UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION III
2443 WARRENVILLE ROAD, SUITE 210
LISLE, IL 60532-4352

August 9, 2010

Mr. Christopher R. Costanzo
Vice President
NextEra Energy Duane Arnold, LLC
3277 DAEC Road
Palo, IA 52324-9785

SUBJECT: DUANE ARNOLD ENERGY CENTER INTEGRATED INSPECTION REPORT 05000331/2010003

Dear Mr. Costanzo:

On June 30, 2010, the U.S. Nuclear Regulatory Commission (NRC) completed an integrated inspection at your Duane Arnold Energy Center. The enclosed report documents the inspection results, which were discussed on July 15, 2010, with Mr. K. Kleinheinz 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, the inspectors identified two findings of very low safety significance which involved violations of NRC requirements and one Severity Level IV violation. Because of their very low safety significance and since the issues were entered into your corrective action program, the NRC is treating the issues as non-cited violations (NCVs) in accordance with Section VI.A.1 of the NRC Enforcement Policy. Additionally, two licensee-identified violations are listed in Section 4OA7 of this report.

If you contest the subject or severity of these NCVs, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the 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 addition, if you disagree with the cross-cutting aspect assigned to any finding in this report, you should provide a response within 30 days of the date of this inspection report, with the basis for your disagreement, to the Regional Administrator, Region III, and the NRC Resident Inspector 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, its enclosure, and your response (if any) will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records System (PARS) component of NRC's document system (ADAMS). ADAMS is accessible from the NRC Website 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 05000331/2010003
w/Attachment: Supplemental Information

cc w/encl: Distribution via ListServe

U.S. NUCLEAR REGULATORY COMMISSION

REGION III

Docket No:	50-331
License No:	DPR-49
Report No:	05000331/2010003
Licensee:	NextEra Energy Duane Arnold, LLC
Facility:	Duane Arnold Energy Center
Location:	Palo, IA
Dates:	April 1 through June 30, 2010
Inspectors:	R. Orlikowski, Senior Resident Inspector R. Murray, Resident Inspector R. Baker, Resident Inspector A. Scarbeary, Reactor Engineer P. Smagacz, Reactor Engineer M. Mitchell, Health Physicist
Observers:	None
Approved by:	Kenneth Riemer, Chief Branch 2 Division of Reactor Projects

Enclosure

TABLE OF CONTENTS

SUMMARY OF FINDINGS

IR 05000331/2010003; 04/01/2010 – 06/30/2010; Duane Arnold Energy Center; Operability Evaluations, Plant Modifications, and Follow-Up of Events and Notices of Enforcement Discretion.

This report covers a three-month period of inspection by resident inspectors and announced baseline inspections by regional inspectors. Two findings and one Severity Level IV violation were identified by the inspectors. These issues were considered non-cited violations (NCVs) of NRC regulations. The significance of most findings is indicated by their color (Green, White, Yellow, Red) using Inspection Manual Chapter (IMC) 0609, "Significance Determination Process" (SDP). Cross-cutting aspects were determined using IMC 0310, "Components Within The Cross-Cutting Areas." 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 4, dated December 2006.

A. NRC-Identified and Self-Revealed Findings

Cornerstone: Barrier Integrity

- Green. A finding of very low safety significance and associated NCV of 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," was identified by the inspectors for the failure of the licensee to follow procedure EN-AA-203-1001, "Operability Determinations/Functionality Assessments," and Administrative Control Procedure (ACP) 110.1, "Conduct of Operations." Specifically, the Shift Manager failed to make an Immediate Operability Determination (IOD) that addressed the impact of the degraded conditions in the drywell cooling system on Primary Containment, and to provide sufficient detail for an independent person to understand the basis for the decision. These actions were contrary to step 4.3 of EN-AA-203-1001 and Attachment 10 of ACP 110.1, and represented a performance deficiency warranting further investigation. The licensee entered the issues into their Corrective Action Program (CAP) as CAP 074069 and Condition Report (CR) 568618, and completed an Operability Recommendation (OPR), OPR 000427, that determined the Primary Containment was operable, but degraded, as a result of the drywell cooling system condition.

  The performance deficiency was determined to be more than minor because, if left uncorrected, failure to properly implement the operability procedures could result in safety-related components being incorrectly declared operable rather than inoperable or operable but degraded or non-conforming (a more significant safety concern).
  The inspectors evaluated the finding using the SDP in accordance with IMC 0609, Table 4a for the Barrier Integrity Cornerstone. Because inspectors answered "No" to all four questions under the Containment Barrier column, the finding was determined to be of very low safety significance (Green). The finding had a cross-cutting aspect in the area of Human Performance, Decision Making, because the licensee did not use conservative assumptions in decision making and adopt a requirement to demonstrate that the proposed action is safe in order to proceed rather than a requirement to demonstrate that it is unsafe in order to disapprove the action. Specifically, not evaluating the degraded condition of the drywell cooling system to determine its impact

on Primary Containment operability was a non-conservative assumption in the IOD. [H.1(b)] (Section 1R15)

Cornerstone: Initiating Events

- Green. A finding of very low safety significance and associated NCV of 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," was identified by the inspectors for the licensee's failure to prescribe a procedure appropriate to the circumstances when Surveillance Test Procedure (STP) 3.0.0-01, Attachment 3, "Reactor Coolant Leakage," was implemented on March 15, 2010 to meet the Technical Specification (TS) definition of Identified Leakage. Specifically, STP 3.0.0-01 did not include a requirement to verify that leakage inside the drywell did not interfere with the leakage detection system prior to reclassifying Unidentified Leakage as Identified Leakage. The licensee entered the issue into their corrective action program as CR 568613.

  The inspectors determined that the issue was a performance deficiency because it was the result of the failure to meet a requirement, and the cause was reasonably within the licensee's ability to foresee and correct, and should have been prevented. The performance deficiency was determined to be more than minor and a finding, because, if left uncorrected, the performance deficiency had the potential to lead to a more significance safety concern. The inspectors applied IMC 0609, Attachment 4, "Phase 1 - Initial Screening and Characterization of Findings" to this finding. Under Table 2, all Reactor Coolant System (RCS) Boundary issues that are not a result of a plant upset will be considered using the Initiating Events Cornerstone. Under Table 4a for the Initiating Events Cornerstone, the finding screened as very low safety significance (Green) because there was no actual RCS leakage that would have exceeded the TS limit, and the finding did not affect other mitigation systems resulting in a total loss of safety function. The inspectors determined that the contributing cause that provided the most insight into the performance deficiency affected the cross-cutting area of Problem Identification and Resolution, relating to the corrective action program components, and involving the aspect associated with the licensee assessing information from the corrective action program in aggregate to identify common cause problems. [P.1(b)] (Section 1R18)

Cornerstone: Miscellaneous

- SL IV. A Severity Level IV NCV of 10 CFR Part 50.73(a)(2)(v)(A) and (D) was identified by the inspectors for the failure of the licensee to report an event or condition that could have prevented the fulfillment of the Turbine Stop Valve Closure and Turbine Control Valve Fast Closure reactor protection system (RPS), and the End-of-Cycle Recirculation Pump Trip (EOC-RPT) safety functions, which are relied upon to shutdown the reactor and maintain it in a shutdown condition, and mitigate the consequences of an accident. The licensee entered the violation into their corrective action program as Action Request (AR) 392462 and CR 568620.

  Violations of 10 CFR 50.73 are considered to be violations that potentially impact the regulatory process and are dispositioned using the traditional enforcement process instead of the Reactor Oversight Process SDP. Because the performance deficiency was not more than minor and not a finding per IMC 0612, Appendix B, "Issue Screening," a cross-cutting aspect was not assigned to this violation. (Section 4OA3.3)

B. Licensee-Identified Violations

Violations of very low safety significance that were identified by the licensee have been reviewed by the inspectors. Corrective actions planned or taken by the licensee have been entered into the licensee's corrective action program. These violations and corrective action tracking numbers are listed in Section 4OA7 of this report.

REPORT DETAILS

Summary of Plant Status

The Duane Arnold Energy Center (DAEC) 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 exception:

- On April 25, 2010, a planned maintenance shutdown was commenced to specifically locate and repair leakage from the drywell cooling system inside the Primary Containment. The outage continued through May 3, 2010, when the main generator was again connected to the grid. The reactor was returned to full power and power ascension was completed on May 5, 2010.

1. REACTOR SAFETY

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

1R01 Adverse Weather Protection (71111.01)

.1 Readiness of Offsite and Alternate AC Power Systems

a. Inspection Scope

The inspectors verified that plant features and procedures for operation and continued availability of offsite and alternate alternating current (AC) power systems during adverse weather were appropriate. The inspectors reviewed the licensee's procedures affecting these areas and the communications protocols between the transmission system operator (TSO) and the plant to verify that the appropriate information was being exchanged when issues arose that could impact the offsite power system. Examples of aspects considered in the inspectors' review included:

- The coordination between the TSO and the plant during off-normal or emergency events;
- The explanations for the events;
- The estimates of when the offsite power system would be returned to a normal state; and
- The notifications from the TSO to the plant when the offsite power system was returned to normal.

The inspectors also verified that plant procedures addressed measures to monitor and maintain availability and reliability of both the offsite AC power system and the onsite alternate AC power system prior to or during adverse weather conditions. Specifically, the inspectors verified that the procedures addressed the following:

- The actions to be taken when notified by the TSO that the post-trip voltage of the offsite power system at the plant would not be acceptable to assure the continued operation of the safety-related loads without transferring to the onsite power supply;

- The compensatory actions identified to be performed if it would not be possible to predict the post-trip voltage at the plant for the current grid conditions;
- A re-assessment of plant risk based on maintenance activities which could affect grid reliability, or the ability of the transmission system to provide offsite power; and
- The communications between the plant and the TSO when changes at the plant could impact the transmission system, or when the capability of the transmission system to provide adequate offsite power was challenged.

Documents reviewed are listed in the Attachment to this report. The inspectors also reviewed CAP items to verify that the licensee was identifying adverse weather issues at an appropriate threshold and entering them into their CAP in accordance with station corrective action procedures.

This inspection activity constituted one readiness of offsite and alternate AC power systems sample as defined in Inspection Procedure (IP) 71111.01-05.

b. Findings

No findings of significance were identified.

.2 Summer Seasonal Readiness Preparations

a. Inspection Scope

The inspectors performed a review of the licensee's preparations for summer weather for selected systems, including conditions that could lead to an extended drought.

During the inspection, the inspectors focused on plant-specific design features and the licensee's procedures used to mitigate or respond to adverse weather conditions. Additionally, the inspectors reviewed the Updated Final Safety Analysis Report (UFSAR) and performance requirements for systems selected for inspection, and verified that operator actions were appropriate as specified by plant-specific procedures. Specific documents reviewed during this inspection are listed in the Attachment to this report. The inspectors also reviewed CAP items to verify that the licensee was identifying adverse weather issues at an appropriate threshold and entering them into their corrective action program in accordance with station corrective action procedures. The inspectors' reviews focused specifically on the following plant systems:

- Main Plant Air Intake System;
- General Service Water System; and
- Control Building Chiller System.

This inspection activity constituted one seasonal adverse weather sample as defined in IP 71111.01-05.

b. Findings

No findings of significance were identified.

1R04 Equipment Alignment (71111.04)

.1 Quarterly Partial System Walkdowns

a. Inspection Scope

The inspectors performed partial system walkdowns of the following risk-significant systems:

- 1K3/1K4 Safety Related Instrument Air Compressors;
- 'A' Standby Diesel Generator (SBDG) with 'B' SBDG Out-of-Service (OOS) for Surveillance Testing;
- 'A' Emergency Service Water (ESW) System prior to removing Standby Liquid Control System from Service for Surveillance Testing; and
- 'B' SBDG with 'A' SBDG OOS for Surveillance Testing.

The inspectors selected these systems based on their risk significance relative to the Reactor Safety Cornerstones at the time they were inspected. The inspectors attempted to identify any discrepancies that could impact the function of the system, and, therefore, potentially increase risk. The inspectors reviewed applicable operating procedures, system diagrams, UFSAR, TS requirements, outstanding work orders (WOs), condition reports, and the impact of ongoing work activities on redundant trains of equipment in order to identify conditions that could have rendered the systems incapable of performing their intended functions. The inspectors also walked down accessible portions of the systems to verify system components and support equipment were aligned correctly and operable. The inspectors examined the material condition of the components and observed operating parameters of equipment to verify that there were no obvious deficiencies. The inspectors also verified that the licensee had properly identified and resolved equipment alignment problems that could cause initiating events or impact the capability of mitigating systems or barriers and entered them into the CAP with the appropriate significance characterization. Documents reviewed are listed in the Attachment to this report.

These inspection activities constituted four partial system walkdown samples as defined in IP 71111.04-05.

b. Findings

No findings of significance were identified.

