UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION IV
611 RYAN PLAZA DRIVE, SUITE 400
ARLINGTON, TEXAS 76011-4005

February 5, 2004

EA-04-012
EA-04-013

Garry L. Randolph, Senior Vice
President and Chief Nuclear Officer
Union Electric Company
P.O. Box 620
Fulton, MO 65251

SUBJECT: CALLAWAY PLANT - NRC INTEGRATED INSPECTION REPORT 05000483/2003006

Dear Mr. Randolph:

On December 31, 2003, the NRC completed an inspection at your Callaway Plant. The enclosed report documents the inspection findings which were discussed on January 6, 2004, with you and other members of your staff.

This inspection examined activities conducted under your license as they relate to safety and compliance with the Commission's rules and regulations and with the conditions of your license. Within these areas, the inspection consisted of selected examination of procedures and representative records, observations of activities, and interviews with personnel.

This report documents one finding concerning a sequencing error in an Emergency Operating Procedure that could result in increased postaccident public radiation dose. This finding has potential safety significance greater than very low significance. This finding did present an immediate safety concern. However, corrective measures have been implemented and the immediate safety hazard no longer exists. Based on the results of this inspection, the NRC has identified four issues that were evaluated under the risk significance determination process as having very low safety significance (Green). The NRC has also determined that violations are associated with three of these issues. These violations are being treated as noncited violations (NCVs), consistent with Section VI.A of the Enforcement Policy. These NCVs are described in the subject inspection report. Two licensee-identified violations which were determined to be of very low safety significance are listed in Section 4OA7 of this report. If you contest the violation or significance of these NCVs, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555-0001, with copies to the Regional Administrator, U.S. Nuclear