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

May 14, 2010

EA-09-215

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

SUBJECT: DUANE ARNOLD ENERGY CENTER INTEGRATED INSPECTION REPORT 05000331/2010002

Dear Mr. Costanzo:

On March 31, 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 April 6, 2010, with you and other members of your staff.

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

Based on the results of this inspection, three NRC-identified and one self-revealed findings of very low safety significance were identified. Three of the four findings involved violations of NRC requirements. However, because of their very low safety significance, and because 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.

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 characterization of 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 and its enclosure will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARS) component of NRC's document system (ADAMS). ADAMS is accessible from the NRC 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 05000331/2010002
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:	05000331/2010002
Licensee:	NextEra Energy Duane Arnold, LLC
Facility:	Duane Arnold Energy Center
Location:	Palo, IA
Dates:	January 1 through March 31, 2010
Inspectors:	R. Orlikowski, Senior Resident Inspector R. Murray, Resident Inspector R. Baker, Resident Inspector M. Mitchell, Health Physicist V. Meghani, Reactor Inspector R. Russell, Emergency Preparedness Inspector P. Smagacz, Reactor Engineer
Observers:	None
Approved by:	Kenneth Riemer, Chief Branch 2 Division of Reactor Projects

Enclosure

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SUMMARY OF FINDINGS

IR 05000331/2010002; 01/01/2010 – 03/31/2010; Duane Arnold Energy Center; Outage Activities, Correction of Emergency Preparedness Weaknesses and Deficiencies, Radiological Hazards Assessment and Exposure Control.

This report covers a three-month period of inspection by resident inspectors and announced baseline inspections by regional inspectors. Three Green findings were identified by the inspectors and one Green finding was self-revealed. Three of the four findings 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: Initiating Events

- Green. A finding of very low safety significance and associated NCV of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," was identified by the inspectors for deficiencies in the design documents for the reactor building crane and the special lifting devices. Specifically, the crane bridge girder rails supporting the trolley were not evaluated for the design basis seismic loads. In the reactor vessel head special lifting device calculation, the licensee did not evaluate the hook pins and the calculated safety factors did not meet the design criteria. In the dryer/separator special lifting device calculation, the licensee used incorrect stress allowable values. The licensee documented the condition in their Corrective Action Programs (CAPs) as CAPs 072917, 072568, 072885 and 072880, and initiated actions for calculation revisions and/or modifications.

  The inspectors determined that not evaluating bridge girder rails for seismic loads in accordance with NUREG-0554, not evaluating the hook pins and accepting safety factors not meeting the design criteria and American National Standards Institute (ANSI) Standard N14.6 on the reactor vessel head special lifting device, and the inadequate calculation of safety factors on the dryer/separator special lifting device in accordance with ANSI N14.6 was a performance deficiency. The finding was more than minor because it was associated with the Initiating Events Cornerstone attribute of Equipment Performance and affected the cornerstone objective to limit the likelihood of those events that upset the plant stability and challenge critical safety functions during shutdown as well as power operations. For the item associated with the crane rail, the Region III Senior Risk Analyst (SRA) performed an SDP Phase 3 risk-assessment for estimating the frequency of occurrence of an Operating Basic Earthquake (OBE) or higher seismic event during use of reactor building crane and concluded that the issue was of very low risk significance (Green). For the item associated with the special lifting devices, the inspectors evaluated the finding using IMC 0609.04, "Phase 1 - Initial Screening and Characterization of Findings," and based on a "No" answer to all the questions in the Initiating Events column of Table 4a, as the licensee demonstrated adequate safety factors on all components through subsequent evaluations, determined the finding to be of very low safety-significance (Green). The inspectors did not identify any cross-cutting aspects associated with this finding because, based on the age of the performance deficiencies, it was not reflective of the current licensee performance. (Section 1R20.1.b(1))

Cornerstone: Mitigating Systems

- Green. A finding of very low safety significance was identified by the inspectors for deficiencies in the design documents for failure to translate the lift height assumptions used in drop load evaluations into field instructions in appropriate rigging procedures. Specifically, calculations for accidental drop during handling of the fuel pool area demineralizer shield plug and of the reactor feed pump motor were based on specific lift heights during rigging; however, no field instructions were provided for limiting the rigging to the specified heights. The licensee documented the condition in CAPs 072551 and 072811 and initiated actions for calculation/procedure revisions.

  The inspectors determined that lack of field instructions or procedures restricting the lift heights was inconsistent with the assumptions used in the drop load analyses and was a performance deficiency. The finding was determined to be more than minor because the finding was associated with the Mitigating Systems Cornerstone attribute of Equipment Performance and affected the cornerstone objective to ensure availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences (i.e., core damage). The inspectors evaluated the finding using the SDP in accordance with IMC 0609, “Significance Determination Process,” Attachment 0609.04, “Phase I - Initial Screening and Characterization of Findings,” Table 4a for the Mitigating Systems. Using the screening questions in Table 4a, the inspectors determined that the finding was of very low safety significance because the deficiency did not result in loss of operability or function. This finding has a cross-cutting aspect in the area of Problem Identification and Resolution because the licensee did not perform a thorough evaluation of CAP 053197 in October 2007 which identified that the lift height assumptions used in the calculation for the stud tensioner load drop were not translated into field instructions or procedures [P.1(c)]. (Section 1R20.1.b(2))

Cornerstone: Emergency Preparedness

- Green. A finding of very low safety significance and associated NCV of 10 CFR Part 50, Appendix E, Section IV.F.2.g, and of the emergency planning standard 10 CFR 50.47(b)(14) was identified by the inspectors for the failure of the critique to identify a planning standard weakness. Specifically, during the 2009 Emergency Response Organization (ERO) Training Drill #2 conducted on May 20, 2009, the licensee’s critique process failed to identify a performance problem associated with communications between the Control Room/Simulator (CRS) and the Technical Support Center (TSC) and, as a result, the deficiency was not corrected. The CRS provided inaccurate information necessary for an Emergency Action Level (EAL) classification to the TSC concerning the reactor water level which prompted a controller injection to stop a potential inaccurate classification. The licensee entered the finding into their corrective action program (CAP 068506 and CE 007572).

  The performance deficiency was determined to be more than minor because the deficiency adversely affected the Emergency Preparedness Cornerstone objective to ensure the licensee is capable of implementing adequate measures to protect the health and safety of the public in a radiological emergency, as demonstrated by the ERO performance in a drill. The inspectors used IMC 0609, Appendix B, and determined the deficiency was similar to the Green example of the drill critique process not properly identifying a weakness resulting from a performance problem associated with a risk significant planning standard 10 CFR 50.47(b)(14). Therefore, the finding was screened to be of very low safety significance (Green). The cause of the finding had a cross -cutting component in the problem identification and resolution area of self and independent assessments [P.3(a)]. (Section 1EP5).

Cornerstone: Occupational Radiation Safety

- Green. A self-revealed finding of very low safety significance and associated NCV of Technical Specification (TS) 5.4.1(a) was identified for failure to establish and implement a procedure for performing decontamination activities associated with a potentially significant decontamination activity. The issue resulted in an event where a radworker became internally contaminated. The event was entered in the licensee's CAP. Additionally, the licensee completed a Human Performance Review Worksheet. The licensee also initiated long-term corrective actions including refuel floor procedure augmentations.

  The finding is more than minor because it affected the Occupational Radiation Safety Cornerstone objective to ensure adequate protection of worker health and safety from exposure to radiation and the corresponding attributes associated with the occupational radiation safety program and processes. The finding was determined to be of very low safety significance because it was not an as-low-as-is-reasonably-achievable (ALARA) planning issue, there was no over-exposure or substantial potential for an overexposure, and the licensee's ability to assess worker dose was not compromised. The finding involved a cross-cutting aspect in the area of human performance related to work control in that the licensee did not coordinate work activities by incorporating actions to address keeping personnel apprised of the operational impact on work activities [H.3.(b)]. (Section 2RSO1)

B. Licensee-Identified Violations

No violations of significance were identified.

REPORT DETAILS

Summary of Plant Status

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 January 4, 2010, the plant unexpectedly increased power to 105 percent due to a faulty circuit card causing both Turbine Bypass Valves (TBVs) to reposition from the full closed to full open position. The plant commenced a fast power reduction to 68 percent in accordance with procedures. The TBVs were declared operable on January 5, 2010, and the plant returned to full power on January 6, 2010.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, and Barrier Integrity

1R01 Adverse Weather Protection (71111.01)

.1 External Flooding

a. Inspection Scope

The inspectors evaluated the design, material condition, and procedures for coping with the design basis probable maximum flood. The evaluation included a review to check for deviations from the descriptions provided in the Updated Final Safety Analysis Report (UFSAR) for features intended to mitigate the potential for flooding from external factors. As part of this evaluation, the inspectors checked for obstructions that could prevent draining, checked that the roofs did not contain obvious loose items that could clog drains in the event of heavy precipitation, and determined that barriers required to mitigate the flood were in place and operable. Additionally, the inspectors performed a walkdown of the protected area to identify any modification to the site which would inhibit site drainage during a probable maximum precipitation event or allow water ingress past a barrier. The inspectors also walked down underground bunkers/manholes subject to flooding that contained multiple train or multiple function risk-significant cables. The inspectors also reviewed the abnormal operating procedure (AOP) for mitigating the design basis flood to ensure it could be implemented as written.

This inspection constituted one external flooding sample as defined in Inspection Procedure (IP) 71111.01-05.

b. Findings

No findings of significance were identified.

.2 Readiness for Impending Adverse Weather Condition – Extreme Cold Conditions

a. Inspection Scope

Since extreme cold conditions were forecast in the vicinity of the facility for February 8, 2010, the inspectors reviewed the licensee's overall preparations/protection for the expected weather conditions. On February 9, 2010, the inspectors walked down the Turbine Building Heating, Ventilation and Air Conditioning (HVAC), Reactor Building HVAC and Circulating Water systems because their safety-related functions could be affected or required as a result of the extreme cold conditions forecast for the facility. The inspectors observed insulation, heat trace circuits, space heater operation, and weatherized enclosures to ensure operability of affected systems. The inspectors reviewed licensee procedures and discussed potential compensatory measures with control room personnel. The inspectors focused on plant management's actions for implementing the station's procedures for ensuring adequate personnel for safe plant operation and emergency response would be available. Specific documents reviewed during this inspection are listed in the Attachment to this report.

This inspection constituted one readiness for impending adverse weather condition 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:

- 'A' Emergency Service Water (ESW) system with the 'B' Standby Diesel Generator (SBDG) out-of-service (OOS) for planned maintenance;
- Reactor Core Isolation Cooling (RCIC) with High Pressure Coolant Injection (HPCI) OOS for planned maintenance; and
- 'B' ESW system with the 'A' SBDG OOS for planned maintenance.

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 corrective action program (CAP) with the appropriate significance characterization. Documents reviewed are listed in the Attachment to this report.

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

b. Findings

No findings of significance were identified.

1R05 Fire Protection (71111.05)

.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) 08; Standby Gas Treatment and Motor Generator (MG) Set Rooms;
- AFP 18, 19, and 20; Turbine Building North and South Ground Floor, Tube Pulling Area, Aux Boiler Room; Emergency Diesel Generator (EDG) and Day Tank Rooms;
- AFP 31 and 32; Intake Structure Pump Rooms and Traveling Screen Areas;
- AFP 01 and 02; Torus Area and North and South Corner Rooms; and
- AFP 03; HPCI, RCIC and Radwaste Tank Rooms.

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 out-of-service, 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 activities constituted five quarterly fire protection inspection samples as defined in IP 71111.05-05.

b. Findings

No findings of significance were identified.

