed simulator based training evolutions on June 11 and June 18. The training simulated a loss of both essential electrical busses followed by a small break loss of coolant accident inside the drywell containment.

Inspectors evaluated the licensee's training evolution conduct and the adequacy of the post-training performance critique to identify weaknesses and deficiencies. The documents listed in the Attachment were used to accomplish the objectives of the inspection procedure. Training evolutions that the licensee had previously scheduled were selected to provide input to the Drill/Exercise PI. Inspection activities included, but were not limited to, the classification of events, notifications to off-site agencies, and drill critiques. Inspector observations were compared with the licensee's observations. Inspectors verified that there were no discrepancies between observed performance and reported PI statistics.

This review represented one inspection sample.

b. Findings

No findings of significance were identified.

2. RADIATION SAFETY

Cornerstone: Public Radiation Safety

2PS2 Radioactive Material Processing and Transportation (71122.02)

.1 Radioactive Waste System

a. Inspection Scope

The inspectors reviewed the liquid and solid radioactive waste system description in the UFSAR for information on the types and amounts of radioactive waste generated and disposed. The inspectors reviewed the scope of the licensee's audit program with regard to radioactive material processing and transportation programs to verify that it met the requirements of 10 CFR 20.1101(c).

This review represented one inspection sample.

b. Findings

No findings of significance were identified.

.2 Radioactive Waste System Walkdown

a. Inspection Scope

The inspectors performed walkdowns of the liquid and solid radwaste processing systems to verify that the systems agreed with the descriptions in the UFSAR and the Process Control Program, and to assess the material condition and operability of the systems. The inspectors reviewed the status of radioactive waste process equipment that was not operational and/or was abandoned in place. The inspectors reviewed the licensee's administrative and physical controls to ensure that the equipment would not contribute to an unmonitored release path or be a source of unnecessary personnel exposure.

The inspectors reviewed changes to the waste processing system to verify the changes were reviewed and documented in accordance with 10 CFR 50.59 and to assess the impact of the changes on radiation dose to members of the public. The inspectors reviewed the current processes for transferring waste resin into shipping containers to determine if appropriate waste stream mixing and/or sampling procedures were utilized. The inspectors also reviewed the methodologies for waste concentration averaging to determine if representative samples of the waste product were provided for the purposes of waste classification in accordance with 10 CFR 61.55.

This review represented one inspection sample.

b. Findings

No findings of significance were identified.

.3 Waste Characterization and Classification

a. Inspection Scope

The inspectors reviewed the licensee's radiochemical sample analysis results for each of the licensee's waste streams, including dry active waste (DAW), spent resins and filters. The inspectors also reviewed the licensee's use of scaling factors to quantify difficult-to-measure radionuclides (e.g., pure alpha or beta emitting radionuclides). The reviews were conducted to verify that the licensee's program assured compliance with 10 CFR 61.55 and 10 CFR 61.56, as required by Appendix G of 10 CFR Part 20. The inspectors also reviewed the licensee's waste characterization and classification program to ensure that the waste stream composition data accounted for changing operational parameters and thus remained valid between the annual sample analysis updates.

This review represented one inspection sample.

b. Findings

No findings of significance were identified.

.4 Shipment Preparation

a. Inspection Scope

The inspectors reviewed shipment packaging, surveying, labeling, marking, placarding, vehicle checks, emergency instructions, disposal manifest, shipping papers provided to the driver, and licensee verification of shipment readiness for selected resins, dry active waste and surface-contaminated object surface contaminated object shipments. The inspectors verified that the requirements of any applicable transport cask Certificate of Compliance were met and verified that the receiving licensee was authorized to receive the shipment packages. The inspectors verified that the licensee's procedures for cask loading and closure procedures were consistent with the vendor's approved procedures. The inspectors interviewed and observed the radiation protection (shipper) and radwaste personnel conducting radioactive waste processing and radioactive material shipping preparation activities. The inspectors determined that shipping personnel had demonstrated adequate skills to accomplish the package preparation requirements for public transport with respect to NRC Bulletin 79-19 and 49 CFR Part 172 Subpart H. During the inspection, the inspectors observed shipping activities of Type-B package (cask shipment) containing dewatered reactor water cleanup/condensate resin.

The inspectors reviewed the training records of personnel responsible for the conduct of radioactive waste processing and radioactive shipment preparation activities. The review was conducted to verify that the licensee's training program provided training consistent with NRC and Department of Transportation requirements.

This review represented one inspection sample.

b. Findings

No findings of significance were identified.

.5 Shipping Records

a. Inspection Scope

The inspectors reviewed five non-excepted package shipment manifests/documents completed in 2006/2007 to verify compliance with NRC and Department of Transportation requirements (i.e., 10 CFR Parts 20, 71, and 49 CFR Parts 172 and 173).

This review represented one inspection sample.

b. Findings

No findings of significance were identified.

.6 Identification and Resolution of Problems

a. Inspection Scope

The inspectors reviewed condition reports, audits and self-assessments that addressed radioactive waste and radioactive materials shipping program deficiencies since the last inspection to verify that the licensee had effectively implemented the corrective action program and that problems were identified, characterized, prioritized, and corrected. The inspectors also verified that the licensee's self-assessment program was capable of identifying repetitive deficiencies or significant individual deficiencies in problem identification and resolution.

The inspectors also reviewed corrective action reports from the radioactive material and shipping programs since the previous inspection, interviewed staff, and reviewed documents to determine if the following activities were being conducted in an effective and timely manner commensurate with their importance to safety and risk:

- Initial problem identification, characterization, and tracking;
- Disposition of operability/reportability issues;
- Evaluation of safety significance/risk and priority for resolution;
- Identification of repetitive problems;
- Identification of contributing causes;
- Identification and implementation of effective corrective actions;
- Resolution of NCVs tracked in corrective action system(s); and
- Implementation/consideration of risk significant operational experience feedback.

This review represented one inspection sample.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA1 Performance Indicator Verification (71151)

Cornerstones: Barrier Integrity

.1 Reactor Safety Strategic Area

The inspectors reviewed the licensee PI submittals. PI guidance and definitions contained in Nuclear Energy Institute Document 99-02, “Regulatory Assessment Performance Indicator Guideline,” were used to verify the accuracy of the PI data. The documents listed in the Attachment were used to accomplish the objectives of the inspection procedure. The inspectors’ review included, but was not limited to, conditions and data from logs, Licensee Event Reports (LERs), condition reports, and calculations for each PI specified.

The following PIs were reviewed for a total of five samples:

- Unplanned Scrams per 7000 Critical Hours, for the period of January 2006 through March 2007;
- Unplanned Scrams with Loss of Normal Heat Removal, for the period of January 2006 through March 2007;
- Unplanned Power Changes per 7000 Critical Hours, for the period of January 2006 through March 2007;
- Reactor Coolant System Specific Activity, for the period of January 2006 through March 2007; and
- Reactor Coolant System Leakage, for the period of January 2006 through March 2007.

b. Findings

No findings of significance were identified.

4OA2 Identification and Resolution of Problems (71152)

Cornerstone: Initiating Events, Mitigating Systems, and Barrier Integrity

.1 Routine Review of Identification and Resolution of Problems

a. Inspection Scope

For inspections performed and documented in previous sections of this report, the inspectors routinely reviewed issues during baseline inspection activities and plant status reviews to verify that they were being entered into the corrective action program at an appropriate threshold, that adequate attention was being given to timely corrective actions, and that adverse trends were identified and addressed. Minor issues entered into the corrective action program as a result of the inspectors' observations are included in the attached list of documents reviewed.

b. Assessment and Observations

No findings of significance were identified.

.2 Daily Corrective Action Program Reviews

a. Inspection Scope

In order to assist with the identification of repetitive equipment failures and specific human performance issues for follow-up, the inspectors performed a daily screening of items entered into the licensee's CAP. This review was accomplished through inspection of the station's daily condition report packages.

b. Assessment and Observations

No findings of significance were identified.

.3 Annual Sample - Semi-Annual Trend Review

a. Inspection Scope

Inspectors performed a review of the licensee's CAPs and associated documents to identify trends that could indicate the existence of a more significant safety issue. This review primarily focused on repetitive equipment issues and maintenance activities, but also considered the results of daily inspector CAP item screening discussed in Section 4OA2.2 above, licensee trending efforts, and licensee human performance results. Nominally, the review considered the six-month period of January 2007 through June 2007, although some examples expanded beyond those dates when the scope of the trend warranted.

The inspectors' semi-annual trend review also included issues documented in major equipment problem lists, repetitive and/or rework maintenance lists, departmental problem/challenges lists, system health reports, quality assurance audit/surveillance reports, self-assessment reports, and Maintenance Rule assessments. The results of this trend review were compared and contrasted with the results contained in the licensee's corrective action program and Nuclear Oversight Department reports. Corrective actions associated with a sample of the trends identified by the licensee were reviewed for adequacy.

Inspectors also evaluated the licensee's trending CAPs against the requirements of the licensee's Corrective Action Program as specified in ACP 114.8, "Action Request Trending." Additional documents reviewed are listed in the attachment.

b. Assessment and Observations

No findings or issues of significance were identified.

.4 Annual Sample - Selected Issues Follow-up: Review of Complex Troubleshooting Processes and Products

a. Inspection Scope

During the performance of the baseline inspection samples for Event Follow-up and Operability Evaluations, the inspectors noted inconsistencies in the licensee's implementation of troubleshooting activities during investigation of equipment issues identified during Event Response activities and complex equipment problems that result from recurring events or negative performance trends. Based on this observation, the inspectors conducted an in-depth review of the licensee's troubleshooting activities performed during the past six-month period. The inspectors assessed the adequacy of procedural requirements for implementing troubleshooting processes during both event response and operational decision-making and issue management. The inspectors also assessed the degree of engineering rigor associated with the troubleshooting plans and

equipment evaluations developed. The assessments included a review of operability and reportability determinations, extent of condition evaluations, cause investigations, and the appropriateness of identified corrective actions. This inspection activity counts as one annual sample.

b. Assessment and Observations

No findings or issues of significance were identified.

