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

May 12, 2009

Mr. Richard L. Anderson
Vice President
Duane Arnold Energy Center
3277 DAEC Road
Palo, IA 52324-9785

SUBJECT: DUANE ARNOLD ENERGY CENTER NRC INTEGRATED INSPECTION REPORT 05000331/2009002

Dear Mr. Anderson:

On March 31, 2009, 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 2, 2009, 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, two NRC-identified and one self-revealed finding of very low safety significance were identified, two of which involved violations of NRC requirements. However, because of the very low safety significance, and because the issues were entered into your corrective action program, the NRC is treating these issues as non-cited violations (NCVs) in accordance with Section VI.A.1 of the NRC Enforcement Policy. Additionally, a licensee identified violation is listed in Section 4OA7 of this report.

If you contest the subject or severity of these NCVs, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555-0001, with a copy to the Regional Administrator, U.S. Nuclear Regulatory Commission - Region III, 2443 Warrenville Road, Suite 210, Lisle, IL 60532-4352; the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001; and the Resident Inspector Office at the Duane Arnold Energy Center. In addition, if you disagree with the 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 Duane Arnold Energy Center. The information you provide will be considered in accordance with Inspection Manual Chapter 0305.

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

Sincerely,

/RA/

Kenneth Riemer, Chief
Branch 2
Division of Reactor Projects

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

Enclosure: Inspection Report 05000331/2009002
w/Attachment: Supplemental Information

cc w/encl: M. Nazar, Senior Vice President and
Chief Nuclear Officer
M. Ross, Managing Attorney
A. Khanpour, Vice President, Nuclear Engineering
D. Curtland, Plant Manager
S. Catron, Manager, Regulatory Affairs
M. Mashhadi, Senior Attorney
T. Jones, Vice President, Nuclear Operations,
Midwest Region
P. Wells, Vice President, Nuclear Safety Assessment
R. Hughes, Director, Licensing and Performance Improvement
D. McGhee, Iowa Dept. of Public Health
Chairman, Linn County, Board of Supervisors
R. McCabe, Chairman, Regional Assistance Committee,
DHS/FEMA Region VII
M. Rasmusson, State Liaison Officer

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Letter to R. Anderson from K. Riemer dated May 12, 2009

SUBJECT: DUANE ARNOLD ENERGY CENTER NRC INTEGRATED INSPECTION REPORT 05000331/2009002

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REGION III

Docket No:	50-331
License No:	DPR-49
Report No:	05000331/2009002
Licensee:	FPL Energy Duane Arnold, LLC
Facility:	Duane Arnold Energy Center
Location:	Palo, IA
Dates:	January 1, through March 31, 2009
Inspectors:	R. Orlikowski, Senior Resident Inspector R. Baker, Resident Inspector M. Holmberg, Senior Reactor Inspector J. Neurauter, Senior Reactor Inspector T. Go, Health Physicist
Observers:	A. Scarbeary, Reactor Engineer
Approved by:	K. Riemer, Chief Branch 2 Division of Reactor Projects

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SUMMARY OF FINDINGS

IR 05000331/2009002; 01/01/2009 – 03/31/2009; Duane Arnold Energy Center; Operability Evaluations and Follow-up of Events.

This report covers a three-month period of inspection by resident inspectors and announced baseline inspections by regional inspectors. Two Green findings were identified by the inspectors and one Green finding was self-revealed. The inspector-identified findings were considered Non-Cited Violations (NCV) 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). 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 was self-revealed when the operators exceeded the operational limit of the cooling tower riser by failing to secure one of the two running circulating water pumps prior to securing flow to the 'A' cooling tower. The inspectors determined that the operators exceeding the operational limit of the 'B' cooling tower west riser was contrary to the guidance for safe operation of plant equipment contained in Administrative Control Procedure (ACP) 110.1, "Conduct of Operations," and therefore was a performance deficiency. No violation of regulatory requirements occurred. The licensee entered this issue into their corrective action program (CAP) as CAP 063426. The 'B' cooling tower riser was repaired, structural support was added to all four cooling tower risers, and operating procedures were revised to preclude operators from operating two circulating water pumps with only one cooling tower in operation.

  The finding was determined to be more than minor because the finding was associated with the Reactor Safety Cornerstone attribute of procedure quality and affected the cornerstone objective of limiting the likelihood of those events that upset plant stability and challenge critical safety functions during shutdown. Specifically, operating the plant in an inappropriate configuration resulted in the loss of the normal plant heat sink, which required the operators to manually scram the reactor and rely on safety-related equipment to cool the plant down. The inspectors determined the finding was of very low safety significance (Green) because the finding only resulted in a reactor scram and did not contribute to the likelihood that mitigation equipment or functions would not be available. This finding has a cross-cutting aspect in the area of Problem Identification and Resolution, Corrective Action, because the licensee did not take appropriate corrective actions to address safety issues and adverse trends in a timely manner. Specifically, maintenance and operations personnel failed to adequately address a known deficiency with a plugged pressure transmitter, which resulted in the control room allowing throttling of the 'A' cooling tower riser valves until they were fully shut, thus exceeding the operational limit of the cooling tower [P.1(d)]. (Section 4OA3.1)

Cornerstone: Mitigating Systems

- Green. A finding of very low safety significance and associated NCV of Technical Specifications (TSs) was identified by the inspectors for the operators failing to perform required actions for existing limiting condition for operation (LCO) conditions, involving TS equipment declared inoperable, during in-vessel fuel movements. The inspectors determined that the failure to perform TS LCO required actions during in-vessel fuel movement was contrary to Refueling Operations TS required actions and therefore was a performance deficiency. The licensee entered this issue into their corrective action program as CAP 064489. The core alterations were suspended to comply with the TSs until the issue was resolved. Actions were taken to ensure that the control rods with the inoperable rod position indicators were fully inserted and to electrically disarm the control rod drives. Once the required actions were completed, the fuel shuffle was recommenced.

  The performance deficiency was determined to be more than minor because the finding was associated with the Mitigating Systems Cornerstone attribute of human performance and adversely affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences (i.e., core damage). Specifically, when changes to in-plant conditions affect previously performed required actions for equipment declared inoperable, the failure to perform the TS LCO required actions for the new plant conditions could lead to a more significant safety concern by unknowingly exceeding allowed outage times established for specific LCOs. This human error could, in turn, challenge mitigating systems' availability, reliability, and capability to respond to initiating events. The inspectors determined that this finding only degraded the reactivity control function of the Mitigating Systems Cornerstone and only affected the safety of a reactor during refueling operations after the entry conditions had been met and shutdown cooling had been initiated. Using IMC 0609, Appendix G, "Shutdown Operations SDP," and Checklist 7, "BWR Refueling Operation with RCS Level > 23'," contained in Attachment 1, the inspectors determined that the finding did not require a quantitative assessment. Using Figure 1, this finding screened as very low safety significance (Green). The inspectors also determined that this finding has a cross-cutting aspect in the area of Human Performance, Decision Making, because the licensee did not adopt a requirement to demonstrate that the proposed action was safe in order to proceed rather than a requirement to demonstrate that it is unsafe in order to disapprove the action. Specifically, the requirements of RFP-403 and IPOI-8 to verify readiness to commence in-vessel fuel movements did not adequately provide for a review of inoperable TS equipment completed LCO actions to ensure core alteration TSs for reactivity control were met during the fuel movements [H.1(b)]. (Section 1R15.1.b)

Cornerstone: Barrier Integrity

- 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 the failure to verify the adequacy of the methodology and design inputs used to support licensee decisions to accept non-conforming systems, structures, and components for continued operation. The licensee entered this issue into its CAP and was able to demonstrate the Primary Containment system and piping subsystems attached to Drywell penetrations to be operable during design basis accident conditions.

The finding was determined to be more than minor because the omission of a design basis load in engineering evaluations used to justify continued operation resulted in a condition where there was reasonable doubt regarding the operability of the Primary Containment system and piping subsystems attached to Drywell penetrations during accident conditions. The inspectors determined the finding was of very low safety significance because it was a design deficiency that did not result in actual loss of safety function. This finding did not have a cross-cutting aspect. (Section 1R15.2.b)

B. Licensee-Identified Violations

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

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 exceptions:

- On January 18, 2009, fuel cycle coastdown began leading to a planned refueling outage, which began on February 1 following a reactor scram from approximately 45 percent power that occurred during plant shutdown. The refueling outage continued through March 3, with the generator connected to the grid on March 6. Power ascension was completed on March 9, when the plant returned to full power.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, and Barrier Integrity

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’ Standby Diesel Generator (SBDG) with the ‘B’ SBDG Out-of-Service (OOS) for Planned Maintenance to Install Permanent Modifications;
- ‘A’ Core Spray (CS) System with the ‘B’ CS System OOS for Planned Preventive Maintenance; and
- ‘A’ Standby Filter Unit with the ‘B’ Standby Filter Unit OOS for Planned Corrective 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, Updated Final Safety Analysis Report (UFSAR), TS requirements, outstanding work orders, condition reports, and the impact of ongoing work activities on redundant trains of equipment in order to identify conditions that could have rendered the systems incapable of performing their intended functions. The inspectors also walked down accessible portions of the systems to verify system components and support equipment were aligned correctly and operable. The inspectors examined the material condition of the components and observed operating parameters of equipment to verify that there were no obvious deficiencies. The inspectors also verified that the licensee had properly identified and resolved equipment alignment problems that could cause initiating events or impact the capability of mitigating systems or barriers and entered them into the CAP with the appropriate significance characterization. Documents reviewed are listed in the Attachment to this report.

These inspection activities constituted three partial system walkdown samples as defined in Inspection Procedure (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) 20, Turbine Building Aux Boiler Room, Emergency Diesel Generator Rooms, and Generator Day Tank Rooms, Elevation 757' 6";
- AFP 17, Turbine Building Condenser Bay, Heater Bay, and Steam Tunnel, Elevations 734' 0" and 757' 6";
- AFP 25, Control Building Cable Spreading Room, Elevation 772' 6";
- AFP 18 & 19, North Turbine Building Ground Floor, Tube Pulling Area, 1A1 Switchgear Room, and South Turbine Building Ground Floor, Elevation 757' 6"; and
- AFP 8; Reactor Building Standby Gas Treatment System and MG Set Rooms, Elevation 786' 0".

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 had 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, the inspectors verified that fire hoses and extinguishers were in their designated locations and available for immediate use; that fire detectors and sprinklers were unobstructed, that transient material loading was within the analyzed limits; and fire doors, dampers, and penetration seals appeared to be in satisfactory condition. The inspectors also verified that minor issues identified during the inspection were entered into the licensee's CAP. Documents reviewed are listed in the Attachment to this report.

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

b. Findings

No findings of significance were identified.

1R07 Annual Heat Sink Performance (71111.07)

a. Inspection Scope

The inspectors reviewed the licensee's testing of the 'A' Residual Heat Removal (RHR) Heat Exchanger following planned maintenance for cleaning during the refueling outage. The inspectors verified that the as-left conditions did not mask the licensee's ability to detect degraded performance, or to identify any common cause issues that had the potential to increase risk, and 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.

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

b. Findings

No findings of significance were identified.

1R08 Inservice Inspection Activities (71111.08)

From February 8, 2009, through February 11, 2009, the inspectors conducted a review of the implementation of the licensee's Inservice Inspection (ISI) Program for monitoring degradation of the reactor coolant system, risk-significant piping and components and containment systems.

The inspections described in Sections 1R08.1 and 1R08.2 below constituted one inspection sample as defined in IP 71111.08-05.

.1 Piping Systems ISI

a. Inspection Scope

The inspectors observed the following nondestructive examinations mandated by the American Society of Mechanical Engineers (ASME) Section XI Code to evaluate compliance with the ASME Code Section XI and Section V requirements and if any indications were detected, to determine if these were dispositioned in accordance with the ASME Code or an NRC approved alternative requirement.

- Magnetic Particle and Visual Examination (VT-3) of Main Steam Line Pipe Support MSA-HA-1; and
- Visual Examination (VT-3) of Reactor Vessel Stabilizers VSW-0AZ and VSW-180AZ.

The inspectors observed the following nondestructive examination conducted as part of the licensee's augmented inspection program for detection of stress corrosion cracking. The inspectors observed this examination to determine if it was conducted in accordance with the licensee's augmented inspection program basis document - Boiling Water Reactor Vessel Internals Program No. 75a "BWR [Boiling Water Reactor] Vessel and Internals Project Technical Basis for Revisions to Generic Letter 88-01 Inspection Schedules" and the associated nondestructive examination procedure. If any indications or flaws were detected during the examination, the inspectors confirmed that the indications were dispositioned in accordance with approved procedures and NRC requirements.

- Ultrasonic Examination of Reactor Recirculation Weld RRB-F002A.

The inspectors reviewed the following examinations completed during the previous outage with relevant/recordable conditions/indications accepted for continued service to determine if acceptance was in accordance with the ASME Code Section XI or an NRC approved alternative.

- Liquid Penetrant Examination Report PT-07-09, Pipe-to-Pipe Weld CSB-F004; and
- Ultrasonic Examination Report UT-07-033, Safe-End-to-Nozzle Weld FWA-J002.

The inspectors reviewed records of the following pressure boundary welds completed for risk-significant systems during the outage to determine if the licensee applied the pre-service, nondestructive examinations and acceptance criteria required by the Construction Code. Additionally, the inspectors reviewed the welding procedure specification and supporting weld procedure qualification records to determine if the weld procedure was qualified in accordance with the requirements of Construction Code and the ASME Code Section IX.

- Welds W2, W3, W5 for replacement of Main Steam Drain Line Inboard Isolation Valve MO 4423; and
- Welds W1, W2, W3 for replacement of Main Steam Drain Line Outboard Isolation Valve MO 4424.

b. Findings

No findings of significance were identified.

.2 Identification and Resolution of Problems

a. Inspection Scope

The inspectors performed a review of ISI related problems entered into the licensee's CAP and conducted interviews with licensee staff to determine if;

- the licensee had established an appropriate threshold for identifying ISI related problems;
- the licensee had performed a root cause (if applicable) and taken appropriate corrective actions; and
- the licensee had evaluated operating experience and industry generic issues related to ISI and pressure boundary integrity.

The inspectors performed these reviews to evaluate compliance with 10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Action," requirements. The corrective action documents reviewed by the inspectors are listed in the Attachment to this report.

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 January 27, 2009, the inspectors observed a crew of licensed operators in the plant's simulator during just-in-time training activities in preparation for shutdown of the reactor plant for a refueling outage. The inspectors observed the training activities 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. Documents reviewed are listed in the Attachment to this report.