.2 Semi-Annual Complete System Walkdown

a. Inspection Scope

On April 6, 2010 the inspectors performed a complete system alignment inspection of the 4160/480/120 Volt Alternating Current (VAC) Systems to verify the functional capability of the system. This system was selected because it was considered both safety significant and risk significant in the licensee's probabilistic risk assessment. The inspectors walked down the system to review mechanical and electrical equipment line ups, electrical power availability, system pressure and temperature indications, as appropriate, component labeling, component lubrication, component and equipment cooling, hangers and supports, operability of support systems, and to ensure that ancillary equipment or debris did not interfere with equipment operation. A review of a sample of past and outstanding WOs was performed to determine whether any deficiencies significantly affected the system function. In addition, the inspectors reviewed the CAP database to ensure that system equipment alignment problems were being identified and appropriately resolved. Documents reviewed are listed in the Attachment to this report.

This inspection activity constituted one complete system walkdown sample as defined in IP 71111.04-05.

b. Findings

No findings of significance were identified.

1R05 Fire Protection (71111.05AQ)

.1 Routine Resident Inspector Tours (71111.05Q)

a. Inspection Scope

The inspectors conducted fire protection walkdowns which were focused on availability, accessibility, and the condition of firefighting equipment in the following risk-significant plant areas:

- Area Fire Plan (AFP) 07, Reactor Building Laydown Area, Corridor and Waste Tank Area and Spent Resin Tank Room, Elevation 786'-0";
- AFP 09, Reactor Building Closed Cooling Water Heat Exchanger Area, Equipment Hatch Area and Jungle Room, Elevation 812'-0";
- AFP 21 and 22, North and South Turbine Operating Floors, Middle Operating Floors, Demineralizer Pits, Elevation 780'-0";
- AFP 24, Essential Switchgear Rooms 1A-3, 1A-4, Elevation 757'-6"; and
- AFP 26 and 27, Control Building Control Room Complex and Heating Ventilation and Air Conditioning (HVAC) Room, Elevation 786'-0" and 800'-4".

The inspectors reviewed areas 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 implemented adequate compensatory measures for OOS, degraded or inoperable fire protection equipment, systems, or features in accordance with the licensee's fire plan. The inspectors selected fire areas based on their overall contribution to internal fire risk as documented in the plant's Individual Plant Examination of External Events with later additional insights, their potential to impact equipment which could initiate or mitigate a plant transient, or their impact on the plant's ability to respond to a security event. Using the documents listed in the Attachment to this report, the inspectors verified that fire hoses and extinguishers were in their designated locations and available for immediate use; that fire detectors and sprinklers were unobstructed; that transient material loading was within the analyzed limits; and fire doors, dampers, and penetration seals appeared to be in satisfactory condition. The inspectors also verified that minor issues identified during the inspection were entered into the licensee's CAP. Documents reviewed are listed in the Attachment to this report.

These inspection activities constituted five quarterly fire protection inspection samples as defined in IP 71111.05-05.

b. Findings

No findings of significance were identified.

.2 Annual Fire Protection Drill Observation (71111.05A)

a. Inspection Scope

On April 19, 2010, the inspectors observed the fire brigade activation for an unannounced drill response to an oil fire in the 'A' SBDG Room. On June 2, 2010, the inspectors observed the fire brigade activation for an unannounced drill response to a fire in the isophase bus duct coolers in the turbine building. On June 14, 2010, the inspectors observed the fire brigade activation for an unannounced drill response to a fire in the mezzanine on the north side of the second floor of the reactor building. Based on these observations, the inspectors evaluated the readiness of the plant fire brigade to fight fires. The inspectors verified that the licensee staff identified deficiencies, openly discussed them in a self-critical manner at the drill debrief, and took appropriate corrective actions. Specific attributes evaluated were:

- proper wearing of turnout gear and self-contained breathing apparatus;
- proper use and layout of fire hoses;
- employment of appropriate fire fighting techniques;
- sufficient firefighting equipment brought to the scene;
- effectiveness of fire brigade leader communications, command, and control;
- search for victims and propagation of the fire into other plant areas;
- smoke removal operations;
- utilization of pre planned strategies;
- adherence to the pre planned drill scenario; and
- drill objectives.

Documents reviewed are listed in the Attachment to this report.

These inspection activities constituted one annual fire protection inspection sample as defined in IP 71111.05-05.

b. Findings

No findings of significance were identified.

1R06 Flood Protection Measures (71111.06)

.1 Underground Vaults

a. Inspection Scope

The inspectors selected underground bunkers/manholes subject to flooding that contained cables whose failure could disable risk-significant equipment. The inspectors determined that the cables were not submerged, that splices were intact, and that appropriate cable support structures were in place. In those areas where dewatering devices were used, such as a sump pump, the device was operable and level alarm circuits were set appropriately to ensure that the cables would not be submerged. In those areas without dewatering devices, the inspectors verified that drainage of the area was available, or that the cables were qualified for submergence conditions. The inspectors also reviewed the licensee's corrective action documents with respect to past submerged cable issues identified in the corrective action program to verify the adequacy of the corrective actions. Documents reviewed are listed in the Attachment to this report. The inspectors performed a walkdown of the following underground bunkers/manholes subject to flooding:

- Manholes 1MH13, 1MH109, and 2MH207.

This inspection activity constituted one underground vaults sample as defined in IP 71111.06-05.

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification Program (71111.11)

.1 Resident Inspector Quarterly Review (71111.11Q)

a. Inspection Scope

On April 22 and 29, May 21, and June 15, 2010, the inspectors observed crews of licensed operators in the plant's simulator during licensed operator requalification examinations to verify that operator performance was adequate, evaluators were identifying and documenting crew performance problems, and training was being conducted in accordance with licensee procedures. The inspectors evaluated the following areas:

- licensed operator performance;
- crew's clarity and formality of communications;
- ability to take timely actions in the conservative direction;
- prioritization, interpretation, and verification of annunciator alarms;
- correct use and implementation of abnormal and emergency procedures;
- control board manipulations;
- oversight and direction from supervisors; and
- ability to identify and implement appropriate TS actions and Emergency Plan actions and notifications.

The crews' performances in these areas were compared to pre-established operator action expectations and successful critical task completion requirements. Documents reviewed are listed in the Attachment to this report.

These inspection activities constituted one quarterly licensed operator requalification program sample as defined in IP 71111.11.

b. Findings

No findings of significance were identified.

1R12 Maintenance Effectiveness (71111.12)

.1 Routine Quarterly Evaluations (71111.12Q)

a. Inspection Scope

The inspectors evaluated degraded performance issues involving the following risk-significant systems:

- 'A' Control Building Chiller (CBC) Trip on April 30, 2010;
- 'B' Reactor Recirculation Pump Motor Generator Set Supply Breaker Inadvertent Closure during Planned Maintenance; and
- Intermittent Bypass Valves Failing Full Open while Operating at 100 percent Reactor Power.

The inspectors reviewed events such as where ineffective equipment maintenance had resulted in valid or invalid automatic actuations of engineered safeguards systems and independently verified the licensee's actions to address system performance or condition problems in terms of the following:

- implementing appropriate work practices;
- identifying and addressing common cause failures;
- scoping of systems in accordance with 10 CFR 50.65(b) of the maintenance rule;
- characterizing system reliability issues for performance;
- charging unavailability for performance;
- trending key parameters for condition monitoring;
- ensuring 10 CFR 50.65(a)(1) or (a)(2) classification or re-classification; and
- verifying appropriate performance criteria for structures, systems, and components (SSCs)/functions classified as (a)(2) or appropriate and adequate goals and corrective actions for systems classified as (a)(1).

The inspectors assessed performance issues with respect to the reliability, availability, and condition monitoring of the system. In addition, the inspectors verified maintenance effectiveness issues were entered into the CAP with the appropriate significance characterization. Documents reviewed are listed in the Attachment to this report.

These inspection activities constituted three quarterly maintenance effectiveness samples as defined in IP 71111.12-05.

b. Findings

No findings of significance were identified.

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

.1 Maintenance Risk Assessments and Emergent Work Control

a. Inspection Scope

The inspectors reviewed the licensee's evaluation and management of plant risk for the maintenance and emergent work activities affecting risk-significant and safety-related equipment listed below to verify that the appropriate risk assessments were performed prior to removing equipment for work:

- Work Week 9018 Risk Management and Assessed Shutdown Risk during Scheduled Maintenance Outage;
- Work Week 9020 and 9021 Emergent Work Activities with the ‘A’ CBC OOS;
- Work Week 9022 Risk Management with the Startup Transformer OOS during 1A201 Breaker Planned Maintenance;
- Work Week 9024 with Emergent Work on the High Pressure Coolant Injection (HPCI) ‘B’ Room Cooler and 1K4 Instrument Air Compressor; and
- Work Week 9026 Risk Management during the 1D6, Division 2 24 VDC Battery Testing.

These activities were selected based on their potential risk significance relative to the Reactor Safety Cornerstones. As applicable for each activity, the inspectors verified that risk assessments were performed as required by 10 CFR 50.65(a)(4) and were accurate and complete. When emergent work was performed, the inspectors verified that the plant risk was promptly reassessed and managed. The inspectors reviewed the scope of maintenance work, discussed the results of the assessment with the licensee's probabilistic risk analyst or shift technical advisor, and verified plant conditions were consistent with the risk assessment. The inspectors also reviewed TS requirements and walked down portions of redundant safety systems, when applicable, to verify risk analysis assumptions were valid and applicable requirements were met. Documents reviewed are listed in the Attachment to this report.

These maintenance risk assessments and emergent work control inspection activities constituted five samples as defined in IP 71111.13-05.

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations (71111.15)

.1 Operability Evaluations

a. Inspection Scope

The inspectors reviewed the following issues:

- Well Water Functionality/Operability (Containment) OPR 000427: Well Water Leaking into Drywell Impact on Primary Containment;
- OPR 000431: Possible Pipe Support Issue on ‘B’ CBC;
- Temperature Element TE4443A Indicating 200 Degrees Fahrenheit Causes a Half Group 1 Isolation Signal;
- Safety-Related Instrument Air Compressor, 1K004, Duty Cycle at 100 percent Due to Air Leak on QEV7602B, Quick Exhaust Valve;
- HPCI Room Cooler, 1VAC014B, Operability during Replacement of HPCI Room Cooler, 1VAC014A; and
- HPCI/Reactor Core Isolation Cooling Equipment Operability with Elevated Room Temperatures.

The inspectors selected these potential operability issues based on the risk significance of the associated components and systems. The inspectors evaluated the technical adequacy of the evaluations to ensure that TS operability was properly justified and the subject component or system remained available such that no unrecognized increase in risk occurred. The inspectors compared the operability and design criteria in the appropriate sections of the TS and UFSAR to the licensee's evaluations to determine whether the components or systems were operable. Where compensatory measures were required to maintain operability, the inspectors determined whether the measures in place would function as intended and were properly controlled. The inspectors determined, where appropriate, compliance with bounding limitations associated with the evaluations. Additionally, the inspectors reviewed a sampling of corrective action documents to verify that the licensee was identifying and correcting any deficiencies associated with operability evaluations. Documents reviewed are listed in the Attachment to this report.

These operability inspection activities constituted six samples as defined in IP 71111.15-05.

b. Findings

(1) Immediate Operability Determination Failed to Address Impact of Degraded Condition on Primary Containment

Introduction: A finding of very low safety significance (Green) and associated NCV of 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," was identified by the inspectors for the failure of the licensee to follow procedure EN-AA-203-1001, "Operability Determinations/ Functionality Assessments," and ACP 110.1, "Conduct of Operations," to adequately assess the impact of the degraded drywell cooling system on Primary Containment.

Description: On March 24, 2010, the inspectors questioned the licensee on the ability of Primary Containment to meet its design functions with an identified leak from the drywell cooling system into the drywell. Specifically, the well water Primary Containment penetrations (part of the drywell cooling system) were designed as Type C penetrations and there was a question on whether the system was meeting those design basis requirements. According to the DAEC UFSAR, Type C penetrations require only one isolation valve outside of Primary Containment since they are part of piping that communicates neither with the reactor vessel, with the Primary Containment free space, nor with the environs. The inspectors were concerned that a leak in the drywell cooling system would allow it to communicate with the Primary Containment free space and the system would no longer be functioning as a Type C penetration. In addition, the degraded condition of the drywell cooling system piping caused the inspectors to question the ability of the system to meet single failure criteria for Primary Containment penetration isolations since the closed loop inside containment for the drywell cooling system acts as one of two barriers for Primary Containment isolation integrity.

The licensee wrote CAP 074069 on March 25, 2010, to document the inspectors' questions. The Operations Shift Manager (OSM) made an IOD and declared the Primary Containment system operable based on meeting all TSs and UFSAR requirements. On March 26, 2010, the inspectors questioned the OSM to describe the basis behind the IOD, considering the description of the drywell cooling system piping in the UFSAR and the TS Bases for T.S. 3.6.2 that support Primary Containment operability.

The licensee's engineering department provided information to the inspector's office on April 2, 2010, to explain the basis for considering the Primary Containment and Primary Containment Isolation Valves operable. On April 6, 2010, after reviewing all information, the inspectors questioned the licensee as to why a Prompt Operability Determination (POD) was not being performed in accordance with procedure EN-AA-203-1001, Operability Determinations/ Functionality Assessments." The OSM requested a functionality assessment on drywell cooling system piping inside containment on April 7, 2010, and a POD on April 8, 2010, (OPR 000427).

