October 20, 2005

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

SUBJECT: DUANE ARNOLD ENERGY CENTER
NRC INTEGRATED INSPECTION REPORT 5000331/2005004

Dear Mr. Van Middlesworth:

On September 30, 2005, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at your Duane Arnold Energy Center. The enclosed integrated inspection report documents the inspection findings which were discussed on October 4, 2005, with you and other members of your staff.

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

Based on the results of this inspection, there was one NRC-identified and two self-revealed findings of very low safety significance, of which two involved a violation of NRC requirements. However, because these violations were of very low safety significance and because the issues were entered into the licensee's corrective action program, the NRC is treating these findings as Non-Cited Violations (NCVs) in accordance with Section VI.A.1 of the NRC's Enforcement Policy. Additionally, a licensee identified violation 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 the basis for your denial, to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555-0001; with a copy to the Regional Administrator, U.S. Nuclear Regulatory Commission - Region III, 2443 Warrenville Road, Suite 210, Lisle, IL 60532-4352; the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001; and the Resident Inspector Office at the Duane Arnold Energy Center.

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Sincerely,

/RA/

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

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

Enclosure: Inspection Report 5000331/2005004
w/Attachment: Supplemental Information

cc w/encl: E. Protsch, Executive Vice President -
Energy Delivery, Alliant;
President, IES Utilities, Inc.
C. Anderson, Senior Vice President, Group Operations
J. Cowan, Executive Vice President and Chief Nuclear Officer
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B. Lacy, Nuclear Asset Manager
Chairman, Linn County Board of Supervisors
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The Honorable Charles W. Larson, Jr.
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U.S. NUCLEAR REGULATORY COMMISSION

REGION III

Docket No:	50-331
License No:	DPR-49
Report No:	05000331/2005004
Licensee:	Nuclear Management Company, LLC
Facility:	Duane Arnold Energy Center
Location:	3277 DAEC Road Palo, Iowa 52324-9785
Dates:	July 1 through September 30, 2005
Inspectors:	G. Wilson, Senior Resident Inspector R. Baker, Resident Inspector R. Langstaff, Acting Senior Resident Inspector G. Gibbs, Reactor Inspector M. Mitchell, Radiation Specialist
Observers:	None
Approved by:	Bruce L. Burgess, Chief Branch 2 Division of Reactor Projects

Enclosure

SUMMARY OF FINDINGS

IR 05000331/2005004; 07/01/2005 - 09/30/2005; Duane Arnold Energy Center, Fire Protection, Post Maintenance Testing, and As Low As Is Reasonably Achievable Planning And Controls.

This report covers a 3-month period of baseline resident inspection and announced baseline inspections of radiation protection and inservice inspection. The inspections were conducted by a Regional Radiation Specialist Inspector and the Resident Inspectors. Three Green findings associated with two non-cited violations were identified. The significance of most findings is indicated by their color (Green, White, Yellow, Red) using Inspection Manual Chapter (IMC) 0609, "Significance Determination Process" (SDP). Findings for which the SDP does not apply may be "Green" or be assigned a severity level after NRC management review. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, "Reactor Oversight Process," Revision 3, dated July 2000.

A. Inspector-Identified and Self-Revealed Findings

Cornerstone: Initiating Events

- Green. A finding of very low safety significance was identified through a self revealing event when an operator failed to adequately verify a valve lineup in the fire protection system. The valve that was inadvertently left open caused partial flooding and contamination of the first floor of the reactor building. The primary cause of this finding was related to the cross-cutting area of Human Performance (Personnel). The licensee entered this issue into their corrective action program and decontaminated the associated floor areas.

  The finding was more than minor because the failure to verify proper a valve lineup prior to restoring the system has the potential to adversely impact plant equipment, thereby affecting plant safety. This finding was determined to be of very low safety significance since it did not impact any mitigating systems capability. Since no 10 Code of Federal Regulations (CFR) 50, Appendix B components were impacted by this finding, no violation of NRC requirements occurred. (Section 1R19)

Cornerstone: Mitigating Systems

- Green. A finding of very low safety significance was identified by the inspectors for a violation of the fire protection license condition. The licensee failed to ensure that travel distance requirements were met for fire extinguishers in the reactor building. Once this issue was identified, the licensee entered the issue into their corrective action program and initiated work requests to provide additional fire extinguishers. The primary cause of this violation was related to the Identification subcategory of the Problem Identification and Resolution cross-cutting area. Licensee fire protection personnel failed to identify that the placement of fire extinguishers did not satisfy fire protection code requirements during a self-assessment of code compliance for fire extinguishers performed in April 2004.

This finding was more than minor because the ability to manually fight a small fire in the area of the spent fuel pool cooling and cleanup pumps was adversely affected. The issue was of very low safety significance due to the limited impact a fire would have in the affected fire zones and the relatively low ignition frequency for the affected fire zones. The finding was a Non-Cited Violation (NCV) of License Condition 2.3.(C) which required the licensee to implement and maintain in effect all provisions of the approved fire protection program as described in Safety Evaluation Report dated June 1, 1978, which specified compliance to the applicable fire protection code for fire extinguishers. (Section 1R05)

Cornerstone: Occupational Radiation Safety

Green. The inspectors reviewed a self-revealing NCV of Technical Specification (TS) 5.4.1 for the failure to follow station as-low-as-reasonably-achievable (ALARA) procedure. During Refueling Outage (RFO) 19, the radiation dose estimate was exceeded by 61 percent and the total was greater than 5 rem on two separate work activities. The control rod drive push/pull and rebuild project was planned with a total dose of 3100 millirem, and the actual dose was 5253 millirem with no revisions to the estimate during the work implementation. The refueling project was estimated at 8500 millirem, and the actual exposure was 13648 millirem. The licensee determined that the work area dose rates were consistent with the plan, but time estimates or person-hours were not consistent with actual work implementation. The finding was entered into the licensee's corrective action program.

The finding was more than minor because it is associated with the Occupational Radiation Safety attribute of exposure control and affected the cornerstone objective of programs and procedures. The occurrence involved a failure to implement procedures needed to achieve occupational doses ALARA and that resulted in an unplanned, unintended occupational collective dose for two work activities. Using the Occupational Radiation Safety Significance Determination Process, the inspectors determined that the finding was of very low safety significance (Green) because while it did involve ALARA planning and controls, (1) the licensee three-year rolling average collective dose was less than 240 person-rem/unit, and it did not involve; (2) an overexposure; (3) a substantial potential for an overexposure; or (4) an impaired ability to assess dose. (Section 2OS1)

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

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

REPORT DETAILS

Summary of Plant Status

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

- On July 30, 2005, the reactor was reduced in power to approximately 60 percent to accomplish rod-pattern adjustments, planned surveillance activities, and various maintenance activities. The return to full power was delayed until August 1, 2005, to affect repairs to the 5B feedwater heater drain valve positioner.

1. REACTOR SAFETY

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

1R04 Equipment Alignment (71111.04)

.1 Partial Walkdown

a. Inspection Scope

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

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

- High Pressure Core Injection (HPCI) with Reactor Core Isolation Cooling (RCIC) out-of-service (OOS) for maintenance during the week ending July 2, 2005;
- 'A' train of the Residual Heat Removal Service Water (RHRSW) system with the 'B' train of RHRSW OOS for maintenance during the week ending August 27, 2005; and
- 'A' train of the Emergency Service Water (ESW) system with the 'B' train of ESW OOS for maintenance during the week ending August 27, 2005.

b. Findings

No findings of significance were identified.