This inspection activity 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:

- Condenser Heat Removal System; and
- Onsite 4160 volt AC Power System.

The inspectors reviewed events such as where ineffective equipment maintenance had resulted in unplanned plant transients and independently verified the licensee's actions to address system performance or condition problems in terms of the following:

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

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

These inspection activities constituted 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:

- RHR/Low Pressure Core Injection (LPCI) Planned Maintenance Work Activities during Work Week 9903;

- Review of Operational Decision Making Instructions (ODMI) for Recirculation Pump Issues;
- Electrical Bus 1B42 Outage Concurrent with Refuel Outage Fuel Shuffle during Work Week 9906;
- Main Turbine Bearing Number 9 Wiped during Startup Forces Plant Shutdown; and
- RHR Logic Functional Test Rescheduled during Work Week 9912.

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)

.1 Operability Evaluations

a. Inspection Scope

The inspectors reviewed the following issues:

- ‘A’ Emergency Service Water (ESW), ‘A’ RHR Service Water (RHRSW) Loop, and ‘A’ SBDG operability following a failure of the ‘A’ ESW/RHRSW pump room ventilation supply fan, 1VSF056A;
- Required actions not performed for planned Control Rod Position Indication LCO conditions, prior to commencing in-vessel fuel movements, following replacements of the control rod position indication probes;
- ‘B’ ESW system operability following discovery that sections of system piping were insulated, but by design should not be insulated; and
- ‘A’ Standby Liquid Control (SBLC) Pump operability following discovery of a pump casing leak during performance of the pump operability surveillance test procedure (STP).

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 also reviewed a sampling of corrective action documents to verify that the licensee was identifying and correcting any deficiencies associated with operability evaluations. Documents reviewed are listed in the Attachment to this report.

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

b. Findings

Introduction: A finding of very low safety significance and associated NCV of TSs was identified by the inspectors for the operators failing to perform TS LCO required actions for existing LCO conditions involving TS equipment declared inoperable during in-vessel fuel movements.

Description: On February 8, 2009, the first fuel shuffle of RFO 21, which involved the in-vessel fuel movements to support control rod drive (CRD) replacements and control rod blade shuffles, was completed. The next day the rod position indications for control rods 14-23 and 22-19 were declared inoperable to support replacement of the position indication probes. The associated TS 3.9.4 LCOs were entered for Condition A and the required actions were met by the control rods being verified fully inserted and electrically disarmed per the clearances and work orders to replace the probes. Additionally, on February 12, due to intermittent 'Full-In' indication, which occurred during under-vessel work in the drywell, control rod 10-11 position indication was declared inoperable and TS 3.9.4 LCO Condition A was entered. The required actions were met by verifying that in-vessel fuel movement and control rod withdrawal were not being performed and all required control rods were fully inserted.

Following replacement of the position indication probes for control rods 14-23 and 22-19, the clearances were removed (controls rods no longer electrically disarmed), but since the required post-maintenance testing to verify that all positions and the 'Full-In' and 'Full-Out' lights operated properly was not scheduled to be performed until subsequent scram time testing was conducted, the rod position indication was not declared operable. The required TS LCO actions were still met because in-vessel fuel movement and control rod withdrawal were not being performed and all required control rods were fully inserted. No further actions were documented. This TS inoperable equipment and the associated TS LCOs continued to be carried forward, tracked in the LCO Notebook, and discussed as shift turnover information.

On February 15, the requirements of Refueling Procedure 403 and IPOI-8 were verified complete and the second fuel shuffle was commenced. Subsequently, during a control room observation on February 17, the inspectors, noting that the TS LCOs for rod position indication of the three control rods was still in effect, asked how the required actions were being met during in-vessel fuel movements. The core alterations were suspended to comply with the TSs until the issue was resolved. The licensee initiated actions to verify the control rods with the inoperable rod position indicators were fully inserted and to electrically disarm the CRDs. Once the required actions were completed, the fuel shuffle was recommenced.

Analysis: The inspectors determined that the failure to perform the required TS LCO actions during in-vessel fuel movements, for TS equipment declared inoperable, was contrary to the TS section for Refueling Operations, was reasonably within the licensee's ability to foresee, correct, and prevent, and was therefore a performance deficiency.

This performance deficiency did not meet any of the conditions requiring traditional enforcement, was not similar to any of the minor examples of IMC 0612 Appendix E, and was therefore compared to the questions in IMC 0612 Appendix B, "Issue Screening." The inspectors determined the performance deficiency to be more than minor because the finding was associated with the Mitigating Systems Cornerstone attribute of human performance and adversely affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences (i.e., core damage). Specifically, when changes to in-plant conditions affect previously performed required actions for equipment declared inoperable, the failure to perform the TS LCO required actions for the new plant conditions could lead to a more significant safety concern by unknowingly exceeding allowed outage times established for specific LCOs. This human error could, in turn, challenge mitigating systems' availability, reliability, and capability to respond to initiating events.

Using Attachment 4 of IMC 0609, "Significance Determination Process," the inspectors determined that the finding only degraded the reactivity control function of the Mitigating Systems Cornerstone and only affected the safety of a reactor during refueling operations after the entry conditions had been met and shutdown cooling had been initiated. Hence, the finding could be evaluated in accordance with IMC 0609, Appendix G, "Shutdown Operations SDP." The inspectors used Checklist 7, "BWR Refueling Operation with RCS Level > 23'," contained in Attachment 1 and determined that the guidelines for reactivity control, which specifically "assumes existing core alteration TSs are being met," was adversely affected. However, the finding did not require a phase 2 or phase 3 analysis because the plant had appropriately met the safety function guidelines for core heat removal and inventory control and the finding did not involve a loss of control associated with inadvertent RCS pressurization or inadvertent loss of 2' of RCS inventory. Using Figure 1, this finding does not require a quantitative assessment and therefore screened as very low safety significance (Green).

The inspectors determined that this finding has a cross-cutting aspect in the area of human performance for decision making because the licensee did not adopt a requirement to demonstrate that the proposed action was safe in order to proceed rather than a requirement to demonstrate that it is unsafe in order to disapprove the action. Specifically, the requirements of RFP-403 and IPOI-8 to verify readiness to commence in-vessel fuel movements did not adequately provide for a review of inoperable TS equipment completed LCO actions to ensure core alteration TSs for reactivity control were met during the fuel movements. [H.1(b)]

Enforcement: Technical Specification Section 3.9, Refueling Operations, LCO 3.9.4, Control Rod Position Indication, requires that during Mode 5, "The control rod 'full-in' position indication for each control rod shall be Operable." Condition A states that for "One or more required control rod position indications inoperable," the licensee will

"Immediately," "(A.1.1) Suspend in-vessel fuel movement; AND (A.1.2) Suspend control rod withdrawal. AND (A.1.3) Initiate action to fully insert all insertable control rods in core cells containing one or more fuel assemblies; OR (A.2.1) Initiate action to fully insert the control rod associated with the inoperable position indicator; AND (A.2.2) Initiate action to disarm the control rod drive associated with the fully inserted control rod."

Contrary to the above, between February 15, and February 17, core alterations were conducted without performing the TS 3.9.4 LCO Condition A required actions for three control rod position indications declared inoperable. Specifically, while in-vessel fuel movements were suspended during CRD exchanges and Control Rod Blade shuffles, the required actions for LCOs entered for control rods 10-11, 14-23, and 22-19 were complete. However, once in-vessel fuel movements were recommenced, without completing actions A.2.1 & A.2.2, the allowed outage times for the LCO actions were exceeded. Because this violation was of very low safety significance and it was entered into the licensee's corrective action program as CAP 064489, this violation may be dispositioned as an NCV, consistent with Section VI.A.1 of the NRC Enforcement Policy. (NCV 05000331/2009002-03)

.2 <u>(Closed) Unresolved Item (URI) 05000331/2005002-02: Failure to Include the Analysis of Thermal Movements in Piping Modifications</u>

a. <u>Inspection Scope</u>

The inspectors reviewed licensee corrective actions pertaining to URI 05000331/2005002-02. Specifically, the inspectors reviewed licensee documentation that included licensee corrective actions following identification that Drywell thermal movement had not been incorporated into the design basis analysis for the Containment Vent Purge Exhaust piping subsystem 18"-HLE-023. The licensee documentation reviewed included extent of condition, operability determinations, and plant modifications to restore piping subsystems to compliance with the design basis requirements.

Specific documents reviewed during the inspection are listed in the Attachment to this report.

This inspection did not constitute an inspection sample.

b. <u>Findings</u>

<u>Failure to Consider Design Basis Load in Evaluation for Continued Operation</u>

<u>Introduction</u>: 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 the failure to verify the adequacy of the methodology and design inputs used to support licensee decisions to accept non-conforming piping subsystems and Drywell penetrations for continued operation.

<u>Description</u>: On March 18, 2005, the licensee identified that calculation CAL-080-312 for piping subsystem 18"-HLE-023, "Containment Vent Purge Exhaust," did not account for the thermal movement of the Drywell in the analytical stress model. Drywell thermal anchor movements (TAMs) were determined in calculation CAL-003C-F-010,

"Penetration Movement Thermal," generated in 1971. The licensee further identified that calculation CAL-082-312 for piping subsystem ¾"-HLE-023 test line installed under DCR1167 did not account for thermal movement of the Drywell, this test line was attached to the 18"–HLE-023 subsystem, and that installation of this test line included a rigid vertical guide and an anchor piping supports. The licensee initiated CAP 035317 to enter the concern into their CAP. Immediate actions taken by the licensee included a walkdown of the identified piping installations, an operability determination to address functional capability of the identified piping subsystems, and a determination of the likely cause for not evaluating the effect of Drywell thermal movement in the identified piping calculations.

The licensee performed an immediate assessment of the functional capability of the piping subsystem using guidance from NRC Inspection Manual Part 9900, "Operable/Operability: Ensuring the Functional Capability of a System or Component," that accepted criteria in Appendix F of Section III of the ASME Boiler and Pressure Vessel Code. Noting Paragraph F-1310(c), "Only limits on primary stresses are prescribed. Thermal stresses resulting from Level D Service limits need not be considered," the licensee concluded, in-part, that "the non-conformance of not accounting for thermal movement of the Drywell is not an operability concern per Appendix F of Section III of the ASME Code."

The licensee further identified that additional plant modifications installed rigid piping supports to other piping subsystems that were attached to the Drywell. As a part of CAP 035317, the licensee entered an action to perform a condition evaluation, CE 002404 dated March 22, 2005, to determine the extent of piping calculations that did not evaluate the effect of Drywell thermal movement. Based on the results of CE 002404, the licensee initiated an additional corrective action, CA 040021 dated March 30, 2005, to review the remaining Drywell penetrations. Pending the licensee's reviews for extent of condition and overall effect on existing designs, the inspectors considered the issue as unresolved (refer to NRC Integrated Inspection Report 05000331/2005002, dated April 29, 2005, ADAMS Accession Number ML051240313).

Based on a subset of Drywell penetrations reviewed as part of CE 002404, on March 30, 2005, the licensee initiated CA 040021 to review the remaining Drywell penetrations for the thermal anchor movement concern.

Following the completion of their extent of condition review, the licensee documented actions to correct the condition in CA 040134, initiated on April 22, 2005: install modifications for 15 Drywell penetrations in the 2005 refueling outage and complete analysis for 16 Drywell penetrations that required follow-up actions. The licensee also documented that 22 Drywell penetrations required no follow-up actions.

The inspectors reviewed the licensee's operability determination documented in CAP 035317, and noted that secondary stresses (the basic characteristic of a secondary stress is that it is self-relieving) due to Drywell movement were only applicable to the attached piping; stresses induced in the piping supports due to Drywell movement were primary stresses (the basic characteristic of a primary stress is that it is not self-relieving). Therefore, the inspectors concluded that the correct ASME Code classification for stresses induced into piping supporting structures due to Drywell thermal expansion was "primary" stress, and the licensee's operability determination performed under CAP 035317 was non-conservative because the impact due to Drywell

TAM was not considered. In addition, since the licensee had not demonstrated that the affected piping system supporting structures were in compliance with the requirements of Appendix F of ASME Section III, the inspectors concluded that the licensee had not demonstrated the operability of SSCs affected by Drywell thermal movement during design basis accident conditions.

In DAEC letter NG-05-2178, G. Van Middlesworth (Site Vice-President, DAEC) to the NRC, Subject: Additional Information Regarding Unresolved Item 05000331/2005002-02, dated December 12, 2005, the Nuclear Management Company (NMC, the former licensee) provided NRC staff with additional information regarding the DAEC's determination of past operability of Drywell penetrations due to TAM. In the enclosure to DAEC letter NG-05-2178, the licensee described the decision process and steps taken for the operability determination performed under CAP 035317:

- Operability was assessed using NRC Inspection Manual Part 9900 guidance regarding operability for non-conforming conditions in piping systems. Part 9900 guidance states licensee may use the criteria in Appendix F of the ASME Code for operability decisions.
- For the evaluation of operability, the licensee used Appendix F of Section III of the ASME Code (1977 Edition/1978 Summer Addenda). This edition/addenda was the current code of record for DAEC Primary Containment.
- Paragraph F-1310(c) stated that "Only limits on primary stresses are prescribed. Thermal stresses resulting from Level D Service Limits need not be considered." Paragraph F-1370 (Component Supports) does not require consideration of thermally-induced stresses.
- The evaluation documented in CAP 035317 concluded the non-conformance (not accounting for thermal movement of the Drywell on the vent line) was not an operability concern using Part 9900 guidelines and the DAEC code of record.
- The licensee noted that later versions of Appendix F of the ASME Code (e.g., 1989) would require consideration as primary stresses, those from the constraint of free end displacement and anchor point motion, in the evaluation of component supports.
- NMC obtained an opinion from an external peer, recognized as knowledgeable in the ASME Code, regarding which ASME Code Edition/Addenda to use when evaluating operability using Part 9900 guidelines. The external peer concluded that the use of Appendix F from the 1977 code with summer 1978 addenda (DAEC's code of record) was acceptable for the use in the NMC's operability determination.
- NMC concluded that the operability determination performed under CAP 035317 was valid.

On December 14, 2005, the licensee initiated CAP 039338 that identified existing support configurations on HLE-21 and HLE-38 that are connected to drywell penetration X-22 required modification to accommodate thermal movement of the Drywell. Furthermore, the licensee determined these piping subsystems to be operable based on the evaluation performed and documented in CAP 035317, i.e., thermal stresses resulting from Level D Service Limits need not be considered in accordance with Paragraph F-1310(c) of Appendix F of Section III of the ASME Code.