1R06 Flooding (71111.06)

.1 Internal Flooding

a. Inspection Scope

The inspectors reviewed selected risk-important plant design features and licensee procedures intended to protect the plant and its safety-related equipment from internal flooding events. The inspectors reviewed flood analyses and design documents, including the UFSAR, engineering calculations, and abnormal operating procedures to identify licensee commitments. The specific documents reviewed are listed in the Attachment to this report. In addition, the inspectors reviewed licensee drawings to identify areas and equipment that may be affected by internal flooding caused by the failure or misalignment of nearby sources of water, such as the fire suppression or the circulating water systems. The inspectors also reviewed the licensee's corrective action documents with respect to past flood-related items identified in the CAP to verify the adequacy of the corrective actions. The inspectors performed a walkdown of the following plant area(s) to assess the adequacy of watertight doors and verify drains and sumps were clear of debris and were operable, and that the licensee complied with its commitments:

- HPCI and RCIC Rooms.

This inspection constituted one internal flooding sample as defined in IP 71111.06-05.

b. Findings

No findings of significance were identified.

1R07 Heat Sink Performance (71111.07)

.1 Annual Review of Heat Sink Performance

a. Inspection Scope

The inspectors reviewed the licensee's testing of the 'B' SBDG heat exchangers to verify that potential deficiencies did not mask the licensee's ability to detect degraded performance, to identify any common cause issues that had the potential to increase risk, and to ensure that the licensee was adequately addressing problems that could result in initiating events that would cause an increase in risk. The inspectors reviewed the licensee's observations as compared against acceptance criteria, the correlation of scheduled testing and the frequency of testing, and the impact of instrument inaccuracies on test results. Inspectors also verified that test acceptance criteria considered differences between test conditions, design conditions, and testing conditions. Documents reviewed for this inspection are listed in the Attachment to this report.

This annual heat sink performance inspection constituted one sample as defined in IP 71111.07-05.

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 February 16 and March 9, 2010, the inspectors observed a crew 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 crew's performance in these areas was compared to pre-established operator action expectations and successful critical task completion requirements. Documents reviewed are listed in the Attachment to this report.

This inspection 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:

- Main Steam Line Turbine Building Steam Leak Detection High Temperature Indicating Switch TIS-4478 High Resistance; and
- Primary Containment Ventilation System.

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.

This inspection constituted two 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)

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:

- Emergent work on the Electro-Hydraulic Control System during Work Week 9002;
- Emergent Work due to Increase in Unidentified Drywell Leakage during Work Weeks 9005 and 9006;
- Planned 2 Year Maintenance Inspection of the B SBDG during Work Week 9006;
- Downpower for Rod Sequence Exchange during Work Week 9012 and 9013; and
- Work Week 9013 Risk Assessment.

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.

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

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations (71111.15)

a. Inspection Scope

The inspectors reviewed the following issues:

- Operability of Bypass Valves BPV-1 and BPV-2;
- Operability Evaluation (OPR) 000419, Standby Liquid Control (SBLC) Pipe Support Discrepancy;
- ‘B’ SBDG Jacket Water Cooling Heat Exchanger;
- OPR 000422, Structural Bolting Questions in Northwest Corner Room; and
- Operability of Primary Containment Isolations for the Well Water System.

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.

This operability inspection constituted five samples as defined in IP 71111.15-05.

b. Findings

No findings of significance were identified.

1R18 Plant Modifications (71111.18)

.1 Temporary Plant Modifications

a. Inspection Scope

The inspectors reviewed the following temporary modification:

- Jumper Installation and Removal for Electro-Hydraulic Control A61 Card Replacement.

The inspectors compared the temporary configuration changes and associated 10 CFR 50.59 screening and evaluation information against the design basis, the UFSAR, and the TS, as applicable, to verify that the modification did not affect the operability or availability of the affected system. The inspectors also compared the licensee's information to operating experience information to ensure that lessons learned from other utilities had been incorporated into the licensee's decision to implement the temporary modification. The inspectors, as applicable, performed field verifications to ensure that the modifications were installed as directed; the modifications operated as expected; modification testing adequately demonstrated continued system operability, availability, and reliability; and that operation of the modifications did not impact the operability of any interfacing systems. Lastly, the inspectors discussed the temporary modification with operations, engineering, and training personnel to ensure that the individuals were aware of how extended operation with the temporary modification in place could impact overall plant performance. Documents reviewed in the course of this inspection are listed in the Attachment to this report.

This inspection constituted one temporary modification sample as defined in IP 71111.18-05.

b. Findings

No findings of significance were identified.

1R19 Post-Maintenance Testing (71111.19)

a. Inspection Scope

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

- 'B' SBDG Operability Test Following 2 Year Maintenance Inspection;
- HPCI Operability Test Following Maintenance Outage; and
- 'B' SBLC Operability Following Maintenance.

These activities were selected based upon the structure, system, or component'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.

This inspection constituted three post-maintenance testing samples as defined in IP 71111.19-05.

b. Findings

No findings of significance were identified.

1R20 Outage Activities (71111.20)

.1 Refueling Outage Activities-Crane and Heavy Lifts Inspection (OpESS FY 2007-03)

a. Inspection Scope

During the period from January 19, 2010, through February 5, 2010, the inspectors performed a review of the licensee's control of heavy loads program in accordance with the NRC's Operating Experience Smart Sample (OpESS) FY 2007-03, Revision 2, "Crane And Heavy Lift Inspection, Supplemental Guidance for IP-71111.20."
The inspection included the activities listed below. Documents reviewed during the inspection are listed in the Attachment to this report.

- Reviewed licensee's submittals and commitments related to Generic Letters 80-113 and 81-07, "Control of Heavy Loads";
- Reviewed documents supporting the reactor building crane upgrade to single failure proof;
- Reviewed licensee's preventive maintenance program and the vendor recommendations for the preventive maintenance;
- Reviewed recent crane inspection records;
- Reviewed reactor disassembly and a sampling of other procedures for consistency with commitments;
- Reviewed calculations and inspection/testing for the reactor vessel head lifting device and the dryer/separator lifting device for conformance with the applicable requirements/standards; and
- Reviewed a sample of drop load calculations to verify conformance with the heavy loads procedures.

b. Findings

(1) Inadequate Evaluations for Crane and Special Lifting Devices

Introduction: A finding of very low safety-significance and associated NCV of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," was identified by the inspectors for deficiencies in the design documents for the reactor building crane and the special lifting devices. Specifically, the crane bridge girder rails were not evaluated correctly for seismic loads. In addition, in the reactor vessel head special lifting device calculation, the hook pins were not evaluated and safety factors not meeting the acceptance criteria were accepted, while in the dryer/separator special lifting device calculation, safety factors against shear failure of steel members were not calculated correctly.

Description: The reactor building crane is a Seismic Category I structure described in Section 3.8.4 of the UFSAR. The licensee upgraded the crane to meet the requirements of NUREG-0554, "Single Failure Proof Cranes for Nuclear Power Plants," to satisfy the commitments during the Phase I review of their Control of Heavy Loads program by the NRC. Special lifting devices for the reactor vessel head and dryer/separator are also Seismic Category I equipment subject to the 10 CFR Part 50, Appendix B quality assurance requirements as described in Table 3.2-1 of the UFSAR, and per Section 9.1.4.4 of the UFSAR, they are designed to meet the requirements of ANSI Standard N14.6-1978.

The following deficiencies were identified by the inspectors:

- The licensee did not evaluate the reactor building bridge girder rails supporting the trolley for the design basis seismic loads. The licensee upgraded the crane to meet the NUREG-0554, "Single Failure Proof Cranes for Nuclear Power Plants," by installing a new hoist and trolley system purchased from Ederer in 1985. The new safety-related Ederer System was evaluated in an NRC Safety Evaluation Report of the Generic Licensing Topical Report EDR-1, Revision 3. The site-specific seismic calculation for the trolley was not included in the topical report reviewed by the NRC staff and was not provided to the licensee by the vendor. The licensee seismic evaluations for the bridge girders and the crane support structure documented in calculations CAL-M01-273 and CAL-C01-002 were reviewed by the staff in 2003 as part of a license amendment. In response to questions from the inspectors, the licensee obtained a copy of the trolley seismic calculation from the vendor. After review of the trolley calculation and the bridge girder calculation, the inspectors identified that while the trolley transferred the lateral seismic loads to the bridge girders through the rails, the rails were not evaluated for such loads. The licensee documented the deficiency in their corrective action program as CAP 72917.

- In order to comply with NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants," Section 5.1.6, the licensee upgraded special lifting devices used for rigging the vessel head and the dryer/separator to the design requirements of ANSI Standard N14.6-1978, Section 6, "Special Lifting Devices for Critical Loads."

- The inspectors identified that in calculation 273C036 for the vessel head lifting device, the hook pins were not evaluated. The pins are critical components providing a connection between the lifting device and the reactor building crane hook for load transfer to the crane. In addition, safety factors of 9.76 and 9.55 for stress in some of the welds and for bearing stress at the hook pins, were accepted by the licensee while a safety factor of 10 was required based on the design criteria stated in the calculation, as well as per the ANSI Standard N14.6. The licensee documented the calculation deficiencies and the need for calculation revisions in their corrective action program as CAP 072568 and CAP 072885. Subsequently, the licensee performed a supplemental evaluation for more accurate determination of the safety factors, which indicated that the pins would meet the design requirements and that the actual safety factors would be equal to or greater than 10.

- The inspectors also identified that in the calculation, CAL-M07-047, for the dryer/separator lifting device that was performed to demonstrate compliance with ANSI Standard N14.6, the safety factors against shear failure of steel members were not calculated correctly due to use of incorrect shear allowable values for steel members. The licensee documented the deficiency and the need for calculation revisions and/or modifications in their corrective action program as CAP 72880. Subsequently, the licensee performed additional evaluations with the correct allowable values and with use of the certified material test reports for the hook box steel plates to conclude that adequate safety factors as required by ANSI N14.6 were available and there were no operability concerns.

Analysis: The inspectors determined that not evaluating bridge girder rails for seismic loads in accordance with NUREG-0554, not evaluating the hook pins and accepting safety factors not meeting the design criteria stated in the calculation and the ANSI Standard N14.6 on the reactor vessel head special lifting device, and the inadequate calculation of safety factors on the dryer/separator special lifting device in accordance with ANSI Standard N14.6 was a performance deficiency.

The finding was determined to be more than minor because the finding was associated with the Initiating Events Cornerstone attribute of Equipment Performance and affected the cornerstone objective to limit the likelihood of those events that upset the plant stability and challenge critical safety functions during shutdown, as well as power operations. Specifically, the purpose of the crane and the lifting device design requirements is to limit the likelihood of a component failure in order to ensure safe handling of heavy loads over the reactor core, the spent fuel pool, or safety-related components.

For the item associated with the crane rails, the inspectors performed a Phase I SDP review of this finding using the guidance provided in IMC 0609, Attachment 0609.04, "Phase 1 -- Initial Screening and Characterization of Findings." In accordance with Table 4b, "Seismic, Flooding, or Severe Weather Screening Criteria," the finding screened as potentially risk-significant due to external initiating event core damage sequences. Therefore, the Region III SRA performed an SDP Phase 3 risk-assessment of this performance deficiency. The inspectors determined that only seismic events exceeding the level of an OBE of 0.06g could impact core damage frequency. The SRA used the Risk-Assessment Standardization Project Handbook and estimated the frequency of seismic events for Duane Arnold exceeding this g-level to be 1.3E-4/yr. Assuming the average duration of a crane lift was 30 minutes, the SRA estimated that there would need to be over 100 lifts performed each year with the reactor building crane over critical locations to approach a threshold where this issue could become risk-significant just based on initiating event frequency. The SRA concluded that the risk of a simultaneous occurrence of an OBE or greater magnitude seismic event during such use of the reactor building crane was of very low safety-significance (Green).

For items associated with the lifting devices, 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 Initiating Events Cornerstone. The inspectors answered “no” to all the questions in the Initiating Events column based on the licensee demonstrating adequate safety factors on all components through subsequent evaluations and concluded that the finding was of very low safety-significance (Green).