.2 Complete Walkdown

a. Inspection Scope

During the week ending August 5, 2005, the inspectors performed a complete system alignment inspection of the Fire Protection system for a total of one sample. This system was selected because it was significant for mitigating the effects of a fire, a high contributor towards plant risk. The inspection consisted of the following activities:

- 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, with a focus on the fire protection equipment in the pump house, the RCIC pump room, and the reactor building hatch way deluge system.

The inspectors also reviewed selected issues documented as Corrective Action Process (CAP) records, initiated within the previous year, to determine if they had been properly addressed in the licensee's corrective action program. The documents listed in the Attachment were used by the inspectors to accomplish the objectives of the inspection procedure.

b. Findings

No findings of significance were identified.

1R05 Fire Protection (71111.05AQ)

.1 Quarterly Fire Zone Walkdowns

a. Inspection Scope

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

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

During the week ending July 9, 2005:

- Area Fire Plan (AFP) 69, Main Transformer 1X1;
- AFP 70, Standby Transformer 1X4;
- AFP 71, Startup Transformer 1X3; and
- AFP 72, Auxiliary Transformer 1X2.

During the week ending July 23, 2005:

- AFP-11, Reactor Building Laydown Area - Elevation 833'-6";
- AFP-12, Reactor Building Decay Tank and Condensate Phase Separator Rooms;
- AFP-13, Reactor Building Refueling Floor; and
- AFP-14, North Turbine Building Basement, Reactor Feed Pump Area and Turbine Lube Oil Tank Area.

During the week ending August 6, 2005:

- AFP-03, Reactor Building, High Pressure Coolant Injection, Reactor Core Isolation Cooling, and RadWaste Tank Rooms.

During the week ending August 13, 2005:

- AFP-25, Control Building Cable Spreading Room.

During the week ending August 20, 2005:

- AFP-10, Reactor Building, Main Exhaust Fan Room, Heating Hot Water Pump Room and the Plant Air Supply Fan Room.

b. Findings

Introduction: The inspectors identified a finding of very low safety significance (Green) and a Non-Cited Violation (NCV) of License Condition 2.3.(C) for failing to meet travel distance requirements for fire extinguishers in the reactor building.

Description: The inspectors identified that there were no fire extinguishers located in Fire Zone 04G, "Reactor Building - Fuel Pool Pump Area," and Fire Zone 05B, "Reactor Building - Phase Separator/Skimmer Surge Tank Rooms," which is described in AFP 12. In addition, there were no fire hose stations in Fire Zone 04G or Fire Zone 05B. The closest fire extinguishers were located in Fire Zone 05A, "Reactor Building - Laydown and Hatch Area 833'," and, one floor below, Fire Zone 04A, "Reactor Building - Reactor Building Closed Cooling Water (RBCCW) Heat Exchanger/Chillers." The inspectors noted that the travel distances from much of Fire Zones 04G and 05B to the closest fire extinguisher exceeded the 75 foot maximum travel distance specified by the applicable National Fire Protection Association (NFPA) code of record, NFPA 10-1975, "Portable Fire Extinguishers."

By letter dated January 18, 1977, the licensee committed to the Nuclear Regulatory Commission (NRC) to comply with position E.6, "Portable Extinguishers," of Appendix A to Branch Technical Position APCSB 9.5-1, "Guidelines for Fire Protection for Nuclear Power Plants Docketed Prior to July 1, 1976." Specifically, by item 144 of the licensee correspondence, the licensee committed to provide portable extinguishers compatible with the combustible materials in accordance with NFPA 10 and the requirements of the Occupational Safety and Health Administration. Section 4.3.3 of NRC Fire Protection Safety Evaluation Report, dated June 1, 1978, stated: "Portable dry chemical and carbon dioxide fire extinguishers have been distributed throughout the plant. The fire extinguishers meet the requirements of the NFPA." Section 3-2.1 of NFPA 10-1975 specified that extinguishers shall be located so that the maximum travel distances shall not exceed those specified in Table 3-2.1, except as modified by Section 3-2.3 of NFPA 10-1975. Table 3-2.1 of NFPA 10-1975 specified a maximum travel distance to extinguishers of 75 feet. Section 3-2.3 of NFPA 10-1975 permitted up to one-half of the complement of extinguishers as specified in Table 3-2.1 to be replaced by uniformly spaced small hose stations for use by the occupants of the building. Section 3-2.3 of NFPA 10-1975 also specified that the location of hose stations and the placement of fire extinguishers shall be in such a manner that the hose stations do not replace more than every other extinguisher.

The inspectors reviewed FPE-S02-005, "Fire Extinguisher Code Compliance," which was a self-assessment performed by licensee fire protection personnel in April 2004. The assessment did not identify any discrepant conditions for Fire Zone 04G. The assessment did identify that the travel distances for Fire Zone 05B were not met. However, the assessment concluded that not having met the travel distances was acceptable because uniformly placed hose stations were provided. The inspectors noted that the nearest hose stations were either located near the fire extinguishers or further away than the fire extinguishers and did not contribute towards reducing the travel distances to a fire extinguishing device. In addition, the inspectors noted that the fire hoses installed at the hose stations only had a length of 50 feet. The assessment failed to identify that the location of the hose stations did not satisfy the NFPA-10 requirement that the location of hose stations and the placement of fire extinguishers be in such a manner that the hose stations did not replace more than every other extinguisher. Discussions with the site fire protection engineer indicated that there may likely be other areas, such as in the radwaste building, where the travel distance requirements had not been met.

<u>Analysis</u>: The inspectors determined that the failure to ensure that the placement of fire extinguishers met NFPA requirements was a performance deficiency, therefore a significance evaluation was warranted. The inspectors concluded that the finding was more than minor, in accordance with Inspection Manual Chapter (IMC) 0612, "Power Reactor Inspection Reports," Appendix B, "Issue Screening," issued May 19, 2005,

because the finding affected a cornerstone objective. The finding was associated with the Mitigating System cornerstone attribute of protection against external factors (i.e., fire), and affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events (i.e., fire) to prevent undesirable consequences. Specifically, the spent fuel pool cooling and cleanup pumps were located in one of the affected fire zones, Fire Zone 4G, and could have been adversely affected by a fire. The failure to have fire extinguishers readily available could adversely impact timely manual fire suppression capability of a small fire. The primary cause of this finding was related to the Identification subcategory of the Problem Identification and Resolution cross-cutting area. Specifically, licensee fire protection personnel failed to identify that the placement of fire extinguishers did not satisfy fire protection code requirements during a self-assessment of code compliance for fire extinguishers performed in April 2004.

The inspectors performed a Significance Determination Process (SDP) Phase 1 screening in accordance with IMC 0609, "Significance Determination Process," Appendix A, "Determining the Significance of Reactor Inspection Findings for At-Power Situations," and determined that the finding affected a fire protection defense-in-depth feature. As such, the inspectors determined that a Phase 2 analysis in accordance with IMC 0609, Appendix F, "Fire Protection Significance Determination Process," dated February 28, 2005, was required. As discussed by IMC 0308, Attachment 3, Appendix F, "Technical Basis, Fire Protection Significance Determination Process (IMC 609 App. F) At Power Operations," dated February 28, 2005, the current significance determination process does not address findings which affect the performance of the onsite fire brigade or manual fire suppression capability. As such, the inspectors determined that IMC 609, Appendix F, did not provide guidance with regard to significance evaluation. The inspectors used judgement based on experience, along with NRC management review, to determine the safety significance of the issue. The inspectors determined that the issue was of very low safety significance (Green) due to the limited impact a fire would have in the affected fire zones and the relatively low ignition frequency for the affected fire zones.

Enforcement: License Condition 2.C.(3), "Fire Protection," required Nuclear Management Company to implement and maintain in effect all provisions of the approved fire protection program as described in the Final Safety Analysis Report (FSAR) for the Duane Arnold Energy Center and as approved in the Safety Evaluation Report dated June 1, 1978, and Supplement dated February 10, 1981. Section 4.3.3 of Safety Evaluation Report, dated June 1, 1978, stated that portable dry chemical and carbon dioxide fire extinguishers had been distributed throughout the plant and that the fire extinguishers met NFPA requirements. The NFPA requirements for fire extinguishers were outlined in NFPA 10-1975. NFPA 10-1975 specified that extinguishers shall be located so that the maximum travel distances shall not exceed 75 feet. Contrary to the above, as of August 5, 2005, the travel distances to fire extinguishers exceeded 75 feet for Fire Zones 04G and 05B in the reactor building. Once identified, the licensee entered the issue into their corrective action program as CAP 37407 and initiated work requests to provide additional fire extinguishers. Because this violation was of very low safety significance and it was entered into the licensee's corrective action program, this violation was treated as an NCV, consistent with Section VI.A of the NRC Enforcement Policy. (NCV 05000331/2005004-01)

1R06 Flood Protection Measures (71111.06)

a. Inspection Scope

During the week ending August 20, 2005, the inspectors performed a semi-annual review of flood protection barriers and procedures for coping with internal flooding in the RCIC pump room for a total of one sample. The documents listed in the Attachment were used by the inspectors to accomplish the objectives of the inspection procedure. Inspection activities focused on verifying that flood mitigation plans and equipment were consistent with design requirements and risk analysis assumptions. Inspection activities included, but were not limited to, a review and/or walkdown to assess design measures, seals, drain systems, contingency equipment condition and availability of temporary equipment and barriers, performance and surveillance tests, procedural adequacy, and compensatory measures.

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification Program (71111.11)

a. Inspection Scope

During the week ending August 27, 2005, the inspectors observed a training crew performance on Simulator Exercise Guide (SEG) 404 for a total of one sample. The scenario included a loss of Drywell Cooling and an Electrical Anticipated Transient Without Scram (ATWS). The documents listed in the Attachment were used by the inspectors to accomplish the objectives of the inspection procedure. The inspection activities assessed the licensee's effectiveness in evaluating the requalification program, ensuring that licensed individuals operated the facility safely and within the conditions of their license, and evaluated licensed operators' mastery of high-risk operator actions. Inspection activities included, but were not limited to, a review of high risk activities, emergency plan performance, incorporation of lessons learned, clarity and formality of communications, task prioritization, timeliness of actions, alarm response actions, control board operations, procedural adequacy and implementation, supervisory oversight, group dynamics, interpretations of technical specifications, simulator fidelity, and the licensee critique of performance.

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 3;
- ACP 101.01, "Procedure Use and Adherence," Revision 31; and
- ACP 101.2, "Verification Process and SELF/PEER Checking Practices," Revision 5.

b. Findings

No findings of significance were identified.

1R12 Maintenance Effectiveness (71111.12)

a. Inspection Scope

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

The inspectors performed the following maintenance effectiveness reviews for a total of one sample:

- A function-oriented review of the Standby Diesel Generator (SBDG) System was performed because it was designated as risk-significant under the Maintenance Rule, during the week ending August 13, 2005.

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 for a total of four samples. An evaluation of the performance of maintenance associated with planned and emergent work activities was completed by the inspectors to determine if they were adequately managed. In particular, the inspectors reviewed the program for conducting maintenance risk safety assessments and to ensure that the planning, assessment and management of on-line risk was adequate. The documents listed in the Attachment were used by the inspectors to accomplish the objectives of the inspection procedure. Licensee actions taken in response to increased on-line risk were reviewed including the establishment of compensatory actions, minimizing activity duration, obtaining appropriate management approval, and informing appropriate plant staff. These activities were accomplished when on-line risk was increased due to maintenance on risk-significant structures, systems, and components (SSCs).

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

- The inspectors reviewed the maintenance risk assessment for work planned during the weeks of July 9, 16, August 27, and September 17, 2005.

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations (71111.15)

a. Inspection Scope

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

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

- Operability (OPR) 000293, Seismic Adequacy of 1T070 and 1T073, during the week ending September 10, 2005; and
- OPR 000295, RHRSW System, during the week ending September 17, 2005.

b. Findings

No findings of significance were identified.

1R16 Operator Workarounds (OWA) (71111.16)

.1 Individual Workaround

a. Inspection Scope

The inspectors reviewed two OWAs. Inspectors used the documents listed in the Attachment to accomplish the objectives of the inspection procedure. Inspectors verified that the selected OWA did not impact the functionality of a mitigating system. Inspection activities included, but were not limited to, a review of the selected OWAs to determine if the functional capability of the system or human reliability in responding to an initiating event was affected, including a review of the impact of the OWAs on the operator's ability to execute Emergency Operating Procedures (EOPs).

The inspectors reviewed the following OWA for a total of two samples:

- CAP 36621, Feedwater Programming Problems, during the week ending September 3, 2005; and
- CAP 36660, Corrective Action Does Not Address All Failures, during the week ending September 3, 2005.

b. Findings

No findings of significance were identified.

1R19 Post-Maintenance Testing (71111.19)

a. Inspection Scope

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

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

- Preventative Work Order (PWO) 1131370, Diesel Fire Pump, during the week ending July 15, 2005;
- Corrective Work Order (CWO) A65579, Fire Station, during the week ending September 10, 2005;
- Modification Work Order (MWO) 113133, Relocate Relay 127X/42, during the week ending September 17, 2005; and
- CWO A70340, "B" Standby Gas Treatment System (SBGT), during the week ending September 24, 2005.

b. Findings

Introduction: A finding of very low safety significance (Green), was identified through a self-revealing event when an operator failed to adequately verify valve V 244-0461 was closed prior to re-pressuring the fire system. The open valve resulted in reactor building flooding.

Description: On August 25, 2005, the Operators were restoring the fire system following repairs that were performed on the isolation valve for fire hose station #23. During the restoration, water was realigned to the fire hose supply header. Valve 44-0461 was inadvertently left partially open, when the fire protection system was re-pressurized, thereby allowing water to flow from the valve. The excessive water flow from Valve 44-0461 resulted in radwaste receiving a reactor building floor drain sump high level alarm. It was estimated that 2700 gallons of water flowed through the vent valve prior to isolation. The primary cause of this finding was related to the Cross-Cutting area of Human Performance (Personnel) for the failure to adequately verify proper valve position.

The water from the fire system resulted in floor drains backing up on the first floor of the reactor building, which resulted in contamination levels of 150K smearable. In addition, access was restricted to the north reactor building stairwell due to heavy flooding and to the third floor of the reactor building due to residual water. Initial surveys on the second, fourth, and fifth floors of the reactor building found no contamination.

Analysis: The inspectors determined that the plant operator's failure to adequately verify that the Valve 44-0461 was fully closed was an example of not complying with a standard, that it could have reasonably been foreseen or corrected by the licensee, and was, therefore, a performance deficiency. Since a performance deficiency existed, 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 for the specific finding. As a result, the inspectors compared this performance deficiency to the minor questions contained in Appendix B of IMC 0612. The inspectors determined that the finding was more than minor, since the finding could be reasonably viewed as a precursor to a significant event and if it was left uncorrected, it would become a more significant safety concern. This was based on the fact that the failure to verify a proper valve lineup prior to restoring systems, has the potential to adversely impact plant equipment, thereby affecting plant safety.

As a result, the inspectors reviewed this issue in accordance with IMC Appendix A, Attachment 1, "Significance Determination of Reactor Inspection Findings for At-Power Situations." Using the Phase 1 SDP worksheet for the initiating events cornerstone, the inspectors determined that the finding did not contribute to the likelihood of a primary or secondary system loss of coolant accident initiators; and the finding did not contribute to both the likelihood of a reactor trip and the likelihood that mitigation equipment or functions will not be available. The finding was identified because of an internal flooding issue where 2700 gallons of water flowed into the reactor building. Because the flow path of the water down the stairwell would not affect any risk significant mitigating systems, and core damage scenarios of concerns and factors that increase the frequency of the event were not identified, the finding was of very low safety significance and screened as Green.

Enforcement: The inspectors determined that an operator failed to adequately verify that Valve 44-0461 was closed on August 25, 2005. The failure to fully close the Valve 44-0461 resulted in internal reactor building flooding. However, a violation of NRC requirements did not occur because no 10 CFR 50, Appendix B, components were impacted by the finding (FIN 05000331/2005004-02). This issue was entered into the licensee's corrective action program as CAP 037648.

Corrective actions taken included decontaminating the area. In addition, the operations shift manager performed training following the event to re-enforce standards and expectations.

1R22 Surveillance Testing (71111.22)

a. Inspection Scope

The inspectors reviewed four surveillance test activities. Inspection procedure objectives were accomplished as indicated by the documents listed in the Attachment to this inspection report. Surveillance testing activities were reviewed to assess operational readiness and ensure that risk-significant SSCs were capable of performing their intended safety function. Surveillance activities were selected based upon risk significance and the potential risk impact from an unidentified deficiency or performance degradation that a SSC could impose on the unit if the condition were left unresolved. Inspection activities included, but were not limited to, a review for preconditioning, integration of testing activities, applicability of acceptance criteria, test equipment calibration and control, procedural use, control of temporary modifications or jumpers required for test performance, documentation of test data, TS applicability, impact of testing relative to Performance Indicator (PI) reporting, and evaluation of test data.

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

- Surveillance Test Procedure (STP) 3.3.3.3-17, Main Steam Isolation Valve Functional Test, during the week ending July 9, 2005;
- STP 3.3.5.1-12, Channel Functional Test of Reactor Steam Dome Pressure-Low Low Pressure Coolant Injection ((LPCI) Loss of Offsite Power (LOOP) Select) Instrumentation, during the week ending July 9, 2005;
- STP 3.8.1.-04, 'A' SBDG Operability Test, during the week ending September 03, 2005; and
- STP NS540002, 'B' ESW Operability, during the week ending September 17, 2005.

b. Findings

No findings of significance were identified.

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

.1 Radiological Work Planning.

a. Inspection Scope

The inspectors evaluated the licensee's list of work activities, ranked by estimated exposure, that were in progress and reviewed the following two work activities of highest exposure significance:

- Drywell Control Rod Drive (CRD) Pull/Put and Rebuild Activities (M8)
- Refuel Project (R1)

This review represented one sample.

The inspectors compared the results achieved, including dose rate reductions and person-rem used, with the intended dose established in the licensee's ALARA planning for these two work activities. Reasons for inconsistencies between intended and actual work activity doses were reviewed. This review represented one sample.

b. Findings

Radiation Work Permit Dose Exceeded the Estimate by 61 Percent on Two Separate Work Activities

Introduction: A Green self-revealed finding and associated NCV was identified when the ALARA coordinator reviewed the cumulative doses for Control Rod Drive Push/Pull and Rebuild Radiation Work Permit (RWP) (M8) and Refuel Project RWP (R1) and compared dose projection for each work activity. The final outage RWP dose for the Control Rod Drive Push/Pull and Rebuild was 161 percent of the planned dose for the work scope, for the Refuel Project was also 161 percent of the planned dose, and each was individually greater than 5 Rem total.

Description: CRD project work was done on RWP 40040 Job Steps 6, 7, 8, 9 and 10. The scope of the project was to replace 19 CRDs and rebuild 20 CRDs. This included activities to leak test CRDs, transfer CRD filters to appropriate radwaste containers, provide under vessel support, remove shootout steel, flush LPRMs, delatch CR Blades, remove TIP tubing and other associated work.

In planning for the outage, the total project was compared to time and dose received on previous outages. That study revealed that the index dose rates (total dose over time) had continued to decrease since RFO-13. Upon final review of the outage doses per RWP, the RFO-19 dose index rate for the total CRD project was 4.09 mrem/hr. Also, the total dose per CRD worked had continued to decline. Therefore, it was concluded that the dose rates in the work areas were not a factor in the project going over the estimated dose.

The licensee's evaluation identified that the cause for the low dose estimate for the project was time spent in the radiological area. The dose rate index and the number of radiological workers was accurate. However, the planning failed to accurately estimate the total time for the project.

The licensee identified that failure to adequately communicate work estimates between the operational departments and the ALARA staff during the process of developing ALARA plans was also a contributing factor in the underestimate of the initial dose projections.

The Radiation Protection group identified weaknesses in arriving at accurate dose estimates, or planning. In response, several Corrective Action documents were developed regarding the need for improved dose estimation in the ALARA program. Additionally, the licensee developed an Action Plan for improving radiation exposure estimation and improving the overall ALARA activities at the Duane Arnold Energy Center (DAEC).

In execution of the project ALARA plan, the ALARA staff did not have reviews at a frequency that would assure that the project accumulated dose was progressing as planned. Most of the project and RWP assigned radiation work was completed within a 24 hour period. The ALARA coordination did not consider a more frequent review of actual doses against projections, based on historical ALARA accomplishments. Therefore, the dose exceeded the projection before the ALARA coordinator had the opportunity to conduct a job-in-progress review. Several methods for overcoming schedule complexities like this are available to the licensee but were not considered or implemented during the outage.

A second work activity, the Refueling Project RWP (R1), was estimated at 8500 millirem. The actual exposure was 13648 millirem, 61 percent over the estimate. The original dose estimate for RFO-19 as documented in the Tier 2 ALARA Job Planning Checklist was 8000 millirem with the comment assuming RFO-19 similar scope. The exposure for RFO-18 was 8207 millirem. RFO-19 dose estimate was later revised to 8500 mrem. The ALARA staff confirmed this estimate with the contractor, Framatome ANO, and representatives concurred with this estimate.

Following the discovery of the RWP dose overage, a study of the RWP time and dose was conducted. The area and individual dose rates for this work activity during the past several outages were compared. The licensee found that cavity dose rates for RFO-19 were higher than RFO-18, but lower than the average of the last three outages. The refuel support dose rates continued a downward trend over the latest outages, however, the refuel bridge work showed a slight increase. The increased work area dose rates (cavity and bridge) contributed to the increased actual exposures vs. the dose estimate. The licensee estimated that the index dose rate from RFO-18 to RFO-19 resulted in an additional 1675 mrem during the 1117 person-hours of cavity work and an additional 956 mrem during the 3416 person-hours of bridge work. The licensee concluded the project as a whole took more time than planned and that in conjunction with the slight work area dose increases, resulted in the dose in excess of the RWP cumulative exposure plan.

In execution of the project ALARA plan, the ALARA staff did not respond to dose estimate updates to assure that the project accumulated dose was progressing as planned. The dose exceeded the projection before the ALARA coordinator had the opportunity to conduct a job-in-progress review. Therefore, the job-in-progress review was not conducted.

Analysis: The performance deficiency associated with the planning and execution of these work activities was failure to follow procedure. The finding, which is under the Occupational Radiation Safety Cornerstone, does not involve the application of traditional enforcement because it did not result in actual safety consequences or potential to impact the NRC's regulatory function and was not the result of any willful actions. The finding was more than minor as it could be reasonably viewed as a precursor to a more significant event.

Enforcement: TS 5.4.1, Procedures, references a commitment to Reg Guide 1.33, Quality Assurance Program Requirements, that commits the licensee to procedures for Radiation Protection. DAEC procedure, HPP 3102.02, ALARA Job Planning, Step 4.7.1, In Progress Evaluation of Ongoing Work Activities, requires, for jobs with a duration of greater than eight hours and job dose estimates greater than 400 person-millirem, that the job be evaluated when the actual exposure is about one-half of the estimate. Additionally, the procedure states that the review will be documented on Form HP-60IPE, In-Progress Evaluation. Contrary to the above, for two separate work activities, Control Rod Drive Push/Pull and Rebuild, and Refuel Project, jobs with dose estimates greater than 400 person-millirem, work in-progress evaluations were not conducted. This resulted in the cumulative doses for two separate jobs exceeding the dose estimate by greater than 50 percent and the total dose exceeding 5 rem. Because work was conducted under an approved RWP, and individual doses were monitored by Electronic Personal Dosimeters and did not exceed individual dose goals, the event is of low safety significance and the finding is within the licensee response band. The licensee has entered the issue into their corrective action program as CAP 36762. The associated violation is being treated as an NCV, consistent with Section VI.A of the NRC Enforcement Policy. (NCV 05000331/2005004-03)

4. OTHER ACTIVITIES

4OA2 Identification and Resolution of Problems (71152)

Cornerstone: Initiating Events, Mitigating Systems, and Barrier Integrity

.1 Routine Review of Identification and Resolution of Problems

a. Inspection Scope

For inspections performed and documented in previous sections of this report, the inspectors routinely reviewed issues during baseline inspection activities and plant status reviews to verify that they were being entered into the corrective action program at an appropriate threshold, that adequate attention was being given to timely corrective actions, and that adverse trends were identified and addressed. Minor issues entered into the corrective action program as a result of the inspectors' observations are included in the attached list of documents reviewed. This activity does not count as an annual sample.

b. Findings

A specific issue which involved the failure to identify that the placement of fire extinguishers did not satisfy fire protection code requirements during a self-assessment of code compliance for fire extinguishers performed in April 2004 was identified during this routine review as discussed in Section 1R05.

.2 Daily Corrective Action Program Reviews

a. Inspection Scope

In order to assist with the identification of repetitive equipment failures and specific human performance issues for follow-up, the inspectors performed a daily screening of items entered into the licensee's corrective action program. This review was accomplished through inspection of the station's daily condition report packages. This activity does not count as an annual sample.

b. Findings

A specific issue which involved a Licensee-Identified Violation was identified during this daily review as discussed in Section 4OA7.

.3 Operational Experience Evaluation of Single Failure Identification for Safety Bus Lockout

Introduction: During review of the licensee's evaluation of an external operating issue associated with a single failure that could prevent re-energizing both safety busses, the inspectors noted several issues with the evaluation. The issues included the thoroughness of the review, the timeliness of the corrective actions, and the adequacy of the corrective actions.

The inspectors selected the following Documents for review:

- Operating Experience (OE) 4358, Single Failure that Could Prevent Re-energizing Both Safety Busses, February 1, 2005;
- CAP 35060, Evaluate Potential Vulnerability of Alternate Shutdown Power Supply, February 25, 2005;
- Condition Evaluation (CE) 2368, Evaluate Potential Vulnerability of Alternate Shutdown Power Supply, March 1, 2005;
- Procedure Change Request (PCR) 40026, Abnormal Operating Procedures (AOP)-915 Change, March 31, 2005;
- CAP 36660, Corrective Action Does Not Address all Failure Sequences, June 2, 2005;

- Apparent Cause Evaluation (ACE) 1463, Corrective Action Does Not Address All Failure Sequences, June 8, 2005; and
- CE 2705, Develop Plan and Plant Effect Evaluation, June 7, 2005.

a. Effectiveness of Problem Identification

(1) Inspection Scope

The inspectors reviewed the licensee's CAP entries and actions associated with this issue to verify that the identification of the problems by the licensee were complete, accurate, and timely, and that the considerations for the extent-of-condition, generic implications, common cause, and previous occurrences reviews were adequate.

(2) Issues

The licensee's initial evaluation, which began on February 1, 2005, on OE 4358, concluded that the essential bus lock out relay circuit was independent, with no metering crossties. An additional review was performed due to a corrective action document from the Monticello plant. This resulted in CAP 35060 being written on February 25, 2005, to address the potential vulnerability of the alternate shutdown power supply. Following that review, CE 2368 was written on March 1, 2005, to address a hot short scenario. On June 2, 2005, CAP 36660 was written due to the fact that the corrective actions performed by CAP 35060 did not properly address all failure sequences. Following the identification of the incomplete corrective actions, the licensee performed ACE 1463 to evaluate the entire corrective action effort relevant to this issue. Temporary Modification 05-007 was developed on June 8, 2005, and installed on June 17, 2005, to isolate ammeter and wattmeter indications in the control room, thereby eliminating the hot short lockout. A permanent modification has since been designed to eliminate the hot short scenario. There are some potential issues associated with timeliness and identification of issues associated with this operating experience evaluation. These issues will be further evaluated when Unresolved Item (URI) 05000331/2005004-04 as discussed in section 4OA3.1 is closed.

b. Effectiveness of Corrective Actions

(1) Inspection Scope

The inspectors reviewed the corrective action documents to determine if they addressed generic implications and that corrective actions were appropriately focused to correct the problem.

(2) Issues

As stated earlier, the licensee's initial evaluation concluded that the essential bus lock out relay circuit was independent, with no metering crossties. An additional review was performed, based on operating experience from the Monticello plant, which identified a potential scenario where the essential bus power would be unavailable due to a hot short that could cause a bus lockout. This review resulted in PCR 40026 being initiated to change AOP 915, " Shutdown Outside Control Room." The change provided guidance to the operators to reset the lockout relays for bus 1A4 by manually removing relay paddles. The relay paddles had to be removed, since the metering circuits are not isolated by the operation of the transfer switches, therefore the fault could still cause a bus lockout to occur, after the remote shutdown panel is activated, unless and until the manual actions are completed. This procedure change was then evaluated by the back up Appendix R engineer for review and approval. During that review it was determined that the feasability of completing the manual actions in time to meet the 20 minute requirement to establish alternate shutdown capability had not been fully addressed. On June 2, 2005, it was determined that the remote shutdown panel did not meet design criteria for Appendix R. This resulted in CAP 36660 being written due to the failure to fully address all failure modes. Temporary Modification 05-007 was installed on June 17, 2005, to isolate ammeter and watt meter indications in the control room, thereby eliminating the hot short lockout. A permanent modification has since been designed to eliminate the hot short scenario. There are some potential issues associated with the effectiveness of corrective actions associated with this operating experience review. These issues will be further evaluated when URI 05000331/2005004-04, as discussed in section 4OA3.1 is closed.

4OA3 <u>Event Follow-up (71153)</u>

.1 <u>(Closed) Licensee Event Report (LER) 05000331/2005-001:</u> Failure to Demonstrate the Capability to Achieve and Maintain Safe Shutdown Conditions.

On June 2, 2005, a review of an event reported at Monticello resulted in the determination that Duane Arnold Energy Center was vulnerable to a similar issue in that, the Remote Shutdown Panel did not meet design criteria for Appendix R, because the metering circuits in the control room could cause a lock out of an essential bus. Control room 4160 VAC current sensing and protective relaying circuits for the standby transformer, the startup transformer, and the 1G21 diesel generator, as well as the kilowatt meter for the 1G21 diesel generator has the potential to initiate a 1A4 essential power bus lockout if a "hot short" were to occur in this control room circuitry. The circuitry for these current transformers has a connection for the 1G21 diesel generator to facilitate a single ammeter and a watt meter that are located in the control room. In a potential fire event, it is plausible that an outside voltage source in contact with one or more current transformer phase legs, could force current through the over-current trip relay. The resultant current could cause the over-current relays to trip, thereby resulting in lockout signals at essential bus 1A4 or the 1G21 diesel generator. This would result in essential bus 1A4 being unavailable. Essential bus 1A4 is required to mitigate the consequences of an accident because it supplies 4.16 KV electrical power to safety-related equipment. The inspectors determined that the licensee failed to identify the single failure vulnerability associated with the ammeter circuits when the remote shutdown panel was designed and installed from 1983 to 1985. On June 17, 2005, a temporary modification was implemented that disconnected control room ammeter cables in the 1A4 essential switchgear room removing the possibility of a "hot short" causing a lockout. The inspectors reviewed the licensee's apparent cause evaluation, the corrective actions implemented and planned, and compliance with requirements.

Pending the results of the significance determination evaluation, this issue is being treated as an Unresolved Item. LER 05000331/2005-001-00 is closed to URI 05000331/2005004-04. The licensee entered this issue into their corrective action program as CAP 35060.

.2 (Closed) LER 05000331/2005-002: Both Standby Gas Treatment Trains Briefly Inoperable During Testing.

On June 20, 2005, the licensee identified that the 'B' SBGT train flow-indicating controller was not controlling flow at the required setpoint during the monthly 10-hour run for testing. Therefore, the 'B' SBGT train was declared inoperable. As part of the monthly testing, the 'A' SBGT train Mode Select Switch was placed in Manual with the inlet damper closed, thereby rendering the 'A' SBGT train inoperable. Between discovering the problems associated with the 'B' SBGT controller and the return of the 'A' SBGT train to operable status, both SBGT trains were inoperable for approximately four minutes. Since the issue did not affect other systems needed to mitigate the consequences of an accident, and since both trains for SBGT were only inoperable for four minutes, this issue is of very low safety significance, therefore this event constitutes a violation of minor significance that is not subject to enforcement action in accordance with Section IV of the NRC's Enforcement Policy. The licensee attributed the cause to a failure of a capacitor in the 'B' SBGT flow controller. Corrective actions included the restoration of the 'A' SBGT train to operable status and the repair of the 'B' SBGT flow controller. The licensee is also tracking the replacement of both SBGT flow controllers with a newer more reliable model. The LER was reviewed by the inspectors and no finding of significance was identified. The licensee entered this issue into their corrective action program as CAP 36839.

.3 (Closed) LER 05000331/2005-003: Inadequate Procedure Leads to Unplanned Mode Change while Performing SCRAM Time Testing.

On April 26, 2005, the licensee violated the requirements of TS 3.10.1. Specifically, after the completion of the reactor coolant system hydrostatic test and code required Visual Test - 2 inspections, the licensee remained above 212 degrees Fahrenheit, while conducting control rod SCRAM Time Testing. This resulted in an unplanned transition from Mode 4 to Mode 3 without completing all of the required Mode 3 TS requirements. Since the issue occurred at the end of the refueling outage, the resultant reactor fuel decay heat load was very low. In addition, multiple trains of emergency core cooling systems were available for accident mitigation purposes, therefore the issue was of very low safety significance. The licensee attributed the cause to a failure to recognize that the procedure changes made to allow for SCRAM Time Testing under extended vessel hydro conditions was contrary to TS requirements. Corrective actions included quarantining STP 3.10.1-01 and 3.10.1-02. In addition, the licensee will submit a plant-specific license amendment request to allow this testing in the future. The LER was reviewed by the inspectors and no findings of significance were identified, other than those already discussed in inspection report 5000331/2005011. The licensee documented the issue in CAP 36273.

4OA4 Cross-Cutting Aspects of Findings

.1 A finding described in Section 1R05 of this report had, as its primary cause, a Problem Identification and Resolution performance deficiency (Identification), in that, the Licensee’s fire protection personnel failed to identify that the placement of fire extinguishers did not satisfy fire protection code requirements during a self-assessment of code compliance for fire extinguishers performed in April 2004.

.2 A finding described in Section 1R19 of this report had, as its primary cause, a Human Performance deficiency (Personnel), in that, an operator failed to properly perform and verify a valve lineup during restoration of the fire protection system.

4OA5 Other Activities

.1 Operational Readiness of Offsite Power (Temporary Instruction (TI) 2515/163)

Cornerstone: Initiating Events, Mitigating Systems

The objective of TI 2515/163, "Operational Readiness of Offsite Power," was to confirm, through inspections and interviews, the operational readiness of offsite power (OSP) systems in accordance with NRC requirements. As a follow-up to the review performed May 22 - 25, 2005, the inspectors reviewed licensee procedures and discussed the attributes identified in TI 2515/163 with licensee personnel, for those site specific issues which had a negative response based on the initial review.

4OA6 Meetings

.1 Exit Meeting

The inspectors presented the inspection results to Mr. G. Van Middlesworth and other members of licensee management on October 4, 2005. The licensee acknowledged the findings presented. The inspectors asked the licensee whether any materials examined during the inspection should be considered proprietary. No proprietary information was identified.

.2 Interim Exit Meetings

Interim exit meetings were conducted for:

- Occupational Radiation Safety inspection with Mr. G. Van Middlesworth, Site Vice President, on September 9, 2005.

4OA7 Licensee-Identified Violations

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

Cornerstone: Initiating Event

.1 Appendix A to the NRC Branch Technical Position, requires, in part, that the use of combustible material in safety related areas should be controlled. Use of wood inside buildings containing safety-related systems should be permitted only when suitable non-combustible substitutes are not available. If wood must be used, only fire retardant treated wood should be permitted. 10 CFR 50 Appendix A, requires, in part, that structures, systems, and components important to safety shall be designed and located to minimize, consistent with other safety requirements, the probability and effect of fires and explosions. Non-combustible and heat resistant materials shall be used, whenever practical, throughout the unit, particularly in locations such as containment and control room. Contrary to these requirements, on September 21, 2005, the licensee identified that they failed to use non-combustible materials when they modified the control room supervisor's desk area in the control room. A new raised floor and desk arrangement was installed using plywood and other combustible materials. The additional materials resulted in the licensee exceeding the allowed combustible materials loading in the control room.

Even though the licensee exceeded the allowed combustible loading in the control room, the control room is continuously manned and a low flashpoint combustible liquid was not involved, therefore, the finding is assigned a low degradation rating. Because it is a low degradation rating, the violation is of very low safety significance (Green). Since the licensee identified the problem and took corrective actions, this violation is being treated as an NCV. The licensee documented the issue in CAP 37978.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee
G. Van Middlesworth, Site Vice President
J. Bjorseth, Site Director
D. Curtland, Plant Manager
S. Catron, Nuclear Safety Assurance Manager
S. Haller, Site Engineering Director
B. Kindred, Security Manager
C. Kress, Training Manager
G. Rushworth, Operations Manager
G. Pry, Maintenance Manager
D. Wheeler, Chemistry Manager
J. Windschill, Radiation Protection Manager

Nuclear Regulatory Commission
D. Spaulding, Project Manager, NRR
B. Burgess, Chief, Reactor Projects Branch 2

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

05000331/2005004-01	NCV	Failure to Meet Travel Distance Requirements for Fire Extinguishers in the Reactor Building. (1R05)
05000331/2005004-02	FIN	Failure to Adequately Verify a Valve Lineup in the Fire Protection System. (1R19)
05000331/2005004-03	NCV	Radiation Work Permit Dose Exceeded the Estimate by 61 Percent on Two Separate Work Activities. (2OS2)
05000331/2005004-04	URI	Failure to Demonstrate the Capability to Achieve and Maintain Safe Shutdown Conditions Due to Bus Lockout. (4OA3)

Closed

05000331/2005004-01	NCV	Failure to Meet Travel Distance Requirements for Fire Extinguishers in the Reactor Building. (1R05)
05000331/2005004-02	FIN	Failure to Adequately Verify a Valve Lineup in the Fire Protection System. (1R19)
05000331/2005004-03	NCV	Radiation Work Permit Dose Exceeded the Estimate by 61 Percent on Two Separate Work Activities. (2OS2)

05000331/2005-001	LER	Failure to Demonstrate the Capability to Achieve and Maintain Safe Shutdown Conditions. (4OA3)
05000331/2005-002	LER	Both Standby Gas Treatment Trains Briefly Inoperable During Testing. (4OA3)
05000331/2005-003	LER	Inadequate Procedure Leads to Unplanned Mode Change while Performing SCRAM Time Testing. (4OA3)

Discussed

None.

LIST OF DOCUMENTS REVIEWED

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

1R04 Equipment Alignment

Operating Instruction (OI) 152A4, HPCI System Control Panel Lineup, Revision 0
OI 152A1, HPCI System Electrical Lineup, Revision 0
OI 152, HPCI System Valve Lineup, CRD Flow Control Station Area, Revision 4
OI 152, HPCI System Valve Lineup, HPCI Room, 717' and 732' Area, Revision 4
OI 152A4, HPCI System Control Panel Lineup, Revision 0
BECH-M133<2>, P & ID, Fire Protection, Revision 22
BECH-M133<4>, P & ID, Fire Protection Deluge System, Revision 13
CAP 28243, Adverse trend in “Mispositioning Events”, dated July 16, 2003
CAP 36846, Potential Adverse Trend / Declining Performance Noted During Ops DRUM, dated June 20, 2005
CAP 37272, PA System / Fire & Evac Alarm Strobe Lights Failure Rate Too High, dated July 28, 2005 (NRC Identified)
CAP 37522, BECH-M133<2> & <4> Drawing Discrepancies (NRC Identified)
OI 513A1, Fire Protection System Electrical Lineup, Revision 4
OI 513A2, Fire Protection System Valve Lineup, Revision 4
Operations Department Rollup Meeting Report, dated First Quarter 2005
OI 454 Attachment 6, ESW System Control Panel Lineup, Revision 0
OI 454 Attachment 1, ESW System Electrical Lineup, Revision 1
OI 454A2, ‘A’ ESW System Valve Lineup and Checklist, Revision 6
OI 454 Attachment 3, ‘A’ ESW System Valve Checklist, Revision 6
OI 416 Attachment 6, RHRSW System Control Panel Lineup, Revision 4
OI 416 Attachment 3, ‘A’ RHRSW System Valve Checklist, Revision 6
OI 416 Attachment 1, RHRSW System Electrical Lineup, Revision 2

1R05 Fire Protection

AFP 69, Main Transformer 1X1, Revision 2
AFP 70, Standby Transformer 1X4, Revision 3
AFP 71, Startup Transformer 1X3, Revision 2
AFP 72, Auxiliary Transformer 1X2, Revision 1
AFP-03, Reactor Building HPCI, RCIC & Radwaste Tank Rooms, Revision 23
AFP-10, Reactor Building, Main Exhaust Fan Room, Heating Hot Water Pump Room and the Plant Air Supply Fan Room, Revision 25
AFP-11, Reactor Building Laydown Area - El. 833'-6", Revision 23
AFP-12, Reactor Building Decay Tank and Condensate Phase Separator Rooms, Revision 23

AFP-13, Reactor Building Refueling Floor, Revision 23
AFP-14, North Turbine Building Basement Reactor Feed Pump Area and Turbine Lube Oil Tank Area, Revision 28
AFP-25, Control Building Cable Spreading Room, Revision 25
CAP 37370, Access to fire extinguisher restricted by equipment, dated August 3, 2005 (NRC Identified)
CAP 37374, House Keeping concern in RCIC, dated August 3, 2005 (NRC Identified)
CAP 37379, Safe use of 1 ½ fire hose, dated August 5, 2005 (NRC Identified)
CAP 37407, Travel distances exceeded to portable fire extinguishers, dated August 5, 2005 (NRC Identified)
CAP 37408, Design/Licensing basis concerns with Sprinkler 1 (turbine lube oil tank area), dated August 5, 2005 (NRC Identified)
FPE-S02-005, Fire Extinguisher Code Compliance, Revision 1
RFT007480, Fire Brigade Annual Live Fire Training, dated August 11, 2005 (NRC Identified)
CAP 37569, Deluge Manual Pull Station, August 19, 2005 (NRC Identified)
CAP 37272, PA System / Fire & Evac Alarm Failures, July 26, 2005 (NRC Identified)

1R06 Flood Protection Measures

1249309D-010, Probabilistic Evaluation of Internal Flooding for Duane Arnold Energy Center, Revision 2
CAP 37476, RCIC Room Hatch leaking during a rainstorm, dated August 11, 2005 (NRC Identified)
CAP 37553, Seismic Adequacy of 1T070 and 1T073, dated August 18, 2005 (NRC Identified)
CAP 37590, RCIC Flooding ARP, August 22, 2005 (NRC Identified)
CAP 37591, Single Valve Isolations, August 22, 2005 (NRC Identified)

1R11 Licensed Operator Requalification Program

SEG 404, Loss of Drywell Cooling / Electrical ATWS, Revision 1
EOP 2, Primary Containment Control, Revision 12
EOP 1, Reactor Pressure Control, Revision 11
Emergency Depressurization (ED), Revision 4
EOP-ATWS, ATWS-Reactor Pressure Vessel, Revision 13
Emergency Action Level (EAL) Table 1, Revision 5
ACP 110.1, Conduct of Operations, Revision 3
ACP 101.01, Procedure Use and Adherence, Revision 31
ACP 101.2, Verification Process and SELF / PEER Checking Practices, Revision 5

1R12 Maintenance Effectiveness

July/August 2004 Maintenance Rule Monitoring and Status Report, October 29, 2004
Maintenance Rule Performance Criteria Basis Document for RHR, Revision 4
Maintenance Rule Criteria Values for RHR, September 2004

CAP 32951, 1G031 frequency did not stabilize on 1C08 meter in required band, dated September 10, 2004
CAP 35236, A SBDG as found frequency OOS during STP 3.8.1-06, dated March 11, 2005
CAP 35551, A Diesel failed operability test run per 3.8.1-06, dated March 31, 2005
CAP 37453, Upgrade Maintenance Rule SBDG Criteria, dated August 10, 2005 (NRC Identified)
Performance Criteria Basis Document, Emergency Diesel Generators (EDGs), SUS 23.00, 24.01, 24.02, 24.03, Revision 23

1R13 Maintenance Risk Assessments and Emergent Work Control

Work Procedure Guidelines (WPG) - 2, On-Line Risk Management Guideline, Revision 19
Maintenance Risk Evaluation for Week 27, July 1, 2005 and Revision 1, July 5, 2005
DAEC Online Schedule, Week 9526/9527, July1, 2005
CWO A69901 “A” SBGT 1V-!A Flow Indicator Controller for Train A, July 5, 2005.
Maintenance Risk Evaluation for Week 28, July 8, 2005
DAEC Online Schedule, Week 9527/9528, July 8, 2005
Maintenance Risk Evaluation for Week 34, August 18, 2005 and Revision 1, August 22, 2005
DAEC Online Schedule, Week 9533/9534, August 18, 2005
Maintenance Risk Evaluation for Week 37, September 8, 2005
DAEC Online Schedule, Week 9536/9537, September 8, 2005

1R15 Operability Evaluations

ACP 110.3, Operability Determination, Revision 1
ACP 114.5, Action Request System, Revision 32
OPR 000293, Seismic Adequacy of 1T070 and 1T073, August 18, 2005
OPR 000295, RHRSW System, September 8, 2005

1R16 Operator Workarounds

ACP 1410.12, Operator Burden Program, Revision 0
CAP 36621, Feedwater Programming Problems, May 30, 2005
CAP 36660, Corrective Action Does Not Address all Failures, June 2, 2005

1R19 Post-Maintenance Testing

Maintenance Directive-024, Post Maintenance Testing Program, Revision 31
CAP 37133, 1P049 STP aborted, dated July 15, 2005
CAP 37215, Fire Pump STPs, dated July 25, 2005 (NRC Identified)
CAP 37216, Diesel Fire Pump Test Failure, dated July 25, 2005 (NRC Identified)

STP NS13B009, Diesel Driven Fire Pump Operability Tests and Fuel Oil Supply Verification, Revision 19
PWO 1131370, Annual Engine Inspection
CWO A65579, Fire Station, August 25, 2005
CAP 37648, Inadvertent water discharge from fire system, August 25, 2005
MWO 113133, Relocate Relay 127X/42, August 24, 2005
STP 3.3.8.1-06, Essential Bus Degraded Voltage Relays Logic System Functional Test, Revision 2
CWO A70340, 'B' SBGT, September 18, 2005
STP 3.6.4.3-05, SBGT Operation With Heaters On, Revision 2
CAP 38009, Step of STP 3.6.4.3-05 was N/A'd, September 22, 2005 (NRC Identified)

1R22 Surveillance Testing

STP 3.3.3.3-17, Main Steam Isolation Valve Functional Test, Revision 6
STP 3.3.5.1-12,Channel Functional Test of Reactor Steam Dome Pressure-Low (LPCI LOOP Select) Instrumentation, Revision 2
CW0 A69905, Drain/Test Valve PS4558-V-89 Leaking By, July 5, 2005
General Electric Drawing 791E414RS, Revision 31
STP 3.8.1.-04, 'A' SBDG Operability Test, Revision 20
CAP 37272, Review Practice of N/Aing steps, July 26, 2005 (NRC Identified)
STP NS540002, 'B' ESW Operability, Revision 16

2OS2 As Low As Is Reasonably Achievable Planning And Controls

CA 040337, Develop and Implement an Improved ALARA Work Planning Strategy, dated June 1, 2005
CAP 036257, Collective Radiation Exposure Estimates - Improvements Needed, dated May 2, 2005
CAP 36762, Investigate Exposures Received on M8 Control Rod Drive Project, dated June 13, 2005
CE 002489, Collective Radiation Exposure Estimates - Improvements Needed, dated May 3, 2005
CE 002728, Investigate Exposures Received on M8 Control Rod Drive Project, dated June 13, 2005
CE 002729, Investigate Exposures Received on Refuel Floor Project for Refuel Outage, dated June 13, 2005
HPP 3102.02, ALARA Job Planning, Revision 18
SA 005964, Radiation Protection Group Focused Self-Assessment - ALARA Work Planning, dated June 1, 2005

4OA2 Identification and Resolution of Problems

ACP 114.4, Corrective Action Program, Revision 16
ACP 114.5, Action Request System, Revision 40
CAP 37726, NRC Cross Cutting Finding, August 31, 2005 (NRC Identified)

CAP 37895, Trend in NRC PI&R Cross Cutting Findings, September 15, 2005 (NRC Identified)
CAP 37519, SGI Information not properly secured, August 15, 2005 (NRC Identified)
CAP 37950, Concern about RWP/Jobstep, September 20, 2005 (NRC Identified)
OE 4358, Single Failure that Could Prevent Re-energizing Both Safety Busses, February 1, 2005;
CAP 35060, Evaluate Potential Vulnerability of Alternate Shutdown Power Supply, February 25, 2005;
CE 2368, Evaluate Potential Vulnerability of Alternate Shutdown Power Supply, March 1, 2005;
PCR 40026, AOP-915 Change, March 31, 2005;
CAP 36660, Corrective Action Does Not Address all Failure Sequences, June 2, 2005;
ACE 1463,Corrective Action Does Not Address all Failure Sequences, June 8, 2005; and
CE 2705, Develop Plan and Plant Effect Evaluation, June 7, 2005.

4OA3 Event Follow-up

LER 2005-001, Failure to Demonstrate the Capability to Achieve and Maintain Safe Shutdown Conditions, August 1, 2005
CAP 35060, Corrective Action Does Not Address all Failure Sequences, June 2, 2005
LER 2005-002, Both Standby Gas Treatment Trains Briefly Inoperable During Testing, August 19, 2005
CAP 36839, ‘B’ SBGT train failed to achieve flow per STP 3.6.4.3-01 step 7.2.14.a, June 20, 2005
LER 2005-003, Inadequate Procedure Leads to Unplanned Mode Change while Performing SCRAM Time Testing, August 19, 2005
CAP 36273, SCRAM Time Testing, May 3, 2005
CAP 37489, NRC Reportable Event, August 12, 2005 (NRC Identified)
CAP 37043, Hotwell, CDE and Feedwater Filter Indication, July 8, 2005 (NRC Identified)

4OA7 Licensee-Identified Violations

CAP 37978, Wood in Control Room is not with accordance with approved fire protection program, September 21, 2005

LIST OF ACRONYMS USED

ACE	Apparent Cause Evaluation
ACP	Administrative Control Procedure
AFP	Area Fire Plan
ALARA	As-Low-As-Is-Reasonably-Achievable
AOP	Abnormal Operating Procedures
ATWS	Anticipated Transient Without Scram
CAP	Corrective Action Process
CE	Condition Evaluation
CFR	Code of Federal Regulations
CR	Control Rod
CRD	Control Rod Drive
CWO	Corrective Work Order
DAEC	Duane Arnold Energy Center
EAL	Emergency Action Level
ED	Emergency Depressurization
EDG	Emergency Diesel Generator
EOP	Emergency Operating Procedure
ESW	Emergency Service Water
FIN	Finding
FSAR	Final Safety Analysis Report
HPCI	High Pressure Core Injection
IMC	Inspection Manual Chapter
IP	Inspection Procedure
LER	Licensee Event Report
LOOP	Loss of Offsite Power
LPCI	Low Pressure Coolant Injection
MWO	Modification Work Order
NCV	Non-Cited Violation
NFPA	National Fire Protection Association
NRC	Nuclear Regulatory Commission
OE	Operating Experience
OI	Operating Instruction
OPR	Operability
OOS	Out-of-service
OSP	Offsite Power
OWA	Operator Workaround
PARS	Publicly Available Records
PCR	Procedure Change Request
PI	Performance Indicator
PMT	Post-Maintenance Testing
PWO	Preventative Work Order
RBCCW	Reactor Building Closed Cooling Water
RCIC	Reactor Core Isolation Cooling
RFO	Refueling Outage
RHR	Residual Heat Removal

LIST OF ACRONYMS USED

RHRSW	Residual Heat Removal Service Water
RP	Radiation Protection
RWP	Radiation Work Permit
SBDG	Standby Diesel Generator
SBGT	Standby Gas Treatment
SDP	Significance Determination Process
SEG	Simulator Exercise Guide
SGTS	Standby Gas Treatment System
SSCs	Structures, Systems, and Components
STP	Surveillance Test Procedure
TI	Temporary Instruction
TS	Technical Specification
UFSAR	Updated Final Safety Analysis Report
URI	Unresolved Issue
WPG	Work Procedure Guideline