April 15, 2003

Mr. Mark Peifer
Site Vice-President
Duane Arnold Energy Center
Nuclear Management Company, LLC
3277 DAEC Road
Palo, IA 52324

SUBJECT: DUANE ARNOLD ENERGY CENTER
USNRC INTEGRATED INSPECTION REPORT 50-331/03-03

Dear Mr. Peifer:

On March 31, 2003, the U.S. Nuclear Regulatory Commission (USNRC) completed an inspection at your Duane Arnold Energy Center. The enclosed report documents the inspection findings which were discussed on April 1, 2003 with Mr. J. Bjorseth 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. Specifically, this inspection focused on (reactor, radiation, and/or safeguards) safety.

Based on the results of this inspection, the inspectors identified six issues of very low safety significance (Green). Five of these issues were determined to involve violations of USNRC requirements. However, because of their very low safety significance and because these issues were entered into your corrective action program, the USNRC is treating these issues as Non-Cited Violations in accordance with Section VI.A.1 of the USNRC's Enforcement Policy. Finally, one violation of very low safety significance (Green) was identified by your staff and is listed in Section 4OA7 of this report.

If you contest the subject or severity of a Non-Cited Violation, you should provide a response within 30 days of the date of this inspection report, with a basis for your denial, to the 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, 801 Warrenville Road, Lisle, Il 60532-4351; the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington, DC 20555-0001; and the Resident Inspector Office at the Duane Arnold Energy Center.

In response to the terrorist attacks on September 11, 2001, the USNRC issued an Order and several threat advisories to commercial power reactors to strengthen licensees' capabilities and readiness to respond to a potential attack. The USNRC established a deadline of September 1, 2002, for licensees to complete modifications and process upgrades required by the order. In order to confirm compliance with this order, the USNRC issued Temporary Instruction 2515/148 and over the next year, the USNRC will inspect each licensee in accordance with this Temporary Instruction. The USNRC continues to monitor overall security controls and may issue additional temporary instructions or require additional inspections should conditions warrant.

In accordance with 10 CFR 2.790 of the USNRC's "Rules of Practice," a copy of this letter and its enclosure will be available electronically for public inspection in the USNRC Public Document Room or from the Publicly Available Records (PARS) component of USNRC's document system (ADAMS). ADAMS is accessible from the USNRC Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

/RA/

Bruce L. Burgess, Chief
Branch 2
Division of Reactor Projects

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

Enclosure: Inspection Report 50-331/03-03

cc w/encl: E. Protsch, Executive Vice President -
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J. Cowan, Chief Nuclear Officer
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The Honorable Charles W. Larson, Jr.
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REGION III

Docket No: License No:	50-331 DPR-49
Report No:	50-331/03-03
Licensee:	Alliant, IES Utilities Inc.
Facility:	Duane Arnold Energy Center
Location:	3277 DAEC Road Palo, Iowa 52324-9785
Dates:	December 29, 2002 through March 31, 2003
Inspectors:	G. Wilson, Senior Resident Inspector S. Caudill, Resident Inspector R. Daley, Reactor Inspector D. Schrum, Reactor Inspector
Approved by:	Bruce L. Burgess, Chief Branch 2 Division of Reactor Projects

SUMMARY OF FINDINGS

IR 05000331-03-03, IES Utilities, Inc.; on 12/29/2002-03/31/2003, Duane Arnold Energy Center; Flood Protection Measures, Operability Evaluations, Refueling and Outage Activities, Surveillance Testing, Event Follow-up, and Other Activities.

This report covers a 3-month period of baseline resident inspection. The inspection was conducted by Region III inspectors and the resident inspectors. This inspection identified six Green issues. Five of these issues involved Non-Cited Violations (NCV). 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 U.S. Nuclear Regulatory Commission (USNRC) management review. The USNRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, "Reactor Oversight Process," Revision 3, dated July 2000.

A. <u>Inspection Findings</u>

Cornerstone: Initiating Events

Green. A finding of very low safety significance was identified through a self revealing event when the licensee failed to adequately evaluate the operation of the 5B low pressure feedwater heater dump valve. The continuous operation of the dump valve resulted in a failure of the deflector plate, which caused a condenser tube leak, and a subsequent reactor scram.

The finding was more than minor, since it had an actual impact on safety and resulted in a reactor scram. This finding was determined to be of very low safety significance, since it did not impact any mitigating systems capability. No violation of USNRC requirements occurred. (Section 4AO3)

Cornerstone: Mitigating Systems

Green. A finding of very low safety significance was identified by the inspectors when the licensee's corrective actions failed to adequately address the degraded drains in the southeast corner room. The southeast corner room contains the "A" and "B" Residual Heat Removal Pumps and "A" Core Spray Pump. The corrective actions failed to address the debris that was on the room floor, which was sufficient to clog the drains.

The finding was more than minor, since there was potential that the drain system would be clogged by the floor debris, which would result in the drain system being unable to remove water from the room, thereby potentially affecting the safety-related pumps in the room. The finding was determined to be of very low safety significance, since the licensee has portable pumps available to remove water from the room. A Non-Cited Violation (NCV) of 10CFR50, Appendix B, Criterion XVI, related to inadequate corrective actions was identified by the inspectors. (Section 1R06)

Cornerstone: Mitigating Systems

Green. A finding of very low safety significance was identified by the inspectors when the licensee failed to evaluate the effect of the filter socks on the design of the drywell floor drain sump system during normal plant operation. The filter socks became clogged in three of the six drains.

The finding was more than minor, since there was potential that the remaining three drains could be clogged by debris, which would result in a significantly increased period of time for accumulated water from inside the drywell to over flow into the equipment sump and be measured as leakage. The finding was determined to be of very low safety significance, since other means remained available to detect an increase in unidentified leakage. A NCV of 10 CFR 50, Appendix B, Criterion III, “Design Control,” was identified by the inspectors for the failure to properly review the suitability of the drywell floor drain sump cover socks resulting in the delay of measured leakage from the drywell. (Section 1R20)

Cornerstone: Mitigating Systems

Green. A finding of very low safety significance was identified through a self-revealing event when the licensee failed to have an adequate procedure to perform the calibration of the Reactor Core Isolation Cooling (RCIC) turbine governor. The inadequate procedure resulted in an improper adjustment to the RCIC turbine governor gain and stability potentiometers which resulted in RCIC flow being below the Technical Specifications (TS) limit.

The finding was more than minor since the finding resulted in increased unavailability of the RCIC system. The finding was determined to be of very low safety significance, since the licensee did not exceed the Allowable Outage Time (AOT) and High Pressure Coolant Injection (HPCI) was always available. A NCV of 10 CFR 50, Appendix B, Criterion V, related to the inadequate procedure for performing the RCIC turbine governor calibration was identified through a self-revealing event. (Section 1R22)

Cornerstone: Mitigating Systems

Green. A finding of very low safety significance was identified by the inspectors when the licensee failed to properly check the design adequacy in a safety evaluation that justified elimination of Residual Heat Removal (RHR) pump mechanical seal cooling. The licensee had not evaluated appropriate seal failure mechanisms. This finding affects the mitigating system cornerstone objective because it could affect the capability of the RHR system to respond to initiating events with undesirable consequences.

This finding is considered to have greater than minor safety significance since if left uncorrected, the lack of a program to monitor and clean the RHR Mechanical Seal Heat Exchangers could have resulted in the failure of the heat exchanger to provide cool water to the RHR Pump Mechanical Seals. This could have resulted in the failure of the RHR Pump Mechanical Seals during an accident. A failure of the mechanical seals would have resulted in a failure of the RHR Pump. A NCV of 10 CFR 50, Appendix B, Criterion III, “Design Control,” was identified for the failure to properly review the removal of the RHR mechanical seal cooling for design adequacy by the inspectors. (Section 4A05)

Cornerstone: Barrier Integrity

Green. A finding of very low safety significance was identified by the inspectors when the licensee failed to have an adequate procedure for operating the fuel pool cooling system. The procedure failed to incorporate the new temperature limit associated with the installed Holtec fuel racks in the fuel pool, thereby allowing the licensing limit to be violated.

The finding was more than minor since there was potential for criticality to occur in the fuel pool. The finding was determined to be of very low safety significance, since actual fuel pool temperature never reached 39.2°F, which would have violated the licensing limit bases. A NCV of 10CFR50, Appendix B, Criterion V, related to inadequate procedure for operating the fuel pool cooling system was identified by the inspectors. (Section 1R15)

B. <u>Licensee-Identified Violations</u>

Violations of very low safety significance, which were identified by the licensee have been reviewed by the inspectors. Corrective actions taken or planned by the licensee have been entered into the licensee’s corrective action program. These violations and corrective action tracking numbers are listed in Section 40A7 of this report.

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REPORT DETAILS

Summary of Plant Status

The plant began the inspection period operating at full power. On February 1, 2003, the plant was manually scrammed due to a condenser tube leak. The condenser was repaired and the reactor was taken critical on February 14, 2003. The generator was placed on-line on February 15, 2003. Over the next few days the unit was gradually brought to full power and operated there until the plant was shutdown for a refueling outage on March 23, 2003. The plant remained shutdown for the remainder of the inspection period.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity

1R04 Equipment Alignment (71111.04)

.1 Partial Walkdowns

a. Inspection Scope

The inspectors performed a partial walkdown of the following equipment trains to ensure operability and proper equipment lineup. These systems were selected based upon risk significance, plant configuration, system work or testing, or inoperable or degraded conditions.

- “B” Standby Diesel Generator (SBDG), during the week of January 11, 2003.
- “1D2" 125 Volts Direct Current (Vdc) Battery, during the week of January 25, 2003.
- “1D1" 125 Vdc Battery, during the week of February 01, 2003.
- Reactor Core Isolation Cooling (RCIC), during the week of March 8, 2003.

The inspectors verified the position of critical redundant equipment and looked for any discrepancies between the existing equipment lineup and the required lineup.

- a review of plant procedures (including selected abnormal and emergency procedures), drawings, and the Updated Final Safety Analysis Report (UFSAR) to identify proper system alignment;
- a review of outstanding or completed temporary and permanent modifications to the system; and
- an electrical and mechanical walkdown of the system to verify proper alignment, component accessibility, availability, and current condition.

b. Findings

No findings of significance were identified.