In January 2006, the inspectors reviewed the enclosure to DAEC letter NG-05-2178 that described the licensee's decision process and steps taken for the operability determination performed under CAP 035317. The inspectors further reviewed technical guidance provided in NRC Inspection Manual Part 9900 dated September 26, 2005, with respect to the information the licensee provided in DAEC letter NG-05-2178. The inspectors noted:

- Part 9900, paragraph 3.4, defined an SSC as "not fully qualified," i.e., degraded or non-conforming, when it did not conform to all aspects of its current licensing basis, including all applicable codes and standards, design criteria, safety analyses, assumptions and specifications, and licensing commitments.
- The DAEC UFSAR, Section 3.8, "Design Criteria of Seismic Category I Structures," Revision 12 dated October 1995, indicated the design code applicable to Primary Containment (including penetrations) as ASME, Section III, Class B [now ASME Section III Subsection NE (Class MC Components)].
- The DAEC UFSAR defined a design basis load, "force on structure from the thermal expansion of pipes under accident conditions," (H_A), applicable to Primary Containment.
- The inspectors did not identify in Article 3000, "Design," of Subsection NE of Section III of the ASME Code (1977 Edition/Summer 1978 Addenda) a paragraph that indicated attached piping "constrained free-end displacement and differential support motion effects need not be considered" as in paragraph NF-3231.1.

Since the force on Primary Containment (including penetrations) due to thermal expansion of pipes under accident conditions was a design basis load at the DAEC, the inspectors concluded that the effect Drywell thermal expansion needed to be included in the licensee's determination of SSC operability as directed in NRC Inspection Manual Part 9900 inspector guidance. Therefore, the inspectors concluded that the licensee's operability determination documented in CAP 035317 did not satisfy inspector technical guidance in NRC Inspection Manual Part 9900.

In February 2006, the licensee initiated prompt operability determinations that evaluated the effects of TAM and pressure anchor movement (PAM) for the 16 Drywell penetrations requiring follow-up actions after the spring 2005 refueling outage (CA040134). The licensee determined the affected piping subsystems and Drywell penetrations to be operable but non-conforming.

In DAEC letter NG-06-0305 to the NRC, Subject: Withdrawal of NG-05-2178, dated April 3, 2006, the current licensee, FPL Energy Duane Arnold, LLC,, withdrew its position regarding the use of ASME Section III, Appendix F for the determination of past operability of the DAEC Primary Containment. This letter also indicated that FPL Duane Arnold, LLC would perform new past operability evaluations for those penetrations modified during the spring 2005 outage. In DAEC letter NG-06-0375 to the NRC, Subject: Voluntary Licensee Event Report No. 2006-002-00, dated June 1, 2006, the licensee submitted Voluntary Licensee Event Report (LER) No. 2006-002-00, "Drywell Penetrations Calculations Do Not Account for Thermal Movement" that committed to complete actions to determine if any past operability concerns existed for the 15 Drywell penetrations that were modified in the 2005 refueling outage. In addition to evaluating

TAM, LER No. 2006-002-00 identified that the past operability determinations would also evaluate the effect of Drywell PAM, also not evaluated in the operability determination for the Containment Vent Purge Exhaust line documented in CAP 033317.

The inspectors reviewed the licensee's corrective actions and new past operability evaluations for the 15 Drywell penetrations modified in the 2005 refueling outage. The inspectors reviewed the evaluations to ensure the new operability determinations considered the effect of both Drywell TAM and PAM at accident conditions. The inspectors identified that one operability determination, calculation IE-P108274-610, "Operability Evaluation of Containment Atmosphere Control Piping @ Penetration X-25," Revision 1, inadvertently evaluated piping load combinations that did not include the effect of Drywell movement. The licensee entered this condition into the corrective action program, CAP 057980, initiated corrective action CA 050263, and revised calculation IE-P108274-610 to correct the error. The inspectors reviewed Revision 2 of calculation IE-P108274-610 to ensure the effect of both TAM and PAM was evaluated.

In summary, the licensee completed evaluations related to the operability of piping subsystems and Drywell penetrations affected by Drywell movement during design basis accident conditions. The licensee's extent of condition evaluated all Drywell penetrations. No operability concern related to Drywell movement during design basis accident conditions was identified.

<u>Analysis</u>: The inspectors determined that licensee's failure to consider a design basis load in evaluations of a non-conforming condition to justify continued operation was a performance deficiency. The issue was determined to be more than minor because this performance deficiency also impacted the Barrier Integrity Cornerstone objective to provide reasonable assurance that physical design barriers (containment) protect the public from radionuclide releases caused by accidents or events. Specifically, the omission of the design basis load resulted in a condition where there was reasonable doubt regarding the operability of the Primary Containment and piping subsystems attached to Drywell penetrations during accident conditions.

The inspectors determined the finding could be evaluated using the SDP in accordance with IMC 0609, "Significance Determination Process," Attachment 0609.04, "Phase 1 - Initial Screening and Characterization of Findings," Table 4a for the Barrier Integrity Cornerstone. Specifically, since all four questions under the Containment Barrier column were answered "no," the finding was determined to be Green, of very low safety significance, because it did not represent an actual open pathway in the physical integrity of reactor containment.

This finding did not have a cross-cutting aspect because the cause of the performance deficiency is not reflective of current licensee performance.

<u>Enforcement</u>: Title 10 CFR Part 50, Appendix B, Criterion III, "Design Control," requires, in part, that measures shall be established to assure that applicable regulatory requirements and the design basis are correctly translated into specifications, drawings, procedures, and instructions, and that design control measures provide for verifying or checking the adequacy of design, such as by the performance of design reviews, by the use of alternate or simplified calculation methods, or by the performance of a suitable testing program.

Contrary to the above, on March 18, 2005, and December 14, 2005, the licensee's design control measures failed to verify the adequacy of the design of the non-conforming piping systems attached to the Drywell, in that, the methodology and design inputs used did not include a design basis load, Drywell movement during design basis accident conditions, in engineering evaluations used to justify continued operation. Consequently, the licensee incorrectly concluded that the Primary Containment (Drywell) system and piping subsystems attached to Drywell penetrations satisfied ASME Code requirements for Service Level D loadings. As a result, non-conforming piping subsystems attached to 31 Drywell penetrations were left in-service from March 18, 2005, to the spring 2005 refueling outage, and non-conforming piping subsystems attached to 16 Drywell penetrations were left in-service from the spring 2005 refueling outage to February 2006, without an adequate basis to justify continued operation.

Because this violation was of very low safety significance and it was entered into the licensee's corrective action program, CAP042817 to CAP035317, this violation is being treated as an NCV, consistent with Section VI.A.1 of the NRC Enforcement Policy (NCV 05000331/2009002-04).

Based on the above discussion, URI 05000331/2005002-02 is closed.

1R18 Plant Modifications (71111.18)

.1 Temporary Plant Modifications

a. Inspection Scope

The inspectors reviewed the following temporary modification:

- Engineering Change Package (ECP) 1833, 354 Degree Feedwater Sparger Repair.

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 systems. 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 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.

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

b. Findings

No findings of significance were identified.

.2 Permanent Plant Modifications

a. Inspection Scope

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

- ECP-1865, Refuel Bridge Modification to Connect Air Supply to Instrument Air System.

This modification added a hose reel to connect air for the refueling platform to instrument air, and a connection to the refueling bridge which would allow for a temporary hose to be connected if both the air compressor (1K202) and the hose reel were to fail.

- ECP-1835, 'B' SBDG Voltage Regulator Modification, and ECP-1748, 'B' SBDG Governor Replacement.

This modification replaced the 'B' SBDG governor and the 'B' SBDG voltage regulators to resolve operable but degraded condition OBD 258, "Calculation CAL-E02-003 Shows SBDG Voltage Dips less than UFSAR/RG 1.9 Requirements."

These documents and related documentation were reviewed for adequacy of the associated 10 CFR 50.59 safety evaluation screening, consideration of design parameters, implementation of the modification, post-modification testing, and relevant procedures, design, and licensing documents were properly updated. The inspectors observed ongoing and completed work activities to verify that the installations were consistent with the design control documents.

These inspection activities constituted two permanent plant modification samples 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:

- Refuel Bridge Testing Following Replacement of a Failed Control Joystick;
- 'B' SBDG Governor/Voltage Regulator Tuning and Testing Following Modification Installations;

- ‘A’ Inboard Main Steam Isolation Valve (MSIV) Leak Rate Testing Following Seat and Disc Repair Maintenance;
- Hydraulic Control Unit (HCU) 34-15 and 10-19 Leakage Testing Following Accumulator Rebuilding and Dragon Valve Replacement;
- Main Turbine Overspeed Testing Following Outage Maintenance; and
- Feedwater Level Control Unit Testing Following Replacement of the FY4450F Compensation Module in the Feedwater Level Control System.

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.

These inspection activities constituted six post-maintenance testing samples as defined in IP 71111.19-05.

b. <u>Findings</u>

No findings of significance were identified.

1R20 <u>Outage Activities</u> (71111.20)

.1 <u>Refueling Outage Activities</u>

a. <u>Inspection Scope</u>

The inspectors reviewed the licensee’s Outage Risk Plan (ORP) and contingency plans for refueling outage (RFO) 21, conducted February 1, through March 3, 2009, to confirm that the licensee had appropriately considered risk, industry experience, and previous site-specific problems in developing and implementing a plan that assured maintenance of defense-in-depth. During the RFO, the inspectors observed portions of the shutdown and cooldown processes and monitored licensee controls over the outage activities listed below. Documents reviewed during the inspection are listed in the Attachment to this report.

The inspectors observed all or portions of the following activities:

- Licensee configuration management, including maintenance of defense-in-depth commensurate with the ORP for key safety functions and compliance with the applicable TS when taking equipment out-of-service.
- Implementation of clearance activities and confirmation that tags were properly hung and equipment appropriately configured to safely support the work or testing.
- Installation and configuration of reactor coolant pressure, level, and temperature instruments to provide accurate indication, accounting for instrument error.
- Controls over the status and configuration of electrical systems to ensure that TS and ORP requirements were met, and controls over switchyard activities.
- Monitoring of decay heat removal processes, systems, and components.
- Controls to ensure that outage work was not impacting the ability of the operators to operate the spent fuel pool cooling system.
- Reactor water inventory controls including flow paths, configurations, and alternative means for inventory addition, and controls to prevent inventory loss.
- Controls over activities that could affect reactivity.
- Maintenance of secondary containment as required by TS.
- Refueling activities, including fuel handling and sipping to detect fuel assembly leakage.
- Touring plant areas normally not accessible during power operations for evidence of leakage and integrity of structures, systems, and components, this included a walkdown of the drywell (primary containment) as soon as reasonably possible following shutdown.
- Verify that fuel assemblies were loaded in the reactor core locations specified by the design.
- Startup and ascension to full power operation, tracking of startup prerequisites, walkdown of the drywell to verify that debris had not been left which could block emergency core cooling system suction strainers, and reactor physics testing.
- Licensee identification and resolution of problems related to RFO 21 activities.

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

b. Findings

No findings of significance were identified.

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:

- STP 3.5.3-02, RCIC Operability Test (inservice testing);
- STP 3.6.1.3-03, MSIV Trip/Closure Time Check (inservice testing);
- STP 3.8.1-07B, ‘B’ EDG Loss of Offsite Power (LOOP)/ Loss of Coolant Accident (LOCA) Test (routine);
- STP 3.6.1.1-13, LLRT HPCI/RCIC Valves (containment isolation valve);

- STP 3.8.1-07, ‘A’ EDG LOOP/LOCA Test (routine); and
- STP 3.3.8.1-05B, 1A4 4KV Emergency Transformer Supply Undervoltage Relay Calibration (routine).

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

- did preconditioning occur;
- were the effects of the testing adequately addressed by control room personnel or engineers prior to the commencement of the testing;
- were acceptance criteria clearly stated, demonstrated operational readiness, and consistent with the system design basis;
- plant equipment calibration was correct, accurate, and properly documented;
- as-left setpoints were within required ranges; and the calibration frequencies 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, ASME code, and reference values were consistent with the system design basis;
- where applicable, test results not meeting acceptance criteria were addressed with an adequate operability evaluation or the system or component was declared inoperable;
- where applicable for safety-related instrument control surveillance tests, reference setting data were accurately incorporated in the test procedure;
- where applicable, actual conditions encountering high resistance electrical contacts were such that the intended safety function could still be accomplished;
- prior procedure changes had not provided an opportunity to identify problems encountered during the performance of the surveillance or calibration test;
- equipment was returned to a position or status required to support the performance of its safety functions; and
- all problems identified during the testing were appropriately documented and dispositioned in the CAP.

Documents reviewed are listed in the Attachment to this report.

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

b. Findings

No findings of significance were identified.

2. RADIATION SAFETY

Cornerstones: Occupational Radiation Safety and Public Radiation Safety

2OS1 Access Control to Radiologically Significant Areas (71121.01)

.1 Review of Licensee Performance Indicators for the Occupational Exposure Cornerstone

a. Inspection Scope

The inspectors reviewed the licensee's Occupational Exposure Control Cornerstone performance indicator (PI) to determine whether the conditions resulting in any PI occurrences had been evaluated and whether identified problems had been entered into the licensee's CAP for resolution.

This inspection constitutes one sample as defined in IP 71121.01-05.

b. Findings

No findings of significance were identified.

.2 Plant Walkdowns and Radiation Work Permit Reviews

a. Inspection Scope

The inspectors reviewed licensee controls and surveys in the following radiologically significant work areas within radiation areas, high radiation areas, and airborne radioactivity areas in the plant to determine if radiological controls including surveys, postings, and barricades were acceptable:

- Remove and replace control rod drives;
- Underwater diving work and setup: desludging, inspection and repair of torus coating in a high radiation area;
- In-service examinations on pipes, vessels, snubbers and supports, and flow accelerated corrosion (FAC) exam on pipes in the drywell;
- Refuel outage support work at 360-platform areas; and
- Diving in the torus and reactor cavity.

This inspection constitutes one sample as defined in IP 71121.01-05.

The inspectors reviewed the radiation work permits (RWPs) and work packages used to access these areas and other high radiation work areas. The inspectors assessed the work control instructions and control barriers specified by the licensee. Electronic dosimeter alarm set points for both integrated dose and dose rate were evaluated for conformity with survey indications and plant policy. The inspectors interviewed workers to verify that they were aware of the actions required if their electronic dosimeters noticeably malfunctioned or alarmed.

This inspection constitutes one sample as defined in IP 71121.01-05.

The inspectors walked down and surveyed (using an NRC survey meter) these areas to verify that the prescribed RWP, procedure, and engineering controls were in place; that licensee surveys and postings were complete and accurate; and that air samplers were properly located.