The inspectors did not identify a cross-cutting aspect associated with this finding because, based on the age of the performance deficiencies, it was not reflective of the current licensee performance.

Enforcement: Title 10 CFR Part 50, Appendix B, Criterion III, “Design Control,” requires, in part, that design control measures shall provide for verifying or checking the adequacy of design, such as by the performance of design reviews, by the use of alternate or simplified calculational methods, or by the performance of a suitable testing program.

Contrary to the above, the licensee did not adequately verify or check the adequacy of the design as stated. During the crane upgrade in 1985 and in the reactor building crane girder calculations, CAL-M01-273, Revision 2, dated June 16, 2003, the licensee failed to evaluate the reactor building bridge girder rails for the design basis seismic loads. Specifically, the licensee calculations evaluated the trolley and the girders for seismic loads but did not evaluate the rails that transfer the lateral seismic loads from the trolley to the bridge girders. Also, in reactor vessel head lifting device calculation, 273C036, dated September 13, 1983, and in the dryer/separator lifting device calculation, CAL-M09-047, performed on November 26, 1997, the licensee failed to verify adequacy of all structural components. Specifically, in the vessel head lifting device calculation the licensee failed to evaluate the hook pins and, for some of the components, accepted safety factors that did not meet the design criteria stated in the calculation. In the calculation for the dryer/separator lifting device, safety factors were not adequately calculated due to the use of incorrect shear allowable values for steel members. Because this violation was of very low safety-significance and it was entered into the licensee’s corrective action program as CAPs 072917, 072568, 072885, and 072880, this violation is being treated as a NCV, consistent with Section VI.A.1 of the NRC Enforcement Policy (NCV 05000331/2010002-01, Inadequate Evaluations for Crane and Special Lifting Devices).

(2) Lift Height Assumptions in Drop Load Analyses Not Reflected in Rigging Procedures

Introduction: A finding of very low safety-significance was identified by the inspectors for a failure to translate the lift height assumptions used in drop load evaluations into field instructions in appropriate rigging procedures.

Description: As part of the control of heavy loads program and determination of safe load paths for rigging, the licensee identified locations of safe shutdown equipment and performed drop load analyses for certain areas to ensure that safe shutdown equipment would not be affected by accidental drops. In calculation 273-33, Revision 1, the licensee evaluation concluded that an accidental drop of the fuel pool area demineralizer shield plug during handling will not damage the reactor building floor slab at elevation 812'-0". The evaluation was based on an assumption that the shield plug will not be lifted more than a foot above the floor at elevation 833'-6". The inspectors found that the licensee did not have any field instructions or procedures in place to limit the lift height to one foot above the floor. In calculation 273-17, Revision 1, the licensee evaluated an accidental drop of a reactor feed pump motor during rigging through the turbine building hatch on the mat foundation. The evaluation was based on an assumption that the equipment will not be lifted more than four feet above the operating floor level. The inspectors found that the licensee did not have any field instructions or procedures in place to limit the lift height to four feet above the floor. The inspectors also noted that CAP 053197 issued on October 16, 2007, documented a similar deficiency where the lift height limit assumed in the reactor vessel head stud tensioner load drop analysis was not reflected in field procedures or instructions. The inspectors concluded that a thorough extent of condition review for CAP 053197 by the licensee would have identified the additional deficiencies noted above.

The licensee documented the deficiencies and the need for calculation/procedure revisions in their corrective action program as CAP 072551 and CAP 072811. The licensee reviewed the work requests last used for the above riggings and concluded that the lift heights assumed in the calculations could have been exceeded. The licensee performed additional evaluations using greater lift heights based on feasibility in field and determined that, using the same methodology as used in the existing calculations, the results would be acceptable.

Analysis: The inspectors determined that the lack of field instructions or procedures restricting the lift heights was contrary to the assumptions used in the drop load analyses and was a performance deficiency.

The finding was determined to be more than minor because the finding was associated with the Mitigating Systems Cornerstone attribute of Equipment Performance and affected the cornerstone objective to ensure availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences (i.e., core damage). Specifically, damage from a load drop accident could impact availability or reliability of safe shutdown equipment.

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 Mitigating Systems. A drop load event could adversely impact the safety-related function of the affected equipment. Using the screening questions in Table 4a, the inspectors determined that the deficiency did not result in a loss of operability or function, and, therefore, the finding was of very low safety-significance (Green).

This finding has a cross-cutting aspect in the area of Problem Identification and Resolution, associated with the corrective action program component, because the licensee did not perform a thorough evaluation of CAP 053197 in October 2007 identifying that the lift height assumptions used in the calculation for the stud tensioner load drop was not translated into field instructions or procedures. Specifically, during the CAP 053197 evaluation, the licensee failed to adequately address the extent of condition by not finding similar problems with the drop load calculations described here [P.1(c)].

Enforcement: No violation of regulatory requirements was identified as the inspectors could not relate the finding to a safety-related activity or a procedure subject to the 10 CFR Part 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Processing Plants." Although safe-shutdown equipment could be within the confines of the load path, the lifting equipment used was not safety-related (FIN 05000331/2010002-02, Lift Height Assumptions in Drop Load Analyses Not Reflected in Rigging Procedures).

1R22 Surveillance Testing (71111.22)

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:

- 'A' SBDG Operability Test Using Normal Starting Air (Routine);
- Technical Support Center Diesel Run (Routine);
- HPCI Steam Supply Pressure Low Functional (Routine);
- 'A' Core Spray Operability (IST); and
- Reactor Coolant System (RCS) Leakage Detection Reclassification (RCS Leakage Detection).

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.

This inspection constituted three routine surveillance testing samples, one inservice testing sample, and one reactor coolant system leak detection inspection sample as defined in IP 71111.22, Sections -02 and -05.

b. Findings

No findings of significance were identified.

Cornerstone: Emergency Preparedness

1EP5 Correction of Emergency Preparedness Weaknesses and Deficiencies (71114.05)

.1 (Closed) Unresolved Item (URI) 05000331/200904-03: Adequacy of the Licensee's Critique for the May 20, 2009, EP Drill

a. Inspection Scope

The inspectors reviewed additional information concerning the Unresolved Item 05000331/2009004-03 opened during the 2009 biennial emergency preparedness program inspection. The inspectors reviewed the final drill report from the 2009 ERO Training Drill #2, the drill scenario, the drill objectives and evaluation criteria, the actual timeline from the drill, corrective actions resulting from the drill, and the licensee's assessment for the performance indicator (PI) for drill and exercise performance (DEP). The inspectors reviewed the controllers' and evaluators' logs and observation statements along with the Senior Resident Inspector's observations to determine the circumstances and sequence of events for the drill. The inspectors reviewed the EALs and other licensee procedures. Documents reviewed are listed in the Attachment.

b. Findings

(1) Inadequate Critique for the 2009 ERO Training Drill #2

Introduction: A finding of very low safety significance and associated NCV of 10 CFR Part 50, Appendix E, Section IV.F.2.g and of the emergency planning standard

10 CFR 50.47(b)(14) was identified by the inspectors for the failure of the emergency preparedness drill critique to identify a planning standard weakness. Specifically, during the 2009 ERO Training Drill #2 conducted on May 20, 2009, the licensee's critique process failed to identify a performance problem associated with communications between the CRS and the TSC and, as a result, was not corrected. The CRS provided inaccurate information necessary for an EAL classification to the TSC concerning the reactor water level, which prompted a controller injection to stop a potential inaccurate classification.

Description: On May 20, 2009, the licensee conducted a training drill that involved the CRS, the TSC, the Operational Support Center (OSC), the Emergency Operations Facility (EOF), and the Joint Information Center (JIC). Additionally, Benton County, Linn County, and Iowa Homeland Security participated in the drill. The drill began just after 8:00 a.m.

During the drill, a simulated storm caused a loss of offsite power to both station transformers. The standby diesel generator automatically started in the scenario and supplied power to the station. The TSC properly classified and declared an Alert based upon the plant conditions. As the scenario continued, the simulator had the standby diesel generator fail, which resulted in a loss of both offsite and onsite power.
The facility lead in the TSC began to evaluate the requirements for declaring a Site Area Emergency based on loss of all offsite power and onsite power.

While the TSC was evaluating the EAL conditions, an entry was made on the Electronic Status Board (ESB) stating that reactor water level was +15 inches. The ESB was viewed in all the emergency response facilities. The entry in the log was made in the CRS after an operator read the simulated RPV level using the wide-range Yarway level indication. When the operator reported the RPV level was +15 inches to the Shift Manager, the Shift Manager identified that the reported value was incorrect because the reactor coolant was greater than 250°F, thus, the wide-range Yarway instrument was not reading the simulated reactor water level accurately. Although the Shift Manager identified the RPV level was not +15 inches, the ESB entry was not corrected and no additional ESB entry was made to notify the other organizations outside of the simulator of the error on the ESB. The information on the ESB was viewed at the TSC and EOF.

Based upon the erroneous ESB entry, the decision makers in the TSC and EOF began to look at requirements for declaring a General Emergency (GE) based on loss of all offsite and onsite power and RPV level was less than +15 inches. The Lead Controller in the TSC monitoring the progress of the drill recognized the Emergency Coordinator (EC) was evaluating whether to declare a GE. The controller anticipated the early GE declaration would cause a deviation from the scenario and would affect drill play and offsite demonstration of tasks. The Lead Controller interjected and stated the ESB entry of the RPV level was incorrect. He provided the correct level to the TSC. The lead controller stated the interjection was intended to provide the correct information and maintain the organization on-track with expected actions. After the controller interjected, the EC and other members of the TSC determined the proper classification based on the corrected RPV level.

On May 21, 2009, the lead controllers from each facility performed a critique of the previous day's drill and discussed the controller interjection that occurred. Corrective Action Program 067417 requested a condition evaluation (CE) to evaluate the appropriateness of the controller interject (CE 007464). The CE was completed on June 25, 2009. As part of the CE process, controllers and players from the TSC, CRS, and the EOF were questioned to attain information concerning the drill occurrences. Additional action requests and evaluations were generated to further examine various drill aspects related to the controller intervening and the DEP PI determination. Also, the condition evaluation (CE 007572) resulting from the action request (CAP 068506) was conducted. The condition evaluation and corrective actions did not address the CRS incorrect RPV level communications on the status board that resulted in a controller intervening.

Analysis: The inspectors determined that the licensee's failure to perform an adequate critique of the ERO performance for the 2009 ERO Training Drill #2 conducted on May 20, 2009, to be a performance deficiency. Specifically, the licensee's critique process failed to identify a performance weakness related to the improper reading of the RPV level, the incorrect entry on the ESB and the failure of the CRS to correct the entry, and the use of the incorrect information in the TSC for event classification. The licensee failed to fully evaluate the non-risk significant planning standard weakness related to ineffective communications of plant conditions and the negative effects on the risk-significant planning standard of accident assessment that prompted a controller to intervene and provide correct information. As a result of the failure to perform an adequate and thorough critique, the deficiency was not corrected.

The failure to perform an adequate critique did not meet the criteria for traditional enforcement and was screened using the Emergency Preparedness SDP Appendix. The performance deficiency was determined to be more than minor because the deficiency adversely affected the cornerstone objective to ensure the licensee is capable of implementing adequate measures to protect the health and safety of the public in a radiological emergency as demonstrated by the ERO performance in a drill and was associated with the attribute of the evaluation and correction of deficiencies.

The inspectors evaluated the finding using IMC 0609, Appendix B, and found the deficiency to be similar to the Green finding example of the drill critique process that not properly identifying a weakness resulting from a performance problem associated with risk-significant planning standard in 10 CFR 50.47(b)(14). Specifically, the critique process did not identify the weakness related to the inaccurate communications of the reactor water level to the decision makers in the TSC and the resulting performance problem during the accident assessment in which a controller was prompted to interject. Also, using the SDP, Appendix B, Sheet 1, "Failure to Comply" flowchart, the performance deficiency was evaluated to be a planning standard degraded function and was screened to be of very low safety significance (Green).