.2 Complete Walkdowns

a. Inspection Scope

During the week of March 15, 2003, the inspectors performed a complete system alignment inspection of the Emergency Service Water (ESW) system. This system was selected because it was considered both safety-significant and risk-significant in the licensee's probabilistic risk assessment. The inspection consisted of the following activities:

- a review of plant procedures (including selected abnormal and emergency procedures), drawings, and the UFSAR to identify proper system alignment;
- a review of outstanding or completed temporary and permanent modifications to the system; and
- an electrical and mechanical walkdown of the system to verify proper alignment, component accessibility, availability, and current condition.

b. Findings

No findings of significance were identified.

1R05 Fire Protection (71111.05)

.1 Quarterly Fire Zone Inspections

a. Inspection Scope

The inspectors walked down the following risk significant areas looking for any fire protection issues. The inspectors selected areas containing systems, structures, or components that the licensee identified as important to reactor safety. The following walkdowns were performed:

During the week of January 04, 2003
- Area Fire Plan (AFP) -1; Torus and North Corner Rooms
- AFP-2; South Corner Rooms
- AFP-13; Refuel Floor
- AFP-22; Turbine Building South Floor

During the week of January 11, 2003
- AFP-3; HPCI/RCIC/RAD
- AFP-26; Control Room
- AFP-27; Control Room HVAC
- AFP-34; Radwaste Building , Drum Filling
- AFP-35; Radwaste Treatment

The inspectors reviewed the control of transient combustibles and ignition sources, fire detection equipment, manual suppression capabilities, passive suppression capabilities, automatic suppression capabilities, and barriers to fire propagation.

b. Findings

No findings of significance were identified.

.2 Annual Fire Drill Observation

a. Inspection Scope

On January 16, 2003, the inspectors conducted an annual observation of the station's fire brigade during a drill which simulated an oil fire in the mixed waste storage area of the radwaste building. The inspectors evaluated the readiness of the licensee's personnel to fight fires by verifying that: protective clothing/turnout gear was properly donned; self-contained breathing apparatus equipment was properly worn and used; fire hose lines were capable of reaching all necessary fire hazard locations, the lines were laid out without flow constrictions, the hoses were simulated being charged with water, and the nozzles were pattern (flow stream) tested prior to entering the fire area; the fire area was entered in a controlled manner; sufficient fire fighting equipment was brought to the scene by the fire brigade; the fire brigade leader's directions were thorough, clear, and effective; communications with plant operators and between fire brigade members were efficient and effective; the fire brigade checked for fire victims and for fire propagation into other plant areas; effective smoke removal operations were simulated; fire fighting preplan strategies were utilized; and the drill scenario was followed and the drill objectives met.

b. Findings

No findings of significance were identified.

1R06 Flood Protection Measures (71111.06)

a. Inspection Scope

The inspectors reviewed the licensee's flooding mitigation plans and equipment to determine consistency with design requirements and the risk analysis assumptions related to internal flooding associated with the Southeast Corner Room area during the week of March 1, 2003. The Southeast Corner Room was chosen since it contains the "A" Residual Heat Removal (RHR) pump, the "B" Residual Heat Removal (RHR) pump, and the "A" Core Spray pump. Walkdowns and reviews performed considered design measures, seals, drain systems, contingency equipment condition and availability of temporary equipment and barriers, performance and surveillance tests, procedural adequacy, and compensatory measures.

b. Findings

Introduction

A finding of very low safety significance (Green) and an associated NCV of 10 CFR 50, Appendix B, Criterion XVI, were identified for the failure of the corrective actions to adequately address the degraded drains in the southeast corner room by the inspectors.

Description

While performing the internal flooding inspection in the southeast corner room, the inspectors found the floor full of debris. The debris included wood chips, lagging metal buttons, various pieces of duct tape, lint free rags, pen, pencil, tie wraps, lagging straps, manilla labeling tags, caution tape, bags of insulation and dirt. The amount of debris on the floor was sufficient to clog the drain system. The drain system is used to remove liquid radwaste from the corner room; providing protection for the safety related pumps from leaking fluid sources within the room. Prior to the inspectors walkdown, the licensee had just completed the performance of a Corrective Action Plan (CAP) 025748 and a Condition Evaluation (CE) 000423, to evaluate the effect of paint debris plugging the drains in the southeast corner room. The evaluation stated that, “since only two floor drains were affected, the floor drain system was capable of providing a means to facilitate the rapid and efficient removal of liquid radwaste from the room.” The amount of debris found on the room floor by the inspectors was sufficient to plug all drains in the room. The debris was found by the inspectors after the licensee had completed corrective actions to evaluate plugging of the two floor drains in the corner room. The corrective actions taken by the licensee were not sufficient to ensure that the drain system remained capable of performing its function. The failure to perform effective corrective actions was a performance deficiency. The inspectors then informed plant management of the debris on the floor and questioned them on the corrective actions associated with the drain system in the southeast corner room. Plant management agreed that the housekeeping did not meet their standards and had the southeast corner room cleaned immediately. They also wrote CAP 026056 to capture both the housekeeping issues and the inadequate corrective actions. The inspectors determined that although the licensee’s corrective actions did not sufficiently address the clogging of the drains, portable pumps are available to remove water, therefore this finding was determined to be of very low safety significance.

Analysis

The inspectors reviewed this issue against the guidance contained in Appendix B, “Issue Dispositioning Screening,” of Inspection Manual Chapter (IMC) 0612, “Power Reactor Inspection Reports.” In particular, the inspectors compared this finding to the findings identified in Appendix E, “Examples of Minor Issues,” of IMC 0612 to determine whether the finding was minor. Following that review, the inspectors concluded that the guidance in Appendix E was not applicable or useful for the specific finding. As a result, the inspectors compared this performance deficiency to the minor questions contained in Section C, “Minor Questions,” to Appendix B of IMC 0612. The inspectors concluded that the issue was more than minor since the finding, if left uncorrected, could become a more significant safety concern. This conclusion was based on the fact the drain system is the primary system utilized to remove water from the southeast corner room. The debris would affect the operation of the drain system, which could potentially affect the operability of the safety-related RHR and Core Spray pumps if sufficient leakage were to occur in the room.

The inspectors reviewed this issue in accordance with Manual Chapter 0609, “Significance Determination Process (SDP)," Appendix A, "Significance Determination of Reactor Inspection Findings for At-Power Situations." The inspectors determined that the finding affected the Mitigating Systems Cornerstone; however, because the ineffective corrective actions did not constitute a design deficiency that resulted in a loss of function per Generic Letter (GL) 91-18, did not represent the actual loss of a safety function, did not exceed the Technical Specification (TS) Allowed Outage Time (AOT), did not represent an actual loss of safety function for a non-Tech Spec train, and were not risk significant due to seismic, fire, flooding or severe weather, this finding was screened as Green.

Enforcement

10 CFR 50, Appendix B, Criterion XVI, “Corrective Action,” requires that measures be established to assure that conditions adverse to quality, such as defective material and equipment, deficiencies, and nonconformances are promptly identified and corrected. The pertinent requirements of Appendix B apply to all activities affecting safety-related component operation. On February 28, 2003, the residents identified the failure to remove debris from the floor in the southeast corner room. This violation was identified subsequent to the completion of corrective actions by the licensee on February 26, 2003, associated with two degraded drains. The debris identified on the floor by the residents was sufficient to clog the degraded drains, thereby potentially affecting the operation of the RHR and Core Spray system pumps, which are Appendix B systems. The inadequate corrective actions to remove debris from the southeast corner room was considered an example of where the requirements of 10 CFR 50, Appendix B, Criterion XVI were not met and was a violation. However, because of its low safety significance and because it was entered into the corrective action program, the USNRC is treating this issue as a Non-Cited Violation (NCV 50-331/2003-003-01), in accordance with Section VI.A.1 of the USNRC’s Enforcement Policy. This issue was entered into the licensee’s corrective action program as CAP 026056.

Additional corrective actions taken included a thorough cleaning the southeast corner room, evaluation of the way corrective actions and condition evaluations are performed, and additional training to pertinent staff regarding housekeeping issues.

1R11 Licensed Operator Requalification Program (71111.11)

a. Inspection Scope

On January 14, 2003, the inspectors observed the performance of a training crew during an evaluated simulator scenario of Evaluated Scenario Guide (ESG) 42, which included an Anticipated Transient without a Scram (ATWS) with an Emergency Depressurization

(ED). The inspectors also reviewed licensed operator performance in mitigating the consequences of events.

The inspectors evaluated crew performance in the areas of:

- clarity and formality of communications;
- timeliness of actions, prioritization of activities;
- procedural adequacy and implementation;
- control board manipulations;
- managerial oversight, emergency plan execution; and
- group dynamics.

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

- Administrative Control Procedure (ACP) 110.1, "Conduct of Operations;" Revision 0,
- ACP 101.01, "Procedure Use and Adherence;" Revision 0, and
- ACP 101.2, "Verification Process and SELF/PEER Checking Practices;" Revision 5.

The inspectors assessed whether the crew completed the critical tasks listed in the above guidelines. The inspectors also compared simulator configurations with actual control board configurations. For any weaknesses identified, the inspectors verified that licensee evaluators also noted the same issues and discussed them during the end of session critique.

b. Findings

No findings of significance were identified.

1R12 Maintenance Effectiveness (71111.12)

a. Inspection Scope

The inspectors reviewed the licensee's implementation of the Maintenance Rule (10 CFR 50.65) to ensure rule requirements were met for the selected systems. The following systems were selected based on their being designated as risk significant under the Maintenance Rule, or were designated as being in Maintenance Rule category a(1) requiring increased monitoring:

- Automatic Depressurization System (ADS) during the week of January 18, 2003
- Reactor Protection System (RPS) during the week of March 15, 2003

The inspectors evaluated the licensee's categorization of specific issues, including the evaluation of performance criteria. The inspectors reviewed the licensee's implementation of the Maintenance Rule requirements, including a review of scoping, goal-setting, and performance monitoring; short-term and long-term corrective actions;

functional failure determinations associated with the condition reports reviewed; and current equipment performance status.

b. Findings

No findings of significance were identified.