4OA3 Event Follow-up (71153)

.1 (Closed) LER 05000331/2007002-00: "Loss of Control of Control Building Boundary"

On February 12, 2007, with the reactor in Mode 5 for a refueling outage, testing was performed to determine how the control building envelope was affected by opening two 4-inch penetrations from the turbine building into the cable spreading room. Subsequent to the completion of the testing, it was discovered that three additional penetrations had been opened between the control room and the cable spreading room, rendering the control building boundary inoperable for a period of time longer than allowed by TS 3.7.4, Condition F. Core alterations were in progress at the time and were suspended upon discovery that the control building boundary was inoperable. The LER was reviewed by the inspectors. Section 4OA7.1 describes a licensee-identified violation associated with the closure of this LER. The licensee entered this issue into their corrective action program as CAP 0473145. This LER is closed.

.2 (Closed) LER 05000331/2007004-00: "Severe Weather Causes Grid Disturbance Resulting in Loss of Shutdown Cooling"

On February 24, 2007, while the plant was in Mode 5 for a refueling outage, a severe winter storm brought freezing rain, ice, and high winds to the Duane Arnold Energy Center grid area causing degraded voltage conditions on the essential busses. At 1757, a full scram occurred due to a loss of 'B' Reactor Protection System (RPS) and Neutron Monitoring System Trip on the 'A' RPS, and Groups 1 through 5 isolations (excluding Main Steam Line Isolation Valves) occurred, resulting in a loss of shutdown cooling. At 1824, bus degraded voltage conditions caused both EDGs to load onto their respective busses. Shutdown Cooling was restored at 1826. Grid repair and recovery allowed the essential bus 1A4 power supply to be transferred from the 'B' EDG to the Startup Transformer at 1148 on February 25, 2007. The essential bus 1A3 power supply was transferred from the 'A' EDG to the Startup Transformer at 0049 on February 26, 2007. The licensee entered this issue into their corrective action program as CAP 047825 and determined that, since the event was caused by severe weather, no corrective actions are required.

The LER was reviewed by the inspectors and it was identified that the LER was submitted 61 days after the event date. This is contrary to the requirement of 10 CFR 50.63, that requires LERs be submitted within 60 days of an event. Section IV.A.3 of the NRC's Enforcement Policy states that the severity of an untimely report, in contrast to no report, may be reduced depending on the circumstances surrounding the matter. In this instance, because the LER was

submitted one day late, it was determined that the untimely submittal of the LER did not significantly impact the NRC's regulatory process and therefore was not more than minor. The licensee entered the untimely LER submittal issue into their corrective action program as CAP 049411. This LER is closed.

.3 (Closed) LER 05000331/2007005-00: "Automatic Reactor Scram Due to Scram Discharge Volume High Water Level During Performance of a Surveillance Test"

On March 2, 2007, during a refueling outage, STP NS550002 was being performed for testing of the Control Rod Drive System Back-up Scram Valves. The procedure required insertion of a manual reactor scram, however, the procedure did not require bypassing the Scram Discharge Volume High Level Scram signal prior to resetting the manual scram. Subsequently, an automatic reactor scram occurred at approximately 2332 due to a Scram Discharge Volume high level. All control rods were already fully inserted and no control rod motion occurred from the manual or automatic scram signal during the performance of the surveillance test. The licensee determined that the cause of the event was lack of guidance in the surveillance procedure to direct operators to bypass the Scram Discharge Volume High Level Signal prior to resetting the reactor scram. NRC Inspection report 05000331/2007002 documented a green finding and associated NCV (NCV 05000331/2007002-08) that was associated with this event for an unplanned RPS automatic scram due to an inadequate STP. The LER was reviewed by the inspectors and no additional finding of significance was identified and no additional violations of NRC requirements occurred. The licensee entered this issue into their corrective action program as CAP 048038. This LER is closed.

.4 (Closed) LER 05000331/2007006-00: "Reactor Shutdown as a Result of a Chemistry Excursion"

On March 18, 2007, while operating at 28 percent reactor power, a chemistry excursion occurred. The magnitude of the excursion required operators to shut down the reactor in accordance with abnormal operating procedures and plant chemistry procedures. The cause of the chemistry excursion was determined to be an intrusion of resin from the Condensate Filter Demineralizers into the Condensate System. The LER was reviewed by the inspectors and no finding of significance was identified and no violations of NRC requirements occurred. The licensee entered this issue into their corrective action program as CAP 048498. This LER is closed.

.5 Review of Personnel Performance When the HPCI Discharge Relief Valve Lifted During Planned Surveillance Testing

a. Inspection Scope

The inspectors reviewed the site response and personnel performance during an unplanned event when a relief valve that had recently been installed on the HPCI discharge line as part of a high pressure keep-fill modification, lifted while Operations personnel were performing STP 3.5.1-05, "HPCI System Operability Test," on March 28, 2007. The documents listed in the Attachment were used by the inspectors to accomplish the objectives of the inspection procedure. This review represented one sample.

b. Findings

Introduction: A finding of very low safety significance (Green) and an associated NCV of 10 CFR 50 Appendix B, Criterion 3, was self-revealed when PSV2302, the HPCI discharge pressure relief valve, lifted and remained open during planned testing of the HPCI System. The licensee entered this issue into the corrective action program for resolution. This issue was also related to the decision making component of the human performance cross-cutting area, because engineering personnel failed to conduct an effective review of the safety-significant HPCI keep-fill modification and identify that the relief valve setpoint did not provide sufficient margin to prevent an unintended consequence. Specifically, the lifting of the relief valve due to the peak HPCI system discharge pressure seen during system startup.

Description: On March 28, 2007, Operations personnel were performing STP 3.5.1-05, “HPCI System Operability Test.” Several minutes after starting the HPCI turbine, the control room received an alarm for Torus high water level. Upon further investigation, a Health Physics technician in the Torus area discovered a substantial amount of water on the floor of Torus bay 6. The source of the water was determined to be a lifted relief valve that had recently been installed on the HPCI discharge line as part of a high pressure keep-fill modification. The HPCI test was aborted and the system was declared inoperable. The HPCI discharge line was isolated, resulting in the system being declared unavailable. On April 1, 2007, a temporary modification was completed to remove the HPCI keep-fill modification. The HPCI system was returned to operable status upon successful completion of STP 3.5.1-05.

Subsequent evaluation by the licensee determined that engineering personnel failed to review previous internal operating experience at the Duane Arnold Energy Center (DAEC). The site’s corrective action program had multiple corrective action documents evaluating previous instances where the HPCI discharge pressure momentarily exceeded its design pressure during HPCI pump starts. It was determined that a review of DAEC internal operating experience during preparation and review for ECP 1797, HPCI Keep Fill System, failed to recognize that HPCI pump discharge pressure momentarily exceeded the piping design rating during system startup. This error resulted in the installation of relief valve PSV2302 with too low of a setpoint.

Analysis: The inspectors determined that the failure to provide sufficient margin between the HPCI discharge relief valve setpoint and the peak discharge pressure of the HPCI system upon startup was a performance deficiency warranting further evaluation. The inspectors reviewed this finding using the guidance contained in Appendix B, “Issue Disposition Screening,” of IMC 0612, “Power Reactor Inspection Reports.” The inspectors compared this finding to the findings identified in Appendix E, “Examples of Minor Issues,” of IMC 0612 to determine whether the finding was minor. Example j, of Section 3 for Non-significant Dimensional, Time, Calculation, or Drawing Discrepancies, was germane. Engineering personnel used a non-conservative value for the setpoint of the HPCI discharge relief valve. The issue was more than minor since the engineering calculation error resulted in a condition where there was a reasonable doubt on the operability of the HPCI system.

The inspectors reviewed this finding in accordance with IMC 0609 Appendix A, "Determining the Significance of Reactor Inspection Findings for At-Power Situations." This issue screened as having a very low safety significance (Green) since the finding is a design deficiency confirmed not to result in a loss of operability per the part 9900 technical guidance for operability determination process for operability and functional assessment.

The inspectors also determined that the cause of this finding was related to the decision making component of the human performance cross-cutting area because engineering personnel failed to conduct an effective review of the safety-significant HPCI keep-fill modification and identify that the relief valve setpoint did not provide sufficient margin to prevent an unintended consequence. Specifically, the lifting of the relief valve due to the peak HPCI system discharge pressure seen during system startup.[H.1(b)]

Enforcement: 10 CFR 50 Appendix B, Criterion 3, "Design Control," requires that measures shall be established for the selection and review for suitability of application of materials, parts, equipment, and processes that are essential to the safety-related function of structures, systems, and components. Contrary to this requirement, a relief valve was installed in the HPCI discharge piping that had a relief setpoint below the discharge pressure of the HPCI system upon system startup. This resulted in the relief valve lifting and not reseating upon HPCI startup for testing. Because this violation was of very low safety significance and it was entered into the licensee's corrective action program, this violation is being treated as an NCV consistent with Section VI.A of the NRC Enforcement Policy (NCV 05000331/2007003-01). The licensee entered this into their corrective action program as CAP 048702. A temporary modification was performed to remove the HPCI high pressure keep fill modification from service and the HPCI system was returned to operable status.