The licensee's failure to perform an adequate critique had a cross-cutting component in the Problem Identification and Resolution area of Self and Independent Assessments. The licensee did not conduct the self-assessment and drill critique in sufficient depth to evaluate the effects of the communications of the inaccurate reactor conditions which prompted a controller injection to stop a potential inaccurate classification [P.3(a)].)).

Enforcement: Title 10 CFR 50.47(b)(14) requires, in part, that periodic exercises will be conducted to evaluate major portions of emergency response capabilities and that the deficiencies identified as a result of these exercises or drills will be corrected. In addition, 10 CFR Part 50, Appendix E, Section IV.F.2.g., states, in part, that all training, including exercises, shall provide for formal critiques in order to identify weak or deficient areas that need correction. Contrary to the above requirements, between May 21, 2009, and June 25, 2009, the licensee failed to identify and correct a weakness associated with the 2009 ERO Training Drill #2 conducted on May 20, 2009. Because the finding is of very low safety significance and has been entered into the licensee's corrective action program (CAP 068506 and CE 007572), the violation is being treated as a NCV, in accordance with Section VI.A.1 of the NRC Enforcement Policy (NCV 05000331/2010002-03, Failure to Conduct an Adequate Critique for the May 20, 2009, Drill).

The correction of emergency preparedness weaknesses and deficiencies inspection does not constitute a sample as defined in IP 71114.05-05.

1EP6 Drill Evaluation (71114.06)

.1 Emergency Preparedness Drill Observation

a. Inspection Scope

The inspectors evaluated the conduct of a routine licensee emergency drill on January 27, 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 Emergency Operating Facility to determine whether the event classification, notifications, and protective action recommendations were performed in accordance with procedures. The inspectors also attended the licensee drill critique to compare any inspector-observed weakness with those identified by the licensee staff in order to evaluate the critique 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 package and other documents listed in the Attachment to this report.

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

b. Findings

No findings of significance were identified.

2. RADIATION SAFETY

Cornerstone: Occupational Radiation Safety

2RS01 Radiological Hazards Assessment and Exposure Control (71124.01)

.1 Follow-up of Contamination Event during Refueling Outage 21 (02.01)

a. Inspection Scope

The inspectors reviewed radiological control problems associated with RPV stud decontamination activities on the refuel floor during Refueling Outage 21 (RFO-21) that resulted in internal contamination of an individual.

b. Findings

Introduction: A self-revealed finding of very low safety-significance and an associated NCV of TS 5.4.1(a) was identified for the failure to establish and implement a procedure for performing decontamination activities associated with a potentially significant decontamination activity. This failure resulted in a radiological material intake event on February 6, 2009.

Description: On February 6, 2009, during RFO-21, a radworker was assigned to decontaminate RPV nuts and bolts. The decontamination activity was to take place in a specific work area on the refuel floor. The licensee had designated the work area as a contaminated area (roped off and posted).

Prior to the start of the decontamination activity, a health physics technologist (HPT) briefed the radworker on the radiological conditions in the area and instructed the individual to use a pre-staged five-gallon bucket that was fitted with a HEPA hose at the bottom. This configuration was designed to minimize the spread of contamination during the evolution and was required when power tools were used during the decontamination process. Additionally, the HPT also indicated that the area contained a table that was used for decontaminating items by hand.

Following the briefing, the radworker entered the work area and commenced his work activity. During a work break, the worker attempted to exit the refuel floor “decon” area. The individual passed the refuel floor personnel contamination monitor. However, the worker was unable to pass the radiologically controlled areas access control contamination monitor, the second personnel contamination monitor, because of a detectable facial contamination. The licensee took nasal smears from the worker. An analysis of the smears indicated the presence of manganese-54 (Mn-54) and cobalt-60 (Co-60). A whole body count of the worker showed Co-60 activity. After several showers, the licensee sent the worker home. Prior to being sent home, the licensee instructed the worker to return the following day for another whole body count. The following day, a whole body count did not detect internal contamination, and the individual was assigned a dose of 0.87 millirem from an ingested intake of approximately 0.017 percent Annual Level of Intake (ALI).

The licensee entered the event in the corrective action program as CAP 063690. Additionally, the licensee performed an apparent cause evaluation (ACE) 001923, “nasal contamination received while cleaning RPV bolts.” The ACE determined that the internal contamination event was caused by the licensee’s less-than-adequate task description for the worker working on highly contaminated equipment. This task was left up to the HPT briefing without any additional guidance. The second contributing factor was, according to the licensee, the HEPA hose was found disconnected from the five-gallon bucket setup, and, therefore, failed to remove any airborne contamination being generated away from the worker. Additionally, it was unclear whether the radworker knew the consequence of having the hose disconnected from the bucket. The connection of the HEPA hose to the bucket was relied upon as the sole barrier between the activity and the generation of airborne contamination and subsequent ingestion.

During an interview conducted by the licensee, the radworker indicated that the preferred method (five-gallon plastic bucket with 2000 cfm HEPA hose attached on the bottom for a negative pressure) was used initially to clean RPV nuts and bolts. Consequently, the method, as the worker understood, was too difficult and presented a safety concern. After trying this method, the worker deviated from it. The worker used Scotch Brite pads to scrub the RPV nuts and washers outside the five-gallon bucket on a flat table at the decon area instead.

The licensee's management instructed the individual on the importance of refuel floor personnel adhering to the specific refuel floor procedures and radiation work permit requirements. The licensee also initiated long-term corrective actions including refuel floor procedure augmentations.

<u>Analysis</u>: The inspectors determined that the issue was a performance deficiency because the licensee failed to provide written instructions regarding the proper technique to use while performing an activity that could change radiological conditions. The inspectors determined that the cause of the performance deficiency was reasonably within the licensee's ability to foresee and correct and should have been prevented. The finding was not subject to traditional enforcement since the issue did not have an actual or potentially safety-significant consequence, nor did it impact the NRC's ability to perform its regulatory function, and was not willful.

In accordance with IMC 0612, the inspectors determined the finding is more than minor because it affected the Occupational Radiation Safety Cornerstone objective to ensure adequate protection of worker health and safety from exposure to radiation and the corresponding attributes associated with the occupational radiation safety program and processes. Specifically, the individual did not fully understand the process preferred by the licensee, which resulted in an unplanned intake of radioactive material. The finding was assessed using the Occupational Radiation Safety-Significance Determination Process and was determined to be of very low safety-significance because it was not an ALARA planning issue, there was no over-exposure or substantial potential for an over-exposure, and the licensee's ability to assess worker dose was not compromised.

The finding involved a cross-cutting aspect in the area of human performance related to work control in that the licensee did not coordinate work activities by incorporating actions to address keeping personnel apprised of the operational impact on work activities [H.3.b].

<u>Enforcement</u>: Duane Arnold Energy Center Technical Specification 5.4.1 requires, in part, that written procedures be established, implemented, and maintained covering the applicable procedures recommended in Regulatory Guide 1.33, Revision 2, Appendix A, February 1978.

Section 7.e (4) of Appendix A to Regulatory Guide1.33, "Quality Assurance Program Requirements (Operation)," Revision 2, February 1978, provides, in part, that the licensee establish written procedures for contamination control.

Contrary to the above, on February 6, 2009, the licensee failed to establish and implement a written procedure for an activity recommended in Regulatory Guide 1.33. Specifically, an individual was assigned to perform decontamination activities of highly contaminated components (reactor pressure vessel nuts and studs) without a written procedure or guidance. Consequently, the individual did not fully understand the process preferred by the licensee which resulted in an unplanned intake of radioactive material. Since the failure to comply with the TS was of very low safety-significance, corrective actions were taken, and the issue was entered into the licensee's corrective action program as ACE 001923, the violation is treated as an NCV, consistent with Section VI.A of the NRC Enforcement Policy (NCV 05000331/2010002-04-03, An Internal Contamination Occurred while Cleaning RPV Studs and Washers on the Refuel Floor at Duane Arnold).

2RS06 Radioactive Gaseous and Liquid Effluent Treatment (71124.06)

.1 Inspection Planning and Program Reviews (02.01)

Event Report and Effluent Report Reviews

a. Inspection Scope

The inspectors reviewed the Radiological Effluent Release Reports issued since the last inspection to determine if the reports were submitted as required by the Offsite Dose Calculation Manual/Technical Specification (ODCM/TS). The inspectors reviewed anomalous results, unexpected trends, or abnormal releases identified by the licensee for further inspection to determine if they were evaluated, were entered in the corrective action program, and were adequately resolved.

The inspectors identified radioactive effluent monitor operability issues reported by the licensee as provided in effluent release reports, to review these issues during the onsite inspection, as warranted, given their relative significance and determine if the issues were entered into the corrective action program and adequately resolved.

b. Findings

No findings of significance were identified.

Offsite Dose Calculation Manual and Final Safety Analysis Review (FSAR)

a. Inspection Scope

The inspectors reviewed UFSAR descriptions of the radioactive effluent monitoring systems, treatment systems, and effluent flow paths so they can be verified during inspection walkdowns. The inspectors reviewed changes to the ODCM made by the licensee since the last inspection against the guidance in NUREG-1301, 1302 and 0133, and Regulatory Guides 1.109, 1.21 and 4.1. When differences were identified, the inspectors reviewed the technical basis or evaluations of the change during the onsite inspection, to determine whether they were technically justified and maintain effluent releases ALARA.

The inspectors reviewed licensee documentation to determine if the licensee has identified any non-radioactive systems that have become contaminated as disclosed either through an event report or the ODCM since the last inspection. This review provided an intelligent sample list for the onsite inspection of any 10 CFR 50.59 evaluations and allowed a determination if any newly contaminated systems have an unmonitored effluent discharge path to the environment, whether any required

ODCM revisions were made to incorporate these new pathways and whether the associated effluents were reported in accordance with Regulatory Guide 1.21.

b. Findings

No findings of significance were identified.

Groundwater Protection Initiative (GPI) Program

a. Inspection Scope

The inspectors reviewed reported groundwater monitoring results and changes to the licensee's written program for identifying and controlling contaminated spills/leaks to groundwater.

b. Findings

No findings of significance were identified.

Procedures, Special Reports, and Other Documents

a. Inspection Scope

The inspectors reviewed License Event Reports, event reports and/or special reports related to the effluent program issued since the previous inspection to identify any additional focus areas for the inspection based on the scope/breadth of problems described in these reports. The review included effluent program implementing procedures, particularly those associated with effluent sampling, effluent monitor set-point determinations, and dose calculations. The review also included copies of licensee and third party (independent) evaluation reports of the effluent monitoring program since the last inspection to gather insights into the licensee's program and aid in selecting areas for inspection review (smart sampling).

b. Findings

No findings of significance were identified.

.2 Walkdowns and Observations (02.02)

a. Inspection Scope

The inspectors walked down selected components of the gaseous and liquid discharge systems to verify that equipment configuration and flow paths align with the documents reviewed in Section 02.01 above and to assess equipment material condition. Special attention was made to identify potential unmonitored release points (such as open roof vents in Boiling Water Reactor turbine decks, temporary structures butted against turbine, auxiliary or containment buildings), building alterations which could impact airborne, or liquid, effluent controls, and ventilation system leakage that communicates directly with the environment.

For equipment or areas associated with the systems selected for review that were not readily accessible due to radiological conditions, the inspectors reviewed the licensee's material condition surveillance records, as applicable.

The inspectors walked down those filtered ventilation systems whose test results will be reviewed to verify that there are no conditions, such as degraded HEPA/charcoal banks, improper alignment, or system installation issues that would impact the performance, or the effluent monitoring capability, of the effluent system.

The inspectors observed selected portions of the routine processing and discharge of radioactive gaseous effluent (including sample collection and analysis) to verify that appropriate treatment equipment was used and the processing activities align with discharge permits.