This inspection constitutes one sample as defined in IP 71121.01-05.

The inspectors reviewed RWPs for airborne radioactivity areas to verify barrier integrity and engineering controls performance (e.g., high-efficiency particulate air ventilation system operation) and to determine if there was a potential for individual worker internal exposures in excess of 50 millirem committed effective dose equivalent. During the inspection period there were no airborne radioactivity work areas. Work areas having a history of, or the potential for, airborne transuranics were evaluated to verify that the licensee had considered the potential for transuranic isotopes and had provided appropriate worker protection.

This inspection constitutes one sample as defined in IP 71121.01-05.

The inspectors assessed the adequacy of the licensee's internal dose assessment process for internal exposures in excess of 50 millirem committed effective dose equivalent, and there were no internal exposure greater than 50 millirem committed effective dose equivalent.

This inspection constitutes one sample as defined in IP 71121.01-05.

b. Findings

No findings of significance were identified.

.3 Problem Identification and Resolution

a. Inspection Scope

The inspectors reviewed a sample of the licensee's self-assessments, audits, LERs, and Special Reports related to the access control program to verify that identified problems were entered into the CAP for resolution.

This inspection constitutes one sample as defined in IP 71121.01-05.

The inspectors reviewed corrective action reports related to access controls and any high radiation area radiological incidents (issues that did not count as PI occurrences identified by the licensee in high radiation areas less than 1R/hr). Staff members were interviewed and corrective action documents were reviewed to verify that follow-up activities were being conducted in an effective and timely manner commensurate with their importance to safety and risk based on the following:

- initial problem identification, characterization, and tracking;
- disposition of operability/reportability issues;
- evaluation of safety significance/risk and priority for resolution;

- identification of repetitive problems;
- identification of contributing causes;
- identification and implementation of effective corrective actions;
- resolution of NCVs tracked in the corrective action system; and
- implementation/consideration of risk-significant operational experience feedback.

This inspection constitutes one sample as defined in IP 71121.01-05.

b. Findings

No findings of significance were identified.

.4 Job-In-Progress Reviews

a. Inspection Scope

The inspectors observed the following four jobs that were being performed in radiation areas, airborne radioactivity areas, or high radiation areas for observation of work activities that presented the greatest radiological risk to workers:

- Boilermaker support for N2 nozzle activities in the drywell;
- In-service examinations on pipes, vessels, snubbers and supports, and FAC exam on pipes in the drywell;
- Refuel outage support work at 360-platform areas; and
- Diving in the torus and reactor cavity.

The inspectors reviewed radiological job requirements for these activities, including RWP requirements and work procedure requirements and attended As-Low-As-Is-Reasonably Achievable (ALARA) job briefings.

This inspection constitutes one sample as defined in IP 71121.01-05.

Job performance was observed with respect to the radiological control requirements to assess whether radiological conditions in the work area were adequately communicated to workers through pre-job briefings and postings. The inspectors evaluated the adequacy of radiological controls, including required radiation, contamination, and airborne surveys for system breaches; radiation protection job coverage, including any applicable audio and visual surveillance for remote job coverage; and contamination controls.

This inspection constitutes one sample as defined in IP 71121.01-05.

The inspectors reviewed radiological work in high radiation work areas having significant dose rate gradients to evaluate whether the licensee adequately monitored exposure to personnel and to assess the adequacy of licensee controls. These work areas involved areas where the dose rate gradients were severe; thereby increasing the necessity of providing multiple dosimeters or enhanced job controls.

This inspection constitutes one sample as defined in IP 71121.01-05.

b. Findings

No findings of significance were identified.

.5 High Risk Significant, High Dose Rate, High Radiation Area and Very High Radiation Area Controls

a. Inspection Scope

The inspectors held discussions with the Radiation Protection Manager concerning high dose rate, high radiation area and very high radiation area controls and procedures, including procedural changes that had occurred since the last inspection, in order to assess whether any procedure modifications substantially reduced the effectiveness and level of worker protection.

This inspection constitutes one sample as defined in IP 71121.01-05.

The inspectors discussed with radiation protection supervisors the controls that were in place for special areas of the plant that had the potential to become very high radiation areas during certain plant operations. The inspectors assessed if plant operations required communication beforehand with the radiation protection group, so as to allow corresponding timely actions to properly post and control the radiation hazards.

This inspection constitutes one sample as defined in IP 71121.01-05.

The inspectors conducted plant walkdowns to assess the posting and locking of entrances to high dose rate high radiation areas and very high radiation areas.

This inspection constitutes one sample as defined in IP 71121.01-05.

b. Findings

No findings of significance were identified

.6 Radiation Worker Performance

a. Inspection Scope

During job performance observations, the inspectors evaluated radiation worker performance with respect to stated radiation safety work requirements. The inspectors evaluated whether workers were aware of any significant radiological conditions in their workplace, of the RWP controls and limits in place, and of the level of radiological hazards present. The inspectors also observed worker performance to determine if workers accounted for these radiological hazards.

This inspection constitutes one sample as defined in IP 71121.01-05.

b. Findings

No findings of significance were identified.

.7 Radiation Protection Technician Proficiency

a. Inspection Scope

During job performance observations, the inspectors evaluated radiation protection technician performance with respect to radiation safety work requirements. The inspectors evaluated whether technicians were aware of the radiological conditions in their workplace, the RWP controls and limits in place, and if their performance was consistent with their training and qualifications with respect to the radiological hazards and work activities.

This inspection constitutes one sample as defined in IP 71121.01-05.

b. Findings

No findings of significance were identified.

2OS2 As-Low-As-Is-Reasonably-Achievable (ALARA) Planning And Controls (71121.02)

.1 Inspection Planning

a. Inspection Scope

The inspectors reviewed plant collective exposure history, current exposure trends, and ongoing and planned activities in order to assess current performance and exposure challenges. The inspectors reviewed the plant's current 3-year rolling average for collective exposure in order to help establish resource allocations and to provide a perspective of significance for any resulting inspection finding assessment.

This inspection constituted one required sample as defined in IP 71121.02-05.

The inspectors reviewed the outage work scheduled during the inspection period and associated work activity exposure estimates for the following work activities, which were likely to result in the highest personnel collective exposures:

- Boilermaker support for N2 nozzle activities in the drywell;
- In-service examinations on pipes, vessels, snubbers and supports, and FAC exam on pipes in the drywell;
- Refuel outage support work at "360-platform" in the refuel floor areas; and
- Diving in the torus and reactor cavity.

This inspection constituted one required sample as defined in IP 71121.02-05.

b. Findings

No findings of significance were identified.

.2 Job Site Inspections and ALARA Control Inspection Scope

The inspectors observed the following jobs that were being performed in radiation areas, airborne radioactivity areas, or high radiation areas to evaluate work activities that presented the greatest radiological risk to workers:

- Boilermaker support for N2 nozzle activities in the drywell;
- In-service examinations on pipes, vessels, snubbers, and supports, and FAC exam on pipes in the drywell;
- Refuel outage support work at 360-platform areas; and
- Diving in the torus and reactor cavity.

The inspectors reviewed the licensee's use of ALARA controls for the work activities. The licensee's use of engineering controls to achieve dose reductions was evaluated to verify that procedures and controls were consistent with the licensee's ALARA reviews, that sufficient shielding of radiation sources was provided, and that the dose expended to install/remove the shielding did not exceed the dose reduction benefits afforded by the shielding.

This inspection constituted one required sample as defined in IP 71121.02-05.

.3 Problem Identification and Resolutions

a. Inspection Scope

The inspectors reviewed the licensee's CAP to determine if repetitive deficiencies and/or significant individual deficiencies in problem identification and resolution had been addressed.

This inspection constituted one required sample as defined in IP 71121.02-05.

b. Findings

Introduction: No findings of significance were identified. However, a URI was identified during the inspector's review of a CAP record that described a radiological contamination of a contractor on February 6, 2009. This item will be resolved pending review by the NRC.

Description: During the inspector's 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 reactor pressure vessel (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.

The licensee's management instructed the contractor on the importance of refuel floor personnel to adhere to the specific refuel floor procedure and RWP requirements. The refuel floor procedure and the RWP requirements stated that radiation protection staff must be notified and agree upon any deviation from the job scope prior to continuing with the evolution.

The licensee's initial review of the incident appeared to indicate that the work instructions had been clearly provided to the worker. Based on the information provided by the licensee and additional information obtained during the inspection, this issue remains under review by the NRC and is categorized as a URI, (URI 05000331/2009002-01).

2PS3 Radiological Environmental Monitoring Program And Radioactive Material Control Program (71122.03)

.1 Inspection Planning

a. Inspection Scope

The inspectors reviewed the 2007 Annual Environmental Monitoring Report, sample results obtained in 2008, and licensee assessment results to verify that the Radiological Environmental Monitoring Program (REMP) was implemented as required by TSs and the offsite dose assessment manual (ODAM). The inspectors reviewed the report for changes to the ODAM with respect to environmental monitoring, commitments in terms of sampling locations, monitoring and measurement frequencies, land use census, interlaboratory comparison program, and analysis of data. The inspectors reviewed the ODAM to identify environmental monitoring stations and reviewed licensee self-assessments, audits, LERs, and inter-laboratory comparison program results. The inspectors reviewed the UFSAR for information regarding the environmental monitoring program and meteorological monitoring instrumentation. The inspectors reviewed the scope of the licensee's audit program to verify that it met the requirements of 10 CFR 20.1101(c).

This radiological environmental monitoring program inspection planning constituted one sample as defined in IP 71122.03-05.

b. Findings

No findings of significance were identified.

.2 Onsite Inspection

a. Inspection Scope

The inspectors walked-down 20 percent of the air sampling stations and approximately 10 percent of the thermoluminescence dosimeter (TLD) monitoring stations to determine whether they are located as described in the ODAM and to determine the equipment material condition.

This radiological environmental monitoring program and radioactive material control program onsite equipment location and equipment material condition inspection constituted one sample as defined in IP 71122.03-05.

The inspectors observed the collection and preparation of a variety of environmental samples (e.g., ground and surface water, milk, vegetation, sediment, and soil) and verified that environmental sampling was representative of the release pathways as specified in the ODAM and that sampling techniques were in accordance with procedures.

This environmental sample collection and preparation inspection constituted one sample as defined in IP 71122.03-05.

The inspectors verified that the meteorological instruments were operable, calibrated, and maintained in accordance with guidance contained in the UFSAR, NRC Safety Guide 23, and licensee procedures. The inspectors verified that the meteorological data readout and recording instruments in the control room and at the tower were operable. The inspectors compared readout data (i.e., wind speed, wind direction, and delta temperature) in the control room and at the meteorological tower to identify if there were any line loss differences.

This meteorological instruments inspection constituted one sample as defined in IP 71122.03-05.

The inspectors reviewed each event documented in the Annual Environmental Monitoring Report, which involved a missed sample, inoperable sampler, lost TLD, or anomalous measurement for the cause and corrective actions and conducted a review of the licensee's assessment of any positive sample results (i.e., licensed radioactive material detected above the lower limits of detection (LLDs). The inspectors reviewed the associated radioactive effluent release data that was the likely source of the released material.

This annual environmental monitoring report events inspection constituted one sample as defined in IP 71122.03-05.

The inspectors reviewed significant changes made by the licensee to the ODAM as the result of changes to the land census or sampler station modifications since the last inspection. The inspectors reviewed technical justifications for changed sampling locations. The inspectors verified 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.

This ODAM significant changes review constituted one sample as defined in IP 71122.03-05.

The inspectors reviewed the calibration and maintenance records for two air samplers and composite water samplers. The inspectors reviewed calibration records for the environmental sample radiation measurement instrumentation (i.e., count room). The inspectors verified that the appropriate detection sensitivities with respect to TS/ODAM were utilized for counting samples (i.e., the samples meet the TS/ODAM required LLDs). The inspectors reviewed quality control charts for maintaining radiation measurement instrument status and actions taken for degrading detector performance.

The inspectors reviewed the results of the REMP sample vendor's quality control program including the interlaboratory comparison program to verify the adequacy of the vendor's program and the corrective actions for any identified deficiencies. The inspectors reviewed audits and technical evaluations the licensee performed on the vendor's program. The inspectors reviewed QA audit results of the program to determine whether the licensee met the TS/ODAM requirements.

This radiological environmental monitoring program sampler maintenance records and quality control inspection constituted one sample as defined in IP 71122.03-05.

b. Findings

No findings of significance were identified.

.3 Unrestricted Release of Material from the Radiologically Controlled Area (RCA)

a. Inspection Scope

The inspectors observed several locations where the licensee monitors potentially contaminated material leaving the RCA, and inspected the methods used for control, survey, and release from these areas. The inspectors observed the performance of personnel surveying and releasing material for unrestricted use to verify that the work was performed in accordance with plant procedures.

This inspection constituted one sample as defined in IP 71122.03-05.

The inspectors verified that the radiation monitoring instrumentation was appropriate for the radiation types present and was calibrated with appropriate radiation sources. The inspectors reviewed the licensee's criteria for the survey and release of potentially contaminated material and verified that there was guidance on how to respond to an alarm, which indicates the presence of licensed radioactive material. The inspectors reviewed the licensee's equipment to ensure the radiation detection sensitivities were consistent with the NRC guidance contained in IE Circular 81-07 and IE Information Notice 85-92 for surface contamination and health physics positions (HPPOS-221) for volumetrically contaminated material. The inspectors verified that the licensee performed radiation surveys to detect radionuclides that decay via electron capture. The inspectors reviewed the licensee's procedures and records to verify that the radiation detection instrumentation was used at its typical sensitivity level based on appropriate counting parameters (i.e., counting times and background radiation levels). The inspectors verified that the licensee had not established a "release limit" by altering the instrument's typical sensitivity through such methods as raising the energy discriminator level or locating the instrument in a high radiation background area.

This unrestricted release of material from the RCA inspection constituted one sample as defined in IP 71122.03-05.

b. Findings

No findings of significance were identified.

.4 Identification and Resolution of Problems

a. Inspection Scope

The inspectors reviewed the licensee’s self assessments, audits, LERs, and Special Reports related to the radiological environmental monitoring program since the last inspection to determine if identified problems were entered into the CAP for resolution. The inspectors also verified that the licensee's self-assessment program was capable of identifying repetitive deficiencies or significant individual deficiencies in problem identification and resolution.

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

- initial problem identification, characterization, and tracking;
- disposition of operability/reportability issues;
- evaluation of safety significance/risk and priority for resolution;
- identification of repetitive problems;
- identification of contributing causes;
- identification and implementation of effective corrective actions;
- resolution of NCVs tracked in the corrective action system; and
- implementation/consideration of risk-significant operational experience feedback.