Procedure EN-AA-203-1001, Section 4.3.1 states, in part, that "Operability Determinations must address all plant conditions" and "the OSM or Senior Reactor Operator should document the basis for the IOD in their own words and provide sufficient detail such that an independent person reading it would have sufficient facts and information to understand the basis of the decision." The inspectors determined that the basis for the IOD did not adequately address the impact of the breach of the closed system inside containment on Primary Containment integrity, and did not contain sufficient detail for an independent person to understand the basis of the decision since there were unanswered questions regarding the ability of the system to comply with design and licensing basis requirements.

The DAEC site specific procedure, ACP 110.1, "Conduct of Operations," Attachment 10, "Operability Determinations," Section 3.0 states, in part, the following:

> "3.1 When questions regarding equipment operability arise, the appropriate members of the technical staff shall be contacted to assist in completing a prompt operability determination;" and
>
> "3.2 Operability determinations shall be requested for SSC or components that are required to be operable by Tech Specs or that perform a required support functions as specified by the Tech Specs definition of operability;" and
>
> "3.4 When there is a cause to question the status of a structure, system or component, the process of determining its status is expected to be thorough and prompt."

Based upon these procedure requirements and expectations, the inspectors determined that the degraded condition of the drywell cooling system piping inside containment warranted a thorough, documented evaluation of the impact the drywell cooling system leakage had on Primary Containment integrity in order to establish a basis for operability. The licensee did not perform a POD until inspectors questioned the operability of Primary Containment on several occasions.

Additionally, procedure EN-AA-203-1001, section 4.4 states, "the determination of the need for a POD must be completed in a time frame commensurate with the SSC's safety significance." Technical Specification 3.6.1.1, "Primary Containment," requires, in part, that Primary Containment shall be Operable and requires the plant to be in Mode 3 within 12 hours if Primary Containment cannot be restored to Operable within 1 hour. Contrary to the above, the licensee's decision to request a POD 15 days after the inspectors questioned the ability of Primary Containment to perform its design functions was not commensurate with the safety significance of the system.

On April 14, 2010, OPR 000427 was completed by the licensee and supported Primary Containment operability. The operability determination also calculated an upper limit on drywell leakage from the well water system to be 5 gallons per minute in order to support Primary Containment operability. The licensee shut down the plant for a planned outage on April 25, 2010, to repair the leaks on the drywell cooling system. The 24 hour average drywell leakage at the time of the shutdown was 4.28 gallons per minute.

Analysis: The inspectors determined that the failure of the licensee to address the impact of the degraded conditions in the drywell cooling system on Primary Containment in an IOD, and to provide sufficient detail for an independent person to understand the basis for the decision, was contrary to step 4.3 of EN-AA-203-1001 and Attachment 10 of ACP 110.1, and represented a performance deficiency warranting further investigation.

The performance deficiency was determined to be more than minor because, if left uncorrected, failure to properly implement the operability procedures could result in safety-related components being incorrectly declared operable rather than inoperable or operable but degraded or non-conforming (a more significant safety concern). Specifically, if the licensee had not evaluated the degraded conditions of the drywell cooling system and established criteria for acceptable drywell cooling system leakage, the system could have degraded to a point where Primary Containment would be rendered inoperable. The inspectors concluded this finding was associated with the Barrier Integrity Cornerstone.

The inspectors determined the finding could be evaluated using the SDP in accordance with IMC 0609, "Significance Determination Process," Attachment 0609.04, "Phase 1 - Initial Screening and Characterization of Findings," Table 4a for the Barrier Integrity Cornerstone. Because inspectors answered "No" to all four questions under the Containment Barrier column, the finding was determined to be of very low safety significance (Green).

This finding had a cross-cutting aspect in the area of Human Performance, Decision Making, because the licensee did not use conservative assumptions in decision making and adopt a requirement to demonstrate that the proposed action is safe in order to proceed rather than a requirement to demonstrate that it is unsafe in order to disapprove the action. Specifically, not evaluating the degraded condition of the drywell cooling system to determine its impact on Primary Containment operability was a non-conservative assumption in the IOD. [H.1(b)]

Enforcement: Title 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," requires, in part, that activities affecting quality shall be prescribed by documented procedures of a type appropriate to the circumstances and shall be accomplished in accordance with these procedures. Procedures ACP 110.1, "Conduct of Operations," Revision 24, and EN-AA-203-1001, "Operability Determinations/ Functionality Assessment," Revision 2, establish the licensee's implementing procedures for evaluating operability questions that arise for safety-related SSCs; an activity affecting quality.

Contrary to the above, on March 25, 2010, the licensee failed to follow Attachment 10 of ACP 110.1 and step 4.3 of procedure EN-AA-203-1001. Specifically, the OSM failed to make an IOD that addressed the impact of the degraded condition in the drywell cooling system piping on Primary Containment, a safety-related SSC, and to provide sufficient detail for an independent person to understand the basis for the decision. Because this violation was of very low safety significance and was entered into the licensee's corrective action program as CAP 074069 and CR 568618, the violation is being treated as an NCV, consistent with Section VI.A.1 of the NRC Enforcement Policy (NCV 05000331/2010003-01, Immediate Operability Determination Failed to Address Impact of Degraded Condition on Primary Containment).

1R18 Plant Modifications (71111.18)

.1 Permanent Plant Modifications

a. Inspection Scope

The following engineering design items were reviewed and selected aspects were discussed with engineering personnel:

- 'B' ESW Pump Remote Shutdown Panel Re-Wiring to Preclude Hot-Short Vulnerability; and
- Review of Procedural Documentation, STP 3.4.4-01 Used for Reclassification of Drywell Leakage.

These items and related documentation were reviewed for adequacy of the associated 10 CFR 50.59 safety evaluation screening, consideration of design parameters, implementation of the modification, post-modification testing, and relevant procedures, design, and licensing documents were properly updated. The inspectors observed ongoing and completed work activities to verify that installation was consistent with the design control documents. The modifications were required to: remove a hot-short vulnerability associated with the operation of the 'B' ESW pump from a remote shutdown panel by re-wiring the control power supply cabling to remain outside the cable spreading room; and to implement an STP developed to permit reclassification of unidentified drywell leakage as identified, non-reactor coolant system (RCS) pressure boundary leakage. Documents reviewed in the course of this inspection are listed in the Attachment to this report.

These inspection activities constituted two permanent plant modification samples as defined in IP 71111.18-05.

b. Findings

(1) Surveillance Test Procedure did not Include Appropriate Guidance for Reclassifying Leakage Inside the Drywell

Introduction: A finding of very low safety significance and associated NCV of 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," was identified by the inspectors for the licensee's failure to prescribe a procedure, appropriate to the circumstances for reclassifying leakage inside the drywell, to meet the TS definition of Identified Leakage.

Description: On March 3, 2010, the licensee identified an increasing trend in Unidentified Leakage inside the drywell while calculating Unidentified Leakage using STP 3.0.0-01, "Instrument Checks," Revision 106. The licensee generated CAP 073599 to document the increasing trend. On March 15, 2010, the licensee began to reclassify a portion of the Unidentified Leakage as Identified Leakage using STP 3.4.4-01, "Reclassification of Drywell Leakage," Revision 3, in conjunction with STP 3.0.0-01, which contains guidance on leakage evaluation. Per Attachment 3, Step 4.0 of STP 3.0.0-01, the licensee was allowed to use STP 3.4.4-01 to reclassify Unidentified Leakage as Identified Leakage if all of the following conditions were met (paraphrased): "drywell radiation monitor particulate indication had not increased or shown a step increase as indicated on RR4379A/B, AND if the gamma specification analysis results did not indicate the presence of short-lived radionuclides, OR if there was no step increase or increasing trend by the radiation monitor or the laboratory results indicated no short lived radionuclides; and conductivity of the drywell sump sample did not indicate a decreasing trend by at least 10 percent; and no Na-24 was present (greater than detectable) in the drywell sump sample."

Duane Arnold's TSs define Identified Leakage as: "(1) leakage into the drywell, such as that from pump seals or valve packing, that is captured and conducted to a sump or collecting tank; or (2) leakage into the drywell atmosphere from sources that are both specifically located and known not to interfere with the operation of leakage detection systems." The leakage detection system at Duane Arnold consists of a drywell sump system and a primary containment air sampling system.

On April 8, 2010, operators identified that torus level was rising at a higher rate than normally occurs with changes in temperature. Engineering staff initiated CAP 074440 to document the increasing trend in torus level. On April 13, 2010, a Condition Evaluation (CE) 008159 was assigned to engineering personnel to evaluate the rising trend in torus water level. On April 20, 2010, operators in the control room received the torus high level alarm. The reactor was shut down on April 25, 2010, to address the well water leaks in the drywell cooling system. During the shutdown, engineers walked down the well water leaks and identified that some of the water leaking from well water circuit setter valve V57-0051 was leaking into the downcomer and to the torus. On April 28, 2010, CE 008159 was completed and concluded that "the source of the torus level increase was determined to be due to the well water leak found during the April 2010 forced outage based on a walkdown of the leak."

While reviewing historical changes that were made to STP 3.0.0-01 and STP 3.4.4-01, the inspectors questioned the licensee regarding the adequacy of STP 3.0.0-01. Specifically, STP 3.0.0-01 allowed the plant to reclassify leakage from Unidentified Leakage to Identified Leakage without verifying that the leakage did not interfere with the operation of the leakage detection system. Because of the indications available that torus water level was slowly rising, operations and engineering personnel had reasonable evidence that not all of the water leaking from the well water system was making its way to the drywell sump system. Because the drywell sump system is used to quantify the amount of Identified and Unidentified Leakage, any well water that was going to the torus would not be quantified by Duane Arnold's leakage detection system. Therefore, the well water leak interfered with the operation of the leakage detection system and did not meet the TS definition of Identified Leakage.

Analysis: The inspectors determined that STP 3.0.0-01 did not include a requirement to verify that leakage inside the drywell did not interfere with the leakage detection system prior to reclassifying Unidentified Leakage as Identified Leakage. The failure to define requirements and prescribe a procedure appropriate to the circumstances for reclassifying leakage inside the drywell to meet the TS definition of Identified Leakage was contrary to 10 CFR 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," and was a performance deficiency. The performance deficiency was determined to be more than minor and a finding because, if left uncorrected, the performance deficiency had the potential to lead to a more significance safety concern. Specifically, because STP 3.0.0-01 allowed reclassifying Unidentified Leakage as Identified Leakage without meeting the TS definition of Identified Leakage, operations personnel could have continued to reclassify the leakage, exceeding the TS limit for Unidentified Leakage.

The inspectors applied IMC 0609, Attachment 4, "Phase 1 - Initial Screening and Characterization of Findings" to this finding. Under Table 2, all RCS Boundary issues that are not a result of a plant upset will be considered using the Initiating Events Cornerstone. Under Table 4a for the Initiating Events Cornerstone, the finding screened as Green because there was no actual RCS leakage that would have exceeded the TS limit, and the finding did not affect other mitigation systems resulting in a total loss of safety function. The inspectors determined that the contributing cause that provided the most insight into the performance deficiency affected the cross-cutting area of Problem Identification and Resolution, relating to the corrective action program components, and involving the aspect associated with the licensee assessing information from the corrective action program in aggregate to identify common cause problems. Specifically, operations and engineering did not assess information contained in CAP 074440 that identified a rising trend in torus water level to the condition adverse to quality identified in CAP 073599 relative to the increase in leakage inside the drywell and the station's use of STP 3.0.0-01 and STP 3.4.4-01 to reclassify Unidentified Leakage to Identified Leakage. [P.1(b)]

Enforcement: Title 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," requires, in part, that activities affecting quality shall be prescribed by documented procedures, of a type appropriate to the circumstances and shall be accomplished in accordance with these procedures. Contrary to the above, beginning on March 15, 2010, the licensee failed to prescribe a procedure appropriate to the circumstances for reclassifying leakage inside the drywell. Specifically, STP 3.0.0-01 did not include appropriate guidance to preclude operations personnel from reclassifying Unidentified Leakage as Identified Leakage without meeting the TS definition of Identified Leakage. Because this issue was of very low safety significance and was entered into the licensee's corrective action program as CR 392257 and 568613, this violation is being treated as an NCV, consistent with Section VI.A.1 of the NRC Enforcement Policy (NCV 05000331/2010003-02, Surveillance Test Procedure did not Include Appropriate Guidance for Reclassifying Leakage Inside the Drywell).

1R19 Post-Maintenance Testing (71111.19)

.1 Post-Maintenance Testing

a. Inspection Scope

The inspectors reviewed the following post-maintenance testing (PMT) activities to verify that procedures and test activities were adequate to ensure system operability and functional capability:

- 'A' CBC Oil Pump Replacement;
- Safety Relief Valve, PSV-4401 Solenoid Valve Repair;
- 'A' CBC Compressor Replacement and Unit Flush;
- 'B' ESW Pump Remote Shutdown Panel Modification Testing;
- QEV-7602B Control Air Quick Exhaust Valve Replacement (1K004); and
- Low Pressure Coolant Injection Swing Bus Power Supply Breaker, 1B3401, Replacement.