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

a. Inspection Scope

The inspectors reviewed the licensee's evaluation of plant risk, scheduling, and configuration control. The inspectors also evaluated the performance of maintenance associated with planned and emergent work activities to determine if they were adequately managed. In particular, the inspectors reviewed the licensee's program for conducting maintenance risk safety assessments and to ensure that the licensee's planning, assessment and management of on-line risk was adequate. The inspectors also reviewed that licensee actions to address increased on-line risk during these periods, such as establishing compensatory actions, minimizing the duration of the activity, obtaining appropriate management approval, and informing appropriate plant staff, were accomplished when on-line risk was increased due to maintenance on risk-significant structures, systems, and components (SSCs). The following activities were reviewed:

- Maintenance risk assessment for work planned during the week of January 11, 2003.
- Maintenance risk assessment for work planned during the week of January 25, 2003.
- Maintenance risk assessment for work planned during the week of February 01, 2003.
- Maintenance risk assessment for work planned during the week of February 22, 2003.
- Maintenance risk assessment for work planned during the week of March 1, 2003.

b. Findings

No findings of significance were identified.

1R14 Personnel Performance During Non-routine Plant Evolutions and Events (71111.14)

.1 1D2 125Vdc Safety Related Battery Changeout

a. Inspection Scope

The inspectors observed the replacement of the 1D2 125Vdc safety-related battery that commenced on January 26, 2003. The inspectors reviewed the licensee's applicable procedures, licensing commitments, compensatory actions, personnel briefings, and

Action Requests (AR) generated to understand and resolve the details of this preplanned evolution. In particular, the inspectors reviewed the operators' actions to verify that they were appropriate for the evolution and in accordance with procedures and training. The inspectors performed a detailed walkdown of the job site and activity to ensure that all licensing commitments were performed. The inspectors observed that appropriate loads were transferred and that the 1D2 125Vdc battery was disconnected from the division two distribution bus. The temporary battery was then connected to the distribution bus in accordance with temporary modification 03-007. When the 1D2 125Vdc battery was replaced and tested, the inspectors observed that the temporary battery was disconnected and the permanent battery was connected to the distribution bus.

b. Findings

No findings of significance were identified.

.2 Plant Shutdown due to Condenser Tube Leak

a. Inspection Scope

The inspectors observed the plant shutdown on February 1, 2003, in response to a condenser tube leak. The inspectors also reviewed the licensee's apparent cause evaluation, applicable procedures, and the Action Requests (AR) generated to understand and resolve the details of this event. In particular, the inspectors observed and reviewed the operators' actions to verify that they were appropriate to the event and in accordance with procedures and training. The inspectors observed that, when Chemistry Action Level three was exceeded, an orderly shutdown was commenced. Reactor Water chemistry continued to degrade during the plant shutdown, so the inspectors observed the licensee perform a manual reactor SCRAM from 50 percent reactor power and cooldown the plant to place it in Mode 4, to minimize the potential plant impact of the chemistry excursion.

b. Findings

No findings of significance were identified.

.3 Plant Startup after Chemistry Excursion

a. Inspection Scope

The inspectors observed the plant startup after the chemistry excursion on February 14, 2003. The inspectors reviewed the licensee's applicable procedures, compensatory actions, personnel briefings, and Action Requests (AR) generated to understand and resolve the details of this preplanned evolution. In particular, the inspectors reviewed the operators' actions to verify that they were appropriate for the evolution and in accordance with procedures and training. The inspectors observed the licensee perform various plant system drains and flushes to prepare the systems for plant startup. After the licensee held an Infrequently Performed Test or Evolution (IPTE) briefing, the inspectors observed the licensee perform a reactor and plant startup.

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations (71111.15)

a. Inspection Scope

The inspectors assessed the following operability evaluations:

- Corrective Action Plan (CAP) 025018, “Evaluate Standby Filter”, during the week of January 4, 2003.
- CAP 025038, ”A” Standby Diesel Generator”, during the week of January 4, 2003.
- CAP 025389, “Fuel Pool Temperature Low Out of Spec”, during the week of February 15, 2003.
- CAP 025246, “Main Steam Isolation Valve (MSIV) and Turbine Stop Valve (TSV) Position Switch Analytic Limits for Reactor Protection System (RPS) trip input”, during the week of February 22, 2003.
- AR 32610, “Non-Conservative Temperature Input for Motor Degraded Voltage Calculation”, during the week of February 22, 2003.

The inspectors reviewed the technical adequacy of the evaluation against the TS, UFSAR, and other design information; determined whether compensatory measures, if needed, were taken; and determined whether the evaluations were consistent with the requirements of the licensees ACP-114.5, “Action Request System;” Rev. 32.

b. Findings

Introduction

A finding of very low safety significance (Green) and an associated NCV of 10 CFR 50, Appendix B, Criterion V, related to an inadequate procedure to operate the Fuel Pool Cooling System was identified by the inspectors.

Description

During the week of February 15, 2003, the inspectors evaluated CAP 025389 written for low out of specification fuel pool temperature. The CAP referenced the Operating Instruction (OI) 435, “Fuel Pool Cooling System” as the acceptable basis for low temperature. The inspectors questioned the approved spent fuel pool lower temperature limit stated as 32°Fahrenheit (F) in Operating Instruction (OI) 435, “Fuel Pool Cooling System.” A review of the Updated Final Safety Analysis Report (UFSAR) showed that there are two different types of fuel racks in the spent fuel pool, the PaR racks and the Holtec racks. The PaR racks have an analyzed low temperature limit of 32°F to ensure that Keff $\leq$ 0.95. The Holtec racks have an analyzed low temperature limit of 39.2°F to ensure that Keff $\leq$ 0.95. The TS states that spent fuel storage racks are designed and shall be maintained to ensure that Keff $\leq$ 0.95 if fully flooded with unborated water to ensure that criticality is not achieved. The fuel pool cooling procedure limit did not address the Holtec rack low temperature limit which is more conservative than that of the PaR racks. The failure to address the limiting temperature for the Holtec racks in the procedure, which is a performance deficiency, would potentially allow the fuel pool temperature to be lowered to a temperature that would not ensure that Keff $\leq$ 0.95 and that criticality is not achieved. The inspectors determined that although the licensee's procedure was inadequate to ensure that Keff $\leq$ 0.95, actual fuel pool temperature never reached 39.2°F, therefore this finding was determined to be of very low safety significance.

Analysis

The inspectors reviewed this issue against the guidance contained in Appendix B, "Issue Dispositioning Screening," of Inspection Manual Chapter (IMC) 0612, "Power Reactor Inspection Reports." In particular, the inspectors compared this finding to the findings identified in Appendix E, "Examples of Minor Issues," of IMC 0612 to determine whether the finding was minor. Following that review, the inspectors concluded that the guidance in Appendix E was not applicable or useful for the specific finding. As a result, the inspectors compared this performance deficiency to the minor questions contained in Section C, "Minor Questions," to Appendix B of IMC 0612. The inspectors concluded that the issue was more than minor since the finding, if left uncorrected, could become a more significant safety concern. This conclusion was based on the fact the procedure would allow the licensing bases for the plant to be violated and a potential for criticality to occur in the fuel pool.

The inspectors reviewed this issue in accordance with Manual Chapter 0609, "Significance Determination Process (SDP)," Appendix A, "Significance Determination of Reactor Inspection Findings for At-Power Situations." The inspectors determined that the finding affected the Containment Barriers Cornerstone and since the finding only represented a degradation of the radiological barrier function provided by the spent fuel pool, that the finding was screened as Green.

Enforcement

10 CFR 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," requires that activities affecting quality be prescribed by documented instructions, procedures, or drawings, of a type appropriate to the circumstances and shall be accomplished in accordance with these instructions. The licensing temperature limit for the fuel pool cooling system, which is an appendix B system for cooling purposes, was changed during the installation of the Holtec fuel racks in February 1994. The operating instructions for the fuel pool cooling system were not changed during the installation of the Holtec fuel racks to raise the temperature limit, thereby resulting in an inadequate procedure. The failure to have an adequate procedure to properly operate the Fuel Pool Cooling System to prevent pool criticality was an example where the requirements of 10 CFR 50, Appendix B, Criterion V, were not met and was a violation. However, because

of its low safety significance and because it was entered into the corrective action program, the NRC is treating this issue as a Non-Cited Violation (NCV 50-331/0303-02), in accordance with Section VI.A.1 of the NRC's Enforcement Policy. This issue was entered into the licensee's corrective action program as CAP025523.

Corrective actions taken included rewriting OI-435, "Fuel Pool Cooling System," to make the fuel pool low temperature limit 40°F which will ensure that Keff $\leq$ 0.95 thereby ensuring that the licensing limit is not exceeded.

1R16 Operator Workarounds (OWA) (71111.16)

a. Inspection Scope

The inspectors reviewed Operator Workaround CAP 025365, "Smoke in the Main Control Room," during the week of March 29, 2003, to identify any potential adverse impact on the function of mitigating systems or the ability to implement an abnormal or emergency operating procedure.

b. Findings

No findings of significance were identified.

1R17 Permanent Plant Modifications (71111.17)

a. Inspection Scope

The inspectors reviewed the following modifications to verify that the design basis, licensing basis, and performance capability of risk significant systems were not degraded by the installation of the modification. The inspectors also ensured that the modifications did not place the plant in an unsafe configuration.

- Engineered Maintenance Action (EMA) A58888, Replace ASCO SCRAM Solenoid Valves, during the week of March 22, 2003.

The inspectors considered the design adequacy of the modification by performing a review, or partial review, of the modification's impact on plant electrical requirements, material requirements and replacement components, response time, control signals, equipment protection, operation, failure modes, and other related process requirements.

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. Activities were selected based upon the structure, system, or component's ability to impact risk.

- CWO A60577; Speed Switch Replacement; during the week of January 11, 2003
- CWO A60103; Power Supply Negative Side Failed Downscale; during the week of January 18, 2003
- CWO A50996; 1D1 125 Vdc Battery Change Out; during the week of January 25, 2003
- CWO A53415; 1D2 125 Vdc Battery Change Out; during the week of February 01, 2003
- CWO A60263; Replacement of 45 degree elbow on the Residual Heat Removal Service Water System; during the week of February 08, 2003
- CWO 1124677; Condensate Storage Tank 1T-5B Level (Low Level Suction transfer for High Pressure Coolant Injection (HPCI) and Reactor Core Isolation Cooling (RCIC); during the week of February 15, 2003

The inspectors ensured by witnessing the test or reviewing the test data that post-maintenance testing activities were adequate for the above maintenance activities. The inspectors reviews included, but were not limited to, integration of testing activities, applicability of acceptance criteria, test equipment calibration and control, procedural use and compliance, control of temporary modifications or jumpers required for test performance, documentation of test data, TS applicability, system restoration, and evaluation of test data. Also, the inspectors reviewed that maintenance and post-maintenance testing activities adequately ensured that the equipment met the licensing basis and Updated Final Safety Analysis Report (UFSAR) design requirements.

b. Findings

No findings of significance were identified.