This radiological environmental monitoring program and radioactive material control program problem identification and resolution inspection constituted one sample as defined in IP 71122.03-05.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA1 Performance Indicator Verification (71151)

.1 Data Submission Issue

a. Inspection Scope

The inspectors performed a review of the data submitted by the licensee for the fourth quarter 2008 performance indicators for any obvious inconsistencies prior to its public release in accordance with IMC 0608, “Performance Indicator Program.”

This review was performed as part of the inspectors’ normal plant status activities and, as such, did not constitute a separate inspection sample.

b. Findings

No findings of significance were identified.

.2 Unplanned Scrams per 7000 Critical Hours

a. Inspection Scope

The inspectors sampled licensee submittals for the Unplanned Scrams per 7000 Critical Hours PI for the period from the first quarter of 2008 through the fourth quarter of 2008. 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 5, 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 of 2008 through the fourth quarter of 2008 to validate the accuracy of the submittals. The inspectors also reviewed the licensee's CAP 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.

.3 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 of 2008 through the fourth quarter of 2008. 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 5, 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 of 2008 through the fourth quarter of 2008 to validate the accuracy of the submittals. The inspectors also reviewed the licensee's CAP 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.

.4 Unplanned Power Changes 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 of 2008 through the fourth quarter of 2008. 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 5, 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 of 2008 through the fourth quarter of 2008 to validate the accuracy of the submittals. The inspectors also reviewed the licensee’s CAP 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 power changes per 7000 critical hours sample as defined in IP 71151-05.

b. Findings

No findings of significance were identified.

.5 Occupational Exposure Control Effectiveness

a. Inspection Scope

The inspectors sampled licensee submittals for the Occupational Radiological Occurrences performance indicator for the period from the first quarter 2008 through fourth quarter 2008. To determine the PI 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 5, were used. The inspectors reviewed the licensee’s assessment of the PI for occupational radiation safety to determine if indicator related data was adequately assessed and reported. To assess the adequacy of the licensee’s PI data collection and analyses, the inspectors discussed with radiation protection staff, the scope and breadth of its data review, and the results of those reviews. The inspectors independently reviewed electronic dosimetry dose rate and accumulated dose alarm and dose reports and the dose assignments for any intakes that occurred during the time period reviewed to determine if there were potentially unrecognized occurrences. The inspectors also conducted walkdowns of numerous locked high and very high radiation area entrances to determine the adequacy of the controls in place for these areas. Documents reviewed are listed in the Attachment to this report.

This inspection constituted one occupational radiological occurrences sample as defined in IP 71151-05.

b. Findings

No findings of significance were identified.

.6 Radiological Effluent TS/Offsite Dose Calculation Manual Radiological Effluent Occurrences

a. Inspection Scope

The inspectors sampled licensee submittals for the Radiological Effluent TS (RETS)/ ODAM Radiological Effluent Occurrences performance indicator for the period of January 2008 through December 2008. The inspectors used PI definitions and guidance contained in the NEI Document 99-02, "Regulatory Assessment Performance Indicator Guideline," Revision 5 to determine the accuracy of the PI data reported during those periods. The inspectors reviewed the licensee's issue report database and selected individual reports generated since this indicator was last reviewed to identify any potential occurrences such as unmonitored, uncontrolled, or improperly calculated effluent releases that may have impacted offsite dose. The inspectors reviewed gaseous effluent summary data and the results of associated offsite dose calculations for selected dates between January 2008 and December 2008 to determine if indicator results were accurately reported. The inspectors also reviewed the licensee's methods for quantifying gaseous and liquid effluents and determining effluent dose. Documents reviewed are listed in the Attachment to this report.

This inspection constituted one RETS/ODAM radiological effluent occurrences 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 CAP

a. 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 CAP Reviews

a. 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.

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

.1 Cooling Tower Riser Break Leads to Manual Reactor Scram

a. Inspection Scope

The inspectors reviewed the plant operator's response to a break in the 'B' Cooling Tower West Riser during a planned downpower prior to commencing RFO 21. While preparing to secure the 'B' Cooling Tower per Operating Instruction (OI) 442, "Circulating Water System," operators observed indications of cavitation of both circulating water pumps. They also noted a lowering circulating water pit level. At time 1801, operators inserted a manual reactor scram as directed by Alarm Response Procedure (ARP) 1C06A, D-11, "Circ Water Pit Lo Level." Documents reviewed in this inspection are listed in the Attachment.

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

b. Findings

Introduction: A finding of very low safety significance was self-revealed when the operators exceeded the operational limit of the cooling tower riser by failing to secure one of the two running circulating water pumps prior to securing flow to the 'A' cooling tower.

Description: On February 1, 2009, operators were lowering reactor power in preparation for RFO 21. Using OI 442, operators were preparing to secure the 'A' circulating water pump and the ‘A’ cooing tower. Per the OI, operators were assigned to throttle the cooling tower riser valves on the tower to be removed from service until circulating water discharge pressure was about 35 psig and then secure the circulating water pump. The operators in the control room were monitoring circulating water discharge pressure using computer point F015, which is fed from pressure transmitter PT4205. An operator in the pump house was assigned to monitor the local circulating water pump discharge pressure.

Operators at the cooling tower were responsible for closing the cooling tower riser valves. In coordination with the control room operators, the cooling tower operators bumped the cooling tower riser valves in the closed direction in 10-second intervals. After each bump, the control room operators and the pump house operators monitored the cooling pump discharge pressure at their respective indications. The pump house operators noted a slightly higher discharge pressure than the control room operators and, therefore, the decision was made to use the local indication at the pump house since it was more conservative.

At 1755, the control room ordered the cooling tower operators to give a 5-second close signal to the cooling tower riser valves. After the 5-second close signal, the pump house operator noted that the local circulating water pump discharge pressure still indicated 33 psig, the same reading as before the 5 second close signal. Also, both cooling tower riser valves indicated fully closed after the 5 second close signal. The cooling tower operators reported the valve position to the control room operators and the control room operators then secured the ‘A’ circulating water pump.

Shortly after the cooling tower riser valves were shut, the pump house operator observed signs of circulating water pump cavitation. He also observed a lowering level in the circulating water pit. Operators also observed a lowering circulating water pit level, and at 1801 hours, they inserted a manual reactor scram per the guidance in ARP 1C06A, D-11, since circulating water pit level was less than eight feet and could not be restored.

Following the reactor scram, operators found that the west riser of the ‘B’ cooling tower had catastrophically failed by separating at the slip joint between the riser and the distribution header and the top of the cooling tower. The Root Cause Evaluation (RCE) determined that the cooling towers were not designed to have both circulating water pumps discharging over a single cooling tower. The station determined that the root cause of the event was that OI 442 was inadequate to prevent an inappropriate operational configuration because the procedure did not prevent operators from operating both circulating water pumps over one cooling tower.

The RCE also identified a contributing cause to the event in that PT4205, the pressure transmitter that provided circulating water pump discharge pressure indication to the control room, was plugged. This plugging resulted in the indicated circulating water discharge pressure in the control room being lower than the actual discharge pressure. This resulted in the control room operators allowing for further throttling of the cooling tower riser valves until the riser isolation valves were fully closed. It was also found that there was a Work Request Card (WRC) associated with PT4205 that had been written on November 10, 2008, to address erratic indication associated with the pressure transmitter. This WRC was scheduled to be worked during RFO 21.

Analysis: The inspectors determined that the operators exceeding the operational limit of the ‘B’ cooling tower west riser by failing to secure one of two circulating water pumps prior to securing flow to the ‘A’ cooling tower was contrary to the guidance for safe operation of plant equipment contained in ACP 110.1, “Conduct of Operations,” and therefore was a performance deficiency.

The finding was determined to be more than minor because the finding was associated with the Reactor Safety Cornerstone attribute of procedure quality and affected the cornerstone objective of limiting the likelihood of those events that upset plant stability and challenge critical safety functions during shutdown. Specifically, operating the plant in an inappropriate configuration resulted in the loss of the normal plant heat sink, which required the operators to manually scram the reactor and rely on safety-related equipment to cool the plant down.

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. Because the finding only resulted in a reactor scram and did not contribute to the likelihood that mitigation equipment or functions would not be available, the finding screened as Green.

This finding has a cross-cutting aspect in the area of Problem Identification and Resolution, Corrective Action, because the licensee did not take appropriate corrective actions to address safety issues and adverse trends in a timely manner. Specifically, maintenance and operations personnel failed to adequately address a known deficiency with a plugged pressure transmitter, which resulted in the control room allowing throttling of the ‘A’ cooling tower riser valves until they were fully shut, thus exceeding an operational limit by operating two circulating water pumps with only one cooling tower in service. [P.1(d)]

Enforcement: No violation of regulatory requirements occurred.
(FIN 05000331/2009002-02)

.2 Observation of Personnel Performance During Planned Non-Routine Evolutions: Plant Special Testing for Increased Recirculation System Core Flow and Phase IV Power Uprate

a. Inspection Scope

The inspectors observed personnel performance during planned non-routine evolutions using special testing procedures for increased recirculation system core flow and phase IV power uprate testing from the previous operating limit of 1880 MWth to the licensed thermal power limit of 1912 MWth. The inspectors performed reviews of the special test procedures, and observed expert panel meetings and briefings conducted to review the new plant operating data obtained during the tests. The inspectors also observed the licensed operators performing reactivity manipulations during the testing as well as the plant’s operation at the increased power level. The documents listed in the Attachment were used by the inspectors to accomplish the objectives of the inspection procedure.

These inspection activities constituted one sample as defined in IP 71153-05.

b. Findings

No findings of significance were identified.

.3 (Closed) LER 05000331/2009002-00: Outdoor Liquid Radwaste Storage Tank Radioactive Concentration Limit Exceeded

On February 1, 2009, reactor water was directed from the reactor to the radwaste system in order to adjust the reactor water level. The water was directed to the radwaste tanks via the reactor water clean-up and RHR systems. However, most of the RHR water was inadvertently directed to the radwaste surge tank, IT-88. A subsequent sampling of IT-088 determined that the tank contained elevated levels of radioactivity that exceeded the Technical Specification administrative limit and the ODAM limit. The licensee was able to reduce the activity of the tank to less than the TS administrative limit. As corrective actions, the licensee revised the operation and radwaste procedures and implemented actions to improve communication between operation and radwaste staff in order to prevent recurrences when adjusting reactor water levels. Documents reviewed as part of this inspection are listed in the Attachment to this report. This LER is closed.

A licensee identified violation of very low safety significance was identified and is documented in Section 4OA7.

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

4OA5 Other Activities

.1 Unit 1 Power Uprate-Related Inspection Activities (71004)

a. Inspection Scope

During this inspection period, the inspectors observed several activities related to the power uprate amendment. The inspectors observed the following tests:

- Special Test Procedure (SpTP) 213, "Increased Core Flow and Power Ascension Test to Greater Than 1880 MWth," Revision 0; and
- SpTP 214, "Pressure Regulator Dynamic Tuning," Revision 0.

These inspection activities did not constitute any additional inspection samples. Rather, they were documented as a sample in Section 4OA3 above, as defined in IP 71153-05.

This inspection documents the completion of two surveillance samples. No concerns were identified.

a. Findings

No findings of significance were identified.

.2 (Closed) NRC Temporary Instruction 2515/173 Review of the Industry Ground Water Protection Voluntary Initiative

a. Inspection Scope

An NRC assessment was performed of the licensee's implementation at DAEC of the NEI – Ground Water Protection Initiative (dated August 2007 (ML072610036)). The inspectors verified that the licensee evaluated work practices that could lead to leaks and spills and performed an evaluation of systems, structures, and components that contain licensed radioactive material to determine potential leak or spill mechanisms.

The inspectors verified that the licensee completed a site characterization of geology and hydrology to determine the predominant ground water gradients and potential pathways for ground water migration from onsite locations to offsite locations. The inspectors also verified that an onsite ground water monitoring program had been implemented to monitor for potential licensed radioactive leakage into groundwater and that the licensee had provisions for the reporting of its ground water monitoring results (annual effluent report). (See http://www.nrc.gov/reactors/operating/ops-experience/tritium/plant-info.html)

The inspectors reviewed the licensee's procedures for the decision making process for potential remediation of leaks and spills, including consideration of the long-term decommissioning impacts. The inspectors also verified that records of leaks and spills were being recorded in the licensee's decommissioning files in accordance with 10 CFR 50.75(g).

The inspectors reviewed the licensee's notification protocols to determine whether they were consistent with the Groundwater Protection Initiative. The inspectors assessed whether the licensee identified the appropriate local and state officials and conducted briefings on the licensee's ground water protection initiative. The inspectors also verified that protocols were established for notification of the applicable local and state officials regarding detection of leaks and spills.

b. Findings

No findings of significance were identified.

.3 Quarterly Resident Inspector Observations of Security Personnel and Activities

a. Inspection Scope

During the inspection period, the inspectors conducted observations of security force personnel and activities to ensure that the activities were consistent with licensee security procedures and regulatory requirements relating to nuclear plant security. These observations took place during both normal and off-normal plant working hours.

These quarterly resident inspector observations of security force personnel and activities did not constitute any additional inspection samples. Rather, they were considered an integral part of the inspectors' normal plant status review and inspection activities.

b. Findings

No findings of significance were identified.

4OA6 Management Meetings

.1 Exit Meeting Summary

On April 2, 2009, the inspectors presented the inspection results to Mr. R. Anderson, Site Vice President, 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 Inservice Inspection with Site Vice President, Mr. R. Anderson, on February 11, 2009.

- Access control to radiologically significant areas and as-low-as-is-reasonably-achievable (ALARA) planning and control under the Occupational Radiation Safety Cornerstone with Site Vice President, Mr. R. Anderson, on February 13, 2009.

- REMP and radiological material control program and a review of the implementation of the industry ground water protection voluntary initiative under the public radiation safety cornerstone with Mr. R. Anderson, Site Vice President on March 20, 2009.
- The inspectors presented the results of the inspection review of licensee corrective actions pertaining to URI 05000331/2005005-02 to Licensing Manager, Mr. S. Catron, and other members of the licensee's staff via telephone on March 31, 2009. Licensee personnel acknowledged the inspection results presented.

The inspectors confirmed that none of the potential report input discussed was considered proprietary.

.3 End of Cycle Assessment Results Discussion

On April 2, 2009, directly following the quarterly integrated resident inspection exit meeting, the NRC met with Mr. R. Anderson, Site Vice President, and members of the licensee staff to discuss their performance during the previous four quarters for the 2008 End-of-Cycle assessment, which was continually within the Licensee Response Column of the Action Matrix, in accordance with Section 06.05 of IMC 0305.