These activities were selected based upon the SSC's ability to impact risk. The inspectors evaluated these activities for the following (as applicable): the effect of testing on the plant had been adequately addressed; testing was adequate for the maintenance performed; acceptance criteria were clear and demonstrated operational readiness; test instrumentation was appropriate; tests were performed as written in accordance with properly reviewed and approved procedures; equipment was returned to its operational status following testing (temporary modifications or jumpers required for test performance were properly removed after test completion); and test documentation was properly evaluated. The inspectors evaluated the activities against TS, the UFSAR, 10 CFR Part 50 requirements, licensee procedures, and various NRC generic communications to ensure that the test results adequately ensured that the equipment met the licensing basis and design requirements. In addition, the inspectors reviewed corrective action documents associated with post-maintenance tests to determine whether the licensee was identifying problems and entering them in the CAP and that the problems were being corrected commensurate with their importance to safety. Documents reviewed are listed in the Attachment to this report.

These inspection activities constituted six PMT samples as defined in IP 71111.19-05.

b. Findings

No findings of significance were identified.

1R20 Outage Activities (71111.20)

.1 Other Outage Activities

a. Inspection Scope

The inspectors evaluated outage activities for a planned maintenance outage that began on April 25, 2010, and ended on May 3, 2010, when the main generator was connected to the electrical grid. The inspectors reviewed activities to ensure that the licensee considered risk in developing, planning, and implementing the outage schedule.

The inspectors observed or reviewed the reactor shutdown and cooldown, outage equipment configuration and risk management, electrical lineups, selected clearances, control and monitoring of decay heat removal, control of containment activities, startup and heatup activities, and identification and resolution of problems associated with the outage. Additionally, the inspectors observed activities associated with the identification and repair of previously identified leakage in the well water system that supplies the drywell cooling system. Documents reviewed are listed in the Attachment to this report.

These inspection activities constituted one other outage sample as defined in IP 71111.20-05.

b. Findings

No findings of significance were identified.

1R22 Surveillance Testing (71111.22)

.1 Surveillance Testing

a. Inspection Scope

The inspectors reviewed the test results for the following activities to determine whether risk-significant systems and equipment were capable of performing their intended safety function and to verify testing was conducted in accordance with applicable procedural and TS requirements:

- STP 3.3.3.2-09A, Reactor Water Level and Pressure Instruments-Loops A and C (Routine);
- STP 3.5.1-02B, 'B' Low Pressure Coolant Injection System Operability Tests (In-Service Test);
- STP 3.8.1-04A, 1G21 SBDG Slow Start from Normal Starting Air (Routine);
- STP 3.3.6.1-09, Primary Containment Isolation, Group 2 and 4, Logic System Functional Test (Containment Isolation Valve);
- STP 3.3.8.1-04, 'B' 4 KV Emergency Bus Under Voltage Relay Calibration Check (Routine); and
- STP 3.4.6-01, Reactor Coolant Iodine Activity (Routine).

The inspectors observed in-plant activities and reviewed procedures and associated records to determine the following:

- did preconditioning occur;
- were the effects of the testing adequately addressed by control room personnel or engineers prior to the commencement of the testing;
- were acceptance criteria clearly stated, demonstrated operational readiness, and consistent with the system design basis;
- plant equipment calibration was correct, accurate, and properly documented;
- as-left setpoints were within required ranges; and the calibration frequency were in accordance with TSs, the UFSAR, procedures, and applicable commitments;
- measuring and test equipment calibration was current;
- test equipment was used within the required range and accuracy; applicable prerequisites described in the test procedures were satisfied;

- test frequencies met TS requirements to demonstrate operability and reliability; tests were performed in accordance with the test procedures and other applicable procedures; jumpers and lifted leads were controlled and restored where used;
- test data and results were accurate, complete, within limits, and valid;
- test equipment was removed after testing;
- where applicable for inservice testing activities, testing was performed in accordance with the applicable version of Section XI, American Society of Mechanical Engineers code, and reference values were consistent with the system design basis;
- where applicable, test results not meeting acceptance criteria were addressed with an adequate operability evaluation or the system or component was declared inoperable;
- where applicable for safety-related instrument control surveillance tests, reference setting data were accurately incorporated in the test procedure;
- where applicable, actual conditions encountering high resistance electrical contacts were such that the intended safety function could still be accomplished;
- prior procedure changes had not provided an opportunity to identify problems encountered during the performance of the surveillance or calibration test;
- equipment was returned to a position or status required to support the performance of its safety functions; and
- all problems identified during the testing were appropriately documented and dispositioned in the CAP.

Documents reviewed are listed in the Attachment to this report.

These inspection activities constituted four routine surveillance testing samples, one in-service testing sample, and one containment isolation valve sample as defined in IP 71111.22, Sections -02 and -05.

b. Findings

No findings of significance were identified.

1EP6 Drill Evaluation (71114.06)

.1 Emergency Preparedness Drill Observation

a. Inspection Scope

The inspectors evaluated the conduct of routine licensee emergency drills on April 7 and May 12, 2010, to identify any weaknesses and deficiencies in classification, notification, and protective action recommendation development activities. The inspectors observed emergency response operations in the simulator control room and the technical support center to determine whether the event classification, notifications, and protective action recommendations were performed in accordance with procedures. The inspectors also attended the licensee drill critiques to compare any inspector-observed weaknesses with those identified by the licensee staff in order to evaluate the critiques and to verify whether the licensee staff was properly identifying weaknesses and entering them into the corrective action program. As part of the inspection, the inspectors reviewed the drill packages and other documents listed in the Attachment to this report.

These emergency preparedness drill inspection activities constituted one sample as defined in IP 71114.06-05.

b. Findings

No findings of significance were identified.

2. RADIATION SAFETY

Cornerstones: Public and Occupational Radiation Safety

2RS8 Radioactive Solid Waste Processing and Radioactive Material Handling, Storage, and Transportation (71124.08)

This inspection constituted one radioactive solid waste processing and radioactive material handling, storage, and transportation sample as defined in IP 71124.08-05.

.1 Inspection Planning (02.01)

a. Inspection Scope

The inspectors reviewed the solid radioactive waste system description in the UFSAR, the Process Control Program (PCP), and the recent radiological effluent release report for information on the types, amounts, and processing of radioactive waste disposed.

The inspectors reviewed the scope of any quality assurance (QA) audit in this area since the last inspection to gain insights into the licensee's performance and inform the "smart sampling" inspection planning.

b. Findings

No findings of significance were identified.

.2 Radioactive Material Storage (02.02)

a. Inspection Scope

The inspectors selected three areas where containers of radioactive waste are stored, and determined whether the containers were labeled in accordance with 10 CFR 20.1904, "Labeling Containers," or controlled in accordance with 10 CFR 20.1905, "Exemptions to Labeling Requirements," as appropriate.

The inspectors determined whether the radioactive materials storage areas were controlled and posted in accordance with the requirements of 10 CFR Part 20, "Standards for Protection against Radiation." For materials stored or used in the controlled or unrestricted areas, the inspectors determined whether they were secured against unauthorized removal and controlled in accordance with 10 CFR 20.1801, "Security of Stored Material," and 10 CFR 20.1802, "Control of Material Not in Storage," as appropriate.

The inspectors determined whether the licensee established a process for monitoring the impact of long-term storage (e.g., buildup of any gases produced by waste decomposition, chemical reactions, container deformation, loss of container integrity, or re-release of free-flowing water) that was sufficient to identify potential unmonitored, unplanned releases or nonconformance with waste disposal requirements.

The inspectors selected four containers of stored radioactive materials, and observed the containers for signs of swelling, leakage, and deformation.

b. Findings

No findings of significance were identified.

.3 Radioactive Waste System Walkdown (02.03)

a. Inspection Scope

The inspectors walked down accessible portions of selected radioactive waste processing systems to determine whether the current system configuration and operation agreed with the descriptions in the UFSAR, offsite dose calculation manual, and PCP.

The inspectors reviewed administrative and/or physical controls (i.e., drainage and isolation of the system from other systems) to determine whether the equipment which is not-in-service or abandoned in place would contribute to an unmonitored release path and/or affect operating systems or be a source of unnecessary personnel exposure. The inspectors determined whether the licensee reviewed the safety significance of systems and equipment abandoned in place in accordance with 10 CFR 50.59, “Changes, Tests, and Experiments”.

The inspectors reviewed the adequacy of changes made to the radioactive waste processing systems since the last inspection. The inspectors determined whether changes from what is described in the UFSAR were reviewed and documented in accordance with 10 CFR 50.59, as appropriate and to assess the impact on radiation doses to members of the public.

The inspectors determined whether the waste stream mixing, sampling procedures, and methodology for waste concentration averaging were consistent with the PCP, and provided representative samples of the waste product for the purposes of waste classification as described in 10 CFR 61.55, “Waste Classification” for selected waste processes.

The inspectors evaluated whether the tank recirculation procedures provided sufficient mixing for systems that provide tank recirculation prior to sampling.

The inspectors assessed whether the licensee’s PCP correctly described the current methods and procedures for dewatering and waste stabilization (e.g., removal of freestanding liquid).

b. Findings

No findings of significance were identified.

.4 Waste Characterization and Classification (02.04)

a. Inspection Scope

The inspectors selected the following radioactive waste streams for review:

- Dry Active Waste; and
- Resin.

For the waste streams listed above, the inspectors determined whether the licensee's radiochemical sample analysis results (i.e., "10 CFR Part 61" analysis) were sufficient to support radioactive waste characterization as required by 10 CFR Part 61, "Licensing requirements for Land Disposal of Radioactive Waste." The inspectors evaluated whether the licensee's use of scaling factors and calculations to account for difficult-to-measure radionuclides was technically sound and based on current 10 CFR Part 61 analyses for the selected radioactive waste streams.

The inspectors determined whether changes to plant operational parameters were taken into account to: (1) maintain the validity of the waste stream composition data between the annual or biennial sample analysis update; and (2) assure that waste shipments continued to meet the requirements of 10 CFR Part 61 for the waste streams selected above.

The inspectors evaluated whether the licensee had established and maintained an adequate QA program to ensure compliance with the waste classification and characterization requirements of 10 CFR 61.55 and 10 CFR 61.56, "Waste Characteristics."

b. Findings

No findings of significance were identified.

.5 Shipment Preparation (02.05)

a. Inspection Scope

The inspectors reviewed paperwork for shipment packaging, surveying, labeling, marking, placarding, vehicle checks, emergency instructions, disposal manifest, shipping papers provided to the driver, and licensee verification of shipment readiness because no shipments were in process during the inspection. The inspectors determined whether the requirements of applicable transport cask certificate of compliance had been met. The inspectors evaluated whether the receiving licensee was authorized to receive the shipment packages. The inspectors evaluated whether the licensee's procedures for cask loading and closure procedures were consistent with the vendor's current approved procedures.

The inspectors interviewed radiation workers regarding the conduct of radioactive waste processing and radioactive material shipment preparation and receipt activities. The inspectors determined whether the shippers were knowledgeable of the shipping regulations and whether shipping personnel demonstrated adequate skills to accomplish the package preparation requirements for public transport with respect to:

- The licensee's response to NRC Bulletin 79-19, "Packaging of Low-Level Radioactive Waste for Transport and Burial," dated August 10, 1979; and
- Title 49 CFR Part 172, "Hazardous Materials Table, Special Provisions, Hazardous Materials Communication, Emergency Response Information, Training Requirements, and Security Plans," Subpart H, "Training."

Additionally, due to limited opportunities for direct observation, the inspectors reviewed the technical instructions presented to workers during routine training. The inspectors assessed whether the licensee's training program provided training to personnel responsible for the conduct of radioactive waste processing and radioactive material shipment preparation activities.

b. Findings

No findings of significance were identified.

.6 Shipping Records (02.06)

a. Inspection Scope

The inspectors determined whether the shipping documents indicated the proper shipper name; emergency response information and a 24-hour contact telephone number; accurate curie content and volume of material; and appropriate waste classification, transport index, and UN number for the following radioactive shipments:

- Shipment RSR 09-13; Control Rod Drive Boxes; 05/20/2009;
- Shipment RSR 09-29; Dry Active Waste; 07/16/2009;
- Shipment RSR 09-30; Resin High Integrity Container; 09/08/2009;
- Shipment RSR 09-31; Resin High Integrity Container; 09/14/2009; and
- Shipment RSR 10-03; Dry Active Waste; 02/21/2010.

Additionally, the inspectors assessed whether the shipment placarding was consistent with the information in the shipping documentation.

b. Findings

No findings of significance were identified.

.7 Identification and Resolution of Problems (02.07)

a. Inspection Scope

The inspectors determined whether problems associated with radioactive waste processing, handling, storage, and transportation, were being identified by the licensee at an appropriate threshold, were properly characterized, and were properly addressed for resolution in the licensee corrective action program. Additionally, the inspectors evaluated whether the corrective actions were appropriate for a selected sample of problems documented by the licensee that involve radioactive waste processing, handling, storage, and transportation.

The inspectors reviewed results of selected audits performed since the last inspection of this program and evaluated the adequacy of the licensee's corrective actions for issues identified during those audits.

Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA1 Performance Indicator Verification (71151)

1. Reactor Coolant System Specific Activity

a. Inspection Scope

The inspectors sampled licensee submittals for the RCS Specific Activity performance indicator (PI) for the period from the 2nd quarter, 2009, through the 1st quarter, 2010. To determine the accuracy of the PI data reported during those periods, PI definitions and guidance contained in the Nuclear Energy Institute (NEI) Document 99-02, "Regulatory Assessment Performance Indicator Guideline," Revision 6, was used. The inspectors reviewed the licensee's RCS chemistry samples, TS requirements, issue reports, event reports and NRC Integrated Inspection Reports for the period of April, 2009, through March, 2010, to validate the accuracy of the submittals. The inspectors also reviewed the licensee's issue report database to determine if any problems had been identified with the PI data collected or transmitted for this indicator and none were identified. In addition to record reviews, the inspectors observed a chemistry technician obtain and analyze a reactor coolant system sample. Documents reviewed are listed in the Attachment to this report.

These inspection activities constituted one reactor coolant system specific activity sample as defined in IP 71151-05.

b. Findings

No findings of significance were identified.

2. Reactor Coolant System Leakage

a. Inspection Scope

The inspectors sampled licensee submittals for the RCS Leakage PI for the period from the 2nd quarter, 2009, through the 1st quarter, 2010. To determine the accuracy of the PI data reported during those periods, PI definitions and guidance contained in the NEI Document 99-02, "Regulatory Assessment Performance Indicator Guideline," Revision 6, was used. The inspectors reviewed the licensee's operator logs, RCS leakage tracking data, issue reports, event reports and NRC Integrated Inspection Reports for the period April, 2009, through March, 2010, to validate the accuracy of the submittals. The inspectors also reviewed the licensee's issue report database to determine if any problems had been identified with the PI data collected or transmitted for this indicator and none were identified. Documents reviewed are listed in the Attachment to this report.

This inspection activity constituted one reactor coolant system leakage sample as defined in IP 71151-05.

b. Findings

No findings of significance were identified.

4OA2 Identification and Resolution of Problems (71152)

Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity, Emergency Preparedness, Public Radiation Safety, Occupational Radiation Safety, and Physical Protection

1. Routine Review of Items Entered into the Corrective Action Program

a. Inspection Scope

As part of the various baseline inspection procedures discussed 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 licensee's CAP at an appropriate threshold, that adequate attention was being given to timely corrective actions, and that adverse trends were identified and addressed. Attributes reviewed included: the complete and accurate identification of the problem; that timeliness was commensurate with the safety significance; that evaluation and disposition of performance issues, generic implications, common causes, contributing factors, root causes, extent-of-condition reviews, and previous occurrences reviews were proper and adequate; and that the classification, prioritization, focus, and timeliness of corrective actions were commensurate with safety and sufficient to prevent recurrence of the issue. Minor issues entered into the licensee's CAP as a result of the inspectors' observations are included in the Attachment to this report.

These routine reviews for the identification and resolution of problems did not constitute any additional inspection samples. Instead, by procedure they were considered an integral part of the inspections performed during the quarter and documented in Section 1 of this report.

b. Findings

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.

These daily reviews were performed by procedure as part of the inspectors' daily plant status monitoring activities and, as such, did not constitute any separate inspection samples.

b. Findings

No findings of significance were identified.

3. Semi-Annual Trend Review

a. Inspection Scope

The inspectors performed a review of the licensee's CAP and associated documents to identify trends that could indicate the existence of a more significant safety issue. The inspectors' review was focused on repetitive equipment issues, 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. The inspectors' review nominally considered the six month period of January 2010, through June 2010, although some examples expanded beyond those dates where the scope of the trend warranted.

The review also included issues documented outside the normal CAP in major equipment problem lists, repetitive and/or reworks maintenance lists, departmental problem/challenges lists, system health reports, quality assurance audit/surveillance reports, self-assessment reports, and Maintenance Rule assessments. The inspectors compared and contrasted their results with the results contained in the licensee's CAP trending reports. Corrective actions associated with a sample of the issues identified in the licensee's trending reports were reviewed for adequacy.

This review constituted one semi-annual trend inspection sample as defined in IP 71152-05.

b. Findings

No findings of significance were identified.

4. Annual Sample: Review of Operator Workarounds (OWAs)

a. Inspection Scope

The inspectors evaluated the licensee's implementation of their process used to identify, document, track, and resolve operational challenges. Inspection activities included, but were not limited to, a review of the cumulative effects of the OWAs on system availability and the potential for improper operation of the system, for potential impacts on multiple systems, and on the ability of operators to respond to plant transients or accidents.

The inspectors performed a review of the cumulative effects of OWAs. The documents listed in the Attachment to this report were reviewed to accomplish the objectives of the inspection procedure. The inspectors reviewed both current and historical operational challenge records to determine whether the licensee was identifying operator challenges at an appropriate threshold, had entered them into their CAP and proposed or implemented appropriate and timely corrective actions which addressed each issue. Reviews were conducted to determine if any operator challenge could increase the possibility of an Initiating Event, if the challenge was contrary to training, required a change from long-standing operational practices, or created the potential for inappropriate compensatory actions. Additionally, all temporary modifications were reviewed to identify any potential effect on the functionality of Mitigating Systems, impaired access to equipment, or required equipment uses for which the equipment was not designed. Daily plant and equipment status logs, degraded instrument logs, and operator aids or tools being used to compensate for material deficiencies were also assessed to identify any potential sources of unidentified operator workarounds.

This review constituted one operator workaround annual inspection sample as defined in IP 71152-05.

b. Findings

No findings of significance were identified.

4OA3 Follow-Up of Events and Notices of Enforcement Discretion (71153)

1. Loss of the Division 1 Non-Essential 480 VAC Supply Transformer (1XR1) and Motor Control Centers Due to a Failed Lightning Arrestor In the Switchyard

a. Inspection Scope

The inspectors reviewed the plant's response to a loss of the 1XR1 Transformer due to a failed lightning arrestor that occurred on April 20, 2010. The lightning arrestor was on a 36 kV line between disconnect 5981 and C5980, located in the switchyard. Power was lost to several nonsafety-related loads, including drywell sump pumps. Documents reviewed in this inspection are listed in the Attachment to this report.

This event follow-up review constituted one sample as defined in IP 71153-05.

b. Findings

No findings of significance were identified.

2. Manual Reactor Scram Due to High Vibrations on the Number 6 Main Turbine Bearing During Plant Shutdown for a Planned Maintenance Outage

a. Inspection Scope

The inspectors reviewed the plant's response to a manual reactor scram inserted by the Operators during a planned shutdown for a maintenance outage. The main turbine was tripped and a manual scram was initiated due to high vibrations on the #6 main turbine bearing. The scram was uncomplicated and all systems responded as anticipated. The licensee entered this event into their CAP as a significant condition adverse to quality (CAP 074740), and performed a root cause evaluation (RCE) of the issue. The inspectors reviewed the licensee's scram report and RCE regarding the potential causes for the high vibration condition as part of this inspection activity. Documents reviewed in this inspection are listed in the Attachment to this report.

This event follow-up review constituted one sample as defined in IP 71153-05.

b. Findings

No findings of significance were identified.

3. (Closed) Licensee Event Report (LER) 05000331/2010002-00: Condition Prohibited by Technical Specifications

a. Inspection Scope

The inspectors reviewed LER 2010-002-00 submitted by the licensee on March 11,2010.

This event, which occurred on January 4, 2010, with the plant operating at 100% reactor power, resulted in the plant unexpectedly increasing power to 105% due to a failed circuit card causing both turbine bypass valves (TBV) to reposition from the full closed to full open position at 03:24 hours. The plant commenced a fast power reduction to 68 percent in accordance with procedures. The TBVs eventually returned to the full closed position at 04:48 hours and the licensee inserted a Minimum Critical Power Ratio (MCPR) penalty in accordance with TS 3.7.7, Condition A. The TBVs again repositioned to the full open position at 07:00 hours and were finally closed at 13:34 hours. On January 5, 2010, the TBVs were declared operable following replacement of a failed circuit card.

On January 11, 2010, the licensee identified a condition prohibited by TS existed during the time of the event. A licensee-identified violation of very low safety significance involving TS 3.3.1.1, Reactor Protection System Instrumentation and 3.3.4.1, EOC-RPT Instrumentation, and the enforcement aspects of the violation are discussed in Section 4OA7.

While reviewing the LER submitted for the condition prohibited by TS, the inspectors identified that the licensee failed to submit a report per 10 CFR 50.73(a)(2)(v)(A) and (D). The characterization of the issue of concern is discussed in Section 4OA3.3b. below.

Documents reviewed in this inspection are listed in the Attachment to this report.

This LER review constituted one sample as defined in IP 71153-05.

This LER is closed; however, the inspectors will review any new, supplemented, or revised information concerning the failure to report per 10 CFR 50.73 once submitted by the licensee.

b. Findings

(1) Failure to Submit LER per 10 CFR 50.73(a)(2)(v)(A) and (D)

Introduction: A Severity Level IV non-cited violation of 10 CFR Part 50.73(a)(2)(v)(A) and (D) was identified by the inspectors for the failure of the licensee to report an event or condition that could have prevented the fulfillment of the Turbine Stop Valve Closure and Turbine Control Valve Fast Closure RPS, and EOC-RPT safety functions, which are relied upon to shutdown the reactor and maintain it in a shutdown condition, and mitigate the consequences of an accident, respectively.

Description: On January 4, 2010, both TBVs unexpectedly opened from 100% reactor power (Inspection Report 05000331/2010002 documented the inspectors' review of this event). On March 11, 2010, the licensee submitted LER 2010-002-00 in accordance with 10 CFR 50.73(a)(2)(i)(B) for a condition that was prohibited by Technical

Specifications. During a review of the event on January 11, 2010, the licensee recognized that if the TBVs were open and inoperable between 26% and 39% reactor power, the Turbine Control Valve Fast Closure and Turbine Stop Valve Closure RPS, and EOC-RPT functions could be unintentionally bypassed due to lowering turbine first stage pressure. These trips are normally automatically bypassed when turbine first stage pressure is less than 26% reactor power.

The licensee concluded that since reactor power was never less than 39%, the trips remained available throughout the event, although they should have been declared inoperable. The licensee also concluded in the LER that the event did not represent a safety system functional failure.

The inspectors consulted NRC headquarters and regional staff to determine whether the inoperability and potential to bypass the RPS and EOC-RPT safety functions between the power levels of 26% and 39% reactor power levels, represented a condition that could have prevented the fulfillment of those safety functions, even if the plant did not actually reach that power range. The inspectors noted that NUREG-1022, "Event Reporting Guidelines 10 CFR 50.72 and 50.73, Revision 2" states, in part, that "The intent of these criteria is to capture those events where there would have been a failure of a safety system to properly complete a safety function, regardless of whether there was an actual demand." NUREG-1022, Revision 2, also states, "The definition of the systems included in the scope of these criteria is provided in the rules themselves. It includes systems required by the TS to be operable to perform one of the four functions (A) through (D) specified in the rule."

Technical Specification 3.3.1.1, "Reactor Protection System Instrumentation," Table 3.3.1.1-1, Functions 8 and 9, Turbine Stop Valve Closure and Turbine Control Valve Fast Closure – Trip Oil Pressure – Low, are required to be operable at all times when Rated Thermal Power is ≥ 26%. The inspectors determined that the Turbine Stop Valve Closure and Turbine Control Valve Fast Closure functions were applicable to 50.73(a)(2)(v)(A) since they are systems needed to shutdown the reactor.

Technical Specification 3.3.4.1, "EOC-RPT Instrumentation," requires two channels per trip system for each EOC-RPT to be operable at all times when Rated Thermal Power is ≥ 26%. The inspectors determined that the EOC-RPT function was applicable to 10 CFR 50.73(a)(2)(v)(D) since it is a system needed to mitigate the consequences of an accident.

The inspectors questioned the licensee regarding their position as to why the condition (two RPS functions and the EOC-RPT function inoperable) was not reported per 10 CFR 50.73(a)(2)(v)(A) and (D). The licensee generated an action request and conducted a condition evaluation to document their position. The licensee's overall position was that because the plant never reached a power range of 26 to 39% reactor power, that the safety functions of the systems were never actually lost. Based on the inspector's discussion with NRC staff, regardless of whether there was an actual demand on the safety systems, the unplanned inoperability of all trains of redundant safety systems was subject to the requirements of 10 CFR 50.73(a)(2)(v).

<u>Analysis</u>: The inspectors determined that the failure to report the condition which could have prevented the fulfillment of the Turbine Stop Valve and Turbine Control Valve Fast Closure RPS and EOC-RPT safety functions in accordance with 10 CFR 50.73(a)(2)(v)(A) and (D) was a performance deficiency. Because violations of

10 CFR 50.73 are considered to be violations that potentially impact the regulatory process, they are dispositioned using the traditional enforcement process instead of the ROP SDP. Because the performance deficiency was not more than minor and not a finding per Inspection Manual Chapter 0612, Appendix B, “Issue Screening,” a cross-cutting aspect was not assigned to this violation. A licensee-identified violation was identified and is discussed in Section 4OA7 of this report and addresses the performance deficiency associated with the cause of the event. Per the NRC Enforcement Policy, Supplement I, Example D.4, a failure to make a required LER is categorized as a Severity Level IV violation.