The inspectors determined if the licensee has made significant changes to their effluent release points, e.g., changes subject to a 10 CFR 50.59 review or require NRC approval of alternate discharge points.

The inspectors observed selected portions of the routine processing and discharge liquid waste (including sample collection and analysis) to verify that appropriate effluent treatment equipment is being used and that radioactive liquid waste is being processed and discharged in accordance with procedure requirements and aligns with discharge permits.

b. Findings

No findings of significance were identified.

.3 Sampling and Analyses (02.03)

a. Inspection Scope

The inspectors selected three effluent sampling activities, consistent with smart sampling, to verify that adequate controls have been implemented to ensure representative samples are obtained (e.g., provisions for sample line flushing, vessel recirculation, composite samplers, etc.).

The inspectors selected three effluent discharges made with inoperable (declared out-of-service) effluent radiation monitors to verify that controls are in place to ensure compensatory sampling is performed consistent with the Radiological Effluent Technical Specification (RETS)/ODCM and that those controls are adequate to prevent the release of unmonitored liquid and gaseous effluents.

The inspectors determined whether the facility is routinely relying on the use of compensatory sampling in-lieu of adequate system maintenance, based on the frequency of compensatory sampling since the last inspection.

The inspectors reviewed the results of the inter-laboratory comparison program to verify the quality of the radioactive effluent sample analyses to verify that the inter-laboratory comparison program include hard-to-detect isotopes as appropriate.

b. Findings

No findings of significance were identified.

.4 Instrumentation and Equipment (02.04)

Effluent Flow Measuring Instruments

a. Inspection Scope

The inspectors reviewed the methodology the licensee uses to determine the effluent stack and vent flow rates to verify that the flow rates are consistent with RETS/ODCM or FSAR values, and that differences between assumed and actual stack and vent flow rates do not affect the results of the projected public doses.

b. Findings

No findings of significance were identified.

Air Cleaning Systems

a. Inspection Scope

The inspectors evaluated whether surveillance test results since the previous inspection for TS required ventilation effluent discharge systems (HEPA and charcoal filtration), such as the Standby Gas Treatment System, meet TS acceptance criteria.

b. Findings

No findings of significance were identified.

.5 Dose Calculations (02.05)

a. Inspection Scope

The inspectors reviewed all significant changes in reported dose values compared to the previous Radiological Effluent Release Report (e.g., a factor of 5, or increases that approach Appendix I Criteria) to evaluate the factors, which may have resulted in the change.

The inspectors reviewed three gaseous waste discharge permits to verify that the projected doses to members of the public were accurate and based on representative samples of the discharge path.

Inspectors evaluated the methods used to determine the isotopes that are included in the source term to ensure all applicable radionuclides are included, within detectability standards. The review included the current 10 CFR Part 61 analyses to ensure hard-to-detect radionuclides are included in the source term.

The inspectors reviewed changes in the licensee's offsite dose calculations since the last inspection to verify the changes are consistent with the ODCM and Regulatory Guide 1.109. Inspectors reviewed meteorological dispersion and deposition factors used in the ODCM and effluent dose calculations to ensure appropriate factors are being used for public dose calculations.

The inspectors reviewed the latest Land Use Census to verify that changes (e.g., significant increases or decreases to population in the plant environs, changes in critical exposure pathways, the location of nearest member of the public or critical receptor, etc.) have been factored into the dose calculations.

For the releases reviewed above, the inspectors assessed whether the calculated doses (monthly, quarterly, and annual dose) are within the 10 CFR Part 50, Appendix I, and TS dose criteria.

The inspectors selected, as available, records of any abnormal gaseous or liquid tank discharges (e.g., discharges resulting from misaligned valves, valve leak-by, etc.) to ensure the abnormal discharge was monitored by the discharge point effluent monitor. There were no abnormal discharges. Discharges made with inoperable effluent radiation monitors, or unmonitored leakages were reviewed to ensure that an evaluation was made of the discharge to satisfy 10 CFR 20.1501 so as to account for the source term and projected doses to the public.

b. Findings

No findings of significance were identified.

.6 GPI Implementation (02.06)

a. Inspection Scope

The inspectors assessed whether the licensee was continuing to implement the Voluntary NEI/Industry GPI since the last inspection. The inspectors reviewed:

- monitoring results of the GPI to determine if the licensee has implemented its program as intended, and to identify any anomalous results; (anomalous results or missed samples were reviewed to determine if the licensee has identified and addressed deficiencies through its corrective action program.);

- identified leakage or spill events and entries made into 10 CFR 50.75 (g) records to assess any remediation actions taken for effectiveness and onsite contamination events involving contamination of ground water to assess whether the source of the leak or spill was identified and mitigated; and

- unmonitored spills, leaks, or unexpected liquid or gaseous discharges, to ensure that an evaluation was performed to determine the type and amount of radioactive material that was discharged, assess whether sufficient radiological surveys were performed to evaluate the extent of the contamination and the radiological source term and verify that a survey/evaluation had been performed to include consideration of hard-to-detect radionuclides.

The inspectors reviewed whether the licensee completed offsite notifications (State, local, and if appropriate, the NRC), as provided in its GPI implementing procedures.

The inspectors reviewed the evaluation of discharges from onsite surface water bodies (ponds, retention basins, lakes) that contain or potentially contain radioactivity, and the potential for ground water leakage from these onsite surface water bodies to determine if licensees are properly accounting for discharges from these surface water bodies as part of their effluent release reports.

The inspectors assessed whether onsite ground water sample results and a description of any significant onsite leaks/spills into ground water for each calendar year was documented in the Annual Radiological Environmental Operating Report for REMP or the Annual Radiological Effluent Release Report for the RETS. For significant, new effluent discharge points (such as significant or continuing leakage to ground water that continues to impact the environment if not remediated), the inspectors determined if the ODCM was updated to include the new release point.

b. Findings

No findings of significance were identified.

.7 Problem Identification and Resolution (02.07)

a. Inspection Scope

Inspectors evaluated whether problems associated with the effluent monitoring and control program are being identified by the licensee at an appropriate threshold and are properly addressed for resolution in the licensee corrective action program. In addition, they assessed appropriateness of the corrective actions for selected sample of problems documented by the licensee involving radiation monitoring and exposure controls.

b. Findings

No findings of significance were identified.

This inspection constituted one sample as defined in IP 71124.06-5.

2RS07 Radiological Environmental Monitoring Program and Radioactive Material Control Program (71124.07)

.1 Inspection Planning (02.01)

a. Inspection Scope

The inspectors reviewed the annual radiological environmental operating reports and the results of any licensee assessments since the last inspection, to verify that the Radiological Environmental Monitoring Program (REMP) was implemented in accordance with the TS and ODCM. This review included report changes to the ODCM with respect to environmental monitoring, commitments in terms of sampling locations, monitoring and measurement frequencies, land use census, inter-laboratory comparison program, and analysis of data.

The inspectors reviewed the ODCM to identify locations of environmental monitoring stations and the FSAR for information regarding the environmental monitoring program and meteorological monitoring instrumentation.

The inspectors reviewed quality assurance audit results of the program to assist in choosing inspection "smart samples" and audits and technical evaluations performed on the vendor laboratory program.

The inspectors reviewed the annual effluent release report and the 10 CFR Part 61, "Licensing Requirements for Land Disposal of Radioactive Waste," report, to determine if the licensee is sampling, as appropriate, for the predominant and dose-causing radionuclides likely to be released in effluents.

b. Findings

No findings of significance were identified.

.2 Site Inspection (02.02)

a. Inspection Scope

The inspectors walked down three of the air sampling stations and five of the thermoluminescent dosimeter (TLD) monitoring stations to determine whether they are located as described in the ODCM and to determine the equipment material condition. Consistent with smart sampling, the air sampling stations were selected based on the locations with the highest X/Q, D/Q wind sectors, and TLDs were selected based on the most risk-significant locations (e.g., those that have the highest potential for public dose impact). For the air samplers and TLDs selected, the inspectors reviewed the calibration and maintenance records to verify that they demonstrate adequate operability of these components. Additionally, the review included the calibration and maintenance records of composite water samplers and evaluation to determine if the licensee has initiated sampling of other appropriate media upon loss of a required sampling station.
The licensee does not use composite samplers.

The inspectors observed the collection and preparation of two to four environmental samples from different environmental media (e.g., ground and surface water, milk, vegetation, sediment, and soil) as available to verify that environmental sampling is representative of the release pathways as specified in the ODCM and that sampling techniques are in accordance with procedures.

By direct observation and review of records, the inspectors evaluated the meteorological instruments to verify they are operable, calibrated, and maintained in accordance with guidance contained in the FSAR, NRC Regulatory Guide 1.23, "Meteorological Monitoring Programs for Nuclear Power Plants," and licensee procedures. Also, the inspectors assessed whether the meteorological data readout and recording instruments in the control room and, if applicable, at the tower were operable.

The inspectors assessed whether missed and or anomalous environmental samples are identified and reported in the annual environmental monitoring report. They selected five events that involved a missed sample, inoperable sampler, lost TLD, or anomalous measurement to verify that the licensee has identified the cause and has implemented corrective actions. The inspectors reviewed the licensee's assessment of any positive sample results (i.e., licensed radioactive material detected above the lower limits of detections and reviewed the associated radioactive effluent release data that was the source of the released material.

Inspectors selected three SSCs that involve, or could reasonably involve, licensed material for which there is a credible mechanism for licensed material to reach ground water, and evaluated whether the licensee has implemented a sampling and monitoring program sufficient to detect leakage of these SSCs to ground water.

The inspectors assessed whether records, as required by 10 CFR 50.75(g), of leaks, spills, and remediation since the previous inspection are retained in a retrievable manner.

The inspectors reviewed any significant changes made by the licensee to the ODCM as the result of changes to the land census, long-term meteorological conditions (3-year average), or modifications to the sampler stations since the last inspection. They reviewed technical justifications for any changed sampling locations to verify that the licensee performed the reviews required to ensure that the changes did not affect its ability to monitor the impacts of radioactive effluent releases on the environment.

The inspectors evaluated whether the appropriate detection sensitivities with respect to TS/ODCM are used for counting samples (i.e., the samples meet the TS/ODCM required lower limits of detections). The inspectors reviewed the results of the vendor's quality control program, including the inter-laboratory comparison program to verify the adequacy of the environmental sample analyses vendor laboratory program and verify inter-laboratory comparison test included the media/nuclide mix was appropriate for the facility.

b. Findings

No findings of significance were identified.

.3 Identification and Resolution of Problems (02.03)

a. Inspection Scope

The inspectors assessed whether problems associated with the REMP are being identified by the licensee at an appropriate threshold and are properly addressed for resolution in the licensee's corrective action program. Additionally, they evaluated the appropriateness of the corrective actions for a selected sample of problems documented by the licensee that involved the REMP.

b. Findings

No findings of significance were identified.

This inspection constituted one sample as defined in IP 71124.07-05.

4. OTHER ACTIVITIES

4OA1 Performance Indicator Verification (71151)

.1 Unplanned Scrams per 7000 Critical Hours

a. Inspection Scope

The inspectors sampled licensee submittals for the Unplanned Scrams per 7000 Critical Hours Performance Indicator (PI) for the period from the first quarter 2009 through the fourth quarter 2009. 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, were used. The inspectors reviewed the licensee's operator narrative logs, issue reports, event reports and NRC Inspection Reports for the period of first quarter 2009 through fourth quarter 2009 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 constituted one unplanned scrams per 7000 Critical Hours sample as defined in IP 71151-05.

b. Findings

No findings of significance were identified.