4OA7 Licensee-Identified Violations

A violation of very low safety significance (Green) was identified by the licensee. This violation meets the criteria of Section VI of the NRC Enforcement Policy, NUREG-1600, for being dispositioned as an NCV.

Cornerstone: Public Radiation Safety

Technical Specification 5.5.8 states "explosive gas and storage tank radioactive monitoring program, Section b., limits the liquid radwaste storage tanks in the low-level radwaste processing and storage facility (LLRPSF) to less than 50 curies." Contrary to this requirement, on February 3, 2009, a radwaste operator discovered that the dose rate at the vicinity of radwaste surge tank IT-088 was elevated. A subsequent sample of IT-088 determined that the tank contained approximately 88 curies of total radioactivity. This exceeded the Technical Specification administrative limit of 50 curies. As a result, the licensee submitted event report (LER No. 2009-002-00) as required by 10 CFR 20.2203(a)(3)(i). On February 3, 2009, ODAM limiting condition for operations 6.1.5 Condition A: "quantity of radioactive material in the tanks exceeding the limit," was entered. The actions associated with Condition A required suspension of all additions of radioactive material to IT-088 tank and to reduce tank concentration limits to less than 50 curies within 48 hours. On February 4, 2009, the IT-088 tank's radioactivity concentration was reduced below 50 curies in less than 48 hours; subsequently, ODAM OLCO 6.1.5 Condition A was exited. This issue was entered into the licensee's corrective action program as CAP 063486. The finding was reviewed using IMC 0609, Appendix D, "Public Radiation Safety Significance Determination Process," and was determined to be of very low safety significance. Specifically, the finding was not a radioactive material control or transportation issue and the finding was not indicative of a failure to implement the effluent control program and did not result in a dose to the public greater than 0.005 rem or in excess of the criterion in Appendix I to 10 CFR Part 50 or CFR 20.1301(e).

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee

R. Anderson, Site Vice President
D. Curtland, Plant General Manager
B. Eckes, NOS Manager
S. Catron, Licensing Manager
J. Cadogan, Engineering Director
B. Kindred, Security Manager
J. Morris, 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
J. Swales, Design Engineering Supervisor
K. Kleinheinz, Maintenance Manager
D. Albrecht, Radwaste Supervisor
G. Park, ISI Program Owner
F. Dohmen, NDE Level III
N. McKenney, General Supervisor Radiation Protection
S. Funk, CHP, REMP Program Manager
D. Johnson, Radwaste Operator/Chem Tech, Radiation Environmental Technician
C. Harberts, Refuel Floor Project Manager
D. Barta, Licensing

Nuclear Regulatory Commission

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

LIST OF ITEMS OPENED, CLOSED AND DISCUSSED

Opened

05000331/2009002-01	URI	An Internal Contamination Occurred while Cleaning RPV Studs and Washers on the Refuel Floor at Duane Arnold (Section 2OS2.3)
05000331/2009002-02	FIN	Cooling Tower Riser Break Leads to Manual Reactor Scram (Section 4OA3.1)
05000331/2009002-03	NCV	Failure to perform required actions for existing LCO conditions during in-vessel fuel movements (Section 1R15.1.b)
05000331/2009002-04	NCV	Failure to Consider Design Basis Load in Evaluation for Continued Operation (1R15.2.b)

Closed

05000331/2009002-02	FIN	Cooling Tower Riser Break Leads to Manual Reactor Scram (Section 4OA3.1)
05000331/2009002-03	NCV	Failure to perform required actions for existing LCO conditions during in-vessel fuel movements (Section 1R15.1.b)
05000331/2009002-00	LER	Outdoor Liquid Radwaste Storage Tank Radioactive Concentration Limit Exceeded
05000331/2005002-02	URI	Failure to Include the Analysis of Thermal Movements in Piping Modifications (1R15.2)
05000331/2009002-04	NCV	Failure to Consider Design Basis Load in Evaluation for Continued Operation (1R15.2.b)

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 1R04

OI 324A1; SBDG 1G-31 System Electrical Lineup; Revision 2
OI 324A3; SBDG 1G-31 System Valve Lineup and Checklist; Revision 9
OI 324A7; SBDG 1G-31 System Control Panel Lineup; Revision 3
STP 3.8.1-07A; 'A' LOOP-LOCA Test; Revision 0
OI 151A1; CS System Electrical Lineup; Revision 3
OI 151A2; CS System Valve Lineup and Checklist; Revision 4
OI 151A4; CS System Control Panel Lineup; Revision 4
OI 730; Control Building Heating, Ventilation and Air Conditioning (HVAC) System; Revision 98
OI 730A1; Control Building HVAC System Electrical Lineup; Revision 2
OI 730A3; Control Building Ventilation System Valve Lineup; Revision 7
OI 730A6; Control Building HVAC System Control Panel Lineup; Revision 9

Section 1R05

FHA-400; Fire Hazards Analysis; Revision 9
AFP-08; Reactor Building Standby Gas Treatment System and MG Set Rooms; Revision 24
AFP-17; Turbine Building Condenser Bay, Heater Bay, and Steam Tunnel; Revision 24
AFP-18; Turbine Building North Turbine Building Ground Floor and Tube Pulling Area; Revision 28
AFP-19; Turbine Building South Turbine Building Ground Floor; Revision 25
AFP-20; Aux Boiler Room, Emergency Diesel Generator Rooms, and Generator Day Tank Rooms; Revision 29
AFP-25; Control Building Cable Spreading Room; Revision 26
CAP 063274; Scaffold Issues for 1G21 'B' SBDG [Standby Diesel Generator] room
Fire Protection Impairment Request FPR-09-7037; Remove Existing Concrete Curbing on Both Sides of Door to 'B' EDG Room and Replace with Removable Steel Curbing

Section 1R07

Heat Exchanger Thermal Performance and Trending Program; Revision 8
General Maintenance Procedure GMP-Mech-26; Heat Exchangers; Revision 10
CWO A66164; Re-repair the Pass Divider Plate on 'A' RHR Heat Exchanger, 1E201A
CAP 055490; CAQ – NS160003 Leakage
PWO 1144176; Perform UT Examination of Heat Exchanger Shell - 1E201A
PWO 1144177; Open, Inspect, and Clean Service Water Side of Heat Exchanger – 1E201A
PWO 1144178; Perform Eddy Current Examination on Designated Tubes – 1E201A

Section 1R08

ACP 1211.5; Nondestructive Examination Procedure Magnetic Particle (Dry or wet Visible) MT-1; Revision 10

ACP 1211.10; Nondestructive Examination Procedure Visual Examination of Component Supports VT-3; Revision 10
Apparent Cause Evaluation (ACE) 001828; Description Error on NIS-2
CAP 047392; Indication Found by Surface Examination
CAP 048040; 45 out of 120 CAP Screws Failed Visual Examination
CAP 048269; CRD 1R215 Leaking 15-20 DPM
CAP 063771; Indication Identified in RRH-F002A Weld
CAP 050093; Trend Fabrication and Welding
CA 045128; Proceduralize Filler Metal Verification
CA 045592; Wrong Welding Procedure Used
CA 045702; Schedule for Non-Overlayed
CA 045834; Evaluate NDE for Underground Pipe
CA 049173; UT 75% of Risk Informed Coverage Achieved
Procedure Qualification Record; GMP 102-311-GS-PQR; dated July 24, 1987
OTH 020420; STP for Visual Exam of Ground Above Buried Pipe
Liquid Penetrant Examination Report PT-07-09; Pipe to Pipe Weld CSB-F004; dated February 16, 2007
Liquid Penetrant Examination Report PT-07-15; RPV Flange CRD CAP Screws; dated March 3, 2007
Procedure Qualification Record; SM-1-1; dated January 2, 1976
Procedure Qualification Record; WP-6; dated January 8, 1991
Radiographic Report and Film; Weld 6 N2G, RRG-F002A; dated November 17, 1978
Radiographic Report and Film; Weld 6 N2H, RRH-F002A; dated November 17, 1978
Radiographic Report and Film; Weld 2 N2G RRG-F002A; dated November 1, 1978
Radiographic Report and Film; MO4423, welds W2, W3 and W5; dated February 8, 2009
Repair/Replacement Plan; MO4424; dated July 29, 2008
STP NS540004B; Visual Examination of Ground Area Above ASME Section XI Service Water Loop B Buried Piping; dated September 21, 2008
Ultrasonic Examination Report UT-07-033; Safe End to Nozzle Weld FWA-J002; dated February 13, 2007
Ultrasonic Examination Report VE-07-013; Safe End to Nozzle Weld RRC-F002; dated February 19, 2007
Weld Checklist; MO4423, welds W1, W2, W3; dated July 23, 2008
Weld Checklist; MO4424, welds W1, W2, W3, W4, W5; dated July 27, 2008
Weld Procedure Specification; FP-PE-B31-P1P1-GTSM-001; dated January 9, 2006
Welding Performance Qualification; B. Scott; dated January 28, 2009
Welding Performance Qualification; G. Beer; dated January 28, 2009
Zetec OmniScanPA; Procedure for Encoded, Manually Driven, Phased Array Ultrasonic Examination of Dissimilar Metal Piping Welds; Revision D

Section 1R11

ACP 110.1; Conduct of Operations; Revision 22
Abnormal Operating Procedure (AOP) 264; Loss of Recirculation Pump(s); Revision 1
OI 264; Reactor Recirculation System; Revision 105

Section 1R12

OI 442; Circulating Water System; Revisions 70 and 71
CAP 063586; NCAQ – Circulating Water Pressure Transmitter Plugged During Circ Pipe Failure Event

CAP 064965; NCAQ – Ineffective Corrective Actions Associated with ‘B’ Cooling Tower Failure
CAP 064267; CAQ – Cooling Tower Replacement Design Error – Lack of Support for the Riser Piping
Calculation 1-B-10; Cooling Tower Risers; Revision 0
Calculation CAL-M09-013; Mechanical Loads on Cooling Tower Risers; Revision 1
CAP 062934; NCAQ – MO4251 (CT 1E-69B West Circ Riser Isol) Unreliability
CAP 063310; NCAQ- Circ Water Startup Challenge
CAP 063426; SCAQ – Unplanned reactor Scram due to Loss of Circ Pit Level
CAP 063740; CAQ – V42-0122 (Circ Water Pressure) Needs to be replaced with a Ball Valve
CAP 064961; NCAQ – Valve Found out of Position During Lineup
ECP-1884; Repair/Replace ‘B’ West Riser
WRC A84341; Identified transmitter as Erratic and Possible Plugged
DAEC System Checklist/Health Report for Condensate Heat Removal System
DAEC Maintenance Rule Program Module 0; Overview; Revision 3
DAEC Maintenance Rule Program Module 0 Attachment 3; Startup Systems Containing Components Performing Maintenance Rule Risk-Significant or Standby Function at DAEC; Revision 3f
DAEC Performance Criteria Basis Document; On-site Distribution SUS 4.00, 5.00, 6.00, 7.00, 17.00, 57.00; Revision 7
DAEC Maintenance Rule System Goals for RED (a)(1) Systems; Revision 34
DAEC System Checklist/Health Report for SUS 4.00 On-Site Distribution 4KV
System Monitoring and Reporting Tool System Report; dated March 16, 2009

Section 1R13

Work Planning Guideline 1; Work Process Guideline; Revisions 26, 27, and 28
Work Planning Guideline 2; On-Line Risk Management Guideline; Revision 45 and 46
WM-AA-1000; Work Activity Risk Management Process; Revision 1
PI-AA-100-1002; Guideline for Failure Investigation Process; Revision 0
CAP 062956; ‘A’ Recirc Motor Generator Scoop Tube Lock Issue ODMI Initiation
Corrective Work Order (CWO) A80328; ‘A’ Recirculation Scoop Tube Locked Up due to ‘High Deviation’ on B31-K49A
CAP 062892; NCAQ – Abnormal Annunciator Activation
CAP 063088; STP 3.4.2-01 ‘B’ Recirc Loop Outside M-Ratio
CAP 062203; CAQ – ‘A’ Recirc Pump Speed Change and Scoop Tube Lockup
CAP 062891; ‘A’ Recirc MG Scoop Tube Lock
CAP 055449; CAQ – ‘B’ Recirc Pump #2 Seal Pressure Trending up
CAP 055211; CAQ – ‘B’ Recirc Pump Seal Pressure Increase
CAP 063057; CAQ – ‘B’ Recirc Pump Seal Number 2 Pressure
CAP 065511; #9 and #10 Main Generator Exciter Bearing
CAP 065526; FIP Team Critique for #9 Bearing High Temperature Event
CAP 065411; Manual Turbine Trip while Starting Up
CAP 065432; Plant Shutdown from Mode One Due to Turbine Bearing Abnormalities
CAP 065440; Evaluate the Time Hydrogen Seal Oil Can be Operated without Turbine Lube Oil in Service
CAP 065463; Evaluate Potential for cocked Turbine Hydrogen Seal
CAP 065486; Issues from Observation of Turbine Generator Bearing #9 Installation
CAP 065489; PI 3106, Turbine Lube Oil Pressure, Low Out of Specification
CWO A100499; Inspect and Repair or replace #9 Bearing
CAP 065897; RHR Logic Surveillance Not Performed as Scheduled Today

Surveillance Work Order S015225; Perform STP 3.3.5.1-37 – RHR Logic System Functional Test-Operating
CWO A100499; Inspect and Repair or replace #9 Bearing
Maintenance Risk Evaluations for Work Week 9903; Revision 0 and 1
DAEC On-line Schedule for Work Week 3
Maintenance Risk Evaluations for Work Week 9904; Revisions 0, 1, 2, and 3
DAEC On-line Schedule for Work Week 4
Maintenance Risk Evaluations for Work Week 9912; Revisions 0 and 1
DAEC On-line Schedule for Work Week 12