1R20 Refueling and Outage Activities (71111.20)

1. Forced Outage for Condenser Tube Leak

a. Inspection Scope

The inspectors observed shutdown activities for the forced outage, to investigate the condenser tube leak, which began on February 1, 2003. The inspectors monitored the licensee's cooldown process and ensured that TS were followed during the transition into Modes three and four. The licensee, as part of the 14-day outage, significant reduced indicated drywell leakage by repairing the 3B drywell cooler. Additionally, the licensee repaired the 5B feed water heater drain valve. The inspectors monitored outage configuration management on a daily basis by verifying that the licensee maintained appropriate defense in depth to address all shutdown safety functions and satisfy TS requirements. Proper operation of the decay heat removal system was reviewed during multiple control room tours and observations. The licensee restarted the reactor on February 14, 2003.

b. Findings

Introduction

A finding of very low significance (Green) and an associated NCV of 10 CFR 50, Appendix B, Criterion III, "Design Control" related to the failure to evaluate the addition of the filter socks on the operation of the drywell floor drain system were identified by the inspectors.

Description

On February 2, 2003, the inspectors toured the drywell during the forced outage to evaluate the leaking 3B drywell cooler. A considerable amount of water on the floor was noticed by the inspectors and the licensee was questioned on the effectiveness of the drywell floor drains. An Apparent Cause Evaluation (ACE) was performed to analyze the issue and it was estimated that approximately 70 gallons of water were residing on the drywell floor at depths up to 3/4 inch. This quantity of water was due to debris that had clogged the drywell floor drain sump cover filter socks in three of the six drains. The clogging of the filter socks restricted water flow through the floor drain to the drywell floor sump. The inspectors then questioned the licensee regarding the design change that would allow the filter socks to be installed during normal plant operation. The ACE identified that the licensee failed to evaluate the impact on system operation with the filter socks installed during normal operation, which is a performance deficiency. The failure to properly evaluate the addition of the filter socks was an example of inadequate design control due to the adverse impact of the filter socks on the drywell floor drain sump system. The inspectors determined that although the drywell floor drain sump flow monitoring system was adversely affected, other multiple independent means for detecting drywell leakage were still available and working as designed, therefore this finding was determined to be of very low safety significance.

Analysis

The inspectors reviewed this issue against the guidance contained in Appendix B, "Issue Dispositioning Screening," of Inspection Manual Chapter (IMC) 0612, "Power Reactor Inspection Reports." In particular, the inspectors compared this finding to the findings identified in Appendix E, "Examples of Minor Issues," of IMC 0612 to determine whether the finding was minor. Following that review, the inspectors concluded that the guidance in Appendix E was not applicable or useful for the specific finding. As a result, the inspectors compared this performance deficiency to the minor questions contained in Section C, "Minor Questions," to Appendix B of IMC 0612. The inspectors concluded that the issue was more than minor since the finding, if left uncorrected, could become a more significant safety concern. This conclusion was based on the fact that there was potential that the remaining three drains could be clogged by debris, which would result in an increased transient time to over flow into the equipment sump. The increased transient time would delay the leak detection. Without adequate detection, a small initial leak could become larger and therefore become a more significant concern prior to its detection by other means.

As a result, the inspectors reviewed this issue in accordance with Inspection Manual Chapter (IMC) 0609 “Significance Determination Process (SDP).” The inspectors determined that the finding affected the Mitigating Systems Cornerstone, however, since the installation of the filter socks was not a design deficiency that resulted in a loss of function per Generic Letter (GL) 91-18, did not represent the actual loss of a safety function, did not exceed the TS Allowed Outage Time (AOT), did not represent an actual loss of safety function for non-Tech Spec train, and was not risk significant due to seismic, fire, flooding or severe weather, that the finding was screened as Green.

Enforcement

10 CFR 50, Appendix B, Criterion III, “Design Control,” requires that measures shall be established for the selection and review for suitability of application of materials, parts, and equipment that are essential to the safety-related functions of structures, systems, and components. The failure to review the suitability of the application of the filter socks for the drywell floor drain sump covers during normal plant operations adversely impacted the response of the drywell floor drain monitoring system, which is an appendix B system, due to debris collecting in the drain filter socks, which were installed in September 2002. The debris caused a significant flow restriction in three of the six drains, thereby delaying the drywell floor drain sump monitoring system ability of measuring and detecting leaks. The failure to analyze the suitability of the drywell floor drain sump cover socks was an example where the requirements of 10 CFR 50, Appendix B, Criterion III, were not met and was a violation. However, because of its low safety significance and because it was entered into the corrective action program, the NRC is treating this issue as a Non-Cited Violation (NCV 50-331/0303-03), in accordance with Section VI.A.1 of the NRC’s Enforcement Policy. This issue was entered into the licensee’s corrective action program as CAP025394.

Corrective actions taken included the removal of the drain socks on the drywell closeout check list, which will ensure that the drains will not get clogged by debris.

2. Refueling Outage Number 18

a. Inspection Scope

The inspectors evaluated outage activities for a Scheduled Refueling Outage Number 18 that began on March 23, 2003, and continued through the end of the inspection period. The inspectors reviewed activities to ensure that the licensee considered risk in developing, planning, and implementing the outage schedule.

The inspectors observed or reviewed the reactor shutdown and cooldown, outage equipment configuration and risk management, electrical lineups, selected clearances, control and monitoring of decay heat removal, control of containment activities, startup and heatup activities, and identification and resolution of problems associated with the outage.

b. Findings

No findings of significance were identified.

1R22 Surveillance Testing (71111.22)

a. Inspection Scope

The inspectors selected the following surveillance test activities for review. Activities were selected based upon risk significance and the potential risk impact from an unidentified deficiency or performance degradation that a system, structure, or component could impose on the unit if the condition were left unresolved.

- Surveillance Test Procedure (STP) 3.3.1.1-24; “Local power Range Monitor Calibration”; during the week of January 4, 2003.
- STP 3.3.5.1-03, “Functional Test of LPCI Loop Select - Reactor Vessel Water Level Low-Low Instrumentation”, during the week of January 4, 2003.
- STP 3.3.6.1-47; “High Pressure Coolant Injection (HPCI) Exhaust Diaphragm Channel Functional Test”, during the week of January 4, 2003.
- STP 3.3.3.2-09, “Reactor Water Level and Pressure Instruments Calibration”, during the week of January 11, 2002.
- STP 3.3.5.1-03, “Functional Test of Low Pressure Coolant Injection (LPCI) LOOP Select”, during the week of January 11, 2002.
- STP 3.5.3-02, “Reactor Core Isolation Cooling (RCIC) System Operability Test”, during the week of March 8, 2003.
- STP 3.6.1.1-06, “Containment Isolation Valve Leak Tightness Test - Type C Penetrations - Feedwater System”, during the week of March 29, 2003.

The inspectors observed or reviewed the performance of surveillance testing activities, including reviews for preconditioning, integration of testing activities, applicability of acceptance criteria, test equipment calibration and control, procedural use, control of temporary modifications or jumpers required for test performance, documentation of test data, TS applicability, impact of testing relative to performance indicator reporting, and evaluation of test data.

b. Findings

Introduction

A finding of very low safety significance (Green) and an associated NCV of 10 CFR 50, Appendix B, Criterion V was identified due to an inadequate maintenance procedure that was used to calibrate the RCIC Turbine Governor through a self-revealing event.

Description

During the week of March 8, 2003, the inspectors reviewed the performance of STP 3.5.3-02, “RCIC System Operability Test.” During the surveillance test, the RCIC pump failed to meet the required 400 gallons per minute (g.p.m.) flow required, and also had a very erratic output, indicating a governor control problem. The licensee commenced troubleshooting the governor control problem and performed an ACE on the failure to meet the required flow. The troubleshooting identified that the method used to conduct the voltage checks during the calibration of the RCIC turbine Governor in accordance with TURBINE-T147-02, “Calibration of RCIC Turbine Governor,” affected the flow output of the RCIC pump. It was identified that when adjustments are performed on the gain and stability potentiometers of the Electronic Governor Module (EGM), these adjustments can change the Electronic Governor Regulator (EGR) voltage and overall stability of the system. The low and erratic flow output of the RCIC pump were attributed to the adjustments made to the gain and stability potentiometers of the EGM. When the voltage checks were performed, the maintenance technician inappropriately performed an initial potentiometer setting instead of the verification check because the procedure was not detailed enough to identify when the appropriate sections were to be performed. The initial setting adjusts the stability and gain potentiometers to verify voltage. The verification check only measures system voltage to verify output voltage is appropriate. The adjustment, which was performed on the potentiometers during the check, resulted in the unexpected output of RCIC and is considered a performance deficiency. The unexpected output resulted in troubleshooting the control circuitry and doubling the scheduled unavailability of RCIC, thereby keeping the plant at an elevated risk condition. This finding was determined to be of very low safety significance, since High Pressure Coolant Injection (HPCI) was always available and the TS Allowed Outage Time (AOT) was not exceeded.

Analysis

The inspectors reviewed this issue against the guidance contained in Appendix B, “Issue Dispositioning Screening,” of IMC 0612, “Power Reactor Inspection Reports.” In particular, the inspectors compared this finding to the findings identified in Appendix E, “Examples of Minor Issues,” of IMC 0612 to determine whether the finding was minor. Following that review, the inspectors concluded that the guidance in Appendix E was not applicable or useful for the specific finding. The inspectors determined that finding was more than minor since it affected the mitigating system attribute of equipment performance by increasing the unavailability of RCIC, thereby increasing overall plant risk.