.6 Review of Personnel Performance During a Lockout of the 1A2 Non-Essential Bus Which Resulted in the Insertion of a Manual Reactor Scram Due to Lowering RPV Level

a. Inspection Scope

The inspectors reviewed the site response and personnel performance during an unplanned event when an isolation of the 1A2 non-essential 4160VAC electrical bus occurred while Maintenance personnel were performing planned preventative maintenance on the 1A2 bus lockout relay. The loss of the 1A2 switchgear resulted in the loss of the 'B' RFP and 'B' Condensate Pump, and a manual reactor scram was inserted due to RPV level approaching 170 inches (automatic scram level). The inspectors observed the operator responses, investigation and repair activities, and the subsequent plant recovery. The documents listed in the Attachment were used by the inspectors to accomplish the objectives of the inspection procedure. This review represented one sample.

b. Findings

Introduction: A finding of very low safety significance (Green) was self-revealed when the Control Room crew, while performing RPV water level recovery actions following the manual scram initiated from 98 percent reactor power on the loss of

the 1A2 non-essential bus, did not recover feedwater flow in a timely manner. The licensee's actions resulted in a second automatic scram due to low RPV water level after the initial manual scram had been reset.

Description: On April 2, 2007, while operating at 98 percent reactor power, the licensee was performing planned preventative maintenance on the 1A2 non-essential 4160VAC electrical bus. At 1125, the 186-2 lockout relay tripped and an isolation of the 1A2 bus occurred. The loss of the 1A2 switchgear resulted in the loss of the 'B' RFP and 'B' Condensate Pump. A manual reactor scram was inserted due to RPV level approaching 170 inches (automatic scram level). Following the scram, the RPV level rose to 211 inches, due to feedwater responding to the low RPV level condition, which caused the 'A' RFP to trip. While performing subsequent recovery actions to restore RPV level control, the reactor operator chose to control feedwater flow with the Startup Feedwater Regulating Valve (FRV) instead of the 'A' FRV, based upon prior training. The normal operational lineup has the Startup FRV aligned to the 'B' RFP discharge and manually isolated from the 'A' RFP. Therefore, as RPV level lowered, the reactor operator started the available 'A' RFP and attempted to control RPV level using the Startup FRV. RPV level continued to lower. Although these actions were observed and corrected by the Operations Shift Manager, who directed the reactor operator to control RPV level by using the 'A' FRV, the untimely Control Room Crew response resulted in a second automatic scram signal due to low RPV level. All control rods were already fully inserted and no control rod motion occurred from the automatic scram signal.

The inspectors reviewed several licensee procedures to assess the adequacy of crew response for control of RPV level following a scram. ACP 101.01, "Procedure Use and Adherence," provides expectations for technical procedure level of use classifications and usage level requirements. The required operator actions are specified in Integrated Plant Operating Instruction (IPOI) 5 Quick Response Card 1, "Reactor Scram Immediate Operator Actions," and Operating Instruction (OI) 644 Quick Response Card 1, "Restoring Feedwater." All of these procedures are designated Reference Use procedures, and as such do not have the same immediate consequence or require the same level of procedure use rigor as Continuous Use procedures. Based upon direct observation during the event, the inspectors determined that the failure to positively take control of RPV level was the result of deficiencies in operator plant awareness, not inappropriate use of the procedures.

Analysis: The inspectors determined that the operator's failure to take positive control of the critical parameter of RPV level following the insertion of the manual scram, which resulted in the second automatic reactor protection system actuation, was a failure to meet a standard to prevent avoidable system actuations, was reasonably within the licensee's ability to foresee, correct, and prevent, and was therefore, a performance deficiency.

The inspectors reviewed this issue against the guidance contained in Appendix B, "Issue Screening," of IMC 0612, "Power Reactor Inspection Reports." In particular, the inspectors compared this finding to the findings identified in Appendix E, "Examples of Minor Issues," of IMC 0612 to determine whether the finding was minor. Example b, of Section 4 for Insignificant Procedural Errors, was germane. The inspectors determined

that the finding was more than minor because it adversely impacted the initiating events cornerstone attribute for human performance which limits the likelihood of events that upset plant stability and challenge critical safety functions. Specifically, the failure to control RPV level following the manual scram resulted in a subsequent automatic reactor protective system actuation.

The inspectors evaluated the finding using IMC 0609, Appendix A, "Significance Determination of Reactor Inspection Findings for At-Power Situations." Using the Phase 1 SDP worksheet for the initiating event cornerstone, transient initiator contributor, the inspectors determined that the finding did not contribute to both the likelihood of a reactor trip and the likelihood that mitigation equipment or functions will not be available. Therefore, the finding screened as very low safety significance (Green). Additionally, the inspectors determined that a cross-cutting aspect was not a significant contributor to this performance deficiency.

Enforcement: The inspectors determined that although the operator responses to verify recovery of feedwater flow were not timely enough to prevent the second automatic RPS actuation, the required procedures were adequate and properly implemented. Therefore, because no 10 CFR 50, Appendix B, components were impacted by the Finding (FIN 05000331/2007003-05), a violation of NRC requirements did not occur. This issue was entered into the licensee's corrective action program as CAP 048784.

.7 (Closed) LER 05000331/2007007-00: "Reactor Scram Due to 1A2 Non-Essential Bus Lockout"

On April 2, 2007, with the plant operating at 98 percent reactor power, the licensee was performing planned preventative maintenance on the non-essential 4160VAC electrical bus 1A2. At 1125, the 186-2 lockout relay tripped and an isolation of the 1A2 bus occurred. This caused a loss of the 'B' RFP and 'B' Condensate Pump. The operators inserted a manual reactor scram due to lowering RPV level, which was approaching 170 inches (automatic scram level). During the subsequent recovery actions, a second automatic RPS actuation occurred prior to reestablishing feedwater flow and positive control of RPV level. All control rods were already fully inserted and no control rod motion occurred from the automatic scram signal. The licensee determined that, although a human performance event (bumping the relay during testing) was the most likely cause of the bus lockout relay tripping, no definitive cause was identified during the root cause investigation. The licensee performed a like for like replacement of the 186-2 lockout relay and five of the six overcurrent relays which provide trip signal inputs to the lockout relay. A Green finding, with no associated violation of NRC requirements, was documented in Section 4OA3.6 of this report. This LER was reviewed by the inspectors and no additional finding of significance was identified and no additional violations of NRC requirements occurred. The licensee entered this issue into their corrective action program as CAP 048780. This LER is closed.

.8 (Closed) LER 05000331/2007008-00: "Condition Prohibited by TSs; 'B' Emergency Diesel Inoperable"

On April 11, 2007, while operating at 98 percent reactor power, a 0.21 gallon per minute lube oil leak was observed coming from the 'B' EDG LOF cover during performance of

STP 3.8.1-04. The STP was aborted and the EDG was shutdown. The licensee performed an apparent cause evaluation and determined that an incorrect LOF cover o-ring had been installed on February 12, 2007, during the cylinder liner replacement maintenance performed during the refueling outage, RFO-20. The licensee also performed a past operability evaluation and determined that the 'B' EDG was inoperable from February 12, 2007 until the leak was repaired and the EDG tested and subsequently declared operable on April 12, 2007. Since the evaluation demonstrated that the EDG would have operated for longer than 24 hours without operator action prior to failure, the event did not result in a loss of safety system function. A Green finding and associated NCV was documented in Section 1R15.b.1 of this report. This LER was reviewed by the inspectors and no additional finding of significance was identified and no additional violations of NRC requirements occurred. The licensee entered this issue into their corrective action program as CAP 049012. This LER is closed.

.9 Observation of Personnel Performance During Non-Routine Planned Evolution: Quarterly Plant Downpower and Control Rod Sequence Exchange

a. Inspection Scope

The inspectors reviewed personnel performance during one sequence of planned downpower evolutions which included performance of a control rod sequence exchange, quarterly operator walkdowns of the feedwater heater and condensate bays, quarterly main turbine STPs, and repair of a steam trap in the condenser bay. The inspectors observed selected evolutions and briefings, and reviewed associated procedures, contingency plans, and records of operator performance. The documents listed in the Attachment were used by the inspectors to accomplish the objectives of the inspection procedure.

This review represented one sample.

b. Findings

No findings of significance were identified.

4OA5 Other Activities

Cornerstone: Initiating Events, Mitigating Systems

.1 Failure to Provide Complete and Accurate Information to the NRC on NRC Form 396

Introduction: The inspectors identified a Severity Level IV NCV of 10 CFR 50.9, "Completeness and Accuracy of Information." The inspectors identified that the facility licensee, on March 30, 2007, submitted to the NRC, an NRC Form 396, "Certification of Medical Examination By Facility Licensee," for a licensed operator applying for renewal of his reactor operator license, that was not complete and accurate in all material respects. Specifically, the NRC Form 396 certified that the licensed operator was not required to have a "corrective lens" restriction on his license. When the NRC questioned the licensee on the accuracy of the date of the most recent biennial medical

examination on the submitted NRC Form 396, the licensee submitted a revised NRC Form 396 on April 19, 2007. The revised NRC Form 396 included a new date for the most recent biennial medical examination, but also showed that the licensed individual was required to have a “corrective lens” restriction added to his license. The finding was determined to be of low safety significance because the license renewal application for the reactor operator was not renewed until complete and accurate information was received on revised NRC Form 396 that showed that a “corrective lens” restriction for the licensed individual.