Enforcement: Title 10 CFR Part 50.73(a)(2)(v) requires, in part, that licensees shall report any event or condition that could have prevented the fulfillment of the safety function of structures or systems that are needed to (A) shutdown the reactor and maintain it in a safe shutdown condition, and (D) mitigate the consequences of an accident. Contrary to this requirement, on March 12, 2010, the licensee failed to report a condition that could have prevented the fulfillment of the Turbine Stop Valve and TurbineControl Valve Fast Closure RPS, and EOC-RPT safety functions. Because this violation was not repetitive or willful, and was entered into the licensee’s corrective action program as AR 392462 and CR 568620, this violation is being treated as a Severity Level IV NCV, consistent with Section VI.A.1 of the NRC Enforcement Policy. (NCV 05000331/2010003-03, Failure to Submit LER per 10 CFR 50.73(a)(2)(v)(A) and (D)).

4OA5 Other Activities

.1 Institute of Nuclear Power Operations (INPO) Plant Assessment Report Review

a. Inspection Scope

The inspectors reviewed the final report for the INPO plant assessment conducted in November 2009. The inspectors reviewed the report to ensure that issues identified were consistent with the NRC perspectives of licensee performance and to verify if any significant safety issues were identified that required further NRC follow-up.

b. Findings

No findings of significance were identified.

4OA6 Management Meetings

.1 Exit Meeting Summary

On July 15, 2010, the inspectors presented the inspection results to Mr. K. Kleinheinz and other members of the licensee staff. The licensee acknowledged the issues presented. The inspectors confirmed that none of the potential report input discussed was considered proprietary.

.2 Interim Exit Meetings

Interim exits were conducted for:

- The results of the radioactive material processing, storage, and transportation program, including closure of an unresolved item with the Plant General Manager, Mr. D. Curtland, on May 21, 2010.

The inspectors confirmed that none of the potential report input discussed was considered proprietary. Proprietary material received during the inspection was returned to the licensee.

4OA7 Licensee-Identified Violations

The following findings of very low safety significance (Green) were identified by the licensee and are violations of NRC requirements, which meet the criteria of Section VI.A.1 of the NRC Enforcement Policy, NUREG-1600, for being dispositioned as NCVs.

Duane Arnold Operating License Condition 2.C.(2) requires, in part, that the licensee shall operate the facility in accordance with the Technical Specifications. Contrary to this requirement, the licensee did not operate the facility in accordance with TS 3.3.1.1, Reactor Protection System Instrumentation, and 3.3.4.1, End of Cycle – Recirculation Pump Trip Instrumentation on January 4, 2010. Ultimately, TS Required Actions to reduce thermal power to <26% rated thermal power were not taken within the associated Completion Times for TS 3.3.1.1, Condition E, and TS 3.3.4.1, Condition C. The facility was in violation of TS 3.3.1.1 and 3.3.4.1 for approximately 1.5 hours and 0.5 hours, respectively. The licensee entered the issues into their corrective action program as CAP 072527. Based on consultation with the NRC Region III Senior Risk Analyst, who determined that the delta core damage frequency for this finding was much less than 1E-6, it was considered to be of very low safety significance (Green).

Title 10 CFR 71.5 requires that each licensee shall comply with the applicable requirements of Department of Transportation regulations in 49 CFR 171 through 180. Title 49 CFR 173.443(c) requires that each transport vehicle used for transporting Class 7 (radioactive) materials as an exclusive use shipment that utilizes the provisions of paragraph (b) of this section be surveyed with an appropriate radiation detection instrument after each use. The vehicle may not be returned to service until the radiation dose rate at each accessible surface is 0.005 mSv per hour (0.5 mrem per hour) or less, and there is no significant removable (non-fixed) radioactive surface contamination as specified in paragraph (a). Contrary to the above, on August 24, 2009, an exclusive use vehicle delivering Class 7 material was allowed to leave the licensee's site without the licensee conducting an appropriate survey. This was identified by the licensee within a few hours and the contract carrier was called to return the vehicle to the site for the required survey. The vehicle returned directly without taking on an additional load and the required survey was conducted. No non-fixed radioactive surface contamination was identified and the radiation dose rates on the accessible surfaces of the vehicle were less than the required limit for free release of the vehicle. The issue was documented in the station's corrective action program (CAP 069233). Corrective actions included changing procedures to require notification of the shift health physics technician and the radwaste shipping coordinator when any and all radioactive shipments arrive on the site.

They are considered subject matter experts in radioactive material shipping requirements and can provide the required technical response. Additionally, warehouse personnel were trained on the provisions for exclusive use as defined in 49 CFR Part 173 and the new procedural requirements.

The finding was more than minor because: an inadequate (not conducted) survey was performed for the vehicle that was released. It was fortuitous that the required follow-up survey concluded that the vehicle did not contain radioactive material indistinguishable from background. While the finding involved control of radioactive material associated with transportation, it was determined to be of very low safety significance because it did not involve a 10 CFR Part 61 finding, package radiation limits in excess of the allowable limits, a breach of a package, a failure to make a required notification or provide emergency information, or a non-compliance with a Certificate of Compliance.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee

C. Costanzo, Site Vice President
D. Curtland, Plant General Manager
B. Eckes, NOS Manager
S. Catron, Licensing Manager
B. Murrell, Licensing Engineer Analyst
K. Kleinheinz, Engineering Director
B. Kindred, Security Manager
B. Simmons, Training Manager
G. Rushworth, Acting Operations Manager
T. Erger, Assistant Operations Manager
P. Giroir, Operations Support Manager
R. Porter, Chemistry & Radiation Protection Manager
M. Davis, Emergency Preparedness Manager
M. Lingenfelter, Design Engineering Manager
C. Kress, Maintenance Manager (Acting)
M. Heermann, Radwaste Shipper in Training
J. Karrich, ALARA Coordinator
R. Schlueter, ALARA Coordinator
W. Render, Instructor, DAEC Operator Training
L. Swenzkinski, Sr. Licensing Engineer

Nuclear Regulatory Commission

K. Feintuch, Project Manager, NRR
K. Riemer, Chief, Reactor Projects Branch 2

LIST OF ITEMS OPENED, CLOSED AND DISCUSSED

Opened

05000331/2010003-01	NCV	Failure to Follow the Procedures for Performing Operability Determinations (1R15)
05000331/2010003-02	NCV	Failure to Adequately Define Requirements and Prescribe a Procedure Appropriate for Plant Conditions (1R18)
05000331/2010003-03	NCV	Failure to Submit LER per 10 CFR 50.73(a)(2)(v)(A) and (D). (4OA3.3)

Closed

05000331/2010003-01	NCV	Failure to Follow the Procedures for Performing Operability Determinations (1R15)
05000331/2010003-02	NCV	Failure to Adequately Define Requirements and Prescribe a Procedure Appropriate for Plant Conditions (1R18)
05000331/2010003-03	NCV	Failure to Submit LER per 10 CFR 50.73(a)(2)(v)(A) and (D). (4OA3.3)

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.

Section 1R01

OP-AA-102-1002 (DAEC); Seasonal Readiness; Revision 2
NG-271K; Plant Return to Normal Operation Checklist; Revision 3
CAP 074921; NCAQ [Condition not Adverse to Quality] – Summer Readiness Period Items not completed
CAP 070674; CAQ [Condition Adverse to Quality] – INPO Identified – Switchyard Concrete Control Cable Troughs are Full of Water
Abnormal Operating Procedure (AOP) 304; Grid Instability; Revision 10

Section 1R04

Operating Instruction (OI) 170A1; Standby Gas Treatment (SBGT) System Electrical Lineup; Revision 8
OI 170A2; "A" SBGT System Valve Lineup and Checklist; Revision 4
OI 170A4; "B" SBGT System Valve Lineup and Checklist; Revision 2
OI 170A6; SBGT System Control Panel Lineup; Revision 3
OI 324A1; SBDG 1G-31 System Electrical Lineup; Revision 2
OI 324A3; SBDG 1G-31 System Valve Lineup and Checklist; Revision 10
OI 324A7; SBDG 1G-31 System Control Panel Lineup; Revision 3
OI 324A10; SBDG Standby/Readiness Condition Checklist; Revision 11
OI 304.1A1; 4160V/480V Nonessential Electrical Distribution System Electrical Lineup; Revision 6
OI 304.2A1; 4160V/480V Essential Electrical Distribution System Electrical Lineup; Revision 1
OI 304.1A3; 4160V/480V Nonessential Electrical Distribution system Startup Transformer Control Panel Lineup; Revision 3
OI 304.2A10; 4160V/480V Essential Electrical Distribution System Control Panel Lineup; Revision 1
OI 317.1A2; 120 VAC Regulated AC Distribution 1Y11 and 1Y21 Electrical Lineup (In-service); Revision 11
OI 317.1A4; 120 VAC Regulated AC Distribution 1D15/1Y1A and 1D25/1Y2A Panel Lineup (In-Service); Revision 1
OI 317.1A6; 120 VAC Regulated AC Distribution 1Y16 and 1Y26 Panel Lineup (In-Service); Revision 1
OI 317.1A8; 120 VAC Regulated AC Distribution 1Y10 and 1Y20 Panel Lineup (In-Service); Revision 3
OI 357A4; 120 VAC Uninterruptible Power Supply System 1D45/1Y4 Panel Lineup (In-Service); Revision 3
CAP 074326; Water Puddle on floor of 1A3 Switchgear Room Under Cable Trough on South Wall; dated April 6, 2010
CAP 074151; NCAQ – Update Walkdown Guidance to Include Water Intrusion Concerns; dated March 29, 2010

CAP 074025; NCAQ – CAP on Water on the Floor in the 1A3 Switchgear was Closed without Evaluation; dated March 23, 2010
CAP 054681; NCAQ – CAP on Water on the Floor in the 1A3 Switchgear was Closed without Evaluation; dated March 30. 2010
CAP 073553; NCAQ – Vulnerabilities to Water Intrusion; dated March 2, 2010
CAP 068364; NCAQ – Water on floor in 1A3 Essential Switchgear Room; dated July 11, 2009
CAP 068274; NCAQ – Electrical Switchgear/MCC Water Intrusion Walkdown Results; dated July 7, 2009
CAP 070228; CAQ – Startup Transformer Duct Leakage; dated October 6, 2009
CAP 070149; CAQ – Water Intrusion Causes Unexpected Alarm: Aux Trans to 1A1 Brkr 1A101 Trip; dated October 1, 2009
Corrective WO A99319; Water Dripping from Bottom of Non-Seg Bus Coming from 1X3 to 1A1 Switchgear Room
OI 454A2; ‘A’ ESW System Valve Lineup and Checklist; Revision 10
OI 324A4; SBDG 1G-21 System Valve Lineup and Checklist; Revision 13
ACP 1410.9; Locked Valve Program; Revision 7
Integrated Plant Operating Instruction (IPOI) 7; Special Operations; Revision 109

Section 1R05

ACP 1203.53; Fire Protection; Revision 14
AFP 09; Reactor Building Closed Cooling Water Heat Exchanger Area, Equipment Hatch Area, and Jungle Room El. 812’-0”; Revision 27
AFP 21; Turbine Building North Turbine Operating Floor; Revision 24
AFP 22; South Turbine Operating Floor, EL. 780’-0”; Revision 25
AFP 24; Control Building 1A4, 1A3 Essential Switchgear Rooms; Revision 28
AFP 07; Reactor Building Laydown Area, Corridor and Waste Tank Area and Spent Resin Tank Room elevation 786’-0”; Revision 30
AFP 26; Control Building Control Room Complex; Revision 32
AFP 27; Control Building Control Room HVAC Room; Revision 25

Section 1R06

CAP 074522; NCAQ – Lifting of 1MH109 & 2MH207 did not have a critical lift
AR 395197; Water Intrusion of 1MH109

Section 1R11

Evaluation Scenario Guide (ESG) DEP-PI; Revision 0
ESG 2010C-055; Revision 0
ESG 133; Revision 0
ACP 110.1; Conduct of Operations; Revision 24
IPOI 5; Reactor Scram; Revision 53
AOP 901; Earthquake; Revision 3
Emergency Operating Procedure 1; [Reactor Pressure Vessel] Control; Revision 16
Emergency Operating Procedure 2; Primary Containment Control; Revision 15
DAEC Emergency Action Level Notification Form; NOTE 5; Revision 12
Emergency Plan Implementing Procedure (EPIP) 1.1; Determination of Emergency Action Levels; Revision 28
Emergency Action Level Matrix – Hot Modes; Revision 8