The failure to have an adequate procedure to perform TURBINE-T147-02, “Calibration of RCIC Turbine Governor” resulted in an unexpected governor and stability response which resulted in additional unavailability of RCIC, warranting further review in accordance with IMC 0609, “Significance Determination Process (SDP).” The inspectors determined that the finding affected the Mitigation Systems Cornerstones; however, the loss of RCIC was not a design deficiency that resulted in a loss of function per GL 91-18, did not represent the actual loss of a safety function, did not exceed the TS Allowed Outage Time (AOT), did not represent an actual loss of safety function for non-Tech

Spec train, and was not risk significant due to seismic, fire, flooding or severe weather. Therefore, the finding was screened as Green.

Enforcement

10 CFR 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," requires in part that activities affecting quality be prescribed by procedures of a type appropriate to the circumstances and shall be accomplished in accordance with these instructions. The failure to have an adequate procedure, due to inadequate details on what section to perform on the gain and stability potentiometers, during the calibration of the RCIC turbine Governor on February 18, 2003, resulted in an increased unavailability of RCIC, which is an Appendix B system, until February 20, 2003, thereby increasing the time period of elevated plant risk. The inadequate procedure to calibrate the RCIC governor is an example where the requirements of 10 CFR 50, Appendix B, Criterion V, were not met and was a violation. However, because of its low safety significance and because it was entered into the corrective action program, the NRC is treating this issue as a Non-Cited Violation (NCV 50-331/02-07-04), in accordance with Section VI.A.1 of the NRC's Enforcement Policy. This issue was entered into the licensee's corrective action program as CAP 025715.

Corrective actions taken included the revision of the Electronic Governor Regulator (EGR) procedure and the evaluation of the calibration.

1R23 Temporary Plant Modifications (71111.23)

a. Inspection Scope

The inspectors reviewed the following temporary modifications (TMOD):

- TMOD 02-061, "Core Spray Pump 1P-211A Motor Bearing Upper Temp" , during the week of January 4, 2003.
- TMOD 03-03, "1D1 125 Vdc Temporary Battery", during the week of January 25, 2003.

The inspectors reviewed the safety screening, design documents, USFAR, and applicable TS to determine that the temporary modification was consistent with modification documents, drawings and procedures. The inspectors also reviewed the post-installation test results to confirm that tests were satisfactory and the actual impact of the temporary modification on the permanent system and interfacing systems were adequately verified.

b. Findings

No findings of significance were identified.

1EP6 Emergency Preparedness Drill Evaluation (71114.06)

a. Inspection Scope

On February 26, 2003, the inspectors observed an operating crew participate in an emergency preparedness simulator drill. The inspectors monitored the operations crews' response to a design basis earthquake, loss of the "B" Reactor Protection System power supply, a Reactor Water Cleanup System leak, a Reactor Core Isolation Cooling steam line leak, and an eventual fuel failure with an off-site radiation release. The inspectors verified that appropriate actions were taken by the operators, the proper emergency procedures were implemented, and that the crew made the proper emergency classifications in a timely manner. The inspectors also attended the licensee's critique to verify that personnel adequately evaluated the crew's emergency plan implementation.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA1 Performance Indicator Verification (71151)

Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity, and Occupational Radiation Safety

.1 Initiating Events, Mitigating Systems and Barrier Integrity Performance Indicator Verification

a. Inspection Scope

The inspectors reviewed Licensee Event Reports (LERs), licensee memoranda, plant logs, and USNRC inspection reports to verify the following performance indicators through the Fourth quarter of 2002.

- Safety System Unavailability, RCIC System, during the week of February 22, 2003.
- Safety System Unavailability, HPCI System, during the week of February 22, 2003.

The inspectors verified that the licensee accurately reported performance as defined by the applicable revision of Nuclear Energy Institute Document 99-02, "Regulatory Assessment Performance Indicator Guideline."

b. Findings

No findings of significance were identified.

4OA2 Identification and Resolution of Problems (71152)

The inspectors selected the issue identified below for additional review.

In conducting the review, the inspectors considered the nature and significance of the issue with respect to safety, risk, and licensee corrective action procedural requirements. Attributes considered during the review of licensee actions included complete and accurate identification of the problem; timeliness was commensurate with the safety significance; 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 the classification, prioritization, focus, and timeliness of corrective actions.

.1 CAP 025009; 125 Volt fuse found in a 480 Volt application for the Reactor Building Crane

a. Introduction

The inspectors selected the corrective actions associated with the 125 Volt fuse that was found installed in a 480 Volt application for the Reactor Building Crane.

b. Effectiveness of Problem Identification

(1) Inspection Scope

The inspectors evaluated whether the licensee's identification of the problem was complete, accurate, and timely, and that the consideration of extent of condition, generic implications, common cause and previous occurrences was adequate.

(2) Issues

The inspectors reviewed the licensee's corrective actions to address the improper fuses that were found in the Reactor Building Crane. During troubleshooting activities associated with the crane, two blown fuses were found. While evaluating the blown fuses, it was discovered that the fuses were 125 Volt 2 amp fuses installed on a 480-Volt system. The proper fuses were procured in accordance with ACP 1408.15, "Control of Replacement Fuses" and installed in the plant. The licensee performed an ACE on the issue. A review of available information associated with the system did not identify any conclusive cause for the fuse problem so the apparent cause was unknown.

The inspectors performed a review of the corrective action data base for additional examples of improperly placed fuses in the plant. The search found five additional examples of improper fuses found in the plant within the last year. In response to inspector questions on trending, the licensee stated that each CAP has factors or codes for trending. When there are a noticeable number of similar issues a trend CAP is written that will examine the extent of condition and generic implications of the issue. Based on the inspectors questions, the licensee wrote CAP 025977 to evaluate the potential trend in the fuse control program. The evaluation showed that all the fuse CAP's had been evaluated as a stand alone items and were not evaluated as potential trends. The licensee also performed a review existing condition reports to evaluate them for cross cutting themes and entered CAP 026297 into their program to evaluate the way trend analyses were performed.

.2 Routine Review of Identification and Resolution of Problems

a. Inspection Scope

As 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 corrective action system at an appropriate threshold, that adequate attention was being given to timely corrective actions, and that adverse trends were identified and addressed. Minor issues entered into the licensee's corrective action system as a result of the inspectors' observations are generally denoted in the report.

b. Findings

A specific issue related to failure to perform adequate corrective actions on the southeast corner room drains was discussed in Section 1R06.

4OA3 Event Follow-up (71153)

.1 (Closed) Licensee Event Report (LER) 50-331/03-01: "Manual Reactor Scram and Reactor Coolant Chemistry Excursion Due to Punctured Main Condenser Tube Caused by Failed Condenser Deflector Plate"

a. Inspection Scope

The inspectors evaluated Licensee Event Report (LER) 50-331/03-01: "Manual Reactor Scram and Reactor Coolant Chemistry Excursion Due to Punctured Main Condenser Tube Caused by Failed Condenser Deflector Plate."

b. Findings

Introduction

A finding of very low safety significance (Green) was identified due to the failure to properly evaluate the long term effect of continually operating the 5B feedwater heater dump valve through a self-revealing event.

Description

In October 2002, plant management made a decision to run the plant for several months by directing the 5B low pressure feedwater heater dump valve directly into the main condenser due to problems with the 5B low pressure feedwater heater drain valve. The design of the dump valve is to handle flow intermittently rather than continuous flow. On January 31, 2003, at 2242 the plant received influent high conductivity alarms. Primary

Plant Chemistry parameters continued to degrade in the plant until the unit was manually scrammed on February 1, 2003. at 0233. The licensee entered the Scram into their corrective action program as CAP 025382 and also performed Root Cause Report 001001, “Reactor SCRAM.” The root cause report identified a punctured condenser tube as the initiating event for the resultant chemistry excursion and associated reactor scram. The condenser tube was punctured by the impact of the deflector plate for the 5B dump valve which had broken free. The dump valve has a deflector plate installed over the condenser penetration to protect tubes from the direct impact of water/steam being dumped into the condenser. The deflector plate broke free due to a failed fillet weld caused by high cycle fatigue failure. The decision to allow flow to go through the dump valve continuously caused the high cycle fatigue failure. The decision to continue to operate the 5B dump valve in an abnormal condition is a performance deficiency that directly resulted in a broken deflector plate and subsequent condenser tube failure causing control room operators to manually scram the plant due to degraded plant chemistry parameters. This finding was determined to be of very low safety significance, since it did not impact any mitigating systems capability.

Analysis

The inspectors reviewed this issue against the guidance contained in Appendix B, “Issue Dispositioning Screening,” of IMC 0612, “Power Reactor Inspection Reports.” In particular, the inspectors compared this finding to the findings identified in Appendix E, “Examples of Minor Issues,” of IMC 0612 to determine whether the finding was minor. Following that review, the inspectors concluded that the guidance in Appendix E was not applicable or useful for the specific finding. The inspectors determined that the finding was more than minor, since it had an actual impact on safety and resulted in a reactor scram.

The failure to evaluate the long term operation of the 5B dump valve resulted in a reactor scram, warranting further review in accordance with IMC 0609, “Significance Determination Process (SDP).” The inspectors determined that the finding affected the Initiating Event Cornerstone; however, the finding did not contribute to the likelihood of a Primary or Secondary Loss of Coolant Accident (LOCA), affect mitigating equipment, or increase the likelihood of a fire or flood. Therefore, the finding was screened as Green.

Enforcement

The inspectors determined that no violations of NRC requirements occurred during the evaluation of continued operation on the 5B dump valve or the resultant condenser tube rupture on February 1, 2003, due to the heater drain system being non safety-related.

Corrective Actions taken include repairing the 5B drain valve, repairing the 5B dump valve deflector plate, additional deflector plate inspections, and various procedure changes for operator actions.

4OA5 Other Activities

.1 (Closed) Non-Cited Violation 50-331-02-11-01(DRS): Corrective Action document CAP 025017 addressed issues related to battery capacity with only 57 battery cells in service (one cell jumpered). DAEC revised their design basis calculations to show that the safety related batteries could perform their function as designed with one cell jumpered out. These revised calculations addressed other design basis conditions, but they did not fully address a Station Blackout (SBO) event. However, as a long term action, CAP 025017 identified that DAEC intends to revise the SBO design bases to demonstrate that the 1D1 battery has the capability to cope with a Station Blackout with one battery cell jumpered out of service. As an interim action until the calculation could be performed, DAEC performed a technical evaluation and attached it to CAP 025017 to demonstrate that the 1D1 battery had the capability to cope with an SBO with a battery cell jumpered out of service. Since one divisional battery will always be available to be able to cope with an SBO, this NCV can be closed.