<u>Section 1R15</u>

ACP 102.17; Pre/Post-Job Briefs and Infrequently Performed Tests and Evolutions; Revision 42
EN-AA-203-1001; Operability Determinations / Functionality Assessments; Revision 1
LI-AA-01; Regulatory Margin Corrective Action Strategy; Revision 1
CAP 062744; CAQ – Deficiencies Noted by NRC in Several Immediate Operability Determinations
STP 3.8.1-02; One Standby Diesel Generator Inoperable; Revision 3
CAP 062919; TS LCO 3.8.1.b Required Action B.3 Exited Prematurely
CAP 062908; CAQ – Unplanned LCOs Due to Failure of 1VSF056A
OTH 037267; Guidance on RHRSE/ESW Availability on Loss of Room Ventilation
CAP 064489; CAQ – CDR Position Indication Logged Inoperative with Core Alterations in Progress
ACE 001925; CAQ – CDR Position Indication Logged Inoperative with Core Alterations in Progress
System Description 856.1; Reactor Manual Control and Rod Position Information Systems; Revision 5
OI 856.3; Rod Position Information System; Revision 8
STP 3.9.1-01; Refueling Interlocks Channel Functional Testing; Revision 10
Operations Electronic Log System Shift Log Entries; dated February 8, 2009 through February 17, 2009
CWO A75306; Troubleshoot and Repair/Replace Rod Position Indication Probe for Control Rod 14-23
CWO A81530; Troubleshoot and Repair/Replace Rod Position Indication Probe for Control Rod 22-19
CAP 064523; CAQ – Potential Trend in LCO Tracking Issues in Operations
ACE 001929; CAQ – Potential Trend in LCO Tracking Issues in Operations
CAP 064521; CAQ – LCO Entry Missed for MO 1905
CAP 063660; CAQ – Failure to Recognize That Loss of 1B42 Affected Operation of Standby Filter Unit
CAP 064162; NCAQ – SRM & IRM Functional STPs for Fuel Shuffle #2 Performance Delayed
CAP 064363; CAQ – Operations Is Not Entering LCOs for Inop Snubbers That Support Required Systems
CAP 064441; CAQ – TRM LCO 3.7.2 Not Entered for Inop Snubber
CAP 065287; NCAQ – Insulation Fell Off of Piping
Operability Recommendation (OPR) 000395; CAQ – Insulation Fell Off of Piping
Operable But Degraded (OBD) 000313; CAQ – Insulation Fell Off of Piping
CAP 065385; CAQ – STP 3.4.2-01 Daily Jet Pump Operability with Recirc Pumps at Minimum Speed
OPR 000396; CAQ – STP 3.4.2-01 Daily Jet Pump Operability with Recirc Pumps at Minimum Speed

CAP 065740; CAQ – ‘A’ SBLC Pump 1P230A Leak
OPR 000397; CAQ – ‘A’ SBLC Pump 1P230A Leak
OBD 000314; CAQ – ‘A’ SBLC Pump 1P230A Leak
CA040021; Corrective Action for CAP035317: HLE-023 Piping Calculations Don’t Include Thermal Movements of the Drywell; dated March 30, 2005
CA040134; Corrective Action for CAP035317: HLE-023 Piping Calculations Don’t Include Thermal Movements of the Drywell; dated April 22, 2005
CA040203; Corrective Action for CAP035317: HLE-023 Piping Calculations Don’t Include Thermal Movements of the Drywell; dated May 10, 2005
CA041828; Corrective Action for CAP035317: HLE-023 Piping Calculations Don’t Include Thermal Movements of the Drywell; dated January 6, 2006
CA042817; Corrective Action for CAP035317: HLE-023 Piping Calculations Don’t Include Thermal Movements of the Drywell; dated April 14, 2006
CA050263; Corrective Action for CAP057980: CAQ – NRC Commitment Not Met in Past Operability Calculation for CA42817; dated July 22, 2008
CAL-080-312; Calculation: 7RB Containment Atmosphere Control - HLE-023; Revision 2
CAP035317; HLE-023 Piping Calculations Don’t Include Thermal Movement of the Drywell; dated March 18, 2005
CAP039338; Two Supports on HLE-21/38 Require Modification Due to Drywell Thermal Movement; dated December 14, 2005
CAP057980; CAQ – NRC Commitment Not Met in Past Operability Calculation for CA42817; dated May 28, 2008
CE002404; Condition Evaluation for CAP035317: HLE-023 Piping Calculations Don’t Include Thermal Movements of the Drywell; dated March 22, 2005
DAEC Letter NG-05-2178 to NRC; Subject: Additional Information Regarding Unresolved Item 500331/2005002-02; dated December 12, 2005
DAEC Letter NG-06-0305 to NRC; Subject: Withdrawal of NG-05-2178; dated April 3, 2006
DAEC Letter NG-06-0375 to NRC; Subject: Voluntary Licensee Event Report No. 2006-002-00; dated June 1, 2006
DGC-M-100; DAEC Engineering Design Guide: Stress Analysis and Support Design of Seismic Category I Piping Systems; Revision 7
Drawing M119AC-04511; Containment Atmosphere Control, Pipe Support, Mark No. HLE-23-H-11; Revision 3
Drawing M119AC-04547; Containment Atmosphere Control, Pipe Support, Mark No. HLE-23-SR-47; Revision 3
Drawing M119AC-11853; Containment Atmosphere Control, Pipe Support, Mark No. HLE-23-SS-47; Revision 1
IE05-P108274-240; Operability Evaluation of Drywell Water Piping at Penetrations X-24A and X-24B; Revision 1
IE05-P108274-241; Operability Evaluation of Drywell Water Piping at Penetrations X-23A and X-23B; Revision 1
IE05-P106464-500; Evaluation of Instrument Piping from Penetration X-108A; Revision 0
IE05-P106464-520; Evaluation of Instrument Piping from Penetration X-108C; Revision 0
IE05-P108274-240; Operability Evaluation of Drywell Water Piping @ Penetrations X-24A and X-24B; Revision 1
IE05-P108274-241; Operability Evaluation of Drywell Water Piping @ Penetrations X-23A and X-23B; Revision 1
IE05-P108274-600; Operability Evaluation of 3”-HEL-31 @ Penetration X48; Revision 1
IE05-P108274-610; Operability Evaluation of Containment Atmosphere Control Piping @ Penetration X25; Revision 1

IE05-P108274-610; Operability Evaluation of Containment Atmosphere Control Piping @ Penetration X25; Revision 2
IE05-P108274-620; Operability Evaluation of Instrument Piping @ Penetration X-108C; Revision 1
IE05-P108274-630; Operability Evaluation of Instrument Piping @ Penetration X46E; Revision 1
IE05-P108274-640; Operability Evaluation of Instrument Piping @ Penetration X40D; Revision 1
LER 2006-002-00; Voluntary Licensee Event Report: Drywell Penetrations Calculations Do Not Account for Thermal Movement; dated June 1, 2006
OBD000246; Operable But Degraded Determination for CAP035317: HLE-023 Piping Calculations Don't Include Thermal Movements of the Drywell; dated March 22, 2005
OPR000308; Operability Recommendation: Containment Atmosphere Control System Piping; Revision 0
OPR000313; Operability Recommendation: Well Water Piping from Drywell HVAC (HLE034) Pressure Boundary; Revision 0
OPR000314; Operability Recommendation: RBCCW from Drywell (HLE029) Pressure Boundary; Revision 0
OPR000315 Operability Recommendation: RBCCW to Drywell (HLE028) Pressure Boundary; Revision 0
OPR000316; Operability Recommendation: Nitrogen Piping to Drywell (HLE021) Pressure Boundary; Revision 0
OPR000318; Operability Recommendation: Well Water Piping to Drywell HVAC (HLE032) Pressure Boundary; Revision 0
Reedy Engineering, Inc. Position Paper; Scope: Identification of ASME Code Requirements for Evaluating Operability of Piping Systems Affected by Anchor Movements Caused by Design Conditions of Containment Vessel; dated June 21, 2005

<u>Section 1R18</u>

Modification Work Order 1146469; Installation of ECP-1865 "Changes to the Air Supply for the Refueling Platform 1S081"
ECP-1865; Changes to the Air Supply for the Refueling Platform 1S081
5059SCRN 030620; DAEC 10 CFR 50.59 Screening for ECP-1865
Engineering Fleet Procedure FP-E-MOD-04; Design Inputs; Revision 3
Engineering Fleet Procedure FP-E-MOD-06; Design Description; Revision 3
Engineering Fleet Procedure FP-E-MOD-08; Engineering Change Notices; Revision 2
CAP 062738; Installation of Instrument Air to the Refuel Platform Deficiencies
Engineering Change Notice ECN-1833-06; 315 Degree and 225 Degree Feedwater Sparger Stiffener Modification; Revision 2
ECP-1833; 315 Degree Feedwater Sparger Stiffener Modification
CAP 063796; Indication Report INR-IVVI-09-03 Feedwater Sparger at 315 degrees RFO 21
WM-AA-1000-F02; Data Sheet 2-Site level High Risk Integrated Review; Revision 0; Activity Risk Review to Replace Pin Keeper on the 315 degrees Feedwater Sparger
OBD 000258; CAQ – Calculation CAL-E02-003 Shows SBDG Voltage Dips less than UFSAR/RG 1.9 Requirements (RFO 21)
CAP 063830; ECP 1748 did not Identify need for SBDG Emergency Shutdown
CAP 063870; Construction Acceptance Testing Issues related to ECP 1748
CAP 063845; 'B' EDG Governor MPU Bracket Installation Issues
CAP 063971; 'B' EDG Islanded Test Data not Captured
CAP 064094; 1G021 ('B' EDG) Failed to come up to Rated Speed during LOOP-LOCA Testing
ACP 1403.3; Modification Acceptance Test Control Program; Revision 10

Drawing M015-006,1A.-WIP; Diesel Generator 1G21 Start Circuit A and B Governor Control and Excitation Control; Revision 7A
Modification Acceptance Test (MAT) for ECP-1748; 1G21 – 'B' Emergency Diesel Generator Governor Replacement; Revision 1
CWO A80549; Received Control Power Failure Alarm and SBDG Failed to come up to Speed during LOOP Test
PWO 1145733; Facilitate Tuning of New Governor
PWO 1136141; 1G021/Governor Upgrade – Post Modification and Acceptance Testing
PWO 1145735; 1C094 New Governor Controls and Wiring Installation
MAT for ECP 1835B; 'B' SBDG Voltage Regulator Upgrade
PWO 1145734; 1C094 Old Governor Controls and Wiring Removal
PWO 1144550; Replace Voltage Regulator Card
CAP 063881; Perform Aggregate Review – ECP 1748 – SBDG Governor Modification Activity Issues
CAP 063870; Construction Acceptance Testing Issues Related to ECP 1748
CAP 063970; 1G021/LOF Lube Oil Filter Head has Minor Oil Leak
CAP 064015; 1G021 SBDG Governor Oil too High when Checked on the Operating Check List
CAP 064024; 1G21 Governor Voltage Regulator During Fast Start Operability STP 3.8.1-06B
CAP 064034; 1G21 Tuning Delayed Due to Missing Alarm Card
CAP 063654; 1G021 Governor Modification Wrong Wire
CAP 063955; 'B' SBDG 1G-21 Filed Flashed Unexpectedly during Tuning, Delaying Tuning
CAP 063832; 1C94 Term Block F (TOP) is not the same as Term Block F (located at top of panel)
CAP 063830; ECP-1748 did not identify need for SBDG Emergency Shutdown Switch Cable
CAP 063704; Significant Delays to SBDG Modification Installation due to Insufficient Human Performance Tool use
CAP 063786; 1G021 Wiring for Hydraulic Governor
CAP 063803; Lack of Ownership and Point of Contact for ECP-1748 SBDG Governor Modification

<u>Section 1R19</u>

CAP 063692; Refuel Bridge Main Hoist Raise – Joystick Problem
CAP 063738; North / South Refueling Bridge Joystick
STP 3.9.101; Refueling Interlocks Channel Functional Testing; Revision 11
STP NS810001; Refueling Platform Inspection; Revision 23
PWO 1148592; Perform Repairs to the Refueling Platform Indicating and Control Devices (Load Cell, Z-Positioner, Joystick, etc.)
STP 3.8.1-07B; 'B' LOOP/LOCA Test
OI 324; Standby Diesel Generator System; Revision 90
CWO A93689; Set-up New DRU [Digital Reference Unit] and Replace the DRU
CAP 063881; CAQ – Perform Aggregate Review-ECP 1748- SBDG Governor Modification Activity Issues
CAP 063765; CAQ – CV4413, 'A' Outboard MSIV High Leakage
CAP 064328; CAQ – CV4413 (A Outboard MSIV) Discretionary LLRT Exceeded Leakage Limits
CAP 064440; CAQ – CV4413 A MSIV Steam Line Outboard Isolation-Material Found on Seat
CAP 064626; Main Steam Isolation Valve CV4413 Testing
PWO 1148609; MSIV Inspection and Overhaul
STP 3.6.1.1-04; Containment Isolation Valve Leak Tightness Test-Type C Penetrations-Main Steam System; Revision 18
CAP 065020; CAQ – HCU 34-15 Vent Dragon Valve Broken in Mid-Position

CWO A80812; Establish Freeze Seal and Replace the 1” Dragon Valve Internals or Replace Valve as Required
General Maintenance Procedure GMP-Mech-03; Pipe Freeze Seals; Revision 20
50023,2008C-30; CRD Friction Testing; Revision 0
CWO A100499; Inspect and Repair or Replace Turbine Generator #9 Bearing, as Required
Surveillance Work Order 013275; Perform STP NS930003-Main Turbine Overspeed System Tests
STP NS930003; Main Turbine Overspeed Trip System Tests; Revision 11
CWO A82110; Perform TIFs for Troubleshooting: Replace FY4450F
PWO 1139809; Calibrate FY4450E

Section 1R20

Outage Management Guidelines-7; Outage Risk Management Guidelines; Revision 17; RFO 21 Outage Risk Plan
Integrated Plant Operating Instruction (IPOI) 4 Attachment 1; Operations Manager/Reactor Engineer Recommendations for Shutdown; Revision 92
Shift Orders dated January 29, 2009
Reactivity Management Plan: Plant Shutdown; February 1, 2009
IPOI 5; Shutdown; Revisions 93, 94, and 95
IPOI 8; Outage and Refueling Operations; Revision 56
IPOI 8 Attachment 3; Time-to-Boil Calculation; Revision 56
IPOI 8 Attachment 5; Daily Risk Assessment Checklists; Revision 56
OI 149A8; RHR System Protected System Placards for SDC; Revision 2
Refueling Procedure 403; Performance of Fuel Handling Activities; Revision 32
ACP 1410.5; Clearance Program; Revision 96
IPOI 3; Power Operations (35% to 100% Rated Power); Revision 102
IPOI 7; Special Operations; Revision 107
IPOI 7; Special Operations; Attachment 2, Primary Containment Closeout; Revision 107
IPOI 7; Special Operations; Attachment 1, Primary Containment Entry; Revision 107
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5059SCRN 034303; DAEC 10 CFR 50.59 Screening for Core Modification Package 21 (ECP-1873)
CAP 064640; RFO 21 Core Verification Successful with Two Minor Issues
ACP 103.1; Nuclear Fuel and Core Design Control Program; Revision 30
STP 3.1.4-01; Scram Insertion Time Testing; Revision 17
STP 3.10.1-01; Non Nuclear Heat Class 1 System Leakage Pressure Test; Revision 31
CAP 065122; NCAQ – STP 3.10.1-10 Hydro Difficulties
IPTE Briefing Package for Reactor Startup 09-01; Startup from RFO 21; dated February 27, 2009
ACP 102.17; Pre/Post-Job Briefs and Infrequently Performed Tests and Evolutions; Revision 42
Reactivity Management Plan: Plant Startup; March 3, 2009
Reactor Engineering Department Procedure 14; Hot-Notch and Estimated Critical Position; Revision 6
IPOI 1; Startup Checklist-Section 5.0 Startup Comments; Revision 123
Reactivity Management Plan: Downpower to take Generator Offline; March 5, 2009
Reactivity Management Plan: Plant Startup; February 27, 2009
RFO 21 Feedwater Sparger Repair Infrequently Performed Test Evolution Briefing
Nuclear Policy NP-910; Plant Readiness for operations; Revision 8