Description: By a letter dated March 30, 2007, the facility licensee transmitted license renewal applications to the NRC Region III. The license renewal applications were for three reactor operators (ROs) whose existing licenses would expire on May 1, 2007. The license renewal applications included an NRC Form 396, “Certification of Medical Examination by Facility Licensee,” for each of the three ROs. NRC Form 396 included a block on the form to record the “Most Recent Biennial Medical Examination Date” and included blocks on the form to record the restrictions that were conditioned on the operator’s license. The “Most Recent Biennial Medical Examination Date” block was listed as March 9, 2005, for one RO, March 15, 2005, for the second RO, and March 28, 2005, for the third RO. The NRC Form 396 for each of the three reactor operators was certified as being true and correct by the Site Vice President on March 29, 2007. When received by NRC Region III on April 3, 2007, the Region III Licensing Assistant noted that the “Most Recent Biennial Medical Examination Date” listed on each of the three NRC Form 396's did not meet the requirement per 10 CFR 55.21 for a license holder to have a medical examination by a physician every two years. When the NRC questioned the licensee on the accuracy of the dates of the most recent biennial medical examination on the submitted NRC Form 396's, the licensee submitted revised NRC Form 396's on April 19, 2007. The revised NRC Form 396's included new dates for the most recent biennial medical examinations. The “Most Recent Biennial Medical Examination Date” for the three ROs were, respectively, March 16, 2007, March 20, 2007, and March 23, 2007. However, the revised NRC Form 396 for one licensed individual showed that a “corrective lens” restriction was required to be added to his license. The original NRC Form 396 for this licensed operator submitted on March 30, 2007, incorrectly stated that the only restriction was “Must Take Medication as Prescribed to Maintain Medical Qualifications.” However, the medical examination performed on March 16, 2007, for the license holder showed that his uncorrected near vision did not meet the minimum requirements specified in American National Standards Institute/American Nuclear Society (ANSI/ANS)-3.4 - 1983, “American National Standard Medical Certification and Monitoring of Personnel Requiring Operator Licenses for Nuclear Power Plants,” Section 5.4.5, “Eyes.” Duane Arnold Energy Center was committed to ANSI/ANS-3.4 - 1983. Thus, the licensed operator was required to have an additional restriction that “Corrective Lenses Be Worn When Performing Licensed Duties.”

Since NRC intervention was required to identify that the original submitted NRC Form 396 did not include a “corrective lens” restriction, this violation was considered NRC identified. The incorrect information provided on the original NRC Form 396 could have impacted an NRC licensing decision. The licensed operator could have, without NRC intervention, been issued a license without a “corrective lens” restriction added to

his license, resulting in an incorrect licensing action. Subsequently, additional information was required to allow the NRC to make the appropriate licensing decision.

Analysis: The inspectors determined that the failure to provide complete and accurate information to the NRC regarding the medical examination for the licensed operator was a significant regulatory issue and a violation of 10 CFR 50.9. Because violations of 10 CFR 50.9 are considered to be violations that potentially impede or impact the regulatory process, they are dispositioned using NUREG-1600, "General Statement of Policy and Procedure for NRC Enforcement Actions" (Enforcement Policy), instead of the SDP. Using IMC 0612, Appendix B, “Issue Dispositioning Screening,” the finding was determined to be more than minor because the information associated with the license renewal of the individual was provided to the NRC under a signed statement by the Site Vice President and could have impacted an NRC licensing decision. The finding was determined to be of low safety significance because the license renewal application for the reactor operator was not renewed until complete and accurate information was received on a revised NRC Form 396 that correctly showed a “corrective lens” restriction for the licensed individual. However, the finding was determined to be of significant regulatory importance because the incorrect information was provided under a signed statement to the NRC and could have impacted a licensing decision for the individual. The licensed operator could have, without NRC intervention, been issued a license without a “corrective lens” restriction added to his license. The NRC relies on Form 396 to determine whether an applicant meets the requirements of 10 CFR Part 55 to operate the controls of a nuclear power plant.

Enforcement: Section 50.9 of 10 CFR required that information provided to the Commission by a licensee shall be complete and accurate in all material respects. Section 55.23 of 10 CFR required, in part, that an authorized representative of the facility licensee shall complete and sign Form NRC - 396, "Certification of Medical Examination by Facility Licensee." Form NRC - 396, when signed by an authorized representative of the facility licensee, certifies that a physician conducted a medical examination of the applicant (as required in 10 CFR 55.21), and that the guidance contained in American National Standards Institute/American Nuclear Society (ANSI/ANS)-3.4 - 1983, “Medical Certification and Monitoring of Personnel Requiring Operator Licenses for Nuclear Power Plants,” was followed in conducting the examination and in making the determination of medical qualification.

Contrary to the above, in a letter dated March 30, 2007, the licensee submitted a license renewal application to the NRC for a licensed operator that was not complete and accurate in all material respects. Specifically, the NRC Form 396 certified that the licensed operator was not required to have a “corrective lens” restriction on his license. In fact, the licensed operator was required to have a “corrective lens” restriction added to his license pursuant to the requirements of ANSI/ANS-3.4 - 1983, Section 5.4.5. This information was material to the NRC because the NRC relies on Form 396 to determine whether the applicant meets the requirements of 10 CFR Part 55 to operate the controls of a nuclear power plant.

This finding is considered a violation of 10 CFR 50.9. However, because this issue was not willful, was of very low safety significance, and was entered into the licensee’s

corrective action program (CAP 049165), the issue is being treated as a NCV consistent with Section VI.A.1 of the NRC Enforcement Policy. (NCV 05000331/2007003-02).

.2 Failure to Implement Appropriate Controls Prior to Using Nylon Rope to Store Items in the Spent Fuel Pool

Introduction: A finding of very low safety significance (Green) and an associated NCV of 10 CFR 50 Appendix B, Criterion 5, was identified by the inspectors when licensee staff failed to implement the appropriate controls to properly store underwater lights in the spent fuel pool, thereby increasing the risk of these items potentially falling on the fuel bundles. The licensee entered this issue into the corrective action program for resolution. This issue was also related to the work practices component of the human performance cross-cutting area. Specifically, the aspect related to procedural compliance, as the station procedure that described the appropriate controls for storing items in the pool, was not followed.

Description: On April 10, 2007, the inspectors observed nylon rope being used to secure various equipment in the spent fuel pool and cask areas. Of particular concern, were underwater lights suspended by the ropes above the fuel assembly racks in the spent fuel pool. The inspectors did not notice any tags identifying when or why the lights had been placed. The licensee reactor engineering staff determined that the lights had been informally placed about two weeks earlier, to support upcoming planned activities. This was contrary to the licensee's requirements for storage of items in the spent fuel pool.

Analysis: Station procedure 1407.2, "Material Control in the Spent Fuel Pool and Cask Pool," revision 15, step 3.1(5), in part, prohibited the use of nylon rope to store items in the pool unless the items were used as part of Work-in-Progress activities. Step 3.2(1) of this procedure required, in part, that all stored items (except those exempt by procedure) have an associated storage permit tag. Although the ropes were installed to support planned work, they were placed outside the work control process and therefore did not have a scheduled task for installation and removal of the ropes nor any associated storage permit tags.

As stated in ACP 1407.2, "Nylon rope has the potential to degrade in a radiation environment and to act as a wick when extended into the pool." This was supported by industry experience, notably in NRC Information Notice 90-33, issued in May 1990. The licensee believed that the ropes were inappropriately placed, in part, due to confusing guidance in the procedure. Although the procedure clearly stated that nylon ropes were not to be used unless to support work-in-progress, it implied that underwater lighting was exempt from these requirements. However, the reactor engineering staff, who were responsible for items stored in the pool, stated that the implied exemption was incorrect and that the stated controls over nylon ropes were the intent of the procedure. Subsequently, the licensee assigned a work task to remove the ropes and initiated CAP 49009 to document the issue.

The inspectors determined that the failure to properly store the underwater lights in the spent fuel pool was a performance deficiency warranting further evaluation. The inspectors reviewed this finding using the guidance contained in Appendix B, "Issue

Disposition Screening," of IMC 0612, "Power Reactor Inspection Reports." The inspectors determined that the issue was more than minor because the finding could be reasonably viewed as a precursor to a more significant event. Specifically, the failure to follow the approved process for controlling the use of nylon ropes in the spent fuel pool, could result in the ropes being in place for an extended period of time. This increased the potential for unplanned radiation exposure either due to wicking or from damage to the underlying fuel assemblies, if the ropes degraded causing the lights to fall.

The inspectors reviewed this finding in accordance with IMC 0609, Appendix A "Determining the Significance of Reactor Inspections Findings for At-Power Situations." Specifically, the inspectors performed a Phase I SDP evaluation of this issue. This issue was determined to affect only the fuel cladding function of the Barrier Cornerstone. The finding did not require a phase 2 quantitative assessment and was therefore considered to be of very low safety significance (Green).

The inspectors also determined that the cause of this finding was related to the work practices component of the human performance cross-cutting area. Specifically, the aspect related to procedural compliance, as the station procedure that described the appropriate controls for storing items in the pool, was not followed. [H.4(b)]

Enforcement: 10 CFR 50 Appendix B, Criterion 5, "Instructions, Procedures, and Drawings," requires that activities involving quality be accomplished in accordance with proscribed instructions, procedures, or drawings. Contrary to this requirement, nylon ropes were used to secure items in the spent fuel pool, absent the specific controls stated in station procedure 1407.2. Specifically, the ropes were placed outside the work control process and did not have a scheduled task for installation and removal of the ropes nor any associated storage permit tags. Because this violation was of very low safety significance and it was entered into the licensee's corrective action program, this violation is being treated as a NCV consistent with Section VI.A of the NRC Enforcement Policy (NCV 05000331/2007003-03). The licensee removed the ropes and initiated CAP 049009.

.3 (Closed) Unresolved Item 05000331/2006014-01: "Surveillances and Compensatory Measures for Appendix 'A' Fire Barriers"

The licensee was not conducting surveillances of nor requiring compensatory measures for impairment of fire barriers for the diesel generator rooms. The fire barriers for the diesel generator rooms were explicitly credited in the DAEC fire protection Safety Evaluation Report. This issue was NRC identified.

The inspectors determined that the failure to perform surveillances of fire barriers which were explicitly credited as part of the DAEC fire protection licensing basis was a violation of the DAEC fire protection license condition. The removal of barriers from a surveillance program which were explicitly credited in the fire protection licensing basis was beyond the scope of changes permitted under the fire protection license condition.