.2 Inspection Scope

(Closed) Unresolved Item (URI) 50-331/02-11-02 Review Impact of Removing RHR Pump Seal Cooling Requirements

Introduction

A finding of very low safety significance (Green) and an associated NCV of 10 CFR Part 50, Appendix B, Criterion III, related to the licensee's failure to identify in a 10 CFR 50.59 evaluation that seal cooling water must be available to prevent damage to the RHR mechanical seals was identified by the inspectors.

Description

On August 2, 1999, the licensee performed a 10 CFR 50.59 evaluation (Safety Evaluation (SE) 99-041) to support the decision to not inspect or flow test the cooling water supply to the RHR pumps mechanical seal heat exchangers. The reason for the licensee's evaluation was the inability to perform a flow test due to the configuration of the associated cooling water piping. The four RHR heat exchangers had not been tested or cleaned in the past. Upon completion of SE 99-041, the licensee changed the Duane Arnold FSAR and TS Basis to eliminate the need for the heat exchanger cooling. The licensee assumed it could operate the RHR Pumps with the heat exchangers plugged the remaining life of the plant.

The licensee based SE 99-041 on the Byron Jackson Pump Division (Borg-Warner Corporation) vendor manual, "Installation, Operation, and Maintenance Instructions for Type 'U' Mechanical Seals," issued in 1969, which stated that the mechanical seal components were designed for temperatures up to 450 degrees Fahrenheit (F). However, the licensee failed to take into account and did not address the critical temperature for operating the mechanical seals which is the temperature allowed for the actual sealing surface face and all of the associated potential seal failure mechanisms. The vendor data indicated that the maximum water temperature for these seal faces was

150 degrees F. The inspectors determined that without a cooling water supply to the mechanical seals the water remaining in the seal will not support the sealing surface as a lubricant. The seal would begin to degrade in a relatively short period of time depending on the original condition of the seal. The rate of degradation and hence overall impact on the accident scenario had not been evaluated by the licensee.

The licensee performed an Operability Evaluation (AR#30414), March 28, 2002, for continued plant operations. The inspectors considered the RHR pump seals operable based on the fact that water could still be seen flowing in the RHR Heat Exchanger sight glasses. In addition, it appeared that no RHR seals were damaged during the last shutdown of the plant when the seals were subjected to temperatures above 300 degrees F. The licensee issued AR 30234 dated March 17, 2002, to resolve the issues related to the evaluation of the seals. The NRC reviewed the licensee's evaluation to demonstrate that all seals met their design basis conditions. The licensee subsequently evaluated the most likely seal failure mechanism and concluded that degradation would not be as rapid or severe enough within the applicable time frame to adversely impact the design basis function. While this conclusion would apply to both the RHR and Core Spray Pumps, some engineering judgement was necessary to reach that conclusion and design margin remained a consideration.

Analysis

The inspectors reviewed this issue against the guidance contained in Appendix B, "Issue Dispositioning Screening," of Inspection Manual Chapter (IMC) 0612, "Power Reactor Inspection Reports." Since this deficiency should have been prevented, was reasonably within the licensee's ability to foresee and correct, and has safety significance, it is considered a finding. This finding is associated with the initial design attribute of the mitigating systems cornerstone. This finding affects the mitigating system cornerstone objective because it could affect the capability of the RHR system to respond to initiating events with undesirable consequences. Therefore, this finding is considered to have greater than minor safety significance since if left uncorrected, the lack of a program to monitor and clean the RHR Mechanical Seal Heat Exchangers could have resulted in the failure of the heat exchanger to provide cool water to the RHR Pump Mechanical Seals. This could have resulted in the failure of the RHR Pump Mechanical Seals during an accident. A failure of the mechanical seals would have resulted in a failure of the RHR Pump.

The inspectors reviewed this issue in accordance with Manual Chapter 0609, "Significance Determination Process (SDP)," Appendix A, "Significance Determination of Reactor Inspection Findings for At-Power Situations," and determined that this finding was of very low safety significance because although the RHR Mechanical Seal Heat Exchangers had been removed from the GL 89-13 program and had not been tested or cleaned, they appeared to still perform their function of providing cooling water to the RHR mechanical seals. Water was still identified as flowing in the sight-glasses.

Enforcement

10 CFR Part 50, Appendix B, Criterion III, “Design Control,” states, in part, that measures shall be provided for checking the adequacy of design. Contrary to the above, on August 31, 1999, the licensee failed to adequately check the design adequacy in SE 99-041 in that potential seal failure mechanisms were not adequately considered in the justification to eliminate the need for RHR Pump mechanical seal cooling. This finding is considered a violation of 10 CFR Part 50, Appendix B, Criterion III. Because of the very low safety significance, this violation is being treated as a Non-Cited Violation (NCV 50-331/03-03-06) consistent with Section VI.A.1 of the NRC Enforcement Policy. This issue is in the licensee’s corrective action program as AR 30234. The RHR Pump Mechanical Seal Heat Exchangers were returned to the GL 89-13 program. The licensee has a planned modification to replace the RHR Pump Mechanical Seal Heat Exchangers and to replace the RHR Pump and Core Spray Pump Mechanical Seals with high temperature seals. URI 50-331/2002-011-02 is considered closed.

4. OTHER ACTIVITIES

4OA6 Meetings

.1 Exit Meeting

The inspectors presented the inspection results to Mr. J. Bjorseth and other members of licensee management at the conclusion of the inspection on April 1, 2003. The inspectors asked the licensee whether any materials examined during the inspection should be considered proprietary. No proprietary information was identified.

.2 Interim Exit Meetings

Interim exits were conducted for:

- Review URI 2002-011-02 with Mr. J. Bjorseth on January 28, 2003.

4OA7 Licensee-Identified Violations

The following violations of very low significance were identified by the licensee and are violations of USNRC requirements which meets the criteria of Section VI of the USNRC Enforcement Manual, NUREG-1600, for being dispositioned as NCVs.

Cornerstone: Mitigating Systems

The fire suppression system in the Control Room Heating, Ventilation, and Air Conditioning (HVAC) Equipment Room, which is the #12 sprinkler system, is required to be installed and operate in accordance with National Fire Protection Association (NFPA)-13-1983. On January 25, 2003, licensee personnel identified that two sprinkler’s heads associated with the Control Room HVAC Equipment Room were not installed per NFPA-13-1983. One of the sprinkler heads was not in compliance with Section 4-4.13 of the code which requires that sprinklers be installed beneath ducts that are more than four feet wide. The other sprinkler was not in compliance with section 4-3.1 of the code which requires that sprinkler heads be installed one to twelve inches below a non combustible ceiling. The placement of the sprinkler heads decreased the actuation time of the heads, which are activated by temperature, by positioning them closer to the ceiling or duct. Because the licensee was able to demonstrate that a fire in this area would not adversely affect the safe shutdown capability of the plant, this violation is not more than of very low safety significance, and is being treated as a Non-Cited Violation (50-331/0303-07).

KEY POINTS OF CONTACT

Licensee

M. Peifer, Site Vice-President Nuclear
J. Bjorseth, Plant Manager
D. Curtland, Training Manager
T. Evans, Manager, Engineering
P. Hansen, Operations Manager
B. Kindred, Security Manager
S. Nelson, Manager, Radiation Protection
K. Putnam, Licensing Manager
W. Simmons, Maintenance Manager
D. Wheeler, Chemistry Manager

Nuclear Regulatory Commission

D. Hood, Project Manager, NRR
B. Burgess, Chief, Reactor Projects Branch 2

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

50-331/2003-003-01	NCV	Inadequate Corrective Actions for South East Corner Rooms Drains
50-331/2003-003-02	NCV	Inadequate Fool Pool Cooling System Procedure
50-331/2003-003-03	NCV	Inadequate Design Control on Drywell Floor Drain Sump System
50-331/2003-003-04	NCV	Inadequately performed calibration on the RCIC Turbine Governor
50-331/2003-003-05	FIN	Inadequate evaluation on continued operation of the 5B dump valve resulted in a Manual Reactor SCRAM
50-331/2003-003-06	NCV	The licensee failed to verify design pump seal failure mechanisms in a 10 CFR 50.59 evaluation to eliminate the need for RHR pump cooling.
50-331/2003-003-07	NCV	System 12 Sprinkler System did not meet required NFPA-13-1983 code

Closed

50-331/2003-003-01	NCV	Inadequate Corrective Actions for South East Corner Rooms Drains
50-331/2003-003-02	NCV	Inadequate Fuel Pool Cooling System Procedure
50-331/2003-003-03	NCV	Inadequate Design Control on Drywell Floor Drain Sump System
50-331/2003-003-04	NCV	Inadequately performed calibration on the RCIC Turbine Governor
50-331/2003-003-05	FIN	Inadequate evaluation on continued operation of the 5B dump valve resulted in a Manual Reactor SCRAM
50-331/2002-011-01	NCV	The licensee failed to establish adequate measures to assure that the design requirements in calculations E92-007 and E92-008 were correctly translated into work instructions.
50-331/2003-003-06	NCV	The licensee failed to verify design pump seal failure mechanisms in a 10 CFR 50.59 evaluation to eliminate the need for RHR pump cooling

50-331/2002-011-02	URI	The licensee needs to perform a more exhaustive evaluation and analysis for continued use of the Core Spray Pump Seals without cooling and to ensure that RHR Pump Seals would perform their design function. In addition, more detailed analysis, inspection and calculations are required to support Operability Evaluation (AR#30414)
50-331/2003-003-07	NCV	System 12 Sprinkler System did not meet required NFPA-13-1983 code
50-331/2003-001	LER	Manual Reactor Scram and Reactor Coolant Chemistry Excursion Due to Punctured Main Condenser Tube Caused by Failed Condenser Deflector Plate