.2 Unplanned Scrams with Complications

a. Inspection Scope

The inspectors sampled licensee submittals for the Unplanned Scrams with Complications PI for the period from the first quarter 2009 through the fourth quarter 2009. 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, were used. The inspectors reviewed the licensee's operator narrative logs, issue reports, event reports and NRC Integrated Inspection Reports for the period of first quarter 2009 through the fourth quarter 2009 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 constituted one unplanned scrams with complications sample as defined in IP 71151-05.

b. Findings

No findings of significance were identified.

.3 Unplanned Transients per 7000 Critical Hours

a. Inspection Scope

The inspectors sampled licensee submittals for the Unplanned Transients per 7000 Critical Hours PI for the period from the first quarter 2009 through the fourth quarter 2009. 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, were used. The inspectors reviewed the licensee's operator narrative logs, issue reports, maintenance rule records, event reports and NRC Integrated Inspection Reports for the period of first quarter 2009 through the fourth quarter 2009 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 constituted one unplanned transients per 7000 Critical Hours 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 attached List of Documents Reviewed.

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 Selected Issue Follow-Up Inspection: Review of Licensee's Causal Evaluations for NRC Findings with Cross-Cutting Aspects

a. Inspection Scope

The inspectors chose to follow-up on the licensee's analysis and actions implemented to address two emerging cross-cutting themes. This annual sample was chosen in the first quarter of 2010 as it provided meaningful input into 2009 end-of-cycle assessment process. Duane Arnold Energy Center has received multiple findings with cross-cutting aspects in the areas of Human Performance, Decision Making, and Problem Identification and Resolution, Corrective Action Program.

b. Observations

The licensee initiated CAP 072908, "NRC Finding Cross-Cutting Aspect – H.1.b," and CAP 072909, "NRC Findings with Cross-Cutting Aspects – P.1.c." The inspectors reviewed these CAPs and the causal evaluations performed for each CAP.
Corrective Action Program 072908 was initially screened by the IST and the Management Review Committee (MRC) as a level 'A' CAP using the guidance contained in station procedure PI-AA-204, "Condition Identification and Screening Process."
The IST and MRC recommended that an ACE be performed. The inspector reviewed station procedure LI-AA-200, "Regulatory Margin Corrective Action Strategy," and noted that the procedure stated that "Should the NRC issue a finding or violation with a cross-cutting aspect, the manager who owns the condition report shall…take the following actions: for the third current cross-cutting finding in an aspect, complete a root cause evaluation to determine the cause of the cross-cutting aspect, the extent of

condition, the extent of cause, the cause for the repetitive findings, and the required corrective actions including the milestones and due dates."

The inspector questioned members of the MRC as to why CAP 072908 did not follow the guidance of LI-AA-200, since the CAP identified that DAEC had three findings that all shared the cross-cutting aspect of H.1(b). Management Review Committee personnel stated that the guidance contained in PI-AA-204 was used by the MRC to assign an ACE to CAP 072908. The inspector reviewed the guidance contained in PI-AA-204. While PI-AA-204 did allow for the MRC to assign an ACE instead of a RCE, certain criteria were required to be met and documented for justification. Those criteria included:

- "The issue has been previously evaluated or identified as a result of an extent of condition review under a previous evaluation. The cause is understood and corrective actions are being implemented."
- "The cause, corrective action, and extent of condition are simple and known. This knowledge may be the result of previous assessments."

After reviewing the guidance in PI-AA-204, and the documentation in CAP 072809, the inspector noted that CAP 072908 did not document any justification for non-performance of an RCE as required by PI-AA-204. This was discussed with members of the MRC, and the station wrote CAP 073088, "Need for Clarity of instructions between CAP Procedures." This CAP documented apparent discrepancies between station procedures LI-AA-200 and PI-AA-204.

After additional review of CAP 072908, the MRC rescreened the CAP and changed the required evaluation from an ACE to a RCE. This evaluation was still being performed at the end of this inspection period.

This review constituted one in-depth problem identification and resolution sample as defined in IP 71152-05.

c. Findings

No findings of significance were identified.

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

.1 Both Turbine Bypass Valves Failed Open

a. Inspection Scope

The inspectors reviewed the plant's response to both Turbine Bypass Valves unexpectedly opening from 100 percent power on January 4, 2010. This caused reactor power to spike to 105 percent and the plant commenced a subsequent fast power reduction to 68 percent power. 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 (Closed) Licensee Event Report (LER) 05000331/2010-001-00: Emergency Diesel Generator Start Due to Failed Arrestor in Switchyard

This event, which occurred on January 1, 2010, was initiated by a failed lightning arrestor on the 161 kilovolt (kV) Vinton line in the DAEC switchyard. This caused an undervoltage condition which automatically started the ‘A’ EDG, which did not load onto its 4160 VAC bus since the bus remained powered from its normal power supply throughout the transient. The licensee determined the cause of the event to be a lack of coordination between the EDG automatic start logic and the DAEC switchyard protective relaying. Corrective actions included replacing the Vinton line lightning arrestors and planning modifications to EDG start logic which will coordinate start logic and protective relaying. Documents reviewed as part of this inspection are listed in the Attachment to this report. This LER is closed.

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

4OA5 Other Activities

.1 (Closed) URI 05000331/2009004-03: Adequacy of the Licensee’s Critique for the May 20, 2009, EP Drill

The inspectors reviewed the final drill report from the 2009 ERO Training Drill #2, the drill scenario, the drill objectives and evaluation criteria, the actual timeline from the drill, corrective actions resulting from the drill, and the licensee’s assessment for the DEP PI. The inspectors reviewed the controllers’ and evaluators’ logs and observation statements along with the Senior Resident Inspector’s observations to evaluate the circumstances and sequence of events to determine if the critique met the requirements of 10 CFR Part 50, Appendix E, Section IV.F.2.g and 10 CFR 50.47(b)(14). The inspectors determined a violation of NRC requirements had occurred. An NRC-Identified NCV was documented in 1EP5 of this report and the unresolved issue was closed. Documents reviewed are listed in the Attachment to this report.

.2 (Closed) URI 05000331/2009002-01: An Internal Contamination Occurred while Cleaning RPV Studs and Washers on the Refuel Floor at Duane Arnold

During the inspectors’ review of a corrective action program record (CAP #0063690) that described a positive facial and internal contamination of a contractor who had performed decontamination of the RPV studs and washers on the refuel floor, it was noted that the individual deviated from the instructions provided by radiation protection staff concerning the method that was to be used to decontaminate the above items.

Section 2RS01.1 discusses details of this event and also documents that a Green finding and associated NCV of TS 5.4.1(a) was identified for the failure to establish and implement a procedure for performing decontamination activities associated with a potentially significant decontamination activity. This URI is closed.

4OA6 Management Meetings

.1 Exit Meeting Summary

On April 6, 2010, the inspectors presented the inspection results to Mr. C. Costanzo 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 Radiological Hazards Assessment and Exposure Control, Radioactive Gaseous and Liquid Effluent Treatment, Radiological Environmental Monitoring Program And Radioactive Material Control Program inspection with the Site Vice President, Mr. C. Costanzo, on January 29, 2010.

- The results of the Crane and Heavy Lift Inspection with the Site Vice President, Mr. C. Costanzo, on February 5, 2010.

- An Emergency Preparedness URI inspection interim exit was conducted by phone on February 4, 2010, with Site Vice President, Mr. C. Costanzo.

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.

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
C. Dieckmann, Operations Manager
G. Rushworth, Assistant Operations Manager
P. Giroir, Operations Support Manager
R. Porter, Chemistry & Radiation Protection Manager
M. Davis, Emergency Preparedness Manager
M. Lingenfelter, Design Engineering Manager
M. Ogden, Maintenance Manager
M. Heermann, Radwaste Shipper in Training
N. McKenney, General Supervisor Radiation Protection
J. Karrich, ALARA Coordinator
R. Schlueter, ALARA Coordinator
W. Render, Instructor, DAEC Operator Training
L. Swenzkinski, Sr. Licensing Engineer
J. Dvorski, Plant Engineer
M. Lingenfetter, Design Engineering Manager
F. Lucas, Design Engineer
T. Browning, Licensing Engineer
R. Cole, PI Manager
S. Inghram, Engineering Supervisor
K. Furman, Safety Manager
J. Dubois, Program Engineering Manager
R. Harter, O & S Manager

Nuclear Regulatory Commission

K. Feintuch, Project Manager, NRR
K. Riemer, Chief, Reactor Projects Branch 2
B. C. Dickson, Branch Chief, Plant Support Team

LIST OF ITEMS OPENED, CLOSED AND DISCUSSED

Opened

05000331/2010002-01	NCV	Inadequate Evaluations for Crane and Special Lifting Devices (1R20.1.b(1))
05000331/2010002-02	FIN	Lift Height Assumptions in Drop Load Analyses Not Reflected in Rigging Procedures (1R20.1.b(2))
05000331/2010002-03	NCV	Failure to Conduct an Adequate Critique for the May 20, 2009, Drill (1EP5)
05000331/2010002-04	NCV	An Internal Contamination Occurred while Cleaning RPV Studs and Washers on the Refuel Floor at Duane Arnold (2RSO1.1)

Closed

05000331/2010002-01	NCV	Inadequate Evaluations for Crane and Special Lifting Devices (1R20.1.b(1))
05000331/2010002-02	FIN	Lift Height Assumptions in Drop Load Analyses Not Reflected in Rigging Procedures (1R20.1.b(2))
05000331/2010002-03	NCV	Failure to Conduct an Adequate Critique for the May 20, 2009, Drill (1EP5)
05000331/2010002-04	NCV	An Internal Contamination Occurred while Cleaning RPV Studs and Washers on the Refuel Floor at Duane Arnold (2RSO1.1)
05000331/2009004-03	URI	Adequacy of the Licensee's Critique for the May 20, 2009, EP Drill (1EP5.b2)
05000331/2009002-01	URI	An Internal Contamination Occurred while Cleaning RPV Studs and Washers on the Refuel Floor at Duane Arnold
05000331/2010-001	LER	Emergency Diesel Generator Start Due to Failed Arrestor in Switchyard (4OA3.2)

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

OI 733; Turbine Building HVAC; Revision 22
AOP 904; Extreme Cold Weather (< 0F); Revision 2
NG-270K; Plant Winterization Checklist; Revision 2
OI 442; Circulating Water System; Revision 79
OI 734; Reactor Building HVAC System; Revision 53
AOP-902; Flood; Revision 35
CAP 073778; CAQ-Higher than Normal Water Level Forecast is Greater than AOP 902 of 742 Feet

Section 1R04

ISO-HBD-024-03; Isometric – Water Pumphouse Emergency Service Water; Revision 14
BECH-M146; P&ID [Piping and Instrument diagram] Service Water System Pumphouse; Revision 82
MECH-M113; P&ID RHR Service Water and Emergency Service Water Systems; Revision 65
OI 454A6; ESW System Control Panel Lineup; Revision 2
OI 454A1; ESW System Electrical Lineup; Revision 3
OI 454A2; ‘A’ ESW System valve Lineup and Checklist; Revision 10
OI 150A2; RCIC System Valve Lineup and Checklist; Revision 12
OI 150A4; RCIC System Control Panel Lineup; Revision 3
OI 454A4; “B” ESW System Valve Lineup and Checklist; Revision 11

Section 1R05

AFP-01; Torus Area and North Corner Rooms, Revision 25
AFP 08; Standby Gas Treatment System and MG Set Rooms;
Revision 25
AFP 31; Intake Structure Pump Rooms; Revision 26
AFP 32; Intake Structure Traveling Screen Areas; Revision 27
Fire Plan, Volume 1, Program; Revision 57
ACP 1412.4; Impairments to the Fire Protection Systems; Revision 57
FPIR-09-7320; Fire Protection Impairment Request for Reactor Building 1st Floor – 757
Hourly Fire Watch Surveillance Checklist for February 17, 2010 through February 21, 2010
CAP 073385; CAQ – Inadequacy in Fire Patrols Identified; dated February 23, 2010
CAP 073378; NCAQ Fire Patrol Checklist Initials Omission; dated February 23, 2010
AFP 18; Turbine Building North Turbine Building Ground Floor and Tube Pulling Area; Revision 28
AFP 19; Turbine Building South Turbine Building Ground Floor and Tube Pulling Area; Revision 25