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STP 3.5.3-02; RCIC System Operability Test; Revision 25
STP NS500001; RCIC System Leakage Inspection Walkdown; Revision 4
STP 3.6.103; MSIV Trip/Closure Time Check; Revision 5
CAP 064221; Question Concerning ‘B’ LOOP/LOCA Test Results
CAP 064288; Extended 1A4 Loss during LOOP-LOCA Testing Delayed SBDG Testing
CAP 064189; LOOP-LOCA Discrepancy
STP 3.8.1-07B; ‘B’ LOOP-LOCA Test; Revision 26
STP 3.8.1-07A; ‘A’ LOOP-LOCA Test; Revision 26
RFO 21 ‘A’ EDG LOOP/LOCA STP Results Summary Sheet
RFO 21 ‘B’ EDG LOOP/LOCA STP Results Summary Sheet
STP 3.6.1.1-13; Containment Isolation Valve Leak Tightness Test-Type C Penetrations-HPCI/RCIC Valves; Revision 4
CAP 064487; CAQ – CV-2411 Failed Post-Maintenance LLRT
CAP 064478; CAQ – CV-2410 Cannot be Fully Closed from Control Room
STP 3.3.8.1-05B; 1A4 4KV Emergency Transformer Supply Undervoltage Calibration; Revision 0
CAP 064164; STP 3.3.8.1-01B Not Completed by WPI Drop Dead Date

Section 2OS1

CAP 063636; Worker Inside RCA Without an Active ED
CAP 063694; Missed Post LPCI Full Flow Test Radiation Survey
CAP 063542; Maintenance Personnel Entered Steam Tunnel On Wrong RWP
CAP 063486; Increased dose rates in the radwaste surge tank IT-088
CAP 063094; Turbine Rollup Door Was Opened by Security Without Health Physics Present
RWP 09-30009; All Refuel Outage Support Work; Task 01 -06; Special Maintenance; Revision 3
Duane Arnold Refueling Outage No. 21; Refuel Floor and Project Kick-Off Meeting; dated February 1, 2009
Duane Arnold Refueling Outage No. 21; Radiological Performance Improvement Initiatives; dated February 13, 2009
ACP 1411.27; Rules for Conduct of Work in Radiologically Controlled Areas; Revision 28
Duane Arnold Inter Office Memo; Intake Occurred on February 6, 2009; dated February 12, 2009
Health Physics Procedure (HPP) 3104.01; Control Access to High Radiation Areas and Above; Revision 47
HPP 3104.07; Diving Operation Within the Radiological Areas; Revision 17
HPP 3104.06; Control of Radiography Activities; Revision 16
ACP 1411.13; Control of Locked High Radiation Areas and Above; Revision 22
RWP 09-30023; Diving in the Reactor Cavity; Revision No. 02
RWP 09-50380; Underwater Diving Work and Setup: Desludging, Inspection and Repair of the Torus Coating in the High Radiation Area

Section 2OS2

CAP 063690; Person Got Nasal Contamination While Cleaning Reactor Pressure Vessel Bolts On Reactor Building 5
CAP 058043; Use of Warehouse second Floor Would be Counterproductive to Rad Dose Goals
RFO-21; Dose Versus Goal for the Project: Total Dose

ALARA Job Planning Checklist; Diving in the Reactor Cavity to Repair the Feedwater Sparger-354 and Associated Support
ALARA Job Planning Checklist; Diving in Torus water to Vacuum Sludge and Repair Surface Coating; Perform General Area Work in Torus Proper and Support in Torus Rooms; RWP No. 50380 and 50382; dated February 2, 2009
ACP 1408.30; Control of Diving; Revision 3

Section 2PS3

LER 2009-002-00; DAEC Outdoor Liquid Radwaste Storage Tank Concentration Limit Exceeded; dated March 5, 2009
CAP 065914; REMP Air Sampling Enhancement Opportunity – New Air Sampling Station Located toward SE; dated March 18, 2009
CAP 058271; NCAQ-UFSAR Not Updated with Groundwater Changes Identified from NEI 07-07 Initiative; dated June 11, 2008
OTH 029324; Evaluation and Implementation of a Five-Year Review of SSCs and SCM for GWPP; dated May 28, 2008
OTH 028222; Expanded Buried Pipe Program to Include Tritium Containing Piping; dated April 24, 2008
Memorandum; Memo Describing the Progress Made Towards the Implementation of Groundwater Protection Program at Duane Arnold Site Before August 31, 2008; dated August 29, 2008
FPL- Quick Hit Self-Assessment Checklist Industry Groundwater Protection Initiative-NEI 07-07; dated August 20-22, 2008
CE 006513; NCAQ-UFSAR Not Updated with Groundwater Changes Identified from NEI 07-07; dated June 30, 2008
CE 006500; During A Nuclear Oversight Audit Of The Implementation Of The Ground Water Protection Program, Weakness Were Noted In The Formal Analysis Of System Structures And Components; dated May 2008
OTH 029213; Develop Tech Staff Training on GWPP; This Training Supports Technical Staff Professional Development; dated May 21, 2008
23 Revisions of ODAM Changes Reviewed form January 2007 Through March 2009
ODAM for Gaseous and Liquid Effluents; dated September 4, 2008
ESP-1.0; Radiological Environmental Monitoring Quality Control Program; Revision 11
ESP-4.1.1.1; General Water Quality Sample Collection; Revision 12
ESP-4.1.2; Terrestrial Sampling Procedure; Revision 6
ESP-4.3.1.1; Airborne Particulate and Iodine Sampling; Revision 28
ESP-4.3.1.2; Ambient Radiation Sampling; Revision 17
ESP-4.3.1.3.A; Surface water Sampling Procedure; Revision 19
ESP-4.3.1.5; Ground water Sampling Procedure; Revision 21
ESP-4.3.1.5A; Environmental Sampling Procedure of site Monitoring Wells; Revision 2
ESP-4.3.1.8; Vegetation Sampling Procedure; Revision 20
ESP-4.3.1.16; Special Radiological Environmental Sampling; Revision 9
ESP-4.3.1.17; REMP Surveillance of Site Construction Activities; Revision 1
ESP-4.4; Land Use Census Environmental Procedure; Revision 12
ESP-4.5; Statistical Comparison of TLDs For Direct Radiation Impact; Revision 5
ACP-1411.35 (Draft); The DAEC Groundwater Protection Program; Revision X
ACP-1402.3; Regulatory Reporting Activities; Revision 34
FORM HP-121; Health Physics Radioactive Spill Report Form; 2008
FPL Nuclear Fleet Ground Water Protection Program; Nuclear Program Description; revision No. 0; Effective Date May 5, 2008

Final Report to FPL Energy from Environmental Inc, Midwest Laboratory; REMP for Duane Arnold energy Center; Reporting Period January – December, 2008; dated February 5, 2009
Environmental Inc, Midwest Laboratory; Appendix A; Inter-laboratory Comparison Program Results for 2008; dated February 2009

Section 4OA1

ACP 1402.4; NRC and WANO Performance Indicator Reporting; Revision 13
DAEC PI Report for Unplanned Scrams per 7000 Critical Hours for January 2008 through December 2008
DAEC PI Report for Unplanned Scrams with Complications for January 2008 through December 2008
DAEC PI Report for Unplanned Power Changes per 7000 Critical Hours for January 2008 through December 2008
NEI 99-02; Regulatory Assessment Performance Indicator Guideline; Revision 5
DAEC First Quarter 2008 PI Summary, April 9, 2008
DAEC Second Quarter 2008 PI Summary, July 14, 2008
DAEC Third Quarter 2008 PI Summary, October 9, 2008
DAEC Fourth Quarter 2008 PI Summary, January 14, 2009
NRC PI Data Calculation, Review and Approval for Occupational Exposure Control Effectiveness and RETS/ODAM Radiological Effluent from the 1st Quarter 2008 through 4th Quarter 2008

Section 4OA2

ACP 114.5; Action Request System; Revision 72
PI-AA-204; Condition Identification and Screening Process; Revision 1
PI-AA-105; Condition Evaluation and Corrective Action; Revision 0

Section 4OA3

SpTP 213, “Increased Core Flow and Power Ascension Test to Greater Than 1880 MWth,” Revision 0
SpTP 214, “Pressure Regulator Dynamic Tuning,” Revision 0
Reactivity Management Plan: Rod/Flow Adjustments to Support Increased Core Flow; March 18, 2009
WM-AA-1000-F01; Data Sheet 1-Work Activity Risk Evaluation; Revision 1; Increase Core Flow Portion of SpTP 213 “Increase Core Flow and Power Ascension Test to Greater Than 1880 MWth”
WM-AA-1000-F02; Data Sheet 2-Site level High Risk Integrated Review; Revision 0; Increase Core Flow Portion of SpTP 213 “Increase Core Flow and Power Ascension Test to Greater Than 1880 MWth”
ACP 102.17; Pre/Post-Job Briefs and Infrequently Performed Tests and Evolutions; Revision 42
CAP 065972; Recirc Motor Data Collection Error during Increase Core Flow
CAP 065978; Recirc Pump Speed Monitoring
CAP 066055; Delay in Power Ascension Due to Procedure Issue
NG-02-0187; Startup Test Report for Extended Power Uprate – Phase I; dated March 4, 2002
NG-05-0516; Startup Test Report for Extended Power Uprate – Phase II; dated September 29, 2005
ACP 110.1; Conduct of Operations; Revision 22
CAP 63426; Unplanned Reactor Scram Due to Loss of Circulating Water Pit Level

RCE 1079; ‘B’ Cooling Tower West Riser Failure; Revision 0
CAP 064965; Ineffective Corrective Actions Associated with the ‘B’ Cooling Tower Failure
CAP 062934; NCAQ – MO4251 (Cooling Tower 1E-69B West Circulating Water Riser Isolation) Unreliability
CAP 063310; NCAQ – Circulating Water Startup Challenge
Event Notification 44821; Manual Reactor Scram due to Loss of Condenser Cooling
CAP 049225; PWO 1136642 for CB-5560 not Completed as Scheduled in Week 16
IPOI 5; Reactor Scram; Revision 49

Section 4OA5

SpTP 213, “Increased Core Flow and Power Ascension Test to Greater Than 1880 MWth,” Revision 0
SpTP 214, “Pressure Regulator Dynamic Tuning,” Revision 0
NG-02-0187; Startup Test Report for Extended Power Uprate – Phase I; dated March 4, 2002
NG-05-0516; Startup Test Report for Extended Power Uprate – Phase II; dated September 29, 2005

Section 4OA7

CAP 063486; CAQ - Increased Dose Rates in Radwaste Surge Tank 1T-088

LIST OF ACRONYMS USED

ACE	Apparent Cause Evaluation
ACP	Administrative Control Procedure
AFP	Area Fire Plan
ALARA	As-Low-As-Is-Reasonably-Achievable
AOP	Abnormal Operating Procedure
ARP	Alarm Response Procedure
ASME	American Society of Mechanical Engineers
CAP	Corrective Action Program
CFR	Code of Federal Regulations
CRD	Control Rod Drive
CS	Core Spray
CWO	Corrective Work Order
DAEC	Duane Arnold Energy Center
DC	Direct Current
DRP	Division of Reactor Projects
ECP	Engineering Change Package
EDG	Emergency Diesel Generator
ESW	Emergency Service Water
FAC	Flow Accelerated Corrosion
HCU	Hydraulic Control Unit
HPP	Health Physics Procedure
HVAC	Heating, Ventilation and Air Conditioning
IMC	Inspection Manual Chapter
IP	Inspection Procedure
ISI	Inservice Inspection
LCO	Limiting Condition for Operation
LER	Licensee Event Report
LLRPSF	Low-level Radwaste Processing and Storage Facility
LLRT	Local Leak Rate Testing
LOCA	Loss of Coolant Accident
LOOP	Loss of Offsite Power
LPCI	Low Pressure Coolant Injection
MAT	Modification Acceptance Test
MG	Motor-Generator
MSIV	Main Steam Isolation Valve
NCV	Non-Cited Violation
NEI	Nuclear Energy Institute
NRC	U.S. Nuclear Regulatory Commission
OBD	Operable But Degraded
ODAM	Offsite Dose Assessment Manual
ODCM	Offsite Dose Calculation Manual
ODMI	Operational Decision Making Instruction
OI	Operating Instruction
OOS	Out-of-service
ORP	Outage Risk Plan
PAM	Pressure Anchor Movement
PI	Performance Indicator
PI&R	Problem Identification and Resolution
PWO	Preventative Work Order

RCA	Radiologically Controlled Area
RCE	Root Cause Evaluation
RCIC	Reactor Core Isolation Cooling
RCS	Reactor Coolant System
REMP	Radiological Environmental Monitoring Program
RETS	Radiological Effluent Technical Specification
RFO	Refueling Outage
RFP	Reactor Feed Pump
RHR	Residual Heat Removal
RHRSW	Residual Heat Removal Service Water
RPV	Reactor Pressure Vessel
RWP	Radiation Work Permit
SBDG	Standby Diesel Generator
SBLC	Standby Liquid Control
SDP	Significance Determination Process
SLC	Standby Liquid Control
SpTP	Special Test Procedure
SSC	Structures, Systems, and Components
STP	Surveillance Test Procedure
TAM	Thermal Movement Anchor
TLD	Thermoluminescent Dosimeters
TS	Technical Specification
UFSAR	Updated Final Safety Analysis Report
URI	Unresolved Item
WO	Work Order
WRC	Work Request Card