LIST OF ACRONYMS USED

ACE	Apparent Cause Evaluation
ACP	Administrative Control Procedures
ADAMS	NRC's Document System
ADS	Automatic Depressurization System
AFP	Area Fire Plan
ALARA	As Low As Reasonably Achievable
AOP	Abnormal Operating Procedures
AOT	Allowable Outage Time
ATWS	Anticipated Transient Without a SCRAM
AR	Action Request
ARM	Area Radiation Monitor
BI	Baseline Inspection
CA	Corrective Action
CAMS	Continuous Air Monitor
CAP	Corrective Action Plan
CE	Condition Evaluation
CRD	Control Rod Drive
CFR	Code of Federal Regulations
CS	Core Spray
CST	Condensate Storage Tank
CWO	Corrective Work Order
CY	Calender Year
DAEC	Duane Arnold Energy Center
DOT	Department of Transportation
DP	Differential Pressure
DRP	Division of Reactor Projects
DRS	Division of Reactor Safety
ECP	Engineering Change Package
ED	Emergency Deressurization
EDG	Emergency Diesel Generator
EGM	Electronic Governor Module
EGR	Electronic Governor Regulator
EMA	Engineered Maintenance Action
ESG	Evaluated Scenario Guide
ESW	Emergency Service Water
F	Fahrenheit
FSAR	Final Safety Analysis Report
GL	Generic Letter
GPM	Gallons Per Minute
HIC	High Integrity Container
HPCI	High Pressure Coolant Injection
HRA	High Radiation Area
HRCQ	Highway Route Controlled Quantity
HSAS	Homeland Security Advisory System
HVAC	Heating Ventilation and Air Conditioning
ICDP	Incremental Core Damage Probability

IMC	Inspection Manual Chapter
IPOI	Integrated Plant Operating Instruction
IPTE	Infrequently Performed Test and Evolution
LER	Licensee Event Report
LCO	Limited Condition Of Operation
LOCA	Loss Of Coolant Accident
LPCI	Low Pressure Coolant Injection
MOV	Motor Operated Valve
MSIV	Main Steam Isolation Valve
Mwth	Megawatts Thermal
NCV	Non-Cited Violation
NFPA	National Fire Protection Association
NRC	Nuclear Regulatory Commission
OI	Operating Instruction
OS	Occupational Radiation Safety
OWA	Operator Work Arounds
P&IDs	Piping and Instrumentation Drawings
PARS	Public Availability Records
PDIC	Pressure Differential Input Controller
PI	Performance Indicator
PM	Preventive Maintenance
PWO	Preventive Work Order
PS	Public Radiation Safety
PSID	Pounds Per Square Inch Differential
PSV	Pressure Setpoint Valve
PTAT	Plant Transient Assessment Tree
Radwaste	Radioactive Waste
RB	Reactor Building
RCA	Radiologically Controlled Area
RCIC	Reactor Core Isolation Cooling
RHR	Residual Heat Removal
RHRSW	Residual Heat Removal Service Water
RIS	Regulatory Information Summary
ROP	Reactor Oversight Process
RP	Radiation Protection
RPS	Reactor Protection System
RPT	Radiation Protection Technician
RWCU	Reactor Water Clean Up
RWP	Radiation Work Permit
SBO	Station Blackout
SCBA	Self Contained Breathing Apparatus
SDC	Shutdown Cooling
SDP	Significance Determination Process
SE	Safety Evaluation
SER	Safeguard Event Report
SGI	Safeguards Information
SRA	Senior Reactor Analyst
SSCs	Structure, System, or Components

STP	Surveillance Test Procedure
TEDE	Total Effective Dose Equivalent
TMOD	Temporary Modification
TMP	Temporary Modification Permit
TS	Technical Specification
TSV	Turbine Stop Valve
UFSAR	Updated Final Safety Analysis Report
USNRC	U.S. Nuclear Regulatory Commission
VDC	Volts Direct Current
VOTES	Valve Operation Test and Evaluation System

LIST OF DOCUMENTS REVIEWED

1R04 Equipment Alignment

OI 324A10; SBDG Standby/Readiness Condition Checklist; Revision 2
OI 324A9; SBDG Operating Checklist; Revision 4
OI 302; 125 VDC Power Distribution System; Revision 30
OI 302A2; Division 2, 125 VDC Power Distribution System (Initial Startup); Revision 0
OI 302A4; Division 2, 125 VDC Power Distribution System (In Service); Revision 0
OI 302A1; Division 1, 125 VDC Power Distribution System (Initial Startup); Revision 1
OI 302A3; Division 1, 125 VDC Power Distribution System (In Service); Revision 0
OI 150A1; RCIC System Electrical Lineup; Revision 0
OI 150A2; RCIC System Valve Lineup and Valve Checklist; Revision 2
UFSAR 9.2.3.2.2; Emergency Service Water; Revision 14
STP NS540002; Emergency Service Water Operability Test; Revision 9; Performance data from September 19, 2002, December 20, 2002, and March 17, 2003 Tests
CAP 019350; 1P099B (ESW Pump) Tripped on Thermal Overload during Surveillance; June 23, 2002
CAP 019545; Re-evaluate Acceptability of Operating ESW With Its Straining Function Bypassed; September 21, 2002
OI 454; Emergency Service Water System; Revision 37
OI 454A1; ESW System Electrical Lineup; Revision 0
OI 454A2; “A” ESW System Valve Lineup and Checklist; Revision 2
OI 454A4; “B” ESW System Valve Lineup and Checklist; Revision 3
OI 454A6; ESW System Control Panel Lineup; Revision 0

1R05 Fire Protection

Fire Plan; Volume II - Fire Brigade Organization; Revision 32
Fire Scenario; Radwaste Building, Shipping Lanes; January 16, 2003
AFP-1; Torus and North Corner Rooms; Revision
AFP-2; South Corner Rooms; Revision
AFP-3; HPCI/RCIC/RAD; Revision
AFP-13; Refuel Floor; Revision
AFP-22; Turbine Building South Floor; Revision
AFP-26; Control Room; Revision 23
AFP-27; Control Room HVAC; Revision 22
AFP-34; Radwaste Building , Drum Filling; Revision 23
AFP-35; Radwaste Treatment; Revision 22

1R06 Flood Protection Measures

Individual Plant Examination Section 3.3.6;Internal Flooding Analysis; November 1992
AOP 902; Flood; Revision 19
EOP 3; Secondary Containment Control; Revision 10
CAP 025748; Floor drain in South East Corner Room is plugged; February 21, 2003
CE 000423; Floor drain is plugged; February 27, 2003
CAP 025949; Review adequacy of Page 11 of AOP 913, “Flood”; March 6, 2003
CAP 026056; South East Corner Room Housekeeping; March 11, 2003

1R11 Licensed Operator Requalification Program
ESG 42 Scenario Guide; Revision 1
EOP 1; Reactor Pressure Vessel Control; Revision 9
EOP 2; Primary Containment Control; Revision 9
ATWS; RPV Control; Revision 10
ED; Emergency Depressurization; Revision 2
EAL; Emergency Action List Table 1; Revision 2
ACP 110.1; Conduct of Operations; Revision 0
ACP 101.01; Procedure Use and Adherence; Revision 19
ACP 101.2; Verification Process and SELF/PEER Checking Practices; Revision 5

1R12 Maintenance Effectiveness
NEI 93-01; "Nuclear Energy Institute Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants; Revision 2
AR32589; Target Rock SRV Main Stage Failure; September 17, 2002
CWO 1113092; Overhaul of PSV4400; April 21, 2001
CWO 1113093; Overhaul of PSV4401; April 21, 2001
CWO 1113094; Overhaul of PSV4402; April 21, 2001
CWO 1113097; Overhaul of PSV4405; April 21, 2001
CWO 1113098; Overhaul of PSV4406; April 21, 2001
CWO 1113099; Overhaul of PSV4407; April 21, 2001
Maintenance Rule Data; Automatic Depressurization System; January 13, 2003
Maintenance Rule Data; Reactor Protection System; March 11, 2003
CAP 025665; Local Power Range Monitor Failed High; February 17, 2003

1R13 Maintenance Risk Assessments and Emergent Work Control
Work Planning Guide - 2; On-Line Risk Management Guideline; Revision 12
Online Look-Ahead Agenda; Week of January 11, 2003
Online Look-Ahead Agenda; Week of January 25, 2003
Online Look-Ahead Agenda; Week of February 01, 2003
Online Look-Ahead Agenda; Week of February 22, 2003
Online Look-Ahead Agenda; Week of March 1, 2003

1R14 Personnel Performance During Nonroutine Plant Evolutions and Events
TMOD 03-07; 1D2 125 Vdc Division 2 Battery; January 26, 2003
CAP 025171; Commitment Change Evaluation; January 12, 2003
CWO A53415; 1D2 125 Vdc Battery Change Out; January 24, 2003
STP 3.8.4-01; Battery Pilot Cell Checks; Revision 9
STP 3.8.4-02; Battery Connected Cell Checks; Revision 7
Engineering Test Procedure (ETP); Performance Discharge Test Of Batteries 1D2; Revision 0
Equipment Specific Maintenance Procedure; Battery-C173-01 Batteries; Revision 33
General Maintenance Procedure (GMP) CNST-10; Fire Barrier Penetration Seal Installation and Repair
GMP-ELEC-08; Scotch Brand Tapes And Gray Boot Connections Installation Instructions; Revision 22
EOP 1; Reactor Pressure Vessel Control; Revision 9
EOP 3; Secondary Containment Control; Revision 15

Operating Instruction (OI) 304.1; 4160/480V Nonessential Electrical Distribution System, Revision 39
Abnormal Operating Procedure (AOP) 639; Reactor Water/Condensate High Conductivity; Revision 15
Integrated Plant Operating Instruction (IPOI) 3; Power Operations; Revision 57
IPOI 4, Shutdown; Revision 58
IPOI 5, Reactor SCRAM; Revision 34
ACP 110.1; Conduct of Operations; Revision 0
ACP 101.01; Procedure Use and Adherence; Revision 19
ACP 101.2; Verification Process and SELF/PEER Checking Practices; Revision 5
Annunciator Response Procedure (ARP) 1C80; Condensate Demineralizer Control Panel; Revision 18
IPTE; Plant Startup 03-01, Condenser Tube Leak Outage; February 14, 2003
IPOI 1; Startup Checklist; Revision 87
IPOI 2; Startup; Revision 73

1R15 Operability Evaluations
CAP 025038; "A" Standby Diesel Generator; December 19, 2002
CWO A50583; Troubleshooting Instruction Form "A" Standby Diesel Generator; January 2, 2003
CAP 025018; Standby Filter; December 17, 2002
CAP 025389; Fuel Pool Temperature Low Out of Spec; February 1; 2003
OI 435; Fuel Pool Cooling System; Revision 34
CAP 025246; MSIV and TSV Position Switch Analytic Limits for RPS trip input; January 20, 2003
AR 32610; Non-Conservative Temperature Input for Motor Degraded Voltage Calculation; September 18, 2002

1R16 Operator Workarounds
Operations Department Instructions 004; Identification, Tracking and Resolution of Equipment issues; Revision 8
AOP 913; Fire; Revision 33
AOP 915; Shutdown Outside Control Room; Revision 25
CAP 025365; Fire could cause loss of Control Room habitability; January 31, 2003

1R17 Permanent Modifications
CWO A58888; Replace ASCO SCRAM Solenoid Valves; January 21, 2003
EMA A58888; Replace ASCO SCRAM Solenoid Valves; Revision 1 (January 25, 2003)

1R19 Post-Maintenance Testing
CWO A60577; Speed Switch Replacement; January 2, 2003
CWO A60103; Power Supply Negative Side Failed Downscale; January 14, 2003
CWO A50996; 1D1 125 Vdc Battery Change Out; January 10, 2003
CWO A53415; 1D2 125 Vdc Battery Change Out; January 24, 2003
CWO A60263; Replacement of 45 degree elbow on the Residual Heat Removal Service Water System; February 1,2003
CWO 1124677; Condensate Storage Tank 1T-5B Level (Low Level Suction transfer for HPCI and RCIC); February 13, 2003

1R20 Refueling and Outage

Planned Outage Look Ahead Report; February 2, 2002
Planned Outage Risk Analysis; February 2, 2003
IPOI 8; Outage and Refueling Operations; Revision 30
IPOI 4, Shutdown; Revision 58
Operating Instruction (OI) 149; RHR System; Revision 81
CAP 025394; Several of the Floor drains in Drywell are not draining; February 2, 2003
OMG 7; Outage Risk Management Guidelines; Revision 11
Outage Memorandum; March 12, 2003
Refuel Outage 18 Shutdown Risk; Revision 0
Refuel Outage 18 Schedule; March 16, 2003

1R22 Surveillance Testing

STP 3.3.1.1-24; Local Power Range Monitor Calibration; Revision 10
STP 3.1.3.-01; Control Rod Exercise; Revision 5
STP 3.3.6.1-47; HPCI Exhaust Diaphragm Channel Functional Test; Revision 1
STP 3.3.3.2-09; Reactor Water Level and Pressure Instruments Calibration; Revision 13
STP 3.3.5.1-03; Functional Test of LPCI LOOP Select - Reactor Vessel Water Level Low-Low Instrumentation; Revision 5
STP 3.5.3-02; RCIC System Operability Test; Revision 12
Turbine-T147-02; Calibration of RCIC Turbine Governor; Revision 11
CWO 1122891; Initial Calibration of TG2406; February 20, 2003.
ACE 001075; RCIC Failure to meet required flow; February 21, 2003
CAP 025715; RCIC failure to meet required flow during STP 3.5.3-02; February 19, 2003
CAP 025899; HPCI and RCIC EGR Voltage Checks; March 3, 2003
STP 3.6.1.1-06; Containment Isolation Valve Leak Tightness Test - Type C Penetrations - Feedwater System; Revision 6

1R23 Temporary Modifications

TMOD 02-061; Lift Leads and Remove Temperature Element TE2149A from 1P211A-M Plant Effect Evaluation; October 3, 2002
CWO A52251; Drain Oil from Upper Reservoir, obtain an oil sample, remove damaged TE, install new TE, Refill Oil in Reservoir; October 2, 2002
CWO A60206; Drain Oil from Upper Reservoir, Remove Temp Mod (pipe plug), Refill Oil in Reservoir; October 25, 2002
Condition Based Analysis - Motor, AC Induction (Combined Oil/Vibration analysis and report to support Systems Engineering evaluation of operability due to damaged thermocouple device for 1P211A-M upper bearing); October 3, 2002
TMOD 03-03; 1D1 125 Vdc Division 1 Battery; January 15, 2003
CAP 025171; Commitment Change Evaluation; January 12, 2003
CAP 025165; Evaluate the Requirement for Temporary Batteries; January 10, 2003
CWO A57180; Setup and Perform checks on Temporary Battery; January 13, 2003

1EP6 Drill Evaluation

2003 Red Team Training Drill Scenario; February 26, 2003
Emergency Plan Implementing Procedure (EPIP) 1.1; Emergency Plan Implementing Procedure; Revision 19

EPIP 2.5; Control Room Emergency Response Operation; Revision 14
EAL; Determination of Emergency Action Levels; Revision 2
EOP 1; RPV Control; Revision 9
EOP 2; Primary Containment Control; Revision 9
EOP 3; Secondary Containment Control; Revision 10
AOP 901; Earthquake; Revision 12

4OA1 Performance Indicator Verification
NEI 99-02; Regulatory Assessment Performance Indicator Guideline; Revision 2
Memo; DAEC 4th Quarter 2002 PI Summary; January 21, 2003
Memo; DAEC 3rd Quarter 2002 PI Summary; October 21, 2002
Memo; DAEC 2nd Quarter 2002 PI Summary; July 19, 2002
Memo; DAEC 1st Quarter 2002 PI Summary; April 20, 2002
ACP 1402.4; NRC Performance Indicators Collection and Reporting; Revision 3
ACP 1402.4; NRC Performance Indicators Collection and Reporting, Attachment #1, PI Data Calculation, Review, and Approval; dated CY 2001, 4th Quarter through

4OA2 Identification and Resolution of Problems
ACP 102.1; External Operating Experience; Revision 19
ACP 114.4; Corrective Action Program; Revision 12
ACP 114.3; Root Cause and Apparent Cause Analysis; Revision 12
ACP 102.18; DAEC Self Assessment; Revision 4
ACP 1408.15; Control of Replacement Fuses; Revision 6
CAP 005845; Found incorrect type of fuse installed in "B" Diesel Generator starter box; February 2, 2000
CAP 007247; Potential for Installation of Discrepant Fuses; June 9, 2000
CAP 010535; Installed fuse different than rating on drawing; May 12, 2001
CAP 019395; Control Ros Position Information Cabinet fuse wrong; July 22, 2002
CAP 014513; Found wrong size control power fuse installed; October 2, 2002
CAP 025009; 125 Volt fuse found in a 480 Volt application; December 16, 2002
CAP 025192; Improper fuse found in terminal; January 14, 2003
CAP 025977; Fuse Issues; March 7, 2003
CAP 026297; Failure to Trend; March 23, 2003

4OA3 Event Follow-up
Root Cause Report 001001; Reactor SCRAM; February 8, 2003
Metallurgical Analysis of Failed Baffle Plate; February 27, 2003
CAP 025382; Manual Reactor SCRAM; February 1, 2003
Licensee Event Report (LER) 50-331/03-01; Manual Reactor Scram and Reactor Coolant Chemistry Excursion Due to Punctured Main Condenser Tube Caused by Failed Condenser Deflector Plate; March 27, 2003

4OA5 Other Activities
NRC SER; Safety Evaluation - Station Blackout Rule Conformance Evaluation; November 22, 1991
NRC SER; Supplemental Safety Evaluation - Station Blackout Rule Conformance Evaluation; June 15, 1992

Ltr NG-90-0757; 10 CFR 50.63, “Loss of All Alternating Current Power” Information Submittal, Revision; 1 March 30, 1990
Ltr NG-89-0923; 10 CFR 50.63, “Loss of All Alternating Current Power” Information Submittal; April 17, 1989
Ltr NG-92-0283; Response to Safety Evaluation by NRC-NRR “Station Blackout Evaluation” Iowa Electric Light and Power Company Duane Arnold Energy Center; February 10, 1992
SE 99-041; Change Emergency Service Water Flow Requirements to the RHR Pump Seal Coolers; Revision 1
Memorandum; Alliant Energy Containment Spray Pumps - Upset Conditions; March 28, 2002
Vendor Manual; Installation, Operation, and Maintenance Instructions for Type “U” Mechanical Seals; 1969
Vendor Manual; Borg-Warner High Pressure Heat Exchangers; June 1970
DBD-E13-001; Duane Arnold Energy Center Design Bases Document for Emergency Service Water System; Revision 6
EQR: Qual-L200-03C, 02C, 01C; Environmental Qualification System Component Evaluation Worksheet; Revision 1
QUAL-SC101; Environmental and Seismic Service Conditions; Revision 11
7884-APED-E11-2776-32-2; Technical Manual Residual Heat Removal Pump - Byron Jackson Pump Division; March 21, 1972
EMP-1P099-FV; Emergency Service Water Flow Verification Test; October 22, 1999 and July 6, 1999
B 3.7.3; Emergency Service Water (ESW) System; Amendment 223
UFSAR Change No. 99-030; Change the Values of Emergency Service Water Flow Requirements in UFSAR Table 9.2-1 From 6 GPM to 0 GPM; August 2, 1999
Table 9.2-1; Emergency Service Water Flow Requirements; Revision 15
Memorandum; Duane Arnold Energy Center - Issuance of Amendment Regarding Alternative Source Term (TAC. NO. MB0347); July 31, 2001
Memorandum; Atomic Energy of Canada Limited (AECL) Evaluation of Seal Designs Under Conditions Consistent With the Transient Presented by DBA for Core Spray and RHR Pumps; April 30, 2002
Publication; On the Lubrication of Mechanical Face Seals; H. Lubbinge; January 1999
Draft Response to AR 30234; RHR Mechanical Seal Evaluation, 2001
AR# 30414; Detailed Operability Evaluation - Can Residual Heat Removal Pump Seals Operate Under Accident Conditions Without Seal Cooling?; March 28, 2002
AR# 30785; Operability Evaluation - Can Residual Heat Removal Pump Seals Operate Under Accident Conditions Without Seal Cooling?; May 1, 2002
AR# 31223; DAEC Response to NRC RHR/CS Pump Seal Questions; June 19, 2002
Memorandum; Duane Arnold Energy Center - Issuance of Amendment Regarding Alternate Source Term (TAC No. MB0347); July 31, 2001
AR#27339; Review NE and Sw Corner Rooms, RHR, Core Spray, Heat Load Calculations; February 28, 2002