AFP 20; Aux Boiler Room, Emergency Diesel Generator Rooms and Generator Day Tank Rooms; Revision 29
AFP 03; Reactor Building HPCI, RCIC & Radwaste Tank Rooms; Revision 26

Section 1R07

CAP 061393; NCAQ – Heat Exchanger Leaks Identified During A SBDG STP
CAP 061471; CAQ – 'B' SBDG Jacket Cooling Water Heat Exchanger Leak
CAP 62018; NCAQ – Leak on B EDG Jacket Coolant Heat Exchanger
CAP 066838; CAQ – 1G031 Heat Exchanger Leak
CAP 072204; NCAQ – ESW Leak from 'B' Jacket Cooling Water Heat Exchanger
CAP 072892; NCAQ- Leak of 'B' EDG Heat Exchanger During Maintenance Run
CAP 072819; CAQ - 1E053B1 Inspection Findings
CAP 072842; CAQ – Eddy Current Testing Results on B EDG Heat Exchangers
Maintenance WO 1148982; Perform Eddy Curent Testing on the Diesel Generator Heat Exchangers
CAP 073076; CAQ - EDG Heat Exchanger ESW Leaks Have Occurred Since Tube Bundle Replacement

Section 1R11

ACP 110.1; Conduct of Operations; Revision 24
Integrated Plant Operating Procedure 5; Reactor Scram; Revision 53
Emergency Operating Procedure 1; RPV [Reactor Pressure Vessel] Control; Revision 16
DAEC Emergency Action Level Notification Form; NOTE 5; Revision 12
Emergency Plan Implementing Procedure 1.1; Determination of Emergency Action Levels; Revision 28
Emergency Action Level Matrix – Hot Modes; Revision 7
CAP 073637; NCAQ Security Unavailable for a DEP PI Opportunity in LOR
CAP 073659; NCAQ Attention to Detail Error during LOR Notification Process

Section 1R12

DAEC Corrective Work Order (CWO)/Preventative Work Order (PWO) System Health Report for System 58.02: Primary Containment Isolation; Period 2009-04
DAEC System Checklist/Health Report for System 58.02: Primary Containment Isolation; Period 2009-04
PWO 1150330; Inspect and Check Calibration of Temperature Element TE4478A
PWO 1150331; Inspect and Check Calibration of Temperature Element TE4478B
CWO A101838; TIS [Temperature Indicating Switch] 4478 Reading Unexpectedly High. Went from 134-148 Deg F to 165 Deg F for no apparent reason.
CWO A96383; Channel B1, Group 1 Isolation, Spurious. Suspect TIS4478 is Causing 1/4 Group 1 Isolation
CAP 072457; NCAQ – Unexpected Change in TIS 4478 Ch 1 Temperature Reading to 169 F
CAP 072477; CAQ – Unexpected Half Group 1 Isolation Signal
CAP 072404; NCAQ Main Steam Line Leakage Detection Panels 1C193A-D and 1C194A-D Switch Corrosion
CAP 071275; CAQ – TIS4477 Channel 1 Turbine Building High Temperature Failed Channel Check
CAP 071148; CAQ – Technical Specification Reading TIS 4478 Channel 1 High Out-of-Specification

CAP 046359; TIS4480 at Limit
CAP 033449; TIS 4480 Channel 1 Temperature is elevated at 174 F
CAP 028527; Setpoint Design Control Incomplete for Installed Equipment
CAP 026795; Received Unexpe3cted 'B' Side Group 1 from TIS-4478 Becoming De-energized

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Work Planning Guideline 1; Work Process Guideline; Revisions 37 and 38
Work Planning Guideline 2; On-Line Risk Management Guideline; Revisions 55 and 56
WM-AA-1000; Work Activity Risk Management Process; Revision 4
OP-AA-102-1003; Guarded Equipment; Revision 1
OP-AA-102-1003 (DAEC); Guarded Equipment (DAEC Specific Information); Revisions 3, 4, and 5
OP-AA-104-1007; Online Aggregate Risk; Revision 0
Maintenance Risk Evaluations for Work Week 9002; Revisions 0 and 1
DAEC On-line Schedule for Work Week 2
Maintenance Risk Evaluations for Work Week 9005; Revisions 0, 1, and 2
DAEC On-line Schedule for Work Week 5
Maintenance Risk Evaluations for Work Week 9006; Revision 0
DAEC On-line Schedule for Work Week 6
WO A101823; Both BPVs Fail Open/Closed/Open Several Times
OI 920; Drywell Sump System; Revision 40
CAP 072511; CAQ Increase in Unidentified Drywell Leakage Observed after RCIC STP
CAP 072636; CAQ Pipe Support Discrepancies Found
CAP 072925; CAQ 1C94 D-5 Generator Filed Ground In and Clear on Diesel Start
WO 1152941; March 2010 Control Rod Sequence Exchange
Work Week 9013 Weekly PRA Risk Profile
Work Week 9013 Work Activity High Risk Summary
CAP 073959; NCAQ 1G-201B Recirc MG Set Lube Oil Temperatures LOOS

Section 1R15

CAL-M05-027; Emergency Diesel Generator Heat Exchanger Heat Transfer Calculation; Revision 3
CAP 072842; CAQ – Eddy Current Testing Results on B EDG Heat Exchangers
ANATEC Report FPLE33-DAEC-01; Emergency Diesel Coolers "B" Train HX-1E053B-1, 2, 3; dated February 3, 2010
CAP 072636; CAQ - Pipe Support Discrepancies Found
OPR 419; Pipe Support Discrepancies Found; dated January 26, 2010
STP 3.7.7-03; MCPR [Minimum Critical Power Ratio] Limit Verification; Revision 10
CAP 073296; CAQ Structural Bolting Questions raised by NRC Resident in Northwest Corner Room
M095-067; Seismic Analysis of Air Handling Units; Revision 3
OPR 000422; Structural Bolting Questions raised by NRC Resident in Northwest Corner Room
WO A101823; Both BPVs Fail Open/Closed/Open Several Times
STP 3.7.7-01; Bypass Valve Test; Revision 12

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WO A101823; Both BPVs Fail Open/Closed/Open Several Times

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STP 3.8.1-06B; B Standby Diesel Generator Operability Test (Fast Start); Revision 9
CAP 072910; NCAQ – Data not Recorded during B EDG Fast Start Surveillance STP 3.8.1-06B
CAP 072926; NCAQ – Question about B SBDG Frequency
STP 3.7.7-01; Bypass Valve Test; Revision 12
STP 3.5.-05; HPCI System Operability Test; Revision 48
CAP 073450; NCAQ Failed to Capture ASME Data During STP 3.5.1-05, Due to Burnt Out bulb
CAP 072066; NCAQ – HPCI STP Alignment
CAP 073433; CAQ MO-2318, HPCI MIN FLOW BYPASS valve failed to meet required ASME times
CAP 073432; NCAQ HPCI quarterly STP and HPCI System Operability Test- Null Voltage Test Starts Late
STP 3.1.7-04; SBLC Pump Operability Test and Comprehensive Pump Test; Revision 11

Section 1R20

NG-04-003; NMC Letter to NRC, Clarification of Assumptions Regarding Reactor Building Crane; dated January 7, 2004
NG-02-1106; NMC Letter to NRC, Single Failure Proof Status of Reactor Building Crane; dated December 4, 2002
NG-01-1428; NMC Letter to NRC, Clarification of Assumptions Regarding Reactor Building Crane; dated December 21, 2001
LDR-81-342; Iowa Electric Light and Power Company Letter to NRC, Supplemental Response, Control of Heavy Loads; dated December 15, 1981
NG-82-2530; Iowa Electric Light and Power Company Letter to NRC, Supplement and Clarifications to the Report on Control of Heavy Loads; dated December 2, 1982
OE 009533; NRC RIS2005-25, Clarification of NRC Guidelines for Control of Heavy Loads; dated November 15, 2005
CAP 049731; Use of Intermediate Hoists Not Addressed By Site Heavy Loads Procedures
OE019864; Perform OE Evaluation – NRC RIS-2005-25, Sup 1 Heavy Loads; May 31, 2007
CAP 059560; NCAQ - NEI 08-05 Requires Heavy Loads Actions
NRC Letter; Acceptance for Referencing of Licensing Topical Report EDR-1(P), Revision 3, “Ederer Nuclear Safety-Related Extra Safety and Monitoring (X-SAM) Cranes, Cecil O. Thomas to C. William Clark Jr., Ederer Corporation; dated August 26, 1983
BECH-MRS-M023A; Technical Specification for Modification to the Reactor Building Crane; Revision 2
Calculation CAL-M01-273; Reactor Building Crane Girder Check; Revision 2
Calculation 273-17; Effects of Impact of Reactor Feed Pump Motor with Turbine Building Mat Foundation; Revision 1
Calculation 273-33; Fuel Pool Demineralizer Plugs; Revision 1
Calculation CAL-M09-047; Remote Steam Dryer/Separator Strongback Structural Analysis; Revision 0
Calculation 273C036; Reactor Vessel Head Strongback Modified to Meet NUREG-0612 Requirements; Revision 0
Procedure ACP 1408.19; Control of Generic Heavy Loads; Revision 20
DWG APED-F13-006 (2); Modification of the Reactor Vessel Head Strongback; Revision 2
Procedure ACP 1203.55; Control of Heavy Loads Analyses, Revision 6
Procedure Reactor Feed Pump 110; Reactor Pressure Vessel Disassembly; Revision 22

CAP 071410; NCAQ-Establish Correct Weight for The Reactor Vessel Head
CAP 072492; NCAQ-ECH-LHR Drawings in MDL are Illegible
CAP 072530; CAQ – Revisit NEI 08-05 Requirements for UFSAR
CAP 072538; NCAQ-Review Reactor Feed Pump 110 and 210 for Heavy Load Restrictions
CAP 072550; CAQ-UFSAR Should Be Enhanced To Clearly Call Out COHL Design Basis Info
CAP 072551; NCAQ-Fuel Pool Demin Area Shield Plugs Need Lift Height Limitations Proceduralized
CAP 072568; NCAQ-Factor of Safety Less Than 10 in CAL-273-C-036 for DW Head Strbck TOOL-E189
CAP 072570; NCAQ-Determine the Weight of the Rx Vessel Head Strongback
CAP 072811; NCAQ-FP Motor Maximum Lift Height Not Proceduralized
CAP 072812; NCAQ-CAL-M09-047 Listed Misstated Allowable Tensile as 5.0 Instead of 7.5 ksi
CAP 072838; NCAQ Section C of GMP-MECH-06 May Not Perform an Adequate NDE
CAP 072880; NCAQ CAL-M09-047 Incorrectly Calculates Allowable Shear Stress for TOOL-E206
CAP 072885; NCAQ CAL-273-C036 Rx Vessel Strongback Calc Does Not Explicitly Evaluate Hook Pins
CAP 072886; NCAQ Reactor Building Crane Cold Proof Test Continued Compliance Not Met
CAP 072912; CAQ NES calc 2941-130 Incorrectly Calculates Allowable Shear Stress for TOOL-E195
CAP 072917; NCAQ Seismic Evaluation for Rx Bldg Crane Trolley Rail
CAP 072924; NCAQ Generate A List of Loads That Have Restrictions On Max Lift Heights

