July 21, 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
NRC INTEGRATED INSPECTION REPORT 50-331/03-04

Dear Mr. Peifer:

On June 30, 2003, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at your Duane Arnold Energy Center. The enclosed report documents the inspection findings which were discussed on June 30, 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 to 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 were three NRC-identified and three self-revealing findings of very low safety significance (Green). All of these findings were determine to involve violations of NRC requirements. However, because of their very low safety significance and because these issues were entered into your corrective action program, the NRC is treating these issues as Non-Cited Violations in accordance with Section VI.A.1 of the NRC's Enforcement Policy. Additionally, licensee identified violations are 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, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001; and the Resident Inspector Office at the Duane Arnold Energy Center.

Since the terrorist attacks on September 11, 2001, NRC has issued five Orders and several threat advisories to licensees of commercial power reactors to strengthen licensee capabilities, improve security force readiness, and enhance controls over access authorization. In addition to applicable baseline inspections, the NRC issued Temporary Instruction 2515/148, "Inspection of Nuclear Reactor Safeguards Interim Compensatory Measures," and its subsequent revision, to audit and inspect licensee implementation of the interim compensatory measures required by order. Phase 1 of TI 2515/148 was completed at all commercial nuclear power plants during calender year 2002 and the remaining inspection activities for Duane Arnold Energy Center were completed in May 2003. The NRC will continue to monitor overall safeguards and security controls at the Duane Arnold Energy Center.

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

Sincerely,

***/RA/***

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

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

Enclosure: Inspection Report 50-331/03-04

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REGION III

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| Docket No: | 50-331 |
| License No: | DPR-49 |
| Report No: | 50-331/03-04 |
| Licensee: | Alliant, IES Utilities Inc. |
| Facility: | Duane Arnold Energy Center |
| Location: | 3277 DAEC Road<br>Palo, Iowa 52324-9785 |
| Dates: | April 1, 2003 through June 30, 2003 |
| Inspectors: | G. Wilson, Senior Resident Inspector<br>S. Caudill, Resident Inspector<br>J. Belanger, Senior Physical Security Inspector<br>M. Holmberg, Senior Reactor Inspector<br>M. Kurth, Resident Inspector Quad Cities<br>J. Maynen, Physical Security Inspector<br>G. Pirtle, Physical Security Inspector<br>R. Schmitt, Radiation Specialist<br>D. Schrum, Reactor Inspector |
| Approved by: | Bruce L. Burgess, Chief<br>Branch 2<br>Division of Reactor Projects |

Enclosure

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## SUMMARY OF FINDINGS


IR 05000331-03-04, IES Utilities, Inc.; on 04/01/2003-06/30/2003, Duane Arnold Energy Center; Inservice Inspection Activities, Post Maintenance Testing, Refueling and Outage Activities, Temporary Plant Modifications, and Public Radiation Safety.

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. All 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 (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. <u>Inspector-Identified and Self-Revealed Findings</u>

**Cornerstone: Initiating Events**

- Green. The inspectors identified a finding of very low safety significance regarding failure to issue a procedure to examine Code Class 1 welds subject to crevice corrosion.

  This finding was more than minor because if left uncorrected, it could have resulted in failure to examine Code Class 1 welds subject to crevice corrosion and consequently could have allowed flawed Code components to go undetected. Undetected flaws in these areas could lead to failure of Class 1 piping components and result in an increased frequency for a loss of coolant accident. This finding was of very low safety significance because the inspectors identified this issue prior to the first scheduled inspection of components susceptible to crevice corrosion. This finding was determined to be a Non-Cited Violation of 10 CFR 50 Appendix B, Criterion V (Section 1R08).

**Cornerstone: Mitigating Systems**

- Green. The inspectors identified a finding of very low safety significance regarding inadequate qualification of a procedure used to conduct surface examination of safety-related piping system welds. Specifically, the licensee had not demonstrated that the dye penetrant materials used would identify flaws in safety-related welds at the expanded temperature ranges allowed in this procedure.

  This finding was more than minor because if left uncorrected, it could have adversely affected the licensee’s ability to perform an adequate inspection of safety-related piping welds. This finding was of very low safety significance because the licensee confirmed that this procedure had not been used on piping welds at the lower temperature ranges where it would not have adequately detected flaws. This finding was determined to be a Non-Cited Violation of 10 CFR 50.55a(g)4 (Section 1R1908).

- Green. A finding of very low safety significance was identified through a self revealing event when the licensee failed to adequately test the pilot solenoid valve replacement on Pressure Setpoint Valve (PSV) 4405 of the Automatic Depressurization System (ADS), during post maintenance testing. The inadequate testing procedure resulted in exceeding the required technical specification condition with the valve being inoperable. The valve was inoperable due to a wiring error during the installation of the pilot solenoid valve. The primary cause of this issue was related to the cross-cutting area of human performance. The licensee failed to adequately ensure that the PSV-4405 was operable prior to entering conditions that it required.

  The issue was more than minor because PSV-4405 was rendered inoperable for the ADS function. The issue was determined to be of very low safety significance, since the other ADS and Low Level Set (LLS) valves were available to perform the relief function. An NCV of 10 CFR 50, Appendix B, Criterion V, related to an inadequate test procedure for post maintenance testing of the ADS system pilot valve replacement was identified through a self-revealing event. (Section 1R19)

- Green. A finding of very low safety significance was identified by the inspectors when the licensee failed to have an adequate procedure for the primary containment closeout. The procedure did not adequately address the evaluation of debris left inside containment to ensure that the Emergency Core Cooling Systems (ECCS) strainers were not impacted.

  The issue was more than minor because if left uncorrected, it could become a more significant safety concern since the failure to perform an evaluation could result in exceeding the assumptions utilized in the ECCS strainer design calculations, thereby potentially degrading the ECCS strainers and affecting the plants mitigating systems. The issue was determined to be of very low safety significance, since the amount of debris left in the primary containment did not exceed the assumptions in the design criteria for the ECCS strainers. An NCV of 10 CFR 50, Appendix B, Criterion V, related to an inadequate procedure to closeout primary containment was identified by the inspectors. (Section 1R20)

- Green. A finding of very low safety significance was identified through a self revealing event when the licensee failed to follow procedures during plant equipment manipulations on the 1D15 120 VAC instrument inverter. The failure to follow procedures resulted in a blown fuse, thereby rendering the 1D15 invertor unavailable. The primary cause of this issue was related to the cross-cutting area of Human performance.

  The issue was more than minor because the failure to follow procedures resulted in a blown fuse that made the 1D15 120 VAC instrument inverter unavailable. The issue was determined to be of very low safety significance, since the 1Y1A regulating transformer supplied power to the division one instrument bus after the 1D15 invertor was made unavailable. An NCV of 10 CFR 50, Appendix B, Criterion V, related to the failure to follow procedures during plant equipment manipulations was identified through a self-revealing event. (Section 1R23)

**Cornerstone: Public Radiation Safety**

Green. The licensee identified a self-revealing violation of 10 CFR 20.1802, when the licensee failed to maintain control of licensed radioactive material in an unrestricted area that was not in storage (i.e., eddy current test equipment with a measurable amount of licensed radioactive material [.3 nCi of Co-60 and lesser quantities of Mn-54] which was found upon subsequent evaluation and survey at Point Beach station [the next location of use of this equipment]).

The finding was more than minor because it was associated with the “Program and Process” and “Human Performance” attributes of the Public Radiation Safety Cornerstone and affected the cornerstone objective in ensuring adequate protection of public health and safety from exposure to radioactive materials released into the public domain. This event was caused by human error and involved reaching a non-conservative conclusion during an incomplete evaluation of the presence of radioactive material on the item. However this finding associated with the licensee’s radioactive material control program was of very low safety significance in that public radiation exposure was not greater than 0.005 rem and the licensee did not have more than five radioactive material control occurrences (in the previous 8 quarters). Thus, this finding will be documented as a Non-Cited Violation (NCV) of 10 CFR 20.1802, where the licensee failed to maintain control of licensed radioactive material in an unrestricted area that was not in storage (Section 2PS3).

## B. Licensee-Identified Violations

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 4OA7 of this report.

# REPORT DETAILS

## Summary of Plant Status

The plant began the inspection period in a refueling outage. On April 19, 2003, the reactor was taken critical. The generator was placed on-line on April 20, 2003. Over the next few days the unit was gradually brought to full power and operated at or near full power for the rest of the inspection period.

## 1. REACTOR SAFETY

**Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity**

1R01 Adverse Weather (71111.01)

.1 Hot Weather

a. Inspection Scope

The inspectors conducted a review of the licensee's preparations for summer conditions to verify that the plant's design features and implementation of procedures were sufficient to protect mitigating systems from the effects of adverse weather. Documentation for selected risk-significant systems was reviewed to ensure that these systems would remain functional when challenged by inclement weather. In particular, the inspectors focused on the Pump House Heating Ventilation and Air Conditioning (HVAC) System, Intake Structure HVAC, Reactor Building HVAC System, and Main Generator System Cooling. For these areas, the inspectors reviewed Integrated Plant Operating Instruction (IPOI) 6, "Cold Weather Operations," Revision 26. During the week of May 24, 2003, the inspectors walked down portions of the systems discussed above and verified that the systems had been properly aligned for hot weather operation.

b. Findings

No findings of significance were identified.

.2 Site Conditional

a. Inspection Scope

The inspectors conducted a review of the licensee's preparations for adverse weather conditions to verify that the plant's design features and implementation of procedures were sufficient to protect mitigating systems from the effects of adverse weather. In particular, the inspectors focused on defined operator actions and readiness of essential systems associated with tornados. In particular, the inspectors focused on the Standby Diesel Generators and Refuel Floor operations. For these areas, the inspectors reviewed Abnormal Operating Procedure (AOP) 903, "Tornado," Revision 12. The inspectors walked down portions of the systems discussed and verified that the systems were properly aligned for operation during the week of May 10, 2003.

b. Findings

No findings of significance were identified.

1R04 Equipment Alignment (71111.04)

.1 Partial Walkdowns

a. Inspection Scope

The inspectors performed four partial walkdowns 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.

- Reactor Core Isolation Cooling (RCIC), during the week of April 21, 2003;
- Control Rod Drive System ‘B’ during the week of April 21, 2003;
- Standby Filter Unit during the week of April 26, 2003; and
- Core Spray System ‘A’ during the week of May 10, 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.

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 April 14, 2003

- AFP-17; Condenser Bay, Heater Bay, and Steam Tunnel.

During the week of April 26, 2003

- AFP-6; RHR Valve Room;
- AFP-7; Laydown Area, Corridor and Waste Tank Area, and Spent Resin Tank Room;
- AFP-8; Standby Gas Treatment System and MG Set Rooms; and
- AFP-9; RBCCW Heat Exchanger Area, Equipment Hatch Area, and Jungle Room.

During the week of May 3, 2003

- AFP-4; North Control Rod Drive Area.

During the week of May 10, 2003

- AFP-5; South Control Rod Drive Module Area;
- AFP-14; Reactor Feed Pump Area; and
- AFP-15; Lower Switchgear Room.

In particular, the inspectors verified that all observed transient combustibles were being controlled in accordance with the licensee's administrative control procedures. In addition, the inspectors observed the physical condition of fire suppression devices, such as overhead sprinklers, and verified that any observed deficiencies did not impact the operational effectiveness of the system. The physical condition of portable fire fighting equipment, such as fire extinguishers, was observed. The inspectors also verified that extinguishers were located appropriately, and that access to the extinguishers was unobstructed. Fire hoses were verified to be installed at their designated locations and the physical condition of the hoses was verified to be satisfactory and access unobstructed. The physical condition of passive fire protection features such as fire doors, ventilation system fire dampers, fire barriers, fire zone penetration seals, and fire retardant structural steel coatings was inspected and verified to be properly installed and in good physical condition.

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 seasonal external flooding during the week of May 17, 2003. 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

No findings of significance were identified.

1R07 Heat Sink Performance (71111.07)

a. Inspection Scope

The inspectors observed the licensee's visual inspection of the "B" Residual Heat Removal Heat Exchanger. The inspectors reviewed completed surveillance tests of the eddy current and thermal performance testing. In addition, the inspectors validated the data, by performing independent calculations to ensure that these test results provided adequate heat transfer capability and identified any common cause issues that had the potential to increase risk, during the week of April 5, 2003. The inspectors reviewed the licensee's analysis as compared against the acceptance criteria, the correlation of scheduled testing, the frequency of testing, the impact of instrument inaccuracies, and the impact of test conditions on test results.

b. Findings

No findings of significance were identified.

1R08 Inservice Inspection Activities (71111.08)

a. Inspection Scope

The inspectors evaluated the implementation of the licensee's inservice inspection program for monitoring degradation of the reactor coolant system boundary and risk significant piping system boundaries, based on review of records and in-process observation of nondestructive examinations.