Section 1R12

DAEC Performance Criteria Basis Document: Control Building HVAC; Revision 7
System Overview Report for Control Building HVAC; Period 2010-1
WM-AA-1000; Work Activity Risk Management; Revision 6
ACP 109.3; Complex Troubleshooting Process; Revision 2
PI-AA-100-1002; Guideline for Failure Investigation Process; Revision 1
WO A98774; Remove and Inspect Oil Pump due to Copper Concentration in Oil system
Corrective WO A97371; Replace ‘A’ Chiller Compressor with a Rebuilt Compressor
Apparent Cause Evaluation (ACE) 001980; CAQ – ‘A’ Control Building Chiller 1VCH001A Trip
CAP 074880; CAQ – TREND – Multiple Failures of the ‘A’ Chiller Lube Oil Pump
ACE 002049; CAQ – ‘A’ Control Building Chiller Trip
CAP 074863; CAQ – ‘A’ Control Building Chiller Trip
CAP 075070; NCAQ – 1VCH001A, ‘A’ CBC is Potentially Damaged
Equipment-Specific Maintenance Procedure CKTBKR-G080-02; General Electric Company 4160 Volt Circuit Breaker (Magne Blast) Model AM-4.16; Revision 35
WO 1116572; Reactor Recirculation MG Set 1G-201B
WO 1128255; Reactor Recirculation MG Set 1G-201B
WO 1144074; Reactor Recirculation MG Set 1G-201B
CAP 074793; CAQ – Closure of 1A204 When Racking Breaker Up
CAP 074877; CAQ – 1A104 Breaker Racked-In without Extra Precautions
OI 264; Reactor Recirculation System; Revision 116
DAEC Plant Level Performance Criteria Basis Document; Unplanned Capability Loss Factor; Revision 2
DAEC Unplanned Capability Loss Factor Data; August 1997 through February 2010

Section 1R13

OP-AA-102-1003 (DAEC); Guarded Equipment (DAEC Specific Information); Revision 7
WPG-2; On-Line Risk Management Guideline; Revision 36
IPOI 8; Outage and Refueling Operations; Revision 65
DAEC Planned Outage Risk Profile
OM-AA-101-1000; Shutdown Risk Management; Revision 0
OP-AA-102-1003; Guarded Equipment; Revision 0
OP-AA-102-1003 (DAEC); Guarded Equipment (DAEC Specific Information); Revision 1
OP-AA-104-1007; Online Aggregate Risk; Revision 0
DAEC On-line Schedule for Work Week 9020
Maintenance Risk Evaluations for Work Week 9020; Revisions 0, 1, and 2
DAEC On-line Schedule for Work Week 9021
Maintenance Risk Evaluations for Work Week 9021; Revisions 0, and 1
General Maintenance Procedure; GMP-ELEC-20; Main Transformer Backfeed; Revision 10
Shift Logs for 24-27 May 2010
Work Week 9022 Preview
Maintenance Risk Evaluations for Work Week 9022; Revisions 0, 1, 2, 3, and 4
Work Week 9024 Preview; Revision 0
Work Week 9024 Preview; Revision 1
Work Week 9024 Preview; Revision 2

DAEC On-line Schedule for Work Week 9024
DAEC On-line Schedule for Work Week 9026
Maintenance Risk Evaluations for Work Week 9026; Revisions 0

Section 1R15

EN-AA-203-1001; Operability Determinations/ Functionality Assessments; Revision 2
ACP 110.1; Conduct of Operations; Revision 24
OPR 000427; Well Water Leaking into drywell impact on Primary Containment
Adverse Condition Monitoring and Contingency Plan; Increased Drywell Leakage; Revisions 1, 2, 3
CAP 074069; NCAQ – Well water leaking into drywell impact on Primary Containment
CAP 074958; Possible Pipe Support Issue Identified
OPR 000431; Perform Prompt Operability Determination on Pipe Support Issue with ‘B’ Control Building Chiller Water Essential Loop HBD-159
CAP 072404; NCAQ – MSL [Main Steam Line] Steam Leakage Detection Panels 1C193A-D and 194A-D Switch Corrosion
STP 3.3.6.1-05; Main Steam Line Tunnel High Temperature Channel Functional Test; Revision 7
CAP 074937; CAQ – Spurious Group 1 Isolation Signal
CAP 072477; CAQ – Unexpected Half Group 1 Isolation Signal
WO A94796; Determine if TIS4479 was the Cause of the Spurious Half Group 1 Isolation Documented in CAP 0749367
CAP 059248; CAQ – Main Steam Line Tunnel Temperature Greater than Acceptance Criteria in 3.0.0-01
WO 1382725; QEV- 7602B Control Air Quick Exhaust Valve Leaking By
STP 3.6.4.3-05; Standby Gas Treatment Operation with Heaters on (Post Maintenance Only); Revision 4
OI 170; Standby Gas Treatment System; Revision 56
STP 3.7.9-03; CB/ SBGTS Instrument Air Compressors System Leakage and Capacity Test; Revision 4
AR 393381; 1K004 Duty Cycle is 100%
AR 393382; 1K3 Duty Cycle is 73%
BECH-M176<2>; P&ID Reactor Building Ventilation System; Revision 21
BECH-M173; P&ID & Air Flow diagram Standby Filter Unit Control Building; Revision 56
AR 393732; HPCI Room Upper Level Temps Above Calc Assumptions

Section 1R18

OPR 000427; Well Water Leaking into drywell impact on Primary Containment
Adverse Condition Monitoring and Contingency Plan; Increased Drywell Leakage; Revisions 1, 2, and 3
CAP 072511; CAQ – Increase in Unidentified Drywell Leakage Observed After Reactor Core Isolation Cooling STP
CAP 073599; NCAQ – Increase in Unidentified Drywell Leakage
CAP 073913; CAQ - TREND – Drywell Equipment Drain Sump Leakage Increasing
CAP 074440; CAQ – TREND – Increase in Torus Level
STP 3.0.0-01; Instrument Checks; Revision 106
DAEC 50.59 SCRN#1913; STP 3.0.0-01 – PWRs 19842/19826 and STP 3.4.4-01 – PWR 19843
STP 3.4.4-01; Reclassification of Drywell Leakage; Revision 3

CAP 075107; AOP-915 Vulnerability with Starting ESW Pump 1P099B
WO A104152; Emergency Service Water Pump 1P99B

Section 1R19

WO A101998; Replace Oil Pump 1P283A on A CB Chiller
CAP 074880; CAQ –TREND – Multiple Failures of the A Chiller Lube Oil Pump
WO A102198; Replace SV4401 Due to Through Wall Valve Body Nitrogen Leakage
General Maintenance Procedure; GMP-ELEC-08; ESG Grayboot Connections; Section D; Revision 14
STP 3.4.3-03; Manual Opening and Exercising of the ADS and LLS Relief Valves; Revision 8
STP 3.6.2.1-01; Suppression Pool Water Temperature Surveillance; Revision 5
CAP 074859; CAQ – A104176 Identified N2 Leak on PSV4401
ACP 114.9; Event Response Procedure; Revision 19
WO A98773; Compressor 1K032A Check Valve Inspect, Repair, Replace
WO A99479; Perform an Oil System Flush and Inspection Due to High Levels of Copper Found in Oil Sample
BECH-M169<3>; P&ID Control Building Chillers 1VCH001A and 1VCH001B; Revision 12
STP NS540003B; B Emergency Service Water Operability Test and Comprehensive Pump Test; Revision 6
STP 3.3.3.1-09 ESWB; B ESW Valve Position Indicator Verification-Operating; Revision 0
STP 3.3.3.2-07; Remote Shutdown Panel Functional Test for ESW RWS, 1V-AC-11, 1V-SF-21, and Pump Ammeters; Revision 5
WO 1382725; QEV- 7602B Control Air Quick Exhaust Valve Leaking By
STP 3.6.4.3-05; Standby Gas Treatment Operation with Heaters on (Post-Maintenance Only); Revision 4
OI 170; Standby Gas Treatment System; Revision 56
STP 3.7.9-03; CB/ SBGTS Instrument Air Compressors System Leakage and Capacity Test; Revision 4
AR 393381; 1K004 Duty Cycle is 100%
AR 393382; 1K3 Duty Cycle is 73%
BECH-M176<2>; P&ID Reactor Building Ventilation System; Revision 21
BECH-M173; P&ID & Air Flow diagram Standby Filter Unit Control Building; Revision 56
STP 3.8.7-01; Low Pressure Coolant Injection (LPCI) Swing Bus AC and DC Undervoltage Transfer Test; Revision 10
AR 394432; Unexpected Annunciator 1C08A C-8 Instrument AC 1Y11 Undervoltage
AR 394407; Time to Trip at Max Allowed During Swing Bus STP
WO 1148366 (1282866); Tie Bkr from MCC 1B34A/1B44A and 1B3
AR 394218; 1B3401 Cal & Inspect Power Shield Test Failed

Section 1R20

Reactivity Management Plan for Planned Reactor Shutdown for Drywell Leakage; May 2010
Reactivity Management Plan for Reactor Startup from Planned Outage; May 2010
IPOI 2; Startup; Revision 118
ACP 102.17; Pre/Post-Job Briefs and Infrequently Performed Tests and Evolutions; Revision 43
IPOI 1; Startup Checklist; Revision 125
NextEra Energy Duane Arnold, Letter A-290b; Startup Readiness Review Meeting
Outage Management Guideline 8; Outage Close-Out Guideline; Revision 8
CAP 074533; NCAQ – Plant Readiness for Operations Following 5/10 Planned Outage
DAEC Health Physics (HP) Survey 10-575; Drywell 757’

DAEC HP Survey 10-592; Drywell below 826’
DAEC HP Survey 10-581; Drywell below 821’
DAEC HP Survey 10-578; Drywell 805’
DAEC HP Survey 10-566; Drywell 775’
DAEC HP Survey 10-561; Drywell 742’
DAEC HP Survey 10-576; Drywell Under-Vessel Platform
CAP 074912; Drywell MAL [Material Accountability Log] discrepancies at closeout
IPOI 7 Attachment 2; Primary Containment Closeout; Revision 109
OI 149; Residual Heat Removal; Revision 114
CAP 074740; Initiated Manual Reactor Scram Due to Rising Turbine Vibrations;
CAP 074737; Received Main turbine High Vibration Alarm (1C07B-B2) Alarm During Shutdown
IPOI 8; Outage and Refueling Operations; Revision 65
CAP 074856; Issues Identified During the 4/29 Drywell Inspection
IPOI 5; Reactor Scram; Revision 53
IPOI 3; Power Operations (35% - 100% Rated Power); Revision 117
IPOI 7; Special Operations; Revision 109
CAP 074769; NCAQ – Foundation Anchorage for 1E009 Degraded
CAP 074863; A CB Chiller Risk Assessment for Mode 2

Section 1R22

STP 3.3.3.2-09A; Reactor Water Level and Pressure Instruments (Loops A and C) Calibration; Revision 2
STP 3.5.1-02B; B Low Pressure Coolant Injection System Operability Tests; Revision 4
STP NS490003B; B Residual Heat Removal System Leakage Inspection Walkdown; Revision 0
STP 3.8.1-04A; A Standby Diesel Generator Operability Test (Slow Start From Norm Start Air; Revision 5
OI 324A9; SBDG Operating Checklist; Revision 10
OI 324A10; SBDG Standby/Readiness Condition Checklist; Revision 11
STP 3.3.6.1-09; Primary Containment Isolation, Groups 2 and 4, Logic System Functional Test; Revision 12
WO 1362726; MA 4KV Emergency Bus Undervoltage Relay Calibration ‘B’
STP 3.3.8.1-04B; 1A4 4KV Emergency Bus Undervoltage Relay Calibration; Revision 2
STP 3.4.6-01; Reactor Coolant Iodine Activity; Revision 6

Section 1EP6

Emergency Action Level-01; Emergency Action Level Matrix - Modes 1, 2, 3; Revision 8
Emergency Action Level-02; Emergency Action Level Matrix - Modes 4, 5; Revision 7
EPIP 1.2; Notifications; Revision 40
EPIP 6.1; Drill and Exercise Program; Revision 2

Section 2RS08

Shipment RSR 09-13; Control Rod Drive Boxes; May 20, 2009
Shipment RSR 09-29; Dry Active Waste; July 16, 2009
Shipment RSR 09-30; Resin High Integrity Container; September 8, 2009
Shipment RSR 09-31; Resin High Integrity Container; September 14, 2009
Shipment RSR 10-03; Dry Active Waste; February 20, 2010
ACE 001977; Exclusive Use Vehicle Left Site Without the Required Release Survey, May 15, 2010

AR 391373; Crane Control Room Door is Sticking; May 19, 2010
AR 391376; Faded Radioactive Material Tag on Mixed Waste Container; May 19, 2010
AR 391378; Compressed Gas Cylinders Not Stored Properly; May 19, 2010
AR 391547; Housekeeping Issues Identified During NRC Walkdown; May 20, 2010
AR 391548; Procedure Issues Found during NRC Audit of Radiation Protection and Radwaste Shipping; May 20, 2010
CAP 055031; It Appears That Procedures Do Not Fully Address 49 CFR 173.410; January 25, 2008
CAP 057329; Shipping Containers Purchased From Non-QSL Supplier; July 15, 2008
CAP 059191; Materiel Condition Issues from Nuclear Oversight Walkdowns; August 6, 2008
CAP 060735; Condensate Phase Separator tank Line Plugged; October 3, 2008
CAP 061289; Observed Increases in Reactor Water Cleanup Resin Trap Differential Pressure; October 24, 2008
CAP 061451; Criticality Monitoring During New Fuel Receipt; October 31, 2008
CAP 062148; Resin Transfer Line Plugged; December 5, 2008
CAP 062179; Area Radiation Monitor Alarm during Reactor Water Cleanup Backwash; December 8, 2008
CAP 062370; Defective Bolt Closure Points In Outer Nuclear Fuel Box Lids; December 17, 2008
CAP 065797; Radiological Issues Identified during a Walkdown in the Low level Radwaste Building; March 17, 2009
CAP 069233; Exclusive Use Vehicle Left Site Without the Required Release Survey; August 24, 2009
CAP 070305; Low Level Radwaste House Keeping Issue; October 8, 2009
071642; Unexpected Area radiation Monitor Alarm in Waste Collector Tank Room; December 4, 2009
CAP 071722; Unexpected Area radiation Monitor Alarm in Waste Collector Tank Room; December 10, 2009
CAP 072851; High Radwaste Resin Inventory; February 17, 2010
CAP 074652; Discrepancy in Labeling; April 20, 2010
CAP 074976; High Integrity Container Cage Redesign; May 4, 2010
Health Physics Procedure (HPP) 3103.08; Container and Material Labeling; Revision 7
HPP 3107.02; Surveys for Shipment of Radioactive Material; Revision 9
HPP 3107.05; Release of Items from the Radiologically Controlled Area; revision 12
HPP 3111.24; Radiological Posting Associated With a Radwaste High Integrity Container Evolution; Revision 13
PDA08-007; Duane Arnold Energy Center Nuclear Assurance report: Radioactive Waste Control; March 3, 2008
PDA09-008; Duane Arnold Energy Center Nuclear Assurance report: Radioactive Waste Control; April 17, 2009
RCE 1085; Unsecured Locked high Radiation Area-High Integrity Container Cage Gate; Revision 2
Radwaste Handling (RWH) Procedure 3401.7; Controls for Disposal of Irradiated Components; Revision 5
RWH 3404.1; General Requirements for Cask Handling; Revision 23
RWH 3404.2; Compliance for Radiolytic Gas Generation in Radwaste Containers/Casks; Revision 6
RWH 3404.4; Cask Handling Requirement for 14-210L, 14-210H, and 14-215; Revision 15
RWH 3404.9; Cask Handling Requirements Chem-Nuclear Systems (CNS) 14-195H; Revision 0
RWH 3404.10; Cask Handling Requirements for CNS 14-215H; Revision 0
RWH 3404.11; LWT Cask Operating Procedure; Revision 2
RWH 3405.4; Inspection Handling and Control of High Integrity Containers; Revision 21

RWH 3405.6; Inspection, Handling and Control of Sealand Containers; Revision 3
RWH 3406.1; Waste Classification and Characterization; Revision 8
RWH 3406.6; Characterizing Radioactive Material for Transport; Revision 8
RWH 3406.7; Verification of License To receive Radioactive Material; Revision 4
RWH 3406.10; Placarding of Radioactive Material Loads; Revision 7
RWH 3406.11; Guidelines for Blocking and Bracing Radioactive Material Loads; Revision 6
RWH 3406.12; Documentation for Radioactive Material shipments; Revision 10
RWH 3409.2; Sampling Instruction and Analysis of Radwaste Streams; Revision 11
RWH 3410.1; Process Control Program; Revision 16
RWH 3411.1; Inventory and Control of Hazardous and Mixed Waste Containers; Revision 6
RWH 3413.4; Quarterly Inspection of Interim On-site Storage Vaults; Revision 1
RWH 3411.7; Marking and Labeling of Hazardous Waste Containers Prior to Shipment; Revision 4
08-002R; Radiological Engineering Calculation – 10CFR61 Basis for Duane Arnold Energy Center Dry Active Waste; Revision 0
08-003R; Radiological Engineering Calculation – 10CFR61 Basis for Duane Arnold Energy Center Condensate Resin; Revision 0
Technical Basis Document: Surface Contaminate Object Classification and Characterization; Revision 0
Updated Final Safety Analysis Report/Duane Arnold Energy Center, Section 11; Revision 14

<u>Section 4OA1</u>

ACP 1402.4; NRC and WANO Performance Indicator Reporting; Revision 14
LI-AA-204-1001; NRC Performance Indicator Guidelines; Revision 0
NRC PI Data Calculation, Review and Approval Report for RCS Activity; Report Quarter No. 2 Year 2009; dated July 10, 2009
NRC PI Data Calculation, Review and Approval Report for RCS Activity; Report Quarter No. 3 Year 2009; dated October 6, 2009
NRC PI Data Calculation, Review and Approval Report for RCS Activity; Report Quarter No. 4 Year 2009; dated January 13, 2010
NRC PI Data Calculation, Review and Approval Report for RCS Activity; Report Quarter No. 1 Year 2010; dated April 12, 2010
NRC PI Data Calculation, Review and Approval Report for RCS Leakage; Report Quarter No. 2 Year 2009; dated July 6, 2009
NRC PI Data Calculation, Review and Approval Report for RCS Leakage; Report Quarter No. 3 Year 2009; dated October 5, 2009
NRC PI Data Calculation, Review and Approval Report for RCS Leakage; Report Quarter No. 4 Year 2009; dated January 12, 2010
NRC PI Data Calculation, Review and Approval Report for RCS Leakage; Report Quarter No. 1 Year 2010; dated April 7, 2010
NEI 99-02; Regulatory Assessment Performance Indicator Guideline; Revision 6
DAEC First Quarter 2010 PI Summary; Submitted April 21, 2010

<u>Section 4OA2</u>

CAP 074808; CAQ – TREND – 6 Mispositioning Events Have Occurred Since the Beginning of 2010
CAL-M05-004; HPCI Emergency Room Cooler Heat Transfer Calculation; Revision 8
CAL-M06-007; Room Heat Up Analysis for DAEC During Station Blackout; Revision 1

CAL-234-025; Post Design Basis LOCA HPCI Room Temperature After Loss of 1VAC-14A & B; Revision 1
CAL-VC2-001; Ventilating, Cooling, and Heating Criteria & Data; Revision 1
CAL-VC1-009; Preliminary Information for Space Cooling; Revision 0
QUAL-SC100; Environmental Service Conditions Analysis; Revision 9
QUAL-SC101; Environmental and Seismic Service Conditions; Revision 16
DBD-T23-001; Primary Containment System; Revision 4
CAP 032586; Closure of CAP 028523 Without Specified Corrective Actions Being Completed
CAP 069435; NCAQ – Ambient Temperature in the Diesel Generators Rooms is Approximately 110F
ACE 001879; Apparent Cause Evaluation for APRM Upscale Lights; Revision 1
ACP 1410.12; Operator Burden Program; Revision 20
Operator Burden Issues with Resolution Information; dated April 19, 2010
CAP 074309; Well Water Pressure Manipulations to Limit DW Leakage Constitute an OWA
CAP 073012; Conduct a Common Cause Evaluation for Control Room Degraded Instruments
CAP 073001; Ops Burden/ TSA KPI Recovery Strategy

Section 4OA3

CAP 074645; NCAQ – CB5960 Auto-Recloser Cycled Repeatedly onto Faulted DAEC 36kV line
CAP 074642; CAQ – Loss of 1XR1
CAP 074647; CAQ – Security Alert Declared Due to Power Loss. Reference SEL 10-032.
CAP 074737; Received Main turbine High Vibration Alarm (1C07B-B2) Alarm During Shutdown
CE 008175; Initiated Manual Reactor Scram Due to Rising Turbine Vibrations;
IPOI 4; Shutdown; Revision 102
IPOI 5; Reactor Scram; Revision 53
IPOI 5; Attachment 1; Scram Report; Revision 53
RCE 001090; SCAQ [Significant Condition Adverse to Quality] – Initiated Manual Rx Scram Due to Rising Turbine Vibrations
Section 4OA3:
AR 392462; NRC Questions Regarding LER 2010-002: SSFF
RCE 1087; Turbine Bypass Valves Failed Open; Revision 0
CAP 072125; SCAQ – Both Turbine Bypass Valves Failed Open
CAP 072527; SCAQ – Missed Tech Spec Actions during January Bypass Valve Event
CAP 062246; NCAQ – BV1 False Open Signal- Need to restore Reliability & Implement Bridge Strategy
CAP 072313; CAQ – EOC-RPT MCPR Penalty Was Not Installed When Bypass Valves Failed Open
DAEC LER 2010-002-00; Condition Prohibited By Technical Specifications
AR 392462; NRC Questions Regarding LER 2010-002: SSFF
RCE 1087; Turbine Bypass Valves Failed Open; Revision 0
CAP 062246; NCAQ – BV1 False Open Signal- Need to restore Reliability & Implement Bridge Strategy
CAP 072313; CAQ – EOC-RPT MCPR Penalty Was Not Installed When Bypass Valves Failed Open
DAEC LER 2010-002-00; Condition Prohibited By Technical Specifications

Section 4OA7

AR 389822; Incorporate CAP 069233 into 49CFR Subpart H Requalification; October 21, 2009

CAP 069233; Duane Arnold Energy Center Clock Reset; Failure to Survey Vehicle Prior to Release; August 24, 2009
RWH 3406.5; Exclusive Use Vehicle Inspection; Revision 7
DAEC Radwaste Department Instructions: Receiving and Shipping Radioactive Material Shipments; Revision 0

LIST OF ACRONYMS USED

AC	Alternating Current
ACE	Apparent Cause Evaluation
ACP	Administrative Control Procedure
ADAMS	Agencywide Document Access Management System
AFP	Area Fire Plan
ALARA	As-Low-As-is-Reasonably-Achievable
AOP	Abnormal Operating Procedure
AR	Action Request
CAP	Corrective Action Program
CAQ	Condition Adverse to Quality
CBC	Control Building Chiller
CE	Condition Evaluation
CFR	Code of Federal Regulations
CR	Condition Report
DAEC	Duane Arnold Energy Center
DRP	Division of Reactor Projects
EOC-RPT	End-of-Cycle Recirculation Pump Trip
EPIP	Emergency Planning Implementing Procedure
ESG	Evaluated Scenario Guide
ESW	Emergency Service Water
HP	Health Physics
HPCI	High Pressure Coolant Injection
HPP	Health Physics Procedure
HVAC	Heating Ventilation and Air Conditioning
IOD	Immediate Operability Determination
IMC	Inspection Manual Chapter
INPO	Institute of Nuclear Power Operations
IP	Inspection Procedure
IPOI	Integrated Plant Operating Instruction
LER	Licensee Event Report
LLC	Limited Liability Corporation
MCPR	Minimum Critical Power Ratio
NCAQ	Condition not Adverse to Quality
NCV	Non-Cited Violation
NEI	Nuclear Energy Institute
NRC	U.S. Nuclear Regulatory Commission
OI	Operating Instruction
OOS	Out-of-Service
OPR	Operability Recommendation
OSM	Operations Shift Manager
OWA	Operator Workaround
PARS	Publicly Available Records
PCP	Process Control Program
PI	Performance Indicator
PMT	Post-Maintenance Testing
POD	Prompt Operability Determination
QA	Quality Assurance
RCE	Root Cause Evaluation
RCS	Reactor Coolant System

RPS	Reactor Protection System
RWH	Radwaste Handling Procedure
SBDG	Standby Diesel Generator
SBGT	Standby Gas Treatment
SCAQ	Significant Condition Adverse to Quality
SDP	Significance Determination Process
SSC	Structures, Systems, and Components
STP	Surveillance Test Procedure
TBV	Turbine Bypass Valve
TS	Technical Specification
TSO	Transmission System Operator
UFSAR	Updated Final Safety Analysis Report
VAC	Volt Alternating Current
WO	Work Order

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter, its enclosure, and your response (if any) will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records System (PARS) component of NRC's document system (ADAMS). ADAMS is accessible from the NRC Website 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 05000331/2010003
w/Attachment: Supplemental Information

cc w/encl: Distribution via ListServe

DISTRIBUTION:
See next page

DOCUMENT NAME: G:\1-Secy\1-Work In Progress\DUA 2010 003.docm
☒ Publicly Available ☐ Non-Publicly Available ☐ Sensitive ☒ Non-Sensitive
To receive a copy of this document, indicate in the concurrence box "C" = Copy without attach/encl "E" = Copy with attach/encl "N" = No copy

OFFICE	RIII	E	RIII		RIII		RIII	
NAME	KRiemer:cs							
DATE	08/09/10							

OFFICIAL RECORD COPY

Letter to C. Costanzo from K. Riemer dated August 9, 2010

SUBJECT: DUANE ARNOLD ENERGY CENTER INTEGRATED INSPECTION REPORT 05000331/2010003

DISTRIBUTION:
Susan Bagley
RidsNrrDorlLpl3-1 Resource
RidsNrrPMDuaneArnold Resource
RidsNrrDirsIrib Resource
Steven Reynolds
Steven Orth
Jared Heck
Allan Barker
Carole Ariano
Linda Linn
DRPIII
DRSIII
Patricia Buckley
Tammy Tomczak
ROPreports Resource