Section 1R22

STP 3.8.1-04A; A Standby Diesel Generator Operability Test (Slow Start from Normal Start Air); Revision 5
Maintenance WO 1148081; TSC/ DAC Standby Diesel Generator Engine
GENERA-C170-01; Caterpillar TSC Standby Diesel Generator Engine Inspections
CAP 071387; NCAQ – TSC Diesel Engine Battery Voltage Reads High
CAP 073074; NCAQ TSC Diesel Field Observation
STP 3.3.6.1-45; HPCI Steam Supply Pressure Low Channel Functional Test; Revision 9
CA 53763; SCAQ – CATPR1 – Revise STPs
CA 53724; NCAQ – Potential Scram Reduction Technique
OTH 6693; Review All Instrument STPs for proper restoration steps that prevent perturbations/ transients
PCR 52764; CAQ – FSA SA# 34037, Configuration Control, AFI, Surveillance Line Up Deficiencies; dated June 26, 2009
STP 3.5.1-01A; A Core Spray System Operability Test; Revision 4
CAP 073761; CAQ Received Unexpected Alarm 1C03A(C-9) ‘A’ Core Spray Discharge Line Hi Pressure
NAP-201; Human Performance; Revision 11
STP 3.4.4-01; Reclassification of Drywell Leakage; Revision 3
STP 3.0.0-01; Instrument Checks; Revision 105
CAP 073872; NCAQ Remove Polyphosphates from STP 3.0.001 INSTRUMENT CHECK Section 4.0
Adverse Condition Monitoring Plan; Increased Drywell Leakage, Revision 3; dated March 16, 2010

Section1EP5

NEP # 2009-0010; 2009 ERO Training Drill #2 Final Report; dated June 24, 2009
2009 ERO Training Drill #2 Information - Scenario, Master Sequence of Events, Drill Objectives; Drill Conducted May 20, 2009
Controller, Evaluator, and Players Event Logs; dated May 20, 2009
EPIP 1.1; Determination of Emergency Action Levels; Revision 28
EBD-S; EAL Bases Document; System Malfunction Category; Revision 7
CAP 0067417; NCAQ -09TD2 – Appropriateness of Controller Interjection Questioned; dated May 20, 2009
CAP 072877; Revise Drill Development Process Based on NRC Observation
CAP 067460; NCAQ-09TD2 – WR Yarway Indication in the Simulator Related to MIL and ED Decision
CAP 068506; CAQ – 09TD2 – Evaluate Impact of May 20, 2009 ERO Training Drill Controller Interject
CE 007464; NCAQ -09TD2 - Appropriateness of Controller Interjection Questioned; dated May 28, 2009
CE 007572; CAQ – 09TD2 – Evaluate Impact of May 20, 2009 ERO Training Drill Controller Interject; dated July 23, 2009

Section 1EP6

CAP 072710; NCAQ – Evaluate Potential Delta between reported Reactor Pressure Vessel Level Water Level for General Emergency and Drill
CAP 072673; CAQ 10TD1 – Note 05 Error in Block 7 at Emergency Operation Facility
CAP 072681; NCAQ 10TD1 – How to Correct Note 5 Information
CAP 072715; NCAQ 10TD1 – Revised Note 05 for Block 7 Missing Date in Message Initiated Box
CAP 072692; NCAQ 10TD1 – Invoking 10 CFR 50.54x for Security Related Matters Remains Unclear
2010 Emergency Response Organization Training Drill 1 Final Report; January 27, 2010
Emergency Department Preparedness Manual EPDM 1010; EP PIs; Revision 13

Section 2RS01

ACE 001923; Nasal Contamination Received While Cleaning of Reactor Pressure Vessel Bolts; April 9, 2009
CAP 065072; CAQ Refuel Floor GE Worker Receives Unanticipated Dose Alarm
CAP 065156; NCAQ Issues Found with Administrative Control of LHRA

Section 2RS06

5059SCRN025543; 50.59 Screening for PCP 8.2 per PWR 40948, Kaman Operating Procedures; January 23, 2008
CA 050729; Corrective Action-Rad Monitor Sensor Checks Not Done for Kaman 2; September 25, 2008
CAP 059407; CAQ Condition Adverse to Quality-Increased Inventory Los Rate from the Condensate Storage Tanks; August 8, 2008
CAP 063486; CAQ Liquid Effluent LCO

CE 006979; Condition Adverse to Quality-Loss of Off-Site Dose Assessment Manual Continuous Sampling for Particulate and Iodine (K-12); January 27, 2009
ORTH028262; Determine If Air Sampling Equipment is the Best Tool for the Job; April 28, 2008
ORTH038870; Kaman Noble Gas Conversion Factor Conservatively High; May 26, 2009
ORTH041066; Locate the Calculation Used to Demonstrate Isokinetic Sampling; August 31, 2009
PCP 10.4; Accumulation of Offsite Effluent Dose Calculational Data; Revision 1
PCR053259; Procedure Change Request-No Longer Able to Obtain REMP Sample; September 8, 2009
REC 07-001-C; Radiological Engineering Calculation-Demonstration of the Effect of RFO-20 Radiation Workers on Site Sewage Treatment Plant Effluent tritium Concentrations; October 22, 2007
REC 08-001-R; Radiological Engineering Calculation-10CFR61 Compliance Data Technical Basis for Reactor Water Clean-up Resin; May 1, 2008
STP NS790206; Surveillance Test Procedure Quarterly Off-Gas Stack Flow Monitor Calibration; November 19, 2008
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STP NS790302; Surveillance Test Procedure Liquid Process Rad Monitor Inoperable Sampling and Analysis; January 6, 2010
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STP NS791016; Surveillance Test Procedure Kaman Monitor; October 20, 2009
2007 Annual Radioactive Material Release Report; April 24, 2008
2008 Annual Radioactive Material Release Report; April 27, 2009
Duane Arnold Energy Company Operations Log Book- Limiting Conditions for Operation; January 2009 to January 2010

Section 2RS07

Material Control Program
ACP 1411.356; DAEC Ground Water Protection Program; Revision 1
DAEC-SC-PD-13; Meteorological System Equipment; Revision 0
ESP 1.0; Radiological Environmental Monitoring Quality Control Program; Revision 9
ESP 4.3.1.1; Airborne Particulate and Iodine sampling; Revision 27
ESP 4.3.1.2; Ambient Radiation Sampling; Revision 15
EAP 4.3.1.3; Surface Water Sampling; Revision 17
ESP 4.3.1.5A Sampling Site Monitoring Wells; Revision 2
ESP 4.4; Land Use Census; Revision 9
EV-AA-100; FPL Nuclear fleet Ground Water Protection Program; Revision 0
Off Site Dose Assessment Manual – Gaseous and Liquid Effluents; Revision 26

2008 Annual Radiological Environmental Operating Report; February 5, 2009
REC 09-001-C; Radiological Engineering Calculation-Public Dose Due to the Washout of Tritium during Rain Events; August 1, 2009

Section 4OA1

DAEC PI Report for Unplanned Scrams per 7000 Critical Hours for January 2009 through December 2009
DAEC PI Report for Unplanned Scrams with Complications for January 2009 through December 2009
DAEC PI Report for Unplanned Power Changes per 7000 Critical Hours for January 2009 through December 2009
NEI 99-02; Regulatory Assessment Performance Indicator Guideline; Revision 6

Section 4OA2

CAP 072908; CAQ – NRC Finding Cross-Cutting Aspect – H.1.b
CAP 071962; CAQ – NRC Cross-Cutting Aspects
RCE 001035; Root Cause Analysis of the Events Contributing to the NRC Mid-Cycle Review, DAEC Cross-Cutting Finding in the area of Human Performance
RCE 001084; Root Cause Analysis of Negative Trend in NRC PI&R Cross-Cutting Aspect P.1.c
CAP 072909; CAQ – NRC Findings with Cross-Cutting Aspects – P.1.c
CAP 067166; CAQ – Trend Identified in PI&R Cross-Cutting Area
CAP 037726; NRC Cross-Cutting Finding – HU
CAP 073088; NCAQ - Need for Clarity of Instructions between CAP Procedures
PI-AA-204; Condition Identification and Screening Process; Revision 5
PI-AA-205; Condition Evaluation and Corrective Action; Revision 3
LI-AA-203; Regulatory Issue Management; Revision 1
LI-AA-200; Regulatory Margin Corrective Action Strategy; Revision 2
LI-AA-200-1000-10000; FPL Fleet Licensing Performance Indicators; Revision 01
CCEM; Common Cause Evaluation Manual; Revision 3
ACEM; Apparent Cause Evaluation Manual; Revision 13
RCEM; Root Cause Evaluation Manual; Revision 18
CAP 073796; NCAQ RCE 1088 Data Review Identified Inadequate Correct Action for ACE 1983
CAP 073793; NCAQ RCE 1088 Data Review Identified RCE 1071 Poor Corrective Action Closure Quality

Section 4OA3

AOP 646; Loss of Feedwater Heating; Revision 19
AOP 255.2; Power/Reactivity Abnormal Change; Revision 34
AOP 262 Loss of Reactor Pressure Control; Revision 4
CAP 072125; SCAQ – Both Turbine Bypass Valves Failed Open
LER 2010-001-00; Emergency Diesel Generator Start due to Failed Arrester in Switchyard; dated March 2, 2010

LIST OF ACRONYMS USED

ACE	Apparent Cause Evaluation
ADAMS	Agencywide Documents Access and Management System
AFP	Area Fire Plan
ALARA	As-Low-As-Is-Reasonably-Achievable
ALI	Annual Level of Intake
ANSI	American National Standards Institute
AOP	Abnormal Operating Procedure
ASME	American Society of Mechanical Engineers
CAP	Corrective Action Program
CE	Condition Evaluation
CFR	Code of Federal Regulations
CRS	Control Room/Simulator
CWO	Corrective Work Order
DAEC	Duane Arnold Energy Center
DEP	Drill and Exercise Performance
DRP	Division of Reactor Projects
EAL	Emergency Action Level
EDG	Emergency Diesel Generator
EOF	Emergency Operations Facility
ERO	Emergency Response Organization
ESB	Electronic Status Board
ESW	Emergency Service Water
FSAR	Final Safety Analysis Report
GE	General Emergency
GL	Generic Letter
GPI	Groundwater Protection Initiative
HEPA	High Efficiency Particulate Air
HPCI	High Pressure Coolant Injection
HPT	Health Physics Technologist
HVAC	Heating, Ventilation and Air Conditioning
IMC	Inspection Manual Chapter
IP	Inspection Procedure
JIC	Joint Information Center
IST	Initial Screening Team
LER	Licensee Event Report
LLC	Limited Liability Corporation
LPCI	Low Pressure Core Injection
MG	Motor-Generator
MRC	Management Review Committee
NCV	Non-Cited Violation
NEI	Nuclear Energy Institute
NRC	U.S. Nuclear Regulatory Commission
NRR	Office of Nuclear Reactor Regulations
OBE	Operating Basic Earthquake
ODCM	Offsite Dose Calculation Manual
OI	Operating Instruction
OOS	Out-of-Service
OpESS	Operating Experience Smart Sample
OPR	Operability Evaluation

OSC	Operations Support Center
P&ID	Piping and Instrumentation Diagrams
PARS	Publicly Available Records
PI	Performance Indicator
PI&R	Problem Identification and Resolution
PWO	Preventative Work Order
RCE	Root Cause Evaluation
RCIC	Reactor Core Isolation Cooling
RCS	Reactor Coolant System
REMP	Radiological Environmental Monitoring Program
RETS	Radiological Effluent Technical Specification
RHR	Residual Heat Removal
RPV	Reactor Pressure Vessel
SBDG	Standby Diesel Generator
SBLC	Standby Liquid Control
SDC	Shutdown Cooling
SDP	Significance Determination Process
SRA	Senior Reactor Analyst
SSC	Structures, Systems, and Components
TBV	Turbine Bypass Valve
TI	Temporary Instruction
TLD	Thermoluminescent Dosimeters
TS	Technical Specification
TSC	Technical Support Center
UFSAR	Updated Final Safety Analysis Report
URI	Unresolved Item
UT	Ultrasonic Testing
WO	Work Order

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Sincerely,

/RA/

Kenneth Riemer, Chief
Branch 2
Division of Reactor Projects

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