From March 31, 2003, through April 2, 2003, the inspectors observed:

- Ultrasonic (UT) examination of four inch diameter welds CUB-J005 and CUB-J008 on the reactor water cleanup system inside the main steam tunnel;
- UT examination of 10 inch diameter pipe-to-elbow welds RRE-J005, RRF-J005, and RRG-J005 on the reactor recirculation piping inside containment; and
- UT examination of a 10 inch diameter pipe-to-pipe weld RRC-J004A, on the reactor recirculation piping inside the containment.

From March 31, 2003, through April 3, 2003, the inspectors reviewed:

- Examination reports for dye penetrant (PT) examination of core spray piping welds CSB-F002 and CSB-F004;

- Repair and replacement records for replacement of a 45 degree pipe elbow in the residual heat removal service water system, and repair welding on a spare emergency service water pump; and
- UT examination reports with recordable indications identified during four Class 1 Code component weld examinations during previous outages.

The inspectors reviewed these records and observed these activities to confirm conformance to requirements in the American Society of Mechanical Engineers (ASME) Code, Section III, Section V, Section IX and Section XI. The inspectors performed the review of records in an office inside the Instruction Support Center Building within the site protected area.

b. Findings

.1 Inadequate Procedure For Surface Examination of Code Components

Introduction: Green. The inspectors identified a Non-Cited Violation of 10 CFR 50.55a(g)(4) associated with inadequate qualification of a procedure used to conduct surface examination of safety-related piping system welds. Specifically, the licensee had not demonstrated that the PT materials used would identify flaws in safety-related piping welds at the expanded temperature ranges allowed in this procedure.

Description: On April 1, 2003, the inspectors identified that procedure ACP 1211.3 “NDE [Nondestructive Examination] Procedure For Liquid Penetrant (Visible Dye & Water Washable) PT-1" Revision 6, was not qualified in accordance with the ASME Code. In this procedure, the licensee specified that PT examinations could be completed in a temperature range of 35-150 degrees Fahrenheit, which was in excess of the Code recognized band of 60-125 degrees Fahrenheit. The Code allowed the licensee to select other temperature ranges provided a procedure demonstration was completed to identify flaws and qualify the procedure for the expanded temperature range. The licensee had not qualified the procedure after a change in 1995 which allowed the use of newer types of penetrant (SKL-HF/S) and developer (SKD-NF) materials. These materials had a different chemical composition from that used in the PT materials previously demonstrated by the licensee.

The inspectors were concerned that the newer types of penetrant and developer used by the licensee might not perform as intended at the expanded temperatures specified in the procedure. On April 1, 2003, the inspectors’ concern prompted the licensee to perform a demonstration to qualify procedure ACP 1211.3 with the newer PT materials. The licensee performed this demonstration on quench cracked aluminum comparator blocks at the temperature extremes allowed in the procedure. However, the licensee was not able to identify the fine cracking pattern in the comparator blocks at the lower end of the allowed temperature band using the newer PT materials. The inspectors were concerned that the procedure may not have adequately detected cracking and if it had been used on safety-related piping welds at the lower end of the allowed temperature band.

Analysis: The inspectors reviewed this finding 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 none of the examples listed in Appendix E accurately represented this example. 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 this finding was greater than minor in accordance with IMC 0612, “Power Reactor Inspection Reports,” Appendix B, “Issue Disposition Screening,” issued on April 29, 2002, because the finding if left uncorrected would become a more significant safety concern. Specifically, the licensee’s use of this procedure could have allowed flawed piping welds to go undetected potentially leading to inservice failures in mitigating systems (PT examinations are primarily conducted on Code Class 2 piping welds used in construction of mitigating systems). Therefore, the inspectors concluded that this finding had the potential to impact the Mitigating Systems Cornerstone. The licensee staff reviewed past examinations and confirmed that no Code components had been examined below the Code recognized minimum temperature band since introduction of the newer PT materials. Therefore, this inadequate procedure had not yet impacted system operability.

The inspectors evaluated this finding using Manual Chapter 0609, “Significance Determination Process,” Appendix A, “Significance Determination of Reactor Inspection Findings for At-Power Situations,” Phase 1 screening associated with the Mitigating Systems Cornerstone. The inspectors concluded that this finding was a design or qualification deficiency that did not result in loss of component/system function. Therefore, the inspectors screened this issue as a finding of very low safety significance (Green).

Enforcement: 10 CFR 50.55a(g)4 required in part that throughout the service life of a boiling or pressurized water reactor facility, components classified as ASME Code Class 1, 2 and 3 must meet requirements of Section XI. Section XI, IWA-2222 required that the “Liquid penetrant examination shall be conducted in accordance with Article 6 of Section V.” Paragraph T-647.1 of Article 6, of Section V, of the ASME Code required “When it is not practical to conduct a liquid penetrant examination within the temperature range of 60°F to 125°F, the examination procedure at the proposed lower or higher temperature range requires qualification.”

Contrary to these requirements, as of March 31, 2003, the licensee failed to qualify procedure ACP 1211.3 Revision 6 for the specified temperature range of 35-150°F. The licensee’s failure to qualify the procedure for the newer types of penetrant (SKL-HF/S) and developer (SKD-NF) materials used is an example where the requirements of 10 CFR 50.55a(g)(4) were not met and is a violation. However, because of the very low safety significance and because the issue was entered into the licensee’s corrective action program (CAP 026610), it is being treated as a NCV, consistent with Section VI.A.1 of the Enforcement Policy (NCV 50-331/03-04-01(DRP)).

.2 No Procedure To Implement Examination Of Welds Subject To Crevice Corrosion Introduction

Introduction: Green. The inspectors identified a Non-Cited Violation of 10 CFR 50 Appendix B, Criterion V associated with the licensee's failure to issue a procedure to examine Class 1 Code welds subject to crevice corrosion.

Description: On April 1, 2003, the inspectors identified that the licensee had not issued nor initiated a procedure to conduct volumetric inspections of Class 1 welds subject to crevice corrosion. In January 2003, the licensee obtained NRC approval to implement a risk based Inservice Inspection Program in accordance with EPRI-TR-112657 "Revised Risk-Informed Inservice Inspection Evaluation," in lieu of the ASME Code Section XI requirements for inspection of Class 1 piping welds. In accordance with Table 4-1 of EPRI-TR-112657, the licensee was required to inspect weld configurations (thermal sleeve to pipe welds) subject to crevice corrosion in accordance with expanded weld volumes defined in figures 4-6 and 4-7. These figures defined inspection volumes centered on the thermal sleeve attachment point at the inside of the pipe wall. This examination volume was a distinct and separate inspection volume which the licensee had not included in the existing UT examination procedures. The licensee had identified eight welds as susceptible to crevice corrosion (two on recirculation system, one on core spray and the balance on feedwater systems) and had scheduled inspection of these welds using existing UT procedures during the next refueling outage. However, as of April 1, 2003, the licensee had not issued nor initiated a procedure (or procedure change) to implement the expanded volumetric examinations of the crevice regions for these welds.

Analysis: The inspectors reviewed this finding 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 none of the examples listed in Appendix E accurately represented this example. 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 finding was greater than minor in accordance with IMC 0612, "Power Reactor Inspection Reports," Appendix B, "Issue Disposition Screening," issued on April 29, 2002, because the finding if left uncorrected would become a more significant safety concern. Specifically, the licensee's failure to implement a procedure to conduct inspection of weld volumes subject to crevice corrosion could have allowed flawed Code components to go undetected. Undetected flaws in these areas could have led to failure of Class 1 piping components and increased the frequency of a loss of coolant accident. Therefore, the inspectors concluded that this finding had the potential to impact the Initiating Event Cornerstone. Because the inspectors identified this issue prior to the first scheduled inspection of components susceptible to crevice corrosion there was no impact on system operability.

The inspectors evaluated this finding using Manual Chapter 0609, "Significance Determination Process," Appendix A, "Significance Determination of Reactor Inspection Findings for At-Power Situations," Phase 1 screening associated with the Initiating Event

Cornerstone. The inspectors concluded that this finding; did not contribute to the likelihood of a primary or secondary system loss of coolant accident; did not contribute to both the likelihood of a reactor trip and the likelihood that mitigation equipment or functions would not be available; and did not increase the likelihood of fire or internal/external flooding. Therefore, the inspectors screened this issue as a finding of very low safety significance (Green).

Enforcement: 10 CFR 50.55a(a)3 required proposed alternatives to the requirements of paragraph (g) (e.g., adherence to Section XI Code requirements) may be used when authorized by the Director of the Office of Nuclear Reactor Regulation. By letter dated January 17, 2003, the NRC approved the licensee's use of the risk-informed Inservice Inspection Program in accordance with EPRI TR-112657, Revision B-A as an alternative to the ASME Code Section XI requirements for examination of Class 1 piping welds at the Duane Arnold Nuclear Generating Plant. In Table 4-1 of EPRI TR-112657, the licensee was required to perform examination of weld volumes as defined in figure 4-6 and 4-7 for welds subject to crevice corrosion. 10 CFR 50 Appendix B, Criterion V "Instructions, Procedures, and Drawings" required in part that activities affecting quality shall be prescribed by documented instructions, procedures, or drawings of a type appropriate to the circumstances.

Contrary to these requirements, as of April 1, 2003, the licensee had not issued or identified a procedure appropriate to the circumstances to perform the required volumetric examinations defined in figure 4-6 and 4-7 of EPRI TR-112657 for eight welds (RRF-F002, RRG-F002, CSB-F002, FWA-J002, FWA-J003, FWC-J002, FWC-J003, FWD-J003) susceptible to crevice corrosion. The failure to implement a procedure to conduct the required volumetric examinations of these welds is an example where the requirements of 10 CFR 50 Appendix B, Criterion V was not met and is a violation. However, because of the very low safety significance and because the issue was entered into the licensee's corrective action program (CAP026607, CAP 026608, CAP 026649), it is being treated as a NCV, consistent with Section VI.A.1 of the Enforcement Policy (NCV 50-331/03-04-02(DRP)).

1R11 Licensed Operator Requalification Program (71111.11)

a. Inspection Scope

On April 28, 2003, the inspectors observed a training crew during an evaluated simulator scenario of Simulator Exercise Guide (SEG) 2003C2-1, which included a jet pump failure, fuel damage, and a steam leak outside containment. Licensed operators' performances in mitigating the consequences of events were reviewed by the inspectors.

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 observed licensee evaluators to verify that they also noted the issues and discussed them in the end of the session critique.

b. Findings

No findings of significance were identified.

1R12 Maintenance Effectiveness (71111.12)

.1 Routine Maintenance Effectiveness

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 being designated as risk significant under the Maintenance Rule, or being in the increased monitoring of Maintenance Rule category a(1):

- Feedwater and Condensate Systems during the week of May 3, 2003;
- Radiation Monitors during the week of May 10, 2003; and
- High Pressure Coolant Injection (HPCI) during the week of June 16, 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.

.2 Biennial Periodic Evaluation

a. Inspection Scope

The objective of the inspection was to:

- Verify that the periodic evaluation was completed within the time restraints defined in 10 CFR 50.65 (once per refueling cycle, not to exceed 2 years), ensuring that the licensee reviewed its goals, monitoring, preventive maintenance activities, industry operating experience, and made appropriate adjustments as a result of that review;

- Verify that the licensee balanced reliability and unavailability during the previous refueling cycle, including a review of safety significant Structures, Systems and Components (SSCs);

- Verify that (a)(1) goals were met, corrective action was appropriate to correct the defective condition, including the use of industry operating experience, and (a)(1) activities and related goals were adjusted as needed; and

- Verify that the licensee has established (a)(2) performance criteria, examined any SSCs that failed to meet their performance criteria, or reviewed any SSCs that have suffered repeated maintenance preventable functional failures including a verification that failed SSCs were considered for (a)(1).

The inspectors examined the periodic evaluation report, "Maintenance Rule Program Cycle 17 Periodic Report," completed for the time period of December 1999 - May 2001. To evaluate the effectiveness of (a)(1) and (a)(2) activities, the inspectors examined (a)(1) action plans, justifications for returning SSCs from (a)(1) to (a)(2), and a number of corrective action documents (contained in the list of documents at the end of this report). In addition, Action Requests were reviewed to verify that the threshold for identification of problems was at an appropriate level and the associated corrective actions were appropriate. Also, the maintenance rule program documents were reviewed. The inspectors focused the inspection on the following systems:

- Residual Heat Removal

- Essential Service Water

- High Pressure Coolant Injection

- Emergency Diesel Generator

In addition, the inspectors reviewed licensee self-assessments that addressed the maintenance rule program implementation.

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, configuration control, and performance of maintenance associated with planned and emergent work activities 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, risk management tools, and the 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:

- The inspectors reviewed the maintenance risk assessment for work planned during the week of April 26, May 3, May 10, May 17, May 24, May 31, June 7, and June 14, 2003 for a total of 8 samples.

b. Findings

No findings of significance were identified.