During the original September 2006 inspection of this issue, the concern was that, if left uncorrected, barriers could degrade over time without appropriate surveillance activities. However, the inspectors had not identified any actual degradation of fire barriers during

that inspection. The inspectors noted that FPL Energy Duane Arnold, LLC, the licensee, had committed to adopt the National Fire Protection Association Standard (NFPA) 805 code, “Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants, 2001 Edition,” as endorsed by 10 CFR 50.48(c) for DAEC. As part of the transition to NFPA 805, the licensee will re-evaluate the fire protection program and determine which fire barriers will be credited under the fire protection program. Section 3.2.3(1) of NFPA 805 required that procedures be established for inspection, testing, and maintenance for fire protection systems and features credited by the fire protection program. Since no actual degradation was identified and because the violation will be addressed as part of the licensee’s transition to NFPA 805, this violation is considered minor as discussed in Inspection Manual Chapter 0612, “Power Reactor Inspection Reports.” This Unresolved Item is considered closed.

.4 (Closed) Unresolved Item 05000331/2007002-04: “Control Building Envelope Inoperable”

On February 10, 2007, maintenance personnel identified that work order steps were not followed resulting in two penetrations between the cable spreading room and turbine building being opened and not worked in the order planned. On February 12, 2007, engineering personnel wrote a TIF to determine the effect upon the control building envelope of open penetrations between the cable spreading room and the turbine building. This TIF simulated the previously identified open penetrations by cracking open a door between the cable spreading room and the administrative building and then measuring the control building differential pressure relative to the outside atmosphere. Subsequent to completion of this TIF, it was discovered that three additional penetrations had been opened between the control room and cable spreading room, revealing that the control building boundary was inoperable for a period of time longer than that allowed by TS 3.7.4, Condition F. Section 4OA7.1 describes a licensee-identified violation associated with this URI. This URI is closed.

.5 (Closed) Unresolved Item 05000331/2006004-01: “Licensee Did Not Conduct Periodic Testing of All Simulator Malfunctions Used in Operator Qualification”

During a Licensed Operator Requalification Program inspection documented in Inspection Report 05000331/2006004 (DRP); 05000331/2006015 (DRS), NRC inspectors determined that 10 malfunctions used in the current requalification operations exam did not have simulator testing documentation available for review. The lead inspector opened URI 05000331/2006004-01, “Licensee Did Not Conduct Periodic Testing of All Simulator Malfunctions Used in Operator Qualification,” to track this possible violation of NRC requirements. The licensee then found and provided copies of the original malfunction test documentation, documenting that these malfunctions were tested in the early 1990's. Following discussions with NRC headquarters program office personnel, a clarification of NRC requirements was given for addressing the adequacy of this testing in accordance with Regulatory Guide 1.149, Revision 1. The clarification was that malfunctions in addition to the 25 listed in Section 3.1.2 of ANSI/ANS-3.5-1985 did not require periodic performance testing to ensure simulator fidelity, and only had to be tested prior to initial use. After reviewing the simulator performance test documentation provided for the 10 simulator

malfunctions, it was determined that the testing performed had been tested prior to initial use and no violation of NRC requirements occurred. This item is closed.

4OA6 Meetings

.1 Exit Meeting

The inspectors presented the inspection results to Mr. G. Van Middlesworth and other members of licensee management on 07/12/2007. 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.

.2 Interim Exit Meetings

Interim exit meetings were conducted for:

- Heat sink biennial inspection with Mr. G. Van Middlesworth and other members of licensee management and staff at the conclusion of the inspection, on April 27, 2007.

- Closure of Unresolved Item 05000331/2006014-01 with Mr. D. Curtland, on May 23, 2007.

- Radioactive Material Processing and Transportation Inspection with Mr. J. Bjorseth, Site Director, Mr. D. Curtland, Plant Manager and Mr. C. Dieckmann, Operation Manager, on May 25, 2007.

- Reviewing NCV 05000331/2007003-02 with Mr. J. Morris, Training Manager, and Mr. S. Catron, Regulatory Assurance Manager, on June 12, 2007.

- Licensed Operator Requalification Program Unresolved Item Inspection with Ms. Diane Englehardt, Acting Training Manager, Duane Arnold Energy Center, on July 23, 2007, via telephone.

4OA7 Licensee-Identified Violations

The following violation of very low significance (Green) was identified by the licensee and is a violation of NRC requirements which meets the criteria of Section VI of the NRC Enforcement Manual, NUREG-1600, for being dispositioned as a NCV.

Cornerstone: Barrier Integrity

.1 Technical Specification 3.7.4, Condition F, requires that at least one standby filter unit be operable during movement of irradiated fuel assemblies in the secondary containment during core alterations, or during operations with the potential to drain the reactor vessel. Contrary to this requirement, the licensee discovered on February 12, 2007, that the control building envelope had been inoperable for as

much as 34 hours and 52 minutes with core alterations in progress, a condition prohibited by TSs. Once the condition was identified, core alterations were suspended. Since an actual demand was not imposed upon the standby filter unit system during the period of inoperability and the finding represented only a degradation of the radiological barrier function for the control room, this issue is of very low safety significance. The licensee documented the issue in their corrective action program as CAP 047315.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee
G. Van Middlesworth, Site Vice President
J. Bjorseth, Site Director
D. Curtland, Plant Manager
S. Catron, Licensing Manager
J. Cadogan, Engineering Director
E. Christopher, GL 89-13 Program Owner
P. Collingsworth, System Engineer
D. Englehardt, Acting Training Manager
B. Kindred, Security Manager
J. Morris, Training Manager
C. Dieckmann, Operations Manager
G. Pry, Maintenance Manager
J. Windschill, Chemistry & Radiation Protection Manager
P. Sullivan, Emergency Preparedness Manager
G. Ellis, Program Owner, Fire Protection
J. Kuehl, Supervisor, Programs Engineering
D. Albrecht, Radwaste Supervisor
R. Patrilla, Shipping Coordinator
A. J. Roderick, Principal Mechanical Engineer

Nuclear Regulatory Commission
Karl Feintuck, Project Manager, NRR
Kenneth Riemer, Chief, Reactor Projects Branch 2

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

05000331/2007003-01	NCV	Lifting of HPCI Discharge Relief Valve During Planned Surveillance Testing (Section 4OA3.5)
05000331/2007003-02	NCV	Failure to Provide Complete and Accurate Information to the NRC on NRC Form 396 (Section 4OA5.1)
05000331/2007003-03	NCV	Failure to Implement the Appropriate Procedural Controls Prior to Using Nylon Rope to Secure Underwater Lights in the Spent Fuel Pool (Section 4OA5.2)
05000331/2007003-04	NCV	TS Allowed Outage Time Exceeded for Inoperable EDGs (Section 1R15)
05000331/2007003-05	FIN	Operators Failed to Control a Critical Parameter and Received a Subsequent Automatic Scram Signal (Section 4OA3.6)

Closed

05000331/2007003-01	NCV	Lifting of HPCI Discharge Relief Valve During Planned Surveillance Testing (Section 4OA3.5)
05000331/2007003-02	NCV	Failure to Provide Complete and Accurate Information to the NRC on NRC Form 396 (Section 4OA5.1)
05000331/2007003-03	NCV	Failure to Implement the Appropriate Procedural Controls Prior to Using Nylon Rope to Secure Underwater Lights in the Spent Fuel Pool (Section 4OA5.2)
05000331/2007003-04	NCV	TS Allowed Outage Time Exceeded for Inoperable EDGs (Section 1R15)
05000331/2007003-05	FIN	Operators Failed to Control a Critical Parameter and Received a Subsequent Automatic Scram Signal (Section 4OA3.6)
05000331/2007002-00	LER	Loss of Control of Control Building Boundary (Section 4OA3.1)
05000331/2007004-00	LER	Severe Weather Causes Grid Disturbance Resulting in Loss of Shutdown Cooling (Section 4OA3.2)
05000331/2007005-00	LER	Automatic Reactor Scram Due to Scram Discharge Volume High Water Level During Performance of a Surveillance Test (Section 4OA3.3)
05000331/2007006-00	LER	Reactor Shutdown as a Result of a Chemistry Excursion (Section 4OA3.4)
05000331/2007007-00	LER	Reactor Scram Due to 1A2 Non-essential Bus Lockout (Section 4OA3.7)
05000331/2007008-00	LER	Condition Prohibited by TSs; 'B' Emergency Diesel Inoperable (Section 4OA3.8)
05000331/2006014-01	URI	Surveillances and Compensatory Measures for Appendix 'A' Fire Barriers (Section 4OA5.3)
05000331/2007002-04	URI	Control Building Envelope Inoperable (Section 4OA5.4)
05000331/2006004-01	URI	Licensee Did Not Conduct Periodic Testing of All Simulator Malfunctions Used in Operator Qualification (Section 4OA5.5)

Discussed

None.

LIST OF DOCUMENTS REVIEWED

The following is a list of documents reviewed during the inspection. Inclusion on this list does not imply that the NRC inspectors reviewed the documents in their entirety but rather that selected sections of portions of the documents were evaluated as part of the overall inspection effort. Inclusion of a document on this list does not imply NRC acceptance of the document or any part of it, unless this is stated in the body of the inspection report.

1R01 Adverse Weather

OI 711; Pumphouse HVAC System; Revision 4
OI 711A1; Pumphouse HVAC System Electrical Lineup; Revision 1
ACP 101.16; Midwest ISO Real-Time Operations: Communications and Mitigation Protocols for Nuclear Plant/Electrical Systems Interface; Revision 0
IPOI 6; Weather Impacted Operations; Revision 40
CAP 049885; Red Rubber Gaskets for the EHC Coolers
CAP 049913; Implementation of the NERC Requirements
CAP 049821; 1E058A (EHC Cooler) Inlet, Outlet Cooler Head Could not be Replaced Due to Size Differences
CAP 049873; 1E058A Leak During Post Maintenance Testing under Work Order A75843
CAP 049447; Threaded Connections Used on 2.5 Inch JBD Piping
CAP 049860; General Service Water Ultrasonic Testing Thickness Readings Found Below Minimum Requirements
CWO A75843; Replace Zinc Anodes and End Bell on 1E058A Cooler

1R04 Equipment Alignment

OI 150A1; RCIC System Electrical Lineup; Revision 0
OI 150A2; RCIC System Valve Lineup and Checklist; Revision 10
OI 150A4; RCIC System Control Panel Lineup; Revision 2
Drawing M-124; RCIC System P&ID; Revision 41
Drawing M-125; RCIC System P&ID; Revision 30
OI 416; RHRSW System; Revision 48
OI 416A6; RHRSW System Control Panel Lineup; Revision 4
OI 416A4; ‘B’ RHRSW System Valve Checklist; Revision 9
OI 416A1; RHRSW System Electrical Lineup; Revision 2
OI 454A4; ‘B’ Emergency Service Water System Valve Lineup and Checklist
OI 151; CS System; Revision 52
OI 151A4; ‘B’ CS System Valve Lineup and Checklist; Revision 3
OI 151A6; CS System Control Panel Lineup; Revision 1
OI 151A1; CS System Electrical Lineup; Revision 2
OI 151A2; ‘A’ CS System Valve Lineup and Checklist; Revision 3
CAP 040674; Discrepancy for CS Pump Shutoff Head
CAP 041839; AOP-913 Attachment 1 has Action in Conflict with RB1 20 Minute Action Statement
OI 410; RWS System; Revision 54
OI 410A1; RWS System Electrical Lineup; Revision 8
OI 410A2; ”A” RWS System Valve Lineup and Checklist; Revision 16

1R05 Fire Protection

AFP 1; Reactor Building Torus Area and North Corner Rooms, Elevations 716'9" and 735'7.5"; Revision 24
AFP 3; Reactor Building HPCI, RCIC and Radwaste Tank Rooms, Elevations 716'9" and 747'0"; Revision 25
AFP 4; Reactor Building North Control Rod Drive Module Area and Control Rod Drive Repair Room, Elevation 757'6"; Revision 28
AFP 9; Reactor Building Reactor Building Closed Loop Cooling Water Heat Exchanger Area, Equipment Hatch Area, and Jungle Room, Elevation 812'0"; Revision 27
AFP 14; Turbine Building Reactor Feed Pump Area, Turbine Lube Oil Tank Area, and 1A2 Switchgear Room, Elevation 734'0"; Revision 30
AFP 19; Turbine Building South Turbine Building Ground Floor, Elevation 757'6"; Revision 25
AFP 20; Turbine Building Aux Boiler Room, Emergency Diesel Generator Rooms, and Generator Day Tank Rooms, Elevation 757'6"; Revision 29
AFP 31; Intake Structure Pump Rooms, Elevation 767'0"; Revision 26
AFP 32; Intake Structure Traveling Screen Areas, Elevation 754'0"; Revision 27
AFP 34; Radwaste Building Drum Filling, Storage, and Shipping Area, Elevation 757'6"; Revision 25
AFP 35; Radwaste Building Radwaste Treatment And Access Area, Elevation 773'6"; Revision 24
AFP 36; Radwaste Building Precoat and Access Area, Control Room, and Heating, Ventilation, Air-Conditioning (HVAC) Equipment Room, Elevation 786'0". Revision 25

1R06 Flood Protection Measures

Abnormal Operating Procedure (AOP) 902; Flood; Revision 25

1R07 Heat Sink Performance

Drawings:
FSK-03500; Turbine Bldg-Area 3 Jacket Water Expansion Tank to Generator; Revision 6
FSK-4310; Pump House Piping; Revision 3

Calculations:
CAL-M05-027; Emergency Diesel Generator Heat Exchanger Heat Transfer Calculation; Revision 3
Data Report for RHR B - RHR Heat Exchanger Heat Transfer Test; dated January 9, 2007

CAPs Reviewed:
CAP029862; 50.59 2002-02, RHRSW Strainer Bypass; dated November 20, 2003
CAP036714; River Survey Shows Increasing Sedimentation; dated June 7, 2005
CAP041914; March River Depth Readings Show Sand Build-Up; dated May 1, 2006
CAP044539; Ineffective CATPR from RCE000222; dated November 20, 2006
CAP048570; As Found DP for 1VAC012 was Below the As Left Requirements; dated March 21, 2007
CE1396; Considering Effects of Bryozoa; Revision 1
CE004610; Safety Issue at the Intake Structure; dated December 4, 2006
OTH015647; Cedar River Sediment Monitoring; dated October 11, 2006

SA013789; GL 89-13 - Thermal Performance and Trending Program; dated September 20, 2006

CAPs NRC-Identified:
CAP049347; Zebra Mussel Monitoring; dated April 26, 2007
CAP049353; Questions on Values for RHR HX Duty Labeled as "Historical" in UFSAR; dated April 26, 2007
CAP049360; Evaluation of Biox Material at the Intake Structure; dated April 27, 2007
CAP050975; EDG Combustion Air Temperature Switch TS3277A/B Set Point Concern; dated July 9, 2007

Miscellaneous Documents:
Chron 0034559; Emergency Service Water System Component Design Basis Update-Emergency Diesel Generator; dated August 22, 1990
Evaluation 02-002; RHRSW Strainer Bypass; Revision 1
ECP-1735; Cedar River Spur Dikes (Wing Dams); Revision 0
Heat Exchanger Specification Sheet; Jacket Water Cooler; dated August 29, 1990
Heat Exchanger Specification Sheet; Residual Heat Removal Exchanger; dated November 6, 1969
Instruction Manual; Residual Heat Removal Exchangers; dated September 17, 1971
Letter; Response to GL 89-13; dated January 29, 1990
OPR347; ESW Cooling Water Supply Pipping to RHRSW Pump Motor Coolers Has Thinned Below 87.5 percent of Nominal Wall Thickness; Revision 0
Report: FPL6-DAEC-01; Emergency Diesel "B" HX-1E053B Scavenging Air-B1; dated April 11, 2007
Report: FPL6-DAEC-01; Emergency Diesel "B" HX-1E053B Jacket Cooler-B3; dated April 11, 2007
Report: NMC-DA1-16;1E201B-RHR B; dated April 6, 2003
Root Cause Report for AR No. 32025; "A" RHRSW Strainer High D/P While Running "A" and "C" RHRSW Pumps; dated October 1, 2002
Safety Evaluation 95-02; Biofouling Agent: Zebra Mussel; Revision 4
Task T0400; Containment System Response; Revision 2

Procedures:
ACP 1208.4; GL 89-13 Heat Exchanger Performance and Trending Revision 9
ARP 1C06A; "A" RHRSW/ESW Pit LO Level; Revision 50
DBD-E11-001; Low Pressure Coolant Injection Subsystem; Revision 9
DBD-E12-001; Residual Heat Removal Service Water System; Revision 6
DBD-R43-001; Standby Diesel Generator System; Revision 3
Heat Exchanger Thermal Performance and Trending Program; Revision 7
PCP 9.5; Chlorination/Halogenation of Circulating Water, General Service Water, and RHRSW/ESW; Revision 21
SD-324; Standby Diesel Generator System; Revision 8
STP NS100102; River Water Supply And Screen Wash System Vibration Measurement and Operability test; Revision 20
STP NS540002; Emergency Service Water Operability Test; Revision 27

Work Orders:
PWO 1138770; Inspect and Clean, As Needed, “A” RHRSW/ESW Pit; dated January 3, 2007
CWO A76453AS; Dredge Sand Downstream of Intake in Accordance With CA044213; Revision 0

1R11 Licensed Operator Requalification Program

Simulator Exercise Guide 2007B-01 PM; Revision 0
Emergency Operating Procedure 1; RPV Control; Revision 14
Emergency Operating Procedure 2; Primary Containment Control; Revision 13
Emergency Depressurization; Revision 4
Emergency Action Level Table 1; Revision 7
AOP 573; Primary Containment Control; Revision 1
ACP 110.1; Conduct of Operations; Revision 7
ACP 101.01; Procedure Use and Adherence; Revision 41
ACP 101.2; Verification Process and Self / Peer Checking Practices; Revision 5
Duane Arnold Energy Center Simulator Certification Test Procedure, “ED03 - Power Grid Voltage Transient,” Revision 0, Performed 1/21/1994.
Duane Arnold Energy Center Simulator Certification Test Procedure, “ED06 - Loss of Any Transformer,” Revision 0, Performed 8/17/1992.
Duane Arnold Energy Center Simulator Certification Test Procedure, “PC13 -Break in Discharge Pipe of SRV Into Torus Airspace,” Revision 0, Performed 12/13/1992.
Duane Arnold Energy Center Simulator Certification Test Procedure, “RH01 A, B, C and D -RHR Pump Trip,” Revision 0, Performed 1/21/1994.
Duane Arnold Energy Center Simulator Certification Test Procedure, “RH03 - RHR Pump Discharge Line Break,” Revision 0, Performed 9/03/1994.
Duane Arnold Energy Center Simulator Certification Test Procedure, “RR14 A, B, C and D - Recirc Jet Pump Riser Failure,” Revision 0, Performed 1/29/1994.
Duane Arnold Energy Center Simulator Certification Test Procedure, “RR14 A through H - Recirc Loop Flow Transmitter Failure,” Revision 0, Performed 9/17/1994.
Duane Arnold Energy Center Simulator Certification Test Procedure, “SW01 A, B, and C - General Service Water Pump Trip,” Revision 0, Performed 2/18/1994.
Duane Arnold Energy Center Simulator Certification Test Procedure, “SW21 A, B, C and D - Well Water Pump Trip,” Revision 0, Performed 2/17/1994.
Regulatory Guide 1.149, “Nuclear Power Plant Simulation Facilities For Use In Operator License Examinations,” Revision 1

1R12 Maintenance Effectiveness

Excellence Plan Action Item for ‘B’ Recirculation Pump Speed Changes Without Operator Input
CAP 042926; Upward Step Change in ‘B’ Recirculation Motor Generator on June 25, 2006
CAP 049311; ‘B’ Recirculation Pump Flow Lowered Changing Power about 4 Megawatts Thermal
CAP 046387; Unexplained Change in Indicated Core Flow
CAP 007467; Review GE SIL-628: Core Flow Measurement System Summer Calibrations
CAP 005020; “Spikes” on Indicated Core Flow (Computer Point B012 and FR4528)
CAP 002902; Investigate the “spiking” of Computer Point B012 (Total Reactor Core Flow)
CAP 049178; Abnormal Heating on Fuse in 1C038 (Jet Pump Instrument Vertical Board)
CAP 049087; Discovered ‘B’ Recirculation Motor Generator Set Scoop Tube Brake not Engaged with Scoop Tube Locked

CWO A78426; Perform Temporary Instruction Form (TIF) numbers 1 through 4 to Monitor Jet Pump Instrumentation. Replace FY4525
CAP 040596; Minor Changes in Recirculation Flow Noted
CAP 040446; Minor Changes in Recirculation Flow Noted
CAP 048913; Step Increase in Reactor Power/Core Flow with No Operator Action
CAP 042932; Two 'B' Recirculation Pump Speed Changes on June 27, 2006
System Health Checklist/Health Report for the Nuclear Boiler and Reactor Recirculation System; Quarter one of 2007
ACP 1201.2; Conduct of Systems/Plant Engineering; Revision 13
DAEC Performance Criteria Basis Document for Secondary Containment/ Standby Gas Treatment (SUS 34.00, 70.00, 99.27, 99.28); Revision 1
DAEC System Checklist/Health Report for Reactor Building HVAC and Standby Gas Treatment; May 14, 2007
System Health Action Plan for Reactor Building HVAC and Standby Gas Treatment; May 2007
CAP 049972; 'B' SBGT Failed TS Step of STP 3.6.4.3-04
CAP 049979; Information Clarification for STP 3.6.4.3-04
CAP 049981; Expected Alarm Not Received

<u>1R13 Maintenance Risk Assessments and Emergent Work Control</u>
Work Procedure Guideline WPG-2; On-Line Risk Management Guideline; Revision 32
Maintenance Risk Evaluation for Week 16; April 13, 2007
DAEC Online Schedule, Week 9715-9716; April 12, 2007
CAP 049132; RHR Heat Exchanger 'A' Inlet Valve, MO-2029, Failed While Cycling for STP
Maintenance Risk Evaluation for Week 18; Revision 0; April 27, 2007
Maintenance Risk Evaluation for Week 18; Revision 1; April 30, 2007
Maintenance Risk Evaluation for Week 18; Revision 2; May 1, 2007
Maintenance Risk Evaluation for Week 18; Revision 3; May 3, 2007
Maintenance Risk Evaluation for Week 18; Revision 4; May 4, 2007
DAEC Online Schedule, Week 9717-9718; April 27, 2007
Maintenance Risk Evaluation for Week 19; Revision 0; May 4, 2007
Maintenance Risk Evaluation for Week 19; Revision 1; May 8, 2007
Maintenance Risk Evaluation for Week 19; Revision 2; May 10, 2007
Main Activity Look Ahead, Week 9719; May 1, 2007
DAEC Online Schedule, Week 9718-9719; May 4, 2007
Maintenance Risk Evaluation for Week 22; Revision 0; May 25, 2007
Maintenance Risk Evaluation for Week 22; Revision 1; May 29, 2007
Maintenance Risk Evaluation for Week 22; Revision 2; May 31, 2007
Main Activity Only Look Ahead, Week 9722; May 24, 2007
DAEC Online Schedule, Week 9721-9722; May 24, 2007

<u>1R15 Operability Evaluations</u>
CAP 049276; 5/16 Inch Linear Indication on MO2202 Disk Face Identified During Final PT
CAP 001841; M2202 (HPCI Turbine Steam Supply Isolation) Cracked Hardfacing on Upstream Face of Disk
Other 001986; M2202 (HPCI Turbine Steam Supply Isolation) Cracked Hardfacing on Upstream Face of Disk
CWO A77477; Repair Valve MO2202
Trouble Shooting Control Form for Work Order A77245

CAP 049547; STP Alarm Not received as Expected
CAP 049560; ‘B’ EDG Automatically Shut Down During PMT for Speed Switch
CAP 042927; SS3237A & B Lubrication May Not Be Sufficient
Equipment-Specific Maintenance Procedure I.SS-519-01; Synchro-Start Products Series G-2 Speed Switches; Revision 5
Condition Evaluation 005345; Past Operability Determination for 1G021, ‘B’ EDG
CLTP 1.1; Containment Leakage Testing Program Plan; Revision 1
ACP 1410.7; Guidelines for Primary Containment Valves and Penetrations; Revision 14
CAP 049405; Increased Containment Nitrogen Usage
CAP 049419; Unplanned TS LCO 3.6.1.3, “Non-MSIV, Non-Purge Valve” PCIV Condition A and B
NS590011; ASME In-Service Check Valve Air Testing; Revision 1
CAP 049936; Incorrect or No Weld Filler Material Identified in Review Of Work Order A71766
CAP 049953; Potential Design Change During Maintenance Activities
Operability Recommendation 000356; Potential Design Change During Maintenance Activities
CWO A71766; Replace or Repair 1VAC012 and Return to Service
PWO 11358113; Replace Cylinder Liners Per Fairbanks Morse Owners Group recommendation
CAP 045901; 1G021 Lube Oil Filter Leak Status
CAP 049012; 1G021/LOF Lube Oil Filter Has a 0.5 Gallon Per Minute Leak
Apparent Cause Evaluation 001719; 1G021/LOF Lube Oil Filter Has a 0.5 Gallon Per Minute Leak
Final Past Operability Evaluation for ‘B’ Diesel Generator Lube Oil Leakage; May 1, 2007

1R19 Post-Maintenance Testing

CAP 049547; STP Alarm Not received as Expected
Trouble Shooting Control Form for Work Order A77245
Condition Evaluation 005345; Past Operability Determination for 1G021, ‘B’ EDG
CAP 049548; Missed Surveillance Section for STP 3.8.1-01
CAP 049560; ‘B’ EDG Automatically Shut Down During PMT for Speed Switch
STP 3.8.1-04; Standby Diesel Generators Operability Test (Slow Start from Normal Starting Air); Revision 28
STP 3.5.1-05, HPCI System Operability Test, Revision 36
PWO 1133610; SV2259 - Replace the solenoid Valve
PWO 1136111; Replace Scoop Tube Deviation Relay
CWO A77477; Repair Valve MO2202
CWO A77244; Calibrate SS3237B - ‘B’ EDG Speed Switch
PWO 1139506; Re-calibration of the ‘B’ EDG Speed Sensing Switch
CAP 049334; PWO 1136110 and 1136111 - PMT Needs to be Completed prior to 4/29/07
CWO A73696; Remove and Replace V13-0059 - ‘A’ Core Spray Pump Motor Cooler ESW Inlet Valve

1R20 Outage Activities

IPOI 5; Reactor Scram; Revision 46
Scram Report for Scram 07-02; April 2, 2007
Forced Outage Schedule; April 2007
CAP 048780; Manual Reactor Scram due to Lowering RPV Level - Loss of 1A2
CAP 048793; ‘B’ Feed Pump Lost Oil Lubrication Due to Bus 1A2 Lockout
CAP 048795; “b” Source Range Monitor Spiking

On Shift Analysis for Reactor Scram Number 07-02
IPOI 2; Startup; Revision 96
IPOI 3; Power Operations (35 percent to 100 percent Rated Power); Revision 90

1R22 Surveillance Testing
STP 3.5.1-13; HPCI System Water Fill Test; Revision 2
STP 3.5.1-10; HPCI System Operability Test and Comprehensive Pump Test; Revision 8
STP NS930001; Main Turbine Operational Tests; Revision 20
STP 3.1.7-01; Standby Liquid Control Operability Test; Revision 20
STP 3.3.3.2-04; Remote Shutdown Panel Functional Test for RHR; Revision 5
CAP 050528; STP 3.3.3.2-04 was not Identified as Risk Significant
STP 3.3.1.1-34; Recirculation Flow Unit Calibration; Revision 13
CAP 050410; STP 3.3.1.1-34 & NS640101 Performance Problems Identified with Recommend Resolution
Condition Evaluation 005470; STP 3.3.1.1-34 & NS640101 Performance Problems Identified with Recommend Resolution

1R23 Temporary Plan Modifications
FP-E-MOD-03; Fleet Modification Procedure for Temporary Modifications; Revision 1
NS440101; Feedwater Correction Factor Manual Calculation; Revision 2
CAL-M07-012; Correction Feedwater Flow Indication; Revision 0
CAP 050122; P-1 Check at 75 percent Power Exceeded Mismatch Limits in 2 of 5 Calculations
CAP 048900; Feedwater Correction Factor Cannot Be Put into Service
CAP 048767; Feedwater Correction Factor not able to be Updated
Temporary Modification 07-008; Change VR9019 Hi Alarm Setpoint for #7 Bearing
PWO 1140235; Adjust VR9019 Channel 7, Turbine Bearing 7 Alarm Setpoint from 7 mils to 8.5 mils Increasing
Temporary Modification 07-009; Brass Seal Washer Installed for Stud #7 of the 'A' Reactor Feed Pump
CWO A79504; Install Brass Washer at #7 Stud Location

1EP6 Drill Evaluation
Note 5; DAEC Emergency Action Level Notification Form for Simulator Evaluation on June 11, 2007
Note 5; DAEC Emergency Action Level Notification Form for Simulator Evaluation on June 18, 2007
Emergency Plan Implementing Procedure (EPIP) 1.1; Determination of Emergency Action Levels; Revision 27
EPIP 1.2; Notifications; Revision 35

2PS2 Radioactive Material Processing and Transportation
Radiological Engineering Calculation; 10 CFR Part 61 Compliance Data Technical Basis for DAEC DAW; dated August 11, 2006
Radiological Engineering Calculation; 10 CFR Part 61 Compliance Data Technical Basis for DAEC reactor water clean-up resin; dated April 17, 2006
Update Final Safety Analysis Report, Section 11; Revision 13
SA 015876; Snapshot self-assessment in Radwaste and Transportation; dated April 27, 2007

2005-001-1-006; Biennial required assessment of Radioactive waste Processing elements; dated April 10, 2005
CA 042557; Radioactive shipment package had greater than expected contact dose rate readings; dated April 17, 2006
CA 039808; Issues with radioactive waste shipment RSR-04-38; dated February 18, 2005
CAP 030969; UFSAR contains references to decommissioned/abandoned in place RW equipment. Two examples are RW Evaporator and Conveyor system; dated April 10, 2004
Assessment Decision Worksheet for Second Quarter of 2007; Radioactive Waste Control; dated April 25, 2007
Radioactive Material Manifest 07-14; Low Specific Activity (LSA-II) shipping of Control Rod Drives (CRDs) to GE NE; Wilmington, NC; dated February 27, 2007
Uniform Low Level Radioactive Waste Manifest 07-46; Type B(M) shipping containing dewatered condensate resin to EnergySolutions, Utah; dated May 22, 2007
Uniform Low Level Radioactive Waste Manifest 07-25; Low Specific Activity (LSA-I) shipping containing metal; wood; plastic; rubber; dirt; insulation; asbestos and miscellaneous DAW to Duratek; Oakridge, TN; dated February 2, 2007
Uniform Low Level Radioactive Waste Manifest 07-29; Low Specific Activity (LSA-I and LSA-II) shipping containing metal; wood; plastic; rubber; dirt; insulation; asbestos and miscellaneous DAW to Duratek; Oakridge, TN; dated March 6, 2007
Uniform Low Level Radioactive Waste Manifest 07-37; Low Specific Activity (LSA-I and LSA-II) shipping containing metal; wood; plastic; rubber; dirt; insulation; asbestos and miscellaneous DAW to Durate; Oakridge, TN; dated March 21, 2007
Uniform Low Level Radioactive Waste Manifest 07-38; Low Specific Activity (LSA-II) shipping containing metal; wood; plastic, rubber; dirt; insulation; asbestos and miscellaneous DAW to Duratek; Oakridge, TN; dated April 10, 2007
RWH 3402.17; 1T-205A/B Cleanup Phase Seperator Resin Sampling; Revision 27
RWH 3409.2; Sampling Instructions and Analysis of Radwaste Stream; Revision 11
RWH 3406.6; Characterizing Radioactive Material for Transport; Revision 8
RWH 3406.8; Packaging Radioactive Material for Shipment; Revision 7
SA 015876; QF-0406 Revision 2(ACP 117.4); Snapshot Report of Radwaste and Transportation; dated April 25, 2007
2005-001-004-1-009; Nuclear Oversight Observation Report; Radiation Protection Program Support Element; dated March 18, 2006

4OA1 Performance Indicator Verification

CAP 048489; Unanticipated Conductivity Transient
PCP 1.9; Water Chemistry Guidelines; Revision 38
ACP 1402.4; NRC & WANO Performance Indicators Reporting; Revision 7
AOP 639; Reactor Water/Condensate High Conductivity; Revision 27
DAEC PI Report for Unplanned Scrams per 7000 Critical Hours for January 2006 through March 2007
DAEC PI Report for Unplanned Scrams with Loss of Normal Heat Removal for January 2006 through March 2007
DAEC PI Report for Unplanned Power Changes per 7000 Critical Hours for January 2006 through March 2007
Nuclear Energy Institute 99-02, Regulatory Assessment Performance Indicator Guideline, Revision 4
DAEC PI Report for Reactor Coolant System Activity for January 2006 through March 2007

DAEC PI Report for Reactor Coolant System Leakage for January 2006 through March 2007
DAEC First Quarter 2006 PI Summary, April 3, 2006
DAEC Second Quarter 2006 PI Summary, July 13, 2006
DAEC Third Quarter 2006 PI Summary, October 17, 2006
DAEC Fourth Quarter 2006 PI Summary, January 10, 2007
DAEC First Quarter 2007 PI Summary, April 17, 2007

4OA2 Identification and Resolution of Problems
ACP 114.4; Corrective Action Program; Revision 22
ACP 114.5; Action Request System; Revision 57
ACP 114.8; Action Request Trending; Revision 6
ACP 114.9; Event Response Procedure; Revision 12
ACP 114.12; Operational Decision-Making and Issue Management; Revision 4
ACP 109.3; Troubleshooting Process; Revision 1
CAP 048799; Trend-Problems With Workmanship During the HPCI Mod in RFO20
CAP 049407; Organizational Response Not Meeting Expectations
CAP 046423; Spurious Annunciators During Testing with CAPs Closed to Trend
CAP 046591; Increasing Trend in Number of Errors Across the Site
CAP 047383; Trend of Lost TLDs is Unacceptable
CAP 048276; Trend in Ops HU Issues During RFO 20 - KPI Currently Below Goal
CAP 048860; There is Currently a Backlog of CAPS Waiting to Be Trended
CAP 048865; Reactivity Mgmt KPI Below Monthly Goal and Adverse Trend over nine Months
CAP 050274; Review of CE5316 Has Identified an Additional Condition Adverse to Quality
CAP 050437; EP Snapshot Self-Assessment on RCE Corrective Actions
CAP 050732; Operator Performance in Training is Not Being Tracked with a Controlled Process
CAP 048313; HPCI Overspeed Trip Setting Was Adjusted Three Times
CAP 048591; Spurious Downscale Alarms Were Received on the B and E APRMs
CAP 048762; Div. II Gr. III Isolation Occurred for Unknown Reasons and Can’t be Reset
CAP 050229; Risk Management for Troubleshooting Activities
CAP 050849; Troubleshooting Information Form Paperwork is Routinely Lost
Operability Recommendation 000349; Spurious Downscale Alarms Were Received on the B and E APRMs

4OA3 Event Follow-up
LER 2007002-00; Loss of Control Building Boundary; April 13, 2007
LER 2007004-00; Severe Weather Causes Grid Disturbance Resulting in Loss of Shutdown Cooling; April 26, 2007
LER 2007005-00; Automatic Reactor Scram Due to Scram Discharge Volume High Water Level During Performance of a Surveillance Test; April 26, 2007
LER 2007006-00; Reactor Shutdown as a Result of a Chemistry Excursion; May 17, 2007
LER 2007007-00; Reactor Scram Due to 1A2 Non-essential Bus Lockout; June 1, 2007
LER 2007008-00; Condition Prohibited by TSs; ‘B’ Emergency Diesel Inoperable; June 11, 2007
CAP 050265; Licensed Operator Notification Not Completed as Described in Letter to the NRC
CAP 047825; Severe Winter Ice Storm Causes Grid Disturbance and Plant Transient
CAP 048702; Unplanned HPCI LCO - PSV2302 Stuck Open
CAP 048708; EBB-005 HPCI Discharge Piping Exceeded Design Pressure During STP

CAP 048780; Manual Reactor Scram due to Lowering RPV Level - Loss of 1A2
CAP 048784; Second Scram Signal Received on Low RPV Level During Recovery

4OA5 Other Activities

CA 041423; Revise the Fire Plan to Require Maintenance of Appendix A Barriers; dated November 11, 2005
CAP 046173; Inspection Report 2006-014 contains Unresolved Item on Fire Protection; dated January 3, 2007
Safety Evaluation (SE) 95-03; Safety Evaluation to Support DDC-3151, Revision of FHA-800 to Supersede Appendix ‘A’ Requirements with Appendix ‘R’ Requirements; dated December 18, 1996
STP NS13F001; Fire Barrier Penetration Seal Inspection, Revision 10
STP NS13F002; Fire Door and Frame Inspection; Revision 17
CAP 049165; Untimely Submittal of Operator Physical Information

4OA7 Licensee-Identified Violations

CAP 047315; Control Building Envelope Inoperable

LIST OF ACRONYMS USED

ACP	Administrative Control Procedure
ADAMS	Agency Wide Documents Access and Management System
AFP	Area Fire Plan
ANSI/ANS	American National Standards Institute/American Nuclear Society
AOP	Abnormal Operating Procedure
ASME	American Society of Mechanical Engineers
CAP	Corrective Action Process
CFR	Code of Federal Regulations
CS	Core Spray
CWO	Corrective Work Order
DAEC	Duane Arnold Energy Center
DAW	Dry Active Waste
DRP	Division of Reactor Projects
DRS	Division of Reactor Safety
EDG	Emergency Diesel Generator
HPCI	High Pressure Coolant Injection
HVAC	Heating, Ventilation, Air-Conditioning
IMC	Inspection Manual Chapter
IPOI	Integrated Plant Operating Instructions
LCO	Limiting Condition for Operation
LER	Licensee Event Report
LOF	Lube Oil Filter
NCV	Non-Cited Violation
NFPA	National Fire Protection Association
NRC	Nuclear Regulatory Commission
OI	Operating Instruction
OOS	Out-of-Service
PMT	Post-Maintenance Testing
PWO	Preventative Work Order
RCIC	Reactor Core Isolation Cooling
RO	Reactor Operator
RHR	Residual Heat Removal
RHRSW	Residual Heat Removal Service Water
RPS	Reactor Protection System
RPV	Reactor Pressure Vessel
RWS	River Water Supply
SDP	Significance Determination Process
SSC	Structures, Systems, Components
STP	Surveillance Test Procedure
TS	Technical Specification
UFSAR	Updated Final Safety Analysis Report
URI	Unresolved Item