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

.1 Maintenance on Division Two Emergency Core Cooling Systems (ECCS)

a. Inspection Scope

The inspectors observed the preparations for and management of the maintenance on the Division 2 ECCS which commenced on June 3, 2003. A review of the licensee's applicable procedures, licensing commitments, compensatory actions, personnel briefings, and Corrective Action Plans (CAP) generated to understand and resolve the details of this preplanned evolution was performed by the inspectors. In particular, the inspectors reviewed the operators' contingency actions to verify that they were appropriate for the evolution and in accordance with procedures and training. Detailed walkdowns of the job sites and activities were performed by the inspectors to ensure that all licensing commitments were met. A review of the licensee's risk control associated with these maintenance activities was performed by the inspectors including the verification that the protected Division 1 systems were operable, while the Division 2 systems were undergoing maintenance. The inspectors had several discussions with the evolution coordinator during the week to ensure that the control of work was maintained, as well as how delays were resolved. The inspectors ensured that post maintenance operability tests were adequately performed and that applicable procedures and paperwork were correctly performed prior to the relevant LCO's being exited.

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) 027317, “A” Pump House HVAC failed as left calibration, during the week of May 17, 2003.
- CAP 027209, “A” Emergency Service Water Strainer, during the week of May 17, 2003.
- Operability (OPR) 000226, Control Building Envelope, during the week of May 24, 2003.
- OPR 000225; “B” SBDG Normal Air Start Supply Valve; during the week of May 24, 2003.
- CAP 027012;Two Bolts Missing from Drywell Shielding; during week of May 24, 2003.

The inspectors reviewed the technical adequacy of the evaluation against the Technical Specification, 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;” Revision 32.

b. Findings

No findings of significance were identified

1R16 Operator Workarounds (OWA) (71111.16)

a. Inspection Scope

The inspectors performed a semiannual review of the cumulative effects of operator workarounds, during the week of June 7, 2003. The inspectors reviewed the cumulative effects of workarounds on the reliability, availability, and potential for improper operation of the system. Additionally, reviews were conducted to determine if the workarounds could increase the possibility of an initiating event, affect multiple mitigating systems, or impact the operators’ ability to respond to accidents or transients.

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.

- CWO1122937; Install Spare Emergency Service Water (ESW) Pump; during the week of April 12, 2003;
- CWO1119785; Overhaul Limitorque Operator Motor Operator (MO) 2312 High Pressure Coolant Injection (HPCI); during the week of April 12, 2003;
- CWO1119785; Overhaul Limitorque Operator MO2512 Reactor Core Isolation Cooling (RCIC) Inject Valve; during the week of April 12, 2003;
- CWOA53416; 1D4 Battery Replacement; during the week of April 12, 2003;
- CWO 1119819; "C" Inboard (INBD) Main Steam Isolation Valve (MSIV) Actuator Replacement; during the week of April 19, 2003;
- CWO 1121020; Change out diaphragm on SCRAM outlet valve operator; during the week of April 19, 2003; and
- CWO 1119848; Remove Pilot Valve and Install Spare for Main Steam "C" Automatic Depressurization System (ADS) Relief Valve; during the week of June 7, 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, Technical Specification (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

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 post maintenance test procedure for the replacement of the pilot solenoid valve on Pressure Setpoint Valve (PSV) 4405 of the Automatic Depressurization System (ADS), was identified through a self-revealing event.

Description: On April 18, 2003, while performing Surveillance Test Procedure (STP) 3.4.3-03, "Manual Opening of the Automatic Depressurization System (ADS) and Low Level Set (LLS) Relief Valves" PSV 4405 failed to operate. PSV 4405 is one of the four ADS valves. The STP was being performed as the post maintenance test for the pilot solenoid valve, which was recently replaced for PSV 4405. When the licensee investigated the failure, they found that the pilot solenoid valve was not connected per the electrical termination sheet, since the solenoid wires were not connected to the field wires. In addition it was also determined that the two independent checks, which consisted of a peer check and a quality control verification check, also failed to recognize the connection error. The failure to properly wire the pilot solenoid valve was a human performance deficiency. When PSV 4405 failed to operate, the inspectors evaluated the plant conditions required in Technical Specification (TS) 3.5.1 to those of the actual plant. TS 3.5.1 requires ADS valves to be operable prior to exceeding 100 per square inch gauge (psig) reactor steam dome pressure in Modes 1, 2 and 3. The post maintenance testing occurred at a reactor steam dome pressure of approximately 150 psig. The post maintenance test procedure was inadequate since it failed to ensure proper pilot solenoid valve operation prior to exceeding 100 psig reactor steam dome pressure in accordance with the requirements of TS 3.5.1. The inspectors determined that although PSV 4405 was inoperable for ADS purposes, the other three ADS valves along with the LLS valves were available to perform the relief function; 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 affected the mitigating system cornerstone attributes of equipment performance due to the inability of PSV 4405 to perform its ADS function.

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 Mitigation Systems Cornerstones; however, the finding 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 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 post maintenance test procedure was inadequate since it failed to ensure proper pilot solenoid valve operation for the PSV 4405, which is an Appendix B system, prior to exceeding 100 psig reactor steam dome pressure in accordance with the requirements of TS 3.5.1. The failure to have an adequate procedure to properly perform the post maintenance testing of PSV 4405 pilot solenoid valve 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 50-331/0304-04), 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 CAP027087.

Corrective actions taken included the revision of the lifted lead procedure and a change in the way post testing is performed to allow identification of potential problems prior to plant start up.

1R20 Refueling and Outage Activities (71111.20)

Refueling Outage Number 18

a. Inspection Scope

The inspectors evaluated outage activities for scheduled refueling outage number 18 that was in progress at the beginning of the inspection period and ended on April 20, 2003. The inspectors reviewed activities to ensure that the licensee considered risk in developing, planning, and implementing the outage schedule.

The inspectors observed or reviewed 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

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 perform the primary containment closeout was identified by the inspectors.

Description: During the week of April 19, 2003, the inspectors performed a closeout inspection of the primary containment, after the licensee had completed their closeout. Various items including tape, tags, insulation, rags, pens, tools, fibrous material, wood chips, welding rods, and metal debris were found by the inspectors. The licensee retrieved some of the debris, which filled a bag that was approximately 12 inches by 18 inches, while other debris which was located under the deck grating, on pipes, and on top of I-beams throughout the drywell was not retrieved and remained in the drywell. The inspectors asked to see the evaluation that was performed on the debris that remained in the drywell, to ensure that it was within the amount assumed in the Emergency Core Cooling System (ECCS) strainer analysis. Integrated Plant Operating Instructions (IPOI) 7, "Special Operations," which is the procedure used to perform the primary containment closeout, did not require an analysis of the debris left inside primary containment to be performed. The failure to perform an evaluation of the debris could result in exceeding the assumptions utilized in the ECCS strainer analysis, thereby potentially impacting the ability of the associated mitigating systems. The inadequate procedure resulted in the failure to ensure that the design assumptions used in the ECCS strainer analysis are maintained. CAP 027024 was written by the licensee to document and analyze the debris left in the drywell. The inspectors determined that although the licensee's procedure was inadequate to ensure that debris left inside primary containment was within the ECCS strainer analysis, the amount of debris left inside would not have clogged the ECCS strainers, 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 the procedure did not require an evaluation to be performed on the material left inside primary containment. The failure to perform an evaluation could result in exceeding the assumptions utilized in the design calculations, thereby potentially degrading the ECCS strainers and affecting the associated mitigating systems.

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 Mitigation Systems Cornerstones; however, the finding 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 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 procedure for the primary containment closeout did not ensure that there was an evaluation of the debris left inside primary containment, which could exceed the assumptions utilized in the ECCS strainer calculations, thereby potentially impacting the operation of the associated mitigating systems, which are Appendix B systems. The failure to properly address the impact of the debris resulted in an inadequate procedure. The failure to have an adequate procedure to properly perform a primary containment closeout 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 50-331/0304-03), 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 CAP027208.

Corrective actions taken included the evaluation of the debris left in the drywell and revision to IPOI 7 to include an evaluation of the debris left in the drywell.

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) NS490002, “Low Pressure Coolant Injection (LPCI) Inject Check Valve Full Flow Test,” during the week of April 5, 2003;
- STP 3.6.1.1-04, “Containment Isolation Valve Leak Tightness Test - Type C Penetrations - Main Steam System,” during the week of April 5, 2003;
- STP 3.9.1-01, “Refueling Interlocks Channel Functional Testing,” during the week of April 12, 2003;
- STP 3.7.2-01, “River Water Supply System Simulated Actuation Test,“ during the week of April 12, 2003;
- STP 3.3.5.1-29, “Containment Spray Logic System Functional Test (LSFT) and Residual Heat Removal (RHR) Timer Calibration,” during the week of April 12, 2003;
- STP 3.8.1-07, “Loss of Offsite Power (LOOP) - Loss of Coolant Accident (LOCA) Test,” during the week of April 12, 2003; and
- STP 3.3.5.1-15, “RHR LSFT,” during the week of April 19, 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, Technical Specifications (TS) applicability, impact of testing relative to performance indicator reporting, and evaluation of test data.

b. Findings

No findings of significance were identified.

1R23 Temporary Plant Modifications (71111.23)

a. Inspection Scope

The inspectors reviewed the following temporary modifications (TMOD):

- TMOD 03-033, “Lift the P4 Connector in the Kaman 10 Micro to Allow Kaman 9 to Operate While Troubleshooting is in Progress,” during the week of April 28, 2003;
- TMOD 03-036, “Remove Disk from Valve V13-053 to Allow Proper ESW Flow to CB Chiller,” during the week of June 2, 2003"; and
- TMOD 03-041, “ Monitor 1D15 120 VAC Instrument AC Invertor,” during week of June 28, 2003.

The inspectors reviewed the safety screening, design documents, USFAR, and applicable TS to determine that the temporary modifications were 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

Introduction: A finding of very low safety significance (Green) and an associated NCV of 10 CFR 50, Appendix B, Criterion V, related to the failure to follow the procedure for the temporary modification installed on the 1D15 120 VAC instrument inverter was identified through a self-revealing event.

Description: On June 18, 2003, an Instrumentation and Control Technician noticed that the Yokogawa recorder, which was installed by Temporary Modification 03-041, was not triggering properly. After a discussion with maintenance engineering, a decision was made to adjust the recorder program. Since only the triggering of the recorder was being changed, the adjustment was performed without a procedure. While performing the program adjustment, the recorder appeared to be locked up so the technician turned the recorder off for approximately 2 seconds. When the recorder was turned back on, a fuse blew in the 1D15 invertor resulting in a reverse power transfer to the 1Y1A regulating transformer. The preliminary cause for the blown fuse was the initialization of the recorder, after it was turned back on. The process of shutting off the recorder essentially removed the recorder from the rectifier circuit. Temporary Modification (TMOD) 03-041 contained specific procedure steps for turning the recorder on, which had a plant effect evaluation to ensure no adverse consequence to the system, for removing and installing the recorder. The failure to follow the procedure steps for returning the recorder to service, as described in TMOD 03-041, resulted in a blown fuse and the loss of the 1D15 invertor. The inspectors determined that although the 1D15 120 VAC instrument inverter was made unavailable, the Division One 120 VAC instrument bus remained energized by regulating transformer 1Y1A; 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 affected the mitigating systems cornerstone attributes of equipment performance due to the unavailability of the 1D15 120VAC instrument inverter.

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 Mitigation Systems Cornerstones; however, the finding 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 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 failure to follow the procedure, as described in TMOD 03-041, to reinstall the recorder resulted in a blown fuse in the 1D15 inverter, which is an Appendix B system.  The failure to follow the procedure to reinstall the recorder in accordance with TMOD 03-041 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 50-331/0304-05), 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 CAP027872.

Corrective actions taken included the evaluation of the potential impact of test equipment operation in the field and a reemphasis to all plant personnel that installed test equipment is to be controlled as plant equipment.

1EP6 Emergency Preparedness Drill Evaluation (71114.06)

a. Inspection Scope

On June 11, 2003, the inspectors observed an operating crew participate in an emergency preparedness simulator drill.  The inspectors monitored the operations crews' response to a steam line rupture in the Turbine Building which could not be isolated. This problem was followed by an inadvertent HPCI initiation and several RPS failures resulting in fuel cladding damage, leading to 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.

## 2. RADIATION SAFETY

### Cornerstone:  Occupational Radiation Safety (OS)

2OS1 Access Control to Radiologically Significant Areas (71121.01)

.1 Plant Walkdowns, Radiological Boundary Verifications, Radiation Work Permit (RWP) Reviews, and Observations of Radiation Worker Performance

a. Inspection Scope

The inspectors conducted walkdowns of the radiologically controlled area (RCA) to verify the adequacy of radiological boundaries, postings, and locking devices. Specifically, the inspectors walked down several radiologically significant work area boundaries (i.e., High Radiation Areas [HRA], Locked High Radiation Areas with radiation levels greater than 1,000 mr/hr, and a Very High Radiation Area [VHRA] with potential radiation levels > 500 R/hr) in the Drywell, Reactor Building, RadWaste Building, and Refuel Floor/Spent Fuel Pool areas. Confirmatory radiation measurements were taken to verify that these areas and other selected areas were properly posted and controlled in accordance with 10 CFR Part 20, licensee procedures, and Technical Specifications. The inspectors reviewed radiation work permits (RWPs) (i.e., for routine plant tours, reactor pressure vessel head bolt de-tensioning, inspection/cleaning of drywell torus, and work in the drywell) for engineering, operations, and maintenance activities, in support of Refueling Outage 18 (RFO 18). The RWPs were evaluated for protective clothing requirements, respiratory protection concerns, electronic dosimetry alarm set points, and radiation protection hold points, to verify that work instructions and controls had been adequately specified and that electronic dosimeter set points were in conformity with survey indications. The inspectors also observed radiation workers performing the activities described in Section 2OS2.4, to evaluate their awareness of radiological work conditions, and to verify the implementation of radiological controls specified in applicable radiation work permits.

b. Findings

No findings of significance were identified.

.2 Control of Non-Fuel Materials Stored in the Spent Fuel Pools

a. Inspection Scope

The inspector reviewed the licensee's programmatic controls and practices for the underwater storage of highly activated or contaminated materials (non-fuel) in the spent fuel or other storage pools. Radiation protection and fuel handling procedures were reviewed, involved staff were interviewed, the most recent inventory record for the spent fuel pools was reviewed and a walkdown of the refuel floor was conducted. The inspector assessed the adequacy of the administrative and physical controls for underwater storage of non-fuel materials for consistency with the licensee's procedures and with Regulatory Guide 8.38, Information Notice 90-33 and applicable Health Physics Positions in NUREG/CR-5569.

b. Findings

No findings of significance were identified.

.3 Problem Identification and Resolution

a. Inspection Scope

The inspectors reviewed licensee Action Requests (ARs) written since the last assessment (November 2002) to the date of the current assessment, which focused on access control to radiologically significant areas (i.e., problems concerning activities in HRAs, radiation protection technicians performance, and radiation worker practices). The inspectors specifically reviewed licensee ARs initiated in conjunction with RFO 18. Additionally, the inspectors also reviewed the 4th Quarter 2002 Action Request Radiological Occurrence Trend Reports. The inspectors reviewed these documents to assess the licensee's ability to identify repetitive problems, contributing causes, the extent of conditions, and implement corrective actions to achieve lasting results.

b. Findings

No findings of significance were identified.

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

.1 ALARA Planning

a. Inspection Scope

The inspectors examined the station's procedures for radiological work/ALARA planning and scheduling and evaluated the dose projection methodologies and practices implemented for RFO 18, to verify that sound technical bases for dose estimates existed. The inspectors reviewed the station's collective exposure histories from 1990 to the present, current exposure trends from ongoing plant operations, and completed radiological work activities for RFO 18 to assess current performance and outage radiation exposure challenges. The inspectors evaluated the licensee's effectiveness in exposure tracking for the outage to verify that the licensee could identify problems with its collective exposure and take actions to address them. Additionally, the inspectors reviewed a representative sampling of radiologically significant RWP/ALARA planning packages to verify that adequate person-hour estimates, job history files, lessons learned, and industry experiences were utilized in the ALARA planning process. As part of the reviews of the planning packages, the inspectors reviewed Total Effective Dose Equivalent (TEDE) ALARA evaluations developed for: (1) drywell cooler removal/replacement; (2) recirculation pump seal removal/replacement; and (3) welding on reactor head insulation package. The inspectors examined the TEDE ALARA evaluations to assess the licensee's analysis for the potential use of respiratory protection equipment and to verify the adequacy of the licensee's internal dose assessment processes/program for the aforementioned work evolutions.

b. Findings

No findings of significance were identified.

.2 Radiological Work Planning and ALARA Implementation

a. Inspection Scope

The inspectors selected the following RFO 18 work activities that were of highest exposure significance, or were otherwise conducted in the drywell, and assessed the adequacy of the radiological controls and work planning:

- Reactor disassembly/reassembly and refuel floor activities;
- In-service inspections; and
- Drywell A/B cooler replacements.

The inspectors reviewed the RWPs and the ALARA Reviews developed for each of the aforementioned jobs. The inspector examined the radiological engineering controls and other dose mitigation techniques specified in these documents and reviewed job dose history files to verify that licensee and industry lessons learned were adequately integrated into each work package. The inspectors reviewed the exposure results for the selected activities to evaluate the accuracy of exposure estimates in the ALARA plan.

b. Findings

No findings of significance were identified.

.3 Verification of Exposure Goals and Exposure Tracking System

a. Inspection Scope

The inspectors evaluated the licensee's effectiveness in exposure tracking for RFO 18 to verify that the licensee could identify problems with its collective exposure and take actions to address them. The inspectors reviewed the exposure history for each outage activity to determine if management was monitoring the exposure status, if in-progress ALARA job reviews were being properly performed, if additional engineering/dose controls needed to be established, and if required corrective documents had been generated. The inspectors compared exposure estimates, exposure goals, job dose rates, and person-hour estimates for consistency to verify that the licensee could project, and thus better control radiation exposure. The inspectors examined job dose history files and dose reductions anticipated through the licensee's implementation of lessons learned, from previous outages, to verify that the licensee could accurately forecast exposure dose goals. The inspectors examined the actual RFO 18 radiation dose exposure data to date (i.e., $\approx$24 person-Rem versus the projected dose $\approx$51 person-Rem.).

b. Findings

No findings of significance were identified.

.4 Job Site Inspections, Radiation Worker Performance, and ALARA Controls

a. Inspection Scope

The inspectors observed work activities in the radiologically controlled areas that were performed in radiation areas, HRAs, and locked HRAs to evaluate the use of ALARA controls. Specifically, the inspectors reviewed the adequacy of RWPs, radiological surveys and pre-job radiological briefings packages and assessed job site ALARA controls, in part, for the following work activities:

- Cavity flood-up and installation of Main Steam line plugs;
- Inspection and cleaning of the drywell torus;
- Reactor disassembly/reassembly and refuel floor activities; and
- Drywell A/B cooler replacements.

The inspectors examined worker instruction requirements which included protective clothing, engineering controls to minimize dose exposures, the use of predetermined low dose waiting areas, as well as the on-the-job supervision by the work crew leaders to verify that the licensee had maintained the radiological exposure for these work activities ALARA. The inspectors evaluated radiation protection technician (RPT) performance for each of the aforementioned work evolutions, as well as observed and questioned workers at each job location, to verify that they had adequate knowledge of radiological work conditions and exposure controls. Enhanced job controls including RPT use of electronic enhanced job controls including RPT use of electronic teledosimetry and remotely monitored cameras were also evaluated to assess the licensee's ability to maintain real time doses ALARA in the field.

b. Findings

No findings of significance were identified.

.5 Source Term Reduction and Control

a. Inspection Scope

The inspectors evaluated the licensee's source term reduction program in order to verify that the licensee had an effective program in place and was knowledgeable of plant source term reduction opportunities and that efforts were being taken to address them. Work control mechanisms for RFO 18 were evaluated to ensure that source term reduction plans had been appropriately implemented.

b. Findings

No findings of significance were identified.

.6 Identification and Resolution of Problems

a. Inspection Scope

The inspectors examined the licensee's lessons learned from RFO 17 refueling outage dose goal estimation process and its' subsequent effect on the establishment of the RFO 18 dose goal. The inspectors evaluated selected outage generated ARs, which focused on ALARA planning and controls. The inspectors examined the contents of a briefing package from a recent "Safety/Human Performance Supervisory Standown," which was held for all plant employees. Additionally, the inspectors reviewed the licensee's CY 2002 Radiation Protection Dosimetry summary report. The inspectors evaluated the effectiveness of the licensee's problem identification and resolution program to verify that the licensee could adequately identify individual problems/trends, determine contributing causes, extent of conditions, and develop corrective actions to achieve lasting results.

b. Findings

No findings of significance were identified.

2OS3 Radiation Monitoring Instrumentation and Protective Equipment (71121.03)

.1 Rescue Capabilities During Use of One-Piece Atmosphere Supplying Respiratory Protection Devices

a. Inspection Scope

The inspectors evaluated the licensee's respiratory protection program and the use of respiratory protection equipment to limit the intake of radioactive material. The inspectors examined the licensee's procedures, lesson plans, and related respiratory protection qualification training materials. The inspectors discussed their implementation relative to the requirements of 10 CFR 20.1703(f) for standby rescue persons whenever one-piece atmosphere supplying suits, or any combination of respiratory protection and personnel protective equipment were used which the wearer may have difficulty extricating himself. Specifically, the inspectors reviewed the licensee's work planning process and implementing practices, and interviewed RP staff and a member of the licensee's confined space rescue team regarding the following aspects of 10 CFR 20.1703:

- Designation of an adequate number of standby rescue workers and their training/instruction;
- Presence of equipment staged at the work site for the safety of the rescuer and for extrication of the respiratory equipment user;
- Practices for continuous communication between standby rescuer(s) and the respiratory protection user(s); and
- provisions for immediate availability of the standby rescuer.

The inspectors discussed with RP management its proposal for enhancing the radiation work permit and as-low-as-reasonably achievable (ALARA) planning process and for developing safety plans for those jobs (i.e., not performed in confined space atmospheres, but where limiting the intake of radioactive materials is desirable) to formally address work provisions for standby rescuers.

b. Findings

No findings of significance were identified.

**Cornerstone: Public Radiation Safety**

2PS3 Radiological Environmental Monitoring and Radioactive Material Control Programs (71122.03)

.1 Review of Environmental Monitoring Reports and Data

a. Inspection Scope

The inspectors reviewed the CY 2001 and 2002 Annual Environmental Monitoring Report. Sampling location commitments, monitoring and measurement frequencies, land use census, the vendor laboratory's Interlaboratory Comparison Program, and data analysis were assessed. Anomalous results including data, missed samples, and inoperable or lost equipment were evaluated. The review of the Radiological Environmental Monitoring Program (REMP) was conducted to verify that the licensee's program was implemented as required by the Radiological Environmental Technical Specifications/Offsite Dose Calculation Manual (RETS/ODCM), and associated Technical Specifications, and that changes, if any, did not affect the licensee's ability to monitor the impacts of radioactive effluent releases on the environment. The most recent quality assessment of the licensee's REMP vendor was reviewed to verify that the vendor laboratory performance was consistent with licensee and NRC requirements.

b. Findings

No findings of significance were identified.

.2 Walkdowns of Radiological Environmental Monitoring Stations and Meteorological Tower

a. Inspection Scope

The inspectors conducted a walkdown of selected environmental air, water, vegetation, and soil sampling stations and thermoluminescent dosimeters locations to verify that the locations were consistent with their descriptions in the RETS/ODCM and to evaluate the equipment material condition and operability. The inspectors also conducted a walkdown of the primary and backup meteorological monitoring site to validate that sensors were adequately positioned and operable. The inspectors reviewed the CY 2001 and 2002 Annual Environmental Monitoring Reports to evaluate the onsite meteorological monitoring program's data recovery rates, routine calibration and maintenance activities, and non-scheduled maintenance activities. The review was conducted to verify that the meteorological instrumentation was operable and was calibrated and maintained in accordance with licensee procedures. The inspectors also reviewed indications of wind speed, wind direction, and atmospheric stability measurements to verify that the indications were available in the Control Room and that the instrument indications were operable.

b. Findings

No findings of significance were identified.

.3 Review of REMP Sample Collection and Analysis

a. Inspection Scope

The inspectors accompanied the licensee REMP technician to observe the collection and preparation of air particulate filters and iodine cartridge samples to verify that representative samples were being collected in accordance with procedures and the RETS/ODCM. Additionally, environmental monitoring thermoluminescent dosimeters were collected (and replaced) in accordance with the documents cited previously. The inspectors observed the technician perform air sampler field check maintenance to verify that the air samplers were functioning in accordance with procedures. Selected air sampler calibration and maintenance records for CY 2001 and 2002 were reviewed to verify that the equipment was being maintained as required. The environmental sample collection program was compared with the RETS/ODCM to verify that samples were representative of the licensee's release pathways. Additionally, the inspectors reviewed results of the vendor Interlaboratory Comparison Program to verify that the vendor was capable of making adequate radio-chemical measurements.

b. Findings

No findings of significance were identified.

.4 Unrestricted Release of Material From the Radiologically Controlled Area

a. Inspection Scope

The inspectors evaluated the licensee's controls, procedures, and practices for the unrestricted release of material from radiologically controlled areas and conducted reviews to verify that: (1) radiation monitoring instrumentation used to perform surveys for unrestricted release of materials was appropriate; (2) instrument sensitivities were consistent with NRC guidance contained in Inspection and Enforcement (IE) Circular 81-07 and Health Physics Positions in NUREG/CR-5569 for both surface contaminated and volumetrically contaminated materials; (3) criteria for survey and release conformed to NRC requirements; (4) licensee procedures were technically sound and provided clear guidance for survey methodologies; and (5) radiation protection staff adequately implemented station procedures.

The inspectors reviewed the circumstances of the February 26, 2003 discovery of eddy current test equipment with a measurable amount of licensed radioactive material (.3 nCi of Co-60 and lesser quantities of Mn-54) which was found upon subsequent evaluation and survey at Point Beach station [the next location of use of this equipment]). The test gear had been moved outside the Owner Controlled Area (OCA), and transported from Iowa to Wisconsin by a contractor/vendor. Specifically, the inspectors reviewed the licensee's initial Condition Report (CR), investigative documents (i.e., an Apparent Cause Evaluation), and survey data. The incident was discussed with the radiation protection manager and several members of the RP staff.

b. Findings

Introduction: A self-revealing Green finding and an associated Non-Cited Violation (NCV) were identified for the failure to maintain control of licensed radioactive material in an unrestricted area that was not in storage (i.e., eddy current test equipment with a measurable amount of licensed radioactive material [.3 nCi of Co-60 and Mn-54] which was found upon subsequent evaluation and survey at Point Beach station [the next location of use of this equipment]).

Description: On February 7, 2003 eddy current test equipment used during the February 2003 forced outage was released from the Radiologically Controlled Area (RCA) for unrestricted use following use in the main condenser water box. Subsequent evaluation and survey at Point Beach Nuclear Plant (PBNP) station (next location of use of this equipment) identified detectable radioactive material. This radioactive material was found on a filter inside of the test equipment; the quantity of radioactive material was calculated to be 0.3 nCi.

During the shutdown, due to river water intrusion to the hotwell, an Eddy Current Tester was brought on site and used by a vendor in the river water side of the condenser water boxes. The equipment alarmed the gross gamma counter (at the RCA access point) upon removal from the plant afer the work was completed. The equipment was isotopically analyzed, and Potassium-40 (K-40) (i.e., a naturally occurring isotope) as well as one peak of Cobalt-60 (Co-60), were identified. The Radiation Protection Manager and Instrument Engineer evaluated the report in accordance with HPP 3109.65 "Operation of the Nuclear Data HPGE Counting System", and made a decision to release the equipment from the RCA. The test equipment was then transported, by the vendor, to their warehouse in Two Rivers, Wisconsin.

This event was self-revealing when on February 26, 2003, after concerns about the appropriateness of the release method were raised, the Point Beach Health Physics (HP) department was contacted by the Duane Arnold Energy Center (DAEC) Radiation Protection Manager to obtain the equipment and to get another isotopic analysis performed. The equipment was brought to Point Beach by the vendor. Upon the equipment's delivery to Point Beach, the external surfaces of the equipment was surveyed (i.e., direct frisk and/or wipe tests) and no detectable activity was found. However, after performing several isotopic analyses (i.e., using multiple counting geometries for the equipment) the Point Beach HP department identified both peaks of Co-60 on an inlet filter, which confirmed that the equipment contained radioactive material (i.e., Co-60 and lesser amounts of Mn-54). After disassembly of the unit, the Point Beach Health Physics (HP) department found 20-40 net counts per minute on an internal air filter in the unit. No other detectable activity was found on or in the unit. The equipment was transferred to the Point Beach RCA for temporary storage. The DAEC and PBNP Radiation Protection Managers were notified, as well as the vendor who owns the equipment (Anatech).

The shipping container (internal and external) and the transfer vehicle at the vendor's facility were surveyed, with no detectable activity found. Additionally, the transfer vehicle used by a Point Beach employee was surveyed, with no detectable activity found.

The licensee's apparent cause evaluation found that this event was caused by human error on the part of the Radiation Protection manger. This error involved reaching a non-conservative conclusion during an incomplete evaluation of the presence of radioactive material on the item. Contributing factors cited were a failure to investigate the equipment's history of use, use of longer count times (i.e., for isotopic analyses), and consideration of differing counting geometries for those same isotopic analyses mentioned earlier. Additionally, the licensee cited a need to evaluate the addition of procedural requirements to perform surveys of materials/equipment that had been used at other nuclear facilities to verify that they are "clean" (i.e., prior to use at the DAEC).

Analysis: The inspectors determined that the issue was associated with the "Program and Process" attribute of the Public Radiation Safety Cornerstone and affected the cornerstone objective in ensuring adequate protection of public health and safety from exposure to radioactive materials released into the public domain. Also, the issue involved an occurrence in the licensee's radioactive material control program that is contrary to both NRC regulations and licensee procedures. Therefore, the issue was more than minor and represents a finding which was evaluated using the significance determination process (SDP) for the Public Radiation Safety Cornerstone.

The inspectors determined that the licensee failed to prevent the inadvertent release and/or loss of control of licensed radioactive material to an unrestricted area that could cause an actual or credible radiation dose to a member of the public. As such, the inspectors determined, utilizing Manual Chapter 0609, Appendix D, "Public Radiation Safety SDP," that the finding involved radioactive material control, but transportation was not involved.

However, the public radiation exposure was not greater than 0.005 rem (5 millirem) and the licensee did not have more than 5 radioactive material control occurrences (in the previous 8 quarters). Consequently, the inspectors concluded that the SDP assessment for this finding was of very low safety significance (Green).

Enforcement: Title 10 CFR 20.1802 requires that the licensee shall control and maintain constant surveillance of licensed material that is in a controlled or unrestricted area and that is not in storage. On February 26, 2003, the licensee failed to maintain control of licensed radioactive material in an unrestricted area that was not in storage (i.e., eddy current test equipment with a measurable amount of licensed radioactive material [.3 nCi of Co-60 and lesser quantities of Mn-54] which was found upon subsequent evaluation and survey at Point Beach station [the next location of use of this equipment]). This failure constitutes a violation of 10 CFR 20.1802. However, because the licensee documented this issue in its corrective action program (CA027027) and because the violation is of very low safety significance, it is being treated as an NCV (NCV 50 331/03-04-06).

.5 Identification and Resolution of Problems

a. Inspection Scope

The inspectors reviewed condition reports, the results of the licensee's REMP self-assessment performed during the second quarter of CY 2002, and Nuclear Oversight observation reports addressing the REMP to determine if problems were being identified and entered into the corrective action program for timely resolution. The inspectors also reviewed the licensee's pre-inspection readiness evaluation of the REMP, which evaluated the current state of the program and the completion status of the previous self-assessment items. The inspector also reviewed the licensee's overall management of the REMP, including attention to details of the sampling program and the vendor laboratory, in order to evaluate the effectiveness of the REMP in collection and analysis of samples for the detection of offsite radiological contamination.

b. Findings

No findings of significance were identified.

## 3. SAFEGUARDS

**Cornerstone: Physical Protection**

3PP2 Access Control (Identification, Authorization and Search of Personnel, Packages, and Vehicles) (IP 71130.02)

a. Inspection Scope

The inspectors reviewed the licensee's protected area access control equipment testing and maintenance procedures to determine if testing was performance-based, challenged the detection capabilities of the equipment, and was in accordance with security plan requirements. The inspectors observed licensee testing of all access control equipment to determine if testing and maintenance practices were performance based. On two occasions, during peak ingress periods, the inspector observed in-processing search of personnel, packages, and vehicles to determine if search practices were conducted in accordance with regulatory requirements, and that sufficient security force staffing was available for the search functions.

The inspectors reviewed a sample of licensee security logged events and other security documents for identification and resolution of problems. In addition, the inspectors interviewed security managers to evaluate their knowledge and use of the licensee's corrective action system.

b. Findings

No findings of significance were identified.

3PP3 Response to Contingency Events (71130.03)

a. Inspection Scope

The inspectors reviewed the current Plant Protective Strategy. The inspectors also conducted a walk down of the protected area boundary and alarm system and observed testing of selected protected area alarm zones. The inspectors reviewed licensee drill and exercise critiques pertaining to response to security contingency events.

The inspectors reviewed a sample of licensee security logged events for identification and resolution of problems. In addition, the inspectors interviewed security managers to evaluate their knowledge and use of the licensee's corrective action system.

b. Findings

No findings of significance were identified.

3PP4 Security Plan Changes (71130.04)

a. Inspection Scope

The inspectors reviewed Revision 45 (dated February 10, 2003) to the Duane Arnold Energy Center Physical Security Plan. The review was conducted to verify that the changes did not decrease the effectiveness of the security plan. The revision was submitted in accordance with 10 CFR 50.54(p).

b. Findings

No findings of significance were identified.

## 4. OTHER ACTIVITIES

4OA1 Performance Indicator Verification (71151)

**Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity, and Physical Protection**

.1 Reactor Safety Strategic Area

a. Inspection Scope

The inspectors sampled licensee submittals for the performance indicators (PIs) listed below for the period of January 2002 to March 2003. To verify the accuracy of the PI data reported during that period, PI definitions and guidance contained in the applicable revision of Nuclear Energy Institute Document 99-02, "Regulatory Assessment Performance Indicator Guideline" were used. The following PIs were reviewed:

During the week of May 3, 2003

- Unplanned SCRAMS;
- SCRAMS with a loss of Normal Heat Sink; and
- Unplanned Power Changes.

In addition, the inspectors reviewed Licensee Event Reports (LERs), licensee memoranda, plant logs, and other documents to determine whether the licensee adequately identified the reported data.

b. Findings

No findings of significance were identified.

.2 Safeguards Strategic Area

a. Inspection Scope

The inspectors sampled licensee submittals for the performance indicators (PIs) listed below for the period of April 2002 to March 2003. To verify the accuracy of the PI data reported during that period, PI definitions and guidance contained in the applicable revision of Nuclear Energy Institute Document 99-02, "Regulatory Assessment Performance Indicator Guideline" were used. The following Pis were reviewed:

- Fitness-For-Duty;
- Personnel Screening Program; and
- Protected Area Security Equipment.

In addition, the inspectors sampled plant reports related to security events and other applicable security records.

b. Findings

No findings of significance were identified.

4OA2 Identification and Resolution of Problems (71152)

.1 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

No findings of significance were identified.

.2 Control of Temporary Modifications

Introduction: The inspectors observed an increase in problems related to temporary modifications. The inspectors noticed this trend during routine daily reviews of CAP reports. Accordingly, the inspectors selected the licensee's temporary modification process for a more detailed review with respect to problem identification and resolution. During the week of May 5, 2003, the inspectors searched the licensee's CAP database for the prior 12 month period for problems with the temporary modification program, and found 16 such examples. During the week of June 9, 2003, the inspectors interviewed the Systems Engineering Manager, who owns the corrective action program for the area of Engineering Department Human Performance.

a. Effectiveness of Problem Identification

(1) Inspection Scope

The inspectors evaluated whether the licensee's identification of each 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 16 CAPs found in the database search. With regard to the recent increase in temporary modification problems, the licensee attributed this to the substantially increased workload during the refueling outage. The inspectors noted that four problems with temporary modifications were related to outage work, although one was a minor documentation error. These identified problems included: a temporary modification package without proper instructions for disconnecting an electronic cable connector which resulted in the wrong cable being disconnected; a temporary modification package which did not include a procedurally-required marked up engineering drawing; and a removal of a temporary modification tag without the Shift Superintendent's approval. The inspectors also noted that the licensee's Nuclear Oversight Department conducted an audit of the temporary modifications during the Refueling Outage, which identified the documentation errors. The inspectors also observed that the licensee conducted a fact-finding meeting for the unauthorized temporary modification tag removal.

As part of the interview with Systems Engineering Manager, the inspectors discussed with him the details of OTH014485, "Perform Trend Review of Engineering Department Human Performance Action Requests." During this interview, the inspectors ascertained that the licensee is taking active measures to promptly and correctly identify problems in the temporary modification process, so as to ensure that data for trend analysis is accurate. The inspectors further determined that the Systems Engineering Manager was cognizant of the causes for any temporary modification problem trends and associated corrective actions. The inspectors performed these reviews to ensure compliance with 10 CFR 50 Appendix B, Criterion XVI "Corrective Action" requirements. The specific corrective action documents that were reviewed by the inspectors are listed in the attachment to this report.

4OA3 Event Follow-up (71153)

.1 (Closed) Licensee Event Report (LER) 50-331/03-02: “Inadequate Procedure Leads to Failure to Remove Key from Mode Switch when Locked”

On February 11, 2003, the licensee discovered that they had violated Technical Specifications 3.10.4, “Single Control Rod Withdrawal - Cold Shutdown” and 3.9.2, “Refuel Position One-Rod-Out Interlock” during Control Rod Drive exercises. They had exercised 23 control rod drives without the key being removed from the Reactor Mode Switch while it was in the Refuel Position. This is in contrast with the Technical Specification bases for Surveillance Requirement 3.9.2.1 which defines the Locked in the Refuel Position as the switch in the Refuel Position with the key removed. The removal of the key from the mode switch is an additional administrative control, since it does not provide additional interlocks. The licensee attributed the event to an inadequate procedure, since the procedure only addressed the mode switch being in the refuel position and the key removal was not included. An additional contributor to the event was the lack of familiarity with the bases for SR 3.9.2.1. Corrective actions included initiating Procedure Work Requests to revise identified procedures by adding specific requirements to remove the key from the mode switch. The safety significance of this event was minimal, since the failure to remove the key was an administrative control error, so it 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 LER was reviewed by the inspectors and no findings of significance were identified. The licensee documented the issue in CAP 025551.

.2 (Closed) LER 50-331/03-03: “Reactor Mode Change with a LCO in effect in Violation of Technical Specification 3.0.4”

On April 20, 2003, the licensee discovered that, they had violated Technical Specification 3.0.4, “Mode of applicability when an LCO is not meet” by changing from Mode 2 to Mode 1 with LCO 3.5.1 condition B, “One low pressure ECCS system inoperable” in effect for “B” RHR in suppression pool cooling. “B” RHR had been placed in suppression pool cooling for post maintenance testing of RCIC. The licensee attributed the event to the confusion about the definition of the term “Mode of Applicability” and the failure to adequately communicate the decision of the applicability of the Technical Specification information throughout operations. In addition, the licensee had identified in this LER that a previous event of violating Technical Specification 3.0.4 had occurred on September 2, 2002, but had not been reported to the NRC in accordance with 10 CFR 50.73(a)(2)(i)(B). Corrective actions included placing procedural improvements in place to prevent future events and providing training on the event to all operations department personnel. The inspectors reviewed the LER and associated documents to verify that the cause was identified and that corrective actions proposed by the licensee were reasonable and appropriate. The issue is greater than minor since it effected the mitigating system cornerstone objective of configuration control in an operating equipment lineup, which required the RHR system to be lined up for injection. The issue is of low safety significance since RHR would automatically realign to an injection mode if called upon. A licensee identified violation associated with this issue is documented in Section 4OA7 of this report. The licensee documented the issue in CAP 027106.

.3 (Closed) LER 2003-S01-00: "Unattended Safeguards Information Outside of the Protected Area Caused by Personal Error," May 15, 2003.

10 CFR 73.21 (b)(1)(i) identifies the composite physical security plan for a nuclear site as safeguards information. 10 CFR 73.21(d)(2) requires safeguards information, when unattended, to be stored in a locked security storage container.

Contrary to this requirement, on April 17, 2003, a copy of the complete security plan, to include the security contingency plan (Appendix C) and the security force training and qualification plan (Appendix B) was left unattended in an open office space in the Plant Support Center within the owner controlled area (OCA) for a period of approximately 12 minutes. The security plan was being routed for review of changes made prior to being sent to the NRC.

Although the OCA is outside of the protected area, heightened security measures were in effect because of security advisories and an Order issued since September 11, 2001. Access to the OCA was limited to pre-authorized visitors and employees with badges. The incident, when discovered, was reported to site security who initiated an investigation. The incident was also reported to the NRC. Adequate immediate corrective actions (adequately protecting the security plan) were taken. Additional corrective actions included counseling the individual involved, and changing the process for routing and review of security plan changes.

The inspectors reviewed the LER and verified that the cause was identified and that corrective actions proposed by the licensee were reasonable and appropriate. The issue is greater than minor because the security plan identified specific vital area locations. The issue is of low safety significance because of the short duration of time the security plan was not adequately protected and the increased security measure in place within the OCA at the time of the occurrence. A licensee identified violation associated with this issue is documented in Section 4OA7 of this report. The licensee documented the issue in CAP 27596.

## 4OA4 Cross-Cutting Issues

.1 A finding described in Section 1R19 of this report had, as its primary cause, a human performance deficiency, in that, the licensee, failed to properly connect the pilot solenoid wires for PSV 4405 during valve replacement. In addition it was also determined that the two independent checks, which consisted of a peer check and a quality control verification check, also failed to recognize the connection error.

.2 A finding described in Section 1R23 of this report had, as its primary cause, a human performance deficiency, in that, the licensee, failed to perform procedures steps described in TMOD 03-041 to re-install the recorder in the 1D15 120 VAC Instrument Invertor, thereby blowing a fuse.

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 June 30, 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:

- Inservice Inspection with Mr. J. Bjorseth on April 3, 2003.
- Access Control to Radiologically Significant Areas and ALARA Planning and Control with Mr. J. Bjorseth on April 4, 2003
- Safeguards Inspection with Mr. J. Bjorseth on May 1, 2003.
- REMP with Mr. J. Bjorseth on June 27, 2003.
- Telephone discussion with the Radiation Protection Manager (Acting) on July 10, 2003.
- Maintenance Rule with Mr. J. Bjorseth on June 27, 2003.

4OA7 Licensee-Identified Violations

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

**Cornerstone: Mitigating Systems**

.1 As discussed in Section 4OA3.2 of this report, Technical Specification 3.0.4 requires that entries into a mode shall not be made when an LCO is not met unless it is for an unlimited period of time. Contrary to these requirements, the licensee changed modes from Mode 2 to Mode 1 with LCO 3.5.1 condition B, “One low pressure ECCS system inoperable” in effect for RHR in suppression pool cooling on April 20, 2003. This issue was entered into the licensee’s corrective action program as CAP 027106. Because RHR would automatically realign to an injection mode if called upon, this violation is not more than very low safety significance, and is being treated as a Non-Cited Violation.

**Cornerstone: Occupational Radiation Safety**

.2 Technical Specification 5.4.1 (a) requires, in part, that the licensee establish and implement procedures covering activities recommended in Regulatory Guide 1.33 (Revision 2), Appendix A, February 1978. The Regulatory Guide recommended procedures include radiation protection procedures for access control. The licensee’s radiation protection procedures require that an area with radiation levels of 1000 mrem/hour or greater be locked and posted as a locked high radiation area (LHRA). Additionally, these radiation protection procedures require that all LHRA door and gate keys shall be maintained under administrative control of the shift supervisor, radiation protection manager, or his or her designee. This did not occur on

March 17, 2003, when a LHRA key for steam areas of the Turbine Building was left unattended for a brief period of time in a common office area inside the plant's Administration Building. The problem is described in AR No. 026175. No unqualified workers used the key to enter the steam areas of the Turbine Building while it was improperly controlled. This fact, coupled with the duration of the problem, precluded a substantial potential for an overexposure. Thus, the issue was determined to be of very low safety significance. Consequently, it is being treated as a NCV.

**Cornerstone: Public Radiation Safety**

.3 10 CFR 50.59(c)(1) allows the licensee to make changes in the facility as described in the UFSAR without obtaining a license amendment pursuant to 10 CFR 50.90 if the changes in the facility meet specified criteria. 10 CFR 50.59(d)(1) requires the licensee to maintain records of changes including a written evaluation which provides the bases for the determination that the change does not require a license amendment. Contrary to these requirements, during February 2001, the licensee changed the design of the facility by removing sections of the North and South Turbine Building Bioshield walls as described in the UFSAR in Section 10.2.4 without performing a written evaluation to ensure that a license amendment was not required. The bioshield walls provide additional shielding to minimize annual site boundary doses. The problem was discovered by the licensee on April 2, 2003 and the walls were reinstalled on April 29, 2003. This issue was entered into the licensee's corrective action program as CAP 026642. Because the dose rates with the bioshield walls removed did not exceed those described in the UFSAR Section 10.2.4 this violation is not more than very low safety significance; therefore, this violation of 10 CFR 50.59(c)(1) and 10 CFR0.59(d)(1) was categorized as a Severity Level IV violation. This Severity Level IV violation is being treated as a NCV, consistent with Section VI.A.1 of the NRC Enforcement Policy.

**Cornerstone: Physical Protection**

As discussed in Section 4OA3.2 of this report, 10 CFR 73.21(d)(2) requires safeguards information, when unattended, to be stored in a locked security storage container. Contrary to this requirement, on April 17, 2003, a copy of the complete security plan was left unattended in an open office space in the Plant Support Center within the owner controlled area (OCA) for a period of approximately 12 minutes. This issue was entered into the licensee's corrective action program as CAP 027596. Because of the short duration of the incident and security measures in place for entry into the OCA, this violation is not more than very low safety significance, and is being treated as a Non-Cited Violation.

ATTACHMENT: SUPPLEMENTAL INFORMATION

## KEY POINTS OF CONTACT

Licensee

M. Peifer, Site Vice-President Nuclear
J. Bjorseth, Plant Manager
C. Bleau, Regulatory Assurance
S. Catron, Licensing Manager
D. Curtland, Director Engineering
T. Evans, Operations Manager
S. Funk, Radiological Effluents Coordinator/Manager
P. Hansen, Work Control Manager
B. Kindred, Security Manager
C. Kress, Training Manger
S. Nelson, Manager, Radiation Protection
J. Probst, Maintenance Rule Coordinator
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

| | | |
|---|---|---|
| 50-331/2003-004-01 | NCV | Inadequate Procedure For Surface Examination of Code Components (Section 1R08.b.(1)) |
| 50-331/2003-004-02 | NCV | No Procedure To Implement Examination Of Welds Subject To Crevice Corrosion (Section 1R08.b.(2)) |
| 50-331/2003-004-03 | NCV | Inadequate procedure for post maintenance testing of PSV 4405 (Section 1R19) |
| 50-331/2003-004-04 | NCV | Inadequate procedure to perform Primary Containment Closeout (Section 1R20) |
| 50-331/2003-004-05 | NCV | Failure to follow the temporary modification procedure for 1D15 (Section 1R23) |
| 50-331/03-04-06 | NCV | Failure to maintain control of licensed radioactive material in an unrestricted area and that was not in storage. |

Closed

| | | |
|---|---|---|
| 50-331/2003-004-01 | NCV | Inadequate Procedure For Surface Examination of Code Components (Section 1R08.b.(1)) |

| | | |
|---|---|---|
| 50-331/2003-004-02 | NCV | No Procedure To Implement Examination Of Welds Subject To Crevice Corrosion (Section 1R08.b.(2)) |
| 50-331/2003-004-03 | NCV | Inadequate procedure for post maintenance testing of PSV 4405 (Section 1R19) |
| 50-331/2003-004-04 | NCV | Inadequate procedure to perform Primary Containment Closeout (Section 1R20) |
| 50-331/2003-004-05 | NCV | Failure to follow the temporary modification procedure for 1D15 (Section 1R23) |
| 50-331/2003-002 | LER | Inadequate Procedure Leads to Failure to Remove Key from Mode Switch when Locked in Refuel Position During Control Rod Movement as Required by Technical Specifications |
| 50-331/2003-003 | LER | Reactor Mode Change with a LCO in effect in Violation of Technical Specification 3.0.4 |
| 50-331/03-04-06 | NCV | Failure to maintain control of licensed radioactive material in an unrestricted area and that was not in storage. |
| 50-331/2003-S01-00 | LER | Unattended Safeguards Information Outside of the Protected Area Caused by Personal Error |

## 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 |
| AOV | Air Operated Valve |
| AR | Action Request |
| ASME | American Society of Mechanical Engineers |
| 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 |
| CR | Condition Report |

| | |
|---|---|
| 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 |
| HP | Health Physics |
| 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 |
| MPFF | Maintenance Preventable Functional Failure |
| MSIV | Main Steam Isolation Valve |
| Mwth | Megawatts Thermal |
| nCi | Nano-curies |
| NCV | Non-Cited Violation |
| NDE | Nondestructive Examination |
| NFPA | National Fire Protection Association |
| NRC | Nuclear Regulatory Commission |
| OCA | Owner Controlled Area |
| OI | Operating Instruction |
| OS | Occupational Radiation Safety |
| OWA | Operator Work Arounds |
| P&IDs | Piping and Instrumentation Drawings |

| | |
|---|---|
| PARS | Public Availability Records |
| PBNP | Point Beach Nuclear Plant |
| 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 |
| PT | Dye Penetrant Test |
| PTAT | Plant Transient Assessment Tree |
| Radwaste | Radioactive Waste |
| RB | Reactor Building |
| RCA | Radiologically Controlled Area |
| RCIC | Reactor Core Isolation Cooling |
| REMP | Radiological Environmental Monitoring Program |
| RETS/ODCM | Radiological Environmental Technical Specifications/Offsite Dose Calculation Manual |
| RFO | Refueling Outage |
| 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 |
| UT | Ultrasonic Test |
| VHRA | Very High Radiation Area |
| VDC | Volts Direct Current |
| VOTES | Valve Operation Test and Evaluation System |

## LIST OF DOCUMENTS REVIEWED

1R01 Adverse Weather

Abnormal Operating Procedure (AOP) 903; Tornado; Revision 12
Emergency Plan Implementing Procedure (EPIP) 1.1; Determination of Emergency Classification; Revision 19
EPIP 1.2; Notification; Revision 28
EPIP 1.3; Plant Assembly and Site Evacuation; Revision 9
Operating Instruction (OI) 324; Standby Diesel Generator System; Revision 58
Refueling Procedure (RFP) 403; Core Alterations; Revision 14
IPOI 6; Cold Weather Operations; Revision 26
Operating Instruction (OI) 724; Reactor Building Heating Ventilation and Air Conditioning (HVAC) System; Revision 35
OI 711; Pump House HVAC System; Revision 6
OI 711A1; Pump House HVAC System Electrical Lineup; Revision 0
OI 710; Intake Structure HVAC System; Revision 9
OI 710A1; Intake Structure HVAC System Electrical Lineup; Revision 2
OI 698; Main Generator System; Revision 42
CAP 027491; Reactor Building Heat Pumps not lined up for Summer Operation; May 20, 2003

1R04 Equipment Alignment

OI 150A1; RCIC System Electrical Lineup; Revision 0
OI 150A2; RCIC System Valve Lineup and Checklist; Revision 4
OI 150A4; RCIC System Control Panel Lineup; Revision 1
OI 255A1; Control Rod Drive System Electrical Lineup; Revision 1
OI 255A2; Control Rod Drive System Valve Lineup and Checklist; Revision 1
OI 730A1; Control Building HVAC System Electrical Lineup; Revision 1
OI 730A2; Control Building Ventilation Compressed Air System Valve Lineup; Revision 3
OI 730A2; Control Building Ventilation System Valve Lineup; Revision 4
OI 730A2; Control Building HVAC System Control Panel Lineup; Revision 3
OI 151A1; A Core Spray System Electrical Lineup; Revision 2
OI 151A2; A Core Spray System Valve Lineup and Checklist; Revision 1
OI 151A6; Core Spray System Control Panel Lineup; Revision 1

1R05 Fire Protection

Fire Plan; Volume II - Fire Brigade Organization; Revision 32
AFP 5; South Control Rod Drive Module Area; Revision 22
AFP 14; Reactor Feed Pump Area; Revision 22
AFP 15; Lower Switchgear Room; Revision 22
AFP 17; Condenser Bay, Heater Bay, and Steam Tunnel; Revision 22
AFP 6; RHR Valve Room; Revision 22
AFP 7;Laydown Area, Corridor and Waste Tank Area, and Spent Resin Tank Room; Revision 22
AFP 8; Standby Gas Treatment System; Revision 22
AFP 9; RBCCW Heat Exchanger Area, Equipment Hatch Area, and Jungle Room; Revision 23
AFP 4; North Control Rod Drive Area; Revision 23

1R06 Flood Protection Measures

Individual Plant Examination Section 3.3.6;Internal Flooding Analysis; November 1992
AOP 902; Flood; Revision 19
AR 30421; NRC Information Notice (IN) 2002-12, Submerged Safety-related Electrical Cables; March 29, 2002

1R07 Heat Sink Performance

CWO 1119092; Clean Coils and Inspect Unit; April 2, 2002
Thermal performance Analysis of RHR Heat Exchangers; February 12, 2003
CWO 119093; Perform Eddy Current Testing; April 6, 2003

1R08 Inservice Inspection Activities

Audit

NG-03-0120; Fourth Quarter 2002 Nuclear Oversight Assessment Report; dated February 7, 2003

Corrective Action Process Documents

CAP 025414; Under Deposit Corrosion In RHRSW For RHR Crosstie Piping; dated February 4, 2003
CAP 025375; Pipe Thinning On RHRSW Elbow Less Than ASME Thickness Requirement; dated January 31, 2003
CAP 026385; Evaluate The Adequacy Of The Current Containment Testing Program Schedule; dated March 26, 2003
AR 17687; Indication Indicative Of IGSCC Found On Recirc Nozzle N2E Weld RRF-F002; dated November 10, 1999
AR 17482; Indication Indicative Of IGSCC Found On Recirc Nozzle N2E Weld RRF-F002; dated November 11, 1999
AR 31257; Track Failure Analysis Of 1E244; dated June 6, 2002
AR 33771; Dissimilar Metal Welds And Assure That Each Configuration Is Covered By a PDI Test Sample; dated December 10, 2002
AR 33900; Some Systems Are Not Identified With Their Potential Damage Mechanisms; dated December 13, 2002
AR 33901; Incorporate RI-ISI Piping Program Into The ASME Section XI Administrative Manual; dated December 13, 2002
AR 33902; RI-ISI Program Review- Identify Second and Third Period Examinations Per NRC Comments; dated December 13, 2002
AR 26787; Revise NG-1427 To Add Mode And Style Of Transducer To Data Sheet; dated July 12, 2001
AR 27051; Structural Integrity of ESW Piping; dated August 15, 2001
AR 26968; After Cooler Leak At Welded Joint Near Inlet; dated August 3, 2001
AR 25397; Indication On Vessel Head Dollar Plate Weld; dated April 24, 2001

Corrective Action Documents Issued As a Result of Inspection Activities

AR OTH027277; Pipe Thinning On RHRSW Elbow Less Than ASME Requirements; dated April 3, 2003

CAP 026610; Perform Demonstration For Temperature Range For ACP 1211.3; dated April 1, 2003
CAP 026607; Revise NDE Procedures To Incorporate EPRI RI ISI Document; dated April 1, 2003
CAP 026608; Revise NDE Procedures To Incorporate RI ISI Regarding Socket Welds; dated April 2, 2003
CAP 026649; RI ISI Implementation Did Not Revise All Examination Procedures; dated April 2, 2003
CAP026580; Revise ACP 1211.3 To Reflect ASME Section V, Article 6 Illumination Requirement; dated March 31, 2003
CAP 026650; Flaw Evaluation For Dollar Weld Was Not Submitted To NRC; dated April 2, 2003
CAP 026651; ISI Schedule Listed VT-1/EVT-1 Exam Rather Than VT-3 For B-N-2 Welds; dated April 2, 2003
CE000662; Weld Repair To Spare ESW Pump May Not Meet ASME Repair Criteria; dated April 2, 2003

Code Replacement/Repair Activities

Purchase Order 08729; Repair Of Wasted Area In Column of Spare ESW Pump; dated August 15, 2002
CWO A60263; Replaced RHR SW Elbow; dated February 6, 2003

Nondestructive Examination Reports

PT Examination Core Spray Piping Weld CSB-F002; dated April 28, 2001
PT Examination Core Spray Piping Weld CSB-F004; dated April 28, 2001
UT Examination Recirculation Inlet Nozzle N2F weld RRF-F002; dated December 16, 1999
UT Examination Recirculation Inlet Nozzle N2D weld RRD-F002; dated November 18, 1999
UT Examination Recirculation Inlet Nozzle N2F weld RRF-F002; dated May 5, 2001
UT Examination of Reactor Vessel Head Dollar Weld; dated May 7, 2001
UT Examination Recirculation elbow to pipe weld RRE-J005; dated April 1, 2003
UT Examination Recirculation elbow to pipe weld CUB-J005; dated March 31, 2003

Procedures

2162.1; Nondestructive Examination Procedure- Liquid Penetrant PT-1; Revision 2
ACP 1211.7; NDE Procedure For VT-1 Visual Examinations; Revision 3.
ACP 1211.3; NDE Procedure For Liquid Penetrant (Visible Dye & Water Washable) PT-1; Revision 6
ACP 1211.5; Nondestructive Examination Procedure Magnetic Particle (Dry or Wet Visible) MT-1; Revision 5
ACP 1211.10; Nondestructive Examination Procedure Visual Examination of Component Supports VT-3; Revision 6
ACP 1211.19; Ultrasonic Examination of Ferritic Piping Welds; Revision 3.
ACP 1211.20; Ultrasonic Examination of Austenitic Piping Welds; Revision 4
ACP 1211.36; Reactor Pressure Vessel Inspection Procedure; Revision 0

Miscellaneous Documents

CMTR; Spoolarc 65-Heat No. 065671- 3/32 Inch Diameter X 36 Inch Straight Length, One Flag; dated August 1, 2001
WPS JCP-GTA-2; Revision 3
PQR JCP-GTA-2-PQR-1; Revision 0
NIS 2; Bottom Column (ESW Class 3); dated October 14, 2002
NIS 2; GBC-1-E-22- 45 Degree (RHR SW Class 3); dated March 8, 2003
CMTR; SMLS BW 45 Degree Elbow SH234; dated February 15, 2002
WPS P1-AT-Lh; Revision 8
PQR PrQR-W-2; dated May 16, 1975
PQR PrQR-W-6; dated May 16, 1975
PQR PrQR-W-10; dated May 16, 1975
Letter to Curt Bock (Duane Arnold Energy Center) from Research and Product Development Candu Technology Development; dated February 7, 2003
Letter to Mark Huting (Nuclear Management Company) from G. J Lincina (gbordon@structural integrity.com); dated February 7, 2003
GE Nuclear Technology Document; An Evaluation Of The Effect Of A Water Chemistry Transient On The Duane Arnold Control Rod Drive System; dated February 10, 2003
NDE Procedure Qualification For Procedure 2162.1 Nondestructive Examination Procedure- Liquid Penetrant PT-1; dated January 12, 1995

1R11 Licensed Operator Requalification Program
SEG 2003C2-1; Revision 0
EOP 1; Reactor Pressure Vessel Control; Revision 9
EOP 3; Secondary Containment Control; Revision 15
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
November/December 2002 Maintenance Rule Monitoring and Status Report, April 15, 2003
System Level Performance Criteria, Feedwater and Condensate; Revision 0
CAP 019664, Per Maintenance Rule Module 4, Section 7.B, Feedwater Being Returned to Red, October 21, 2002
AR 22738, SUS 44, Feedwater, Declared Maintenance Rule Red; October 25, 2000
AR 27059, Apparent Cause Evaluation, Additional Events to Feedwater SUS in Maintenance Rule 'Red', August 16, 2001
CAP 025538, RX Feed Pump 1P-1A Aux Lube Oil Pump Failed to Auto Start After Scram, February 11, 2003
CAP 027009, CV-1622 (Startup Feed Regulating Valve) Temporary Repairs, April 16, 2003
Performance Criteria Basis Document, Annunciators SUS 99.31, Revision 3
CWO A62805, Kaman 10 is Erratic, April 27, 2003
CWO A61785, Kaman 10 is Erratic, March 14, 2003
CWO A61596, Radiation Fluctuations Above Normal Levels, March 27, 2003

CWO A61583, RI9177 Still Spiking, Causing Upscale Alarms, March 6, 2003
CWO A61433, RI9177 High Radiation Level Alarm, February 6, 2003
CWO A59392, RI9177 Fail Upscale, July 11, 2002
Health and Status Report for HPCI, June 17, 2003
CAP 027780, “HPCI in Maintenance Rule 10 CFR 50.65(a)(1)(Red);” June 10, 2003
CAP 027762, “TRM Inoperability Versus Maintenance Rule MPFF Requirements;” June 9, 2003
AR 27529, “Leaking Plug on HPCI Steam Supply Drain Steam Trap Failed and Started to Fill HPCI Room with Steam;” September 4, 2001
AR 28272, “HPCI Declared Inoperable Due to Oil Leak on a Threaded Fitting;” October 30, 2001
List of Work Orders for Startup System 52.00 (HPCI) from 2000 to 2003
System Parameter Information for HPCI Unavailability
System Parameter Information for HPCI Functional Failures
Maintenance Rule Program Cycle 16 Periodic Report May 1998 - December 1999; August 28, 2000
Maintenance Rule Program Cycle 17 Periodic Report December 1999 - May 2001; January 11, 2002
Maintenance Rule Program Line Management Self-Assessment Report; June 2001
Results of March 2003 Self-Assessment of the Duane Arnold Energy Center (DAEC) Maintenance Rule Program (10 CFR 50.65 (a)(1) - (a)(3)); April 14, 2003
List of DAEC Maintenance Preventable Functional Failures (MPFFs), Cycle 17 (12/1/99 - 5/26/01); June 23, 2003
Cycle 17 50.65(a)(1) Reviews; June 23, 2003
NUMARC 93-01; Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants; Revision 3
Module 0; Maintenance Rule Program Overview; Revision 3
Module 1; Maintenance Rule Program Scoping; Revision 2
Module 2; Maintenance Rule Program Risk-Significance Determination; Revision 2
Module 3; Maintenance Rule Program Performance Criteria Development; Revision 3
Module 4; Maintenance Rule Program Monitoring Performance, Goal Setting, and EPIX Activities; Revision 9
Module 5; Maintenance Rule Program Preparation of Cyclic Report; Revision 5
Module 6; Maintenance Rule Program Monitoring of Structures; Revision 2
CP 1208.3; Preventive Maintenance Program; Revision 13
Performance Criteria Basis Document; Residual Heat Removal (RHR); Revision 4
Performance Criteria Basis Document; Emergency Diesel Generator (EDG); Revision 3
Performance Criteria Basis Document; Essential Service Water (ESW); Revision 2
Performance Criteria Basis Document; High Pressure Coolant Injection System (HPCI); Revision 3
DAEC Maintenance Rule Unavailability and Failure Data for RHR, EDG, ESW, and HPCI; June 23, 2003
Work Orders RHR, EDG, ESW, and HPCI (12/1/99 to 5/26/03); June 23, 2003
DAEC Preventive Maintenance Improvement Project; March 18, 2003
Expert Panel Meeting Minutes (1998 - 2003)
Maintenance Rule Criteria Values; June 23, 2003
November/December 1999 Maintenance Rule Monitoring and Status Report; June 23, 2003
January/February 2000 Maintenance Rule Monitoring and Status Report; June 23, 2003
March/April 2000 Maintenance Rule Monitoring and Status Report; June 23, 2003

May/June 2000 Maintenance Rule Monitoring and Status Report; June 23, 2003
July/August 2000 Maintenance Rule Monitoring and Status Report; June 23, 2003
September/October 2000 Maintenance Rule Monitoring and Status Report; June 23, 2003
November/December 2000 Maintenance Rule Monitoring and Status Report; June 23, 2003
January/February 2001 Maintenance Rule Monitoring and Status Report; June 23, 2003
March/April 2001 Maintenance Rule Monitoring and Status Report; June 23, 2003
May/June 2001 Maintenance Rule Monitoring and Status Report; June 23, 2003
March/April 2003 Maintenance Rule Monitoring and Status Report; June 23, 2003
List of DAEC Air Operated Valve (AOV) Failures Cycle 17 Start Through Cycle 18 End; June 25, 2003
List of Radiation Monitor Failures From May 2001 to March 2003; June 25, 2003
List of Preventative Maintenance Frequency Changes for 2000; June 26, 2003
List of Preventative Maintenance Frequency Changes for Feedwater System and HPCI; June 24, 2003
AR# 32025; 'A' Residual Heat Removal Service Water (RHRSW) Strainer High Differential Pressure (D/P) While Running 'A' and 'C' RHRSW Pumps; August 5, 2002
AR# 16019; Diesel Driven Fire Pump Inoperable During Surveillance Test Procedure NS13B004; July 21, 1999
AR# 24015; Instrument Air Samples Taken During 1999 and 2000 Exceeded the Maintenance Rule Condition Monitoring Limit for Particles Larger Than 3 Microns in Size; February 22, 2001
AR# 32065; Operation of the RHRSW System Bypassing the RHRSW Strainers; August 8, 2002
AR# 29948; Diesel Generator Transient Loading and Sequencing; February 20, 2002
AR# 27347; Review Whether Maintenance Rule Low Pressure Coolant Injection (LPCI) Unavailability Monitoring Should Capture Torus Cooling Modes Effect on LPCI; August 29, 2001
AR# 27340; Review Possible Effects of Power Uprate on Maintenance Rule Performance Criteria With Expert Panel; August 21, 2001
AR# 27341; Reactor Building Sump Maintenance Rule 10 CFR 50.59; August 29, 2001
AR#27350; Form an Engineering Team to Identify Design Basis Calculation, Modifications, and Programs that Pertain to RHR, RHRSW, ESW and EDG Systems; August 24, 2001
ORT026309; Complete AOV Trend Report for Cycle 18, February 3, 2003
ORT028420; Develop an AOV Margin Improvement Plan to Improve Calculated Margin of AOVs; January 24, 2003
Memorandum; Project Completed a Significant Amount of Work Toward Ensuring Reliable, Continued Operation of Plant AOVs; June 21, 2001
System Health Report Data for EDG, RHR, HPCI, and ESW for 2002 and 2003; June 23, 2003
Cycle 17 AR(s) for EDG, RHR, HPCI, and ESW; June 23, 2003

Corrective Action Program (CAP) Initiated as a Result of Inspection

CAP027991; Review MPFF Examples in Maintenance Rule Module 4; June 26, 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 April 26, 2003
Online Look-Ahead Agenda; Week of May 2, 2003
Online Look-Ahead Agenda; Week of May 10, 2003
Online Look-Ahead Agenda; Week of May 17, 2003
Online Look-Ahead Agenda; Week of May 24, 2003
Online Look-Ahead Agenda; Week of May 31, 2003
Online Look-Ahead Agenda; Week of June 7, 2003
Online Look-Ahead Agenda; Week of June 14, 2003
CAP 027496; ORAM-Sentinel regarding ESW and SBDG Unavailable; May 20, 2003
Memorandum NG-03-0415; Risk Review with 1D15 and “B” SBDG Out Of Service; May 30, 2003
Memorandum NG-03-0346; Risk Analysis for Week 23 Scheduled On-Line Maintenenace; May 9,2003

1R14 Personnel Performance During Nonroutine Plant Evolutions and Events

Week 9322/9323 On-line Maintenance Schedule; May 30, 2003
DAEC Plant Status Report; June 5, 2003
PWO 1123011; Remove and Replace Air Start Check Valves; June 3, 2003
PWO 1123446; Replace Solenoid Valve; June 5, 2003
PWO 1123580; VOTES Diagnostic Test On MO-1934; June 5, 2003
CWO A56889; Replace Pipe Nipple Between Header Line and AV4929D; June 5, 2003
OI 149A1; RHR System Electrical Lineup; Rev. 2
OI 149A2; “A” RHR System Valve Lineup and Checklist; Rev. 5
OI 149A6; RHR System Control Panel Lineup; Rev. 1
OI 324A4; SBDG 1G-21 System Valve Lineup and Checklist; Rev. 2
OI 324A8; SBDG 1G-21 System Control Panel Lineup; Rev. 0
OI 454A1; ESW System Electrical Lineup; Rev. 0
OI 454A2; “A” ESW System Valve Lineup and Checklist; Rev. 3
OI 454A6; ESW System Control Panel Lineup; Rev. 0

1R15 Operability Evaluations

CAP 027317; “A” Pump House HVAC failed as left calibration; May 7, 2003
CAP 027209; “A” Emergency Service Water Strainer; May 12, 2003
OPR 000226; Control Building Envelope; April 29, 2003
STP 3.7.4-04; Control Building Boundary Inoperable; Revision 0
STP 3.7.4-03; Control Room Positive Pressure Test; Revision 5
OPR 000225; “B” SBDG Normal Air Start Supply Valve; April 19, 2003
CAP 027012; 027012;Two Bolts Missing from Drywell Shielding; April 16, 2003
OPR 000222; 027012;Two Bolts Missing from Drywell Shielding; April 16, 2003

1R16 Operator Workarounds

Operations Department Instructions 004; Identification, Tracking and Resolution of Equipment issues; Revision 8
Equipment Issues Assessment Factor; May 17,2003
CAP 027105; Turning Gear Drive didn’t engage; April 20, 2003
CAP 019119; “A” Control Building Chiller tripped; February 5, 2001
OTH 020484; Prepare Modification Package for Control Building Chiller; November 9, 2001

CAP 019106; Received Multiple Division 1 “125Vdc” system trouble alarms; September 6, 2000
OTH 020729; Track the implementation of modification for noise suppression; August 21, 2002
OTH 020981; Track Replacement of SV2436; October 25, 2002
CAP 019337; Cooling Water Supply Basket Strainer High Differential Pressure; June 10, 2002
OTH 020895; Evaluate Silt Removal following Refueling Outage18; October 14, 2002
CAP 025646; Control Rod 10-23 will not withdraw; February 16, 2003
CAP 025639; Mispositioned Control Rod 10-39; February 15, 2003
CE 000368; Action Plan in Response to Smoke in the Control Room; February 18, 2003
CAP 025397; Source Range Monitors spiking; February 2, 2003

1R19 Post-Maintenance Testing
CWO 1122937; Install Spare “B” ESW Pump; March 02, 2003
CAP 026638; Weld Repair to Spare ESW Pump may not meet ASME Repair Criteria; April 2, 2003
STP NS540001; ESW System Class 3 Leakage Inspection; Revision 2
STP NS540002; ESW Operability Test; Revision 10
CWO 1119785; Overhaul Limitorque Operator for HPCI Inject Valve; March 29, 2003
CWO 1123622; VOTES Diagnostic Test; April 7, 2003
STP 3.6.1.1-06; Containment Isolation Valve Leak Tightness Test - Type C Penetrations - Feedwater System; Revision 6
CWO 1119785; Overhaul Limitorque Operator for RCIC Inject Valve; March 29, 2003
CWO A60385; Cable is brittle and damaged, Operator RCIC Inject; April 4, 2003
CWO 112363; VOTES Diagnostic Test; April 8, 2003
STP 3.6.1.1-06; Containment Isolation Valve Leak Tightness Test - Type C Penetrations - Feedwater System; Revision 6
CWO A53416; 1D4 Battery Replacement; March 31, 2003
STP 3.8.4-08; Performance Discharge Test of Battery 1D4; Revision 5
STP 3.8.4-01; Battery Pilot Cell Checks; Revision 9
STP 3.8.4-02; Battery Connected Cell Checks; Revision 7
CWO 1119819; “C” INBD MSIV Actuator Replacement; April 4, 2003
STP 3.3.1.1-18; MSIV Limit Switch Calibration and Inspection; Revision 6
STP NS830002; MSIV Trip/Closure Time Check Refueling; Revision 1
STP 3.6.1.3-03; MSIV Trip/Closure Time Check; Revision 3
STP 3.3.1.1-17; MSIV Functional Test; Revision 4
STP 3.1.4-01; Scram Insertion Time Test; Revision 12
CWO 1121020; Change out diaphragm on SCRAM outlet valve operator; March 29,2003
CWO 1119848; Remove Pilot Valve and Install Spare for Main Steam “C” ADS Relief Valve; April 15, 2003
STP 3.4.3-03; Manual Opening of the ADS and LLS Relief Valves; Revision 5
CAP 027087; Main Steam Line “C” ADS Relief Valve lost indication; April 18, 2003
ACE 001159; Main Steam Line “C” ADS Relief Valve lost indication; April 18, 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 7; Integrated Plant Operating Instruction; Revision 74

IPOI 4, Shutdown; Revision 58
Operating Instruction (OI) 149; RHR System; Revision 81
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
Temporary Modification Permit RO-06; Temporary Power for critical loads when 1A4 and 1B04 are de-energized; March 28, 2003
Temporary Modification Permit RO-07; Temporary Power to Motor Control Center 1B4327 (Fuel Pool Cooling); March 28, 2003
Temporary Modification Permit RO-14; Disable “B” Core Spray Auto Initiation; April 1, 2003
Temporary Modification Permit RO-13; Disable “A” Core Spray Auto Initiation; April 1, 2003
Temporary Modification Permit RO-12; Disable “LPCI” Auto Initiation; April 1, 2003
CAP 027208; Drywell Closeout IPOI #7 improvement; April 28,2003
CAP 027024; Final Closeout Inspection of the drywell; April 16, 2003

1R22 Surveillance Testing
STP 3.6.1.1-04; Containment Isolation Valve Leak Tightness Test - Type C Penetrations - Main Steam System; Revision 6
CAP 026693; Better Guidance needed in STP 3.6.1.1-04; April 3, 2003
STP NS490002; LPCI Inject Check Valve Full Flow Test; Revision 4
STP 3.9.1-01; Refueling Interlocks Channel Functional Testing; Revision 6
IPOI 8; Outage and Refueling Operations; Revision 30
Refueling Procedure 403; Core Alterations; Revision 14
CAP 026624; Refueling Platform monorail hoist; April 1, 2003
STP 3.7.2-01; River Water Supply System Simulated Automatic Auction Test; Revision 4
STP 3.3.5.1-29; Containment Spray Logic System Functional Test (LSFT) and RHR Timer Calibration; Revision 7
STP 3.8.1-07; LOOP-LOCA Test; Revision 9
STP 3.3.5.1-15 RHR LSFT; Revision 3

1R23 Temporary Modifications

CWO A62805; Kaman 10 is Erratic; April 27, 2003
Temporary Modification Permit Number 03-033; Lift the P4 Connector in the Kaman 10 Micro to Allow Kaman 9 to Operate While Troubleshooting is in Progress; April 29, 2003
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AR 25423; Health Physics technician received dose alarm while checking dose rates for radiography; dated February 4, 2003
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AR 26175; NSAO key ring left unattended; dated March 17, 2003
AR 26320; Contract Operating Engineer on wrong RWP to enter High Radiation Area to operate Turbine Building crane; dated March 23, 2003
AR 26359; Violation of Radiography posting; dated March 25, 2003
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ACE 001090; Apparent Cause Evaluation for Release of Contaminated Eddy Current Test Gear

ACP 114.8; Action Request Trending; Revision 3

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ESP 1.0; Environmental Sampling Procedure, Radiological Environmental Monitoring Quality Control Program; Revision 6

ESP 4.3.1.3A; Environmental Sampling Procedure, Surface Water Sampling; Revision 12

ESP 4.3.1.5; Environmental Sampling Procedure, Ground Water Sampling; Revision 16

ESP 4.3.1.6; Environmental Sampling Procedure, Bottom Sediments Sampling; Revision 11

ESP 4.3.1.8; Environmental Sampling Procedure, Vegetation Sampling; Revision 16

ESP 4.3.1.14; Environmental Sampling Procedure, Fish Sampling; Revision 9

ESP 4.3.1.15; Environmental Sampling Procedure, Milk Sampling; Revision 21

ESP 4.3.1.16; Environmental Sampling Procedure, Special Radiological Sampling; Revision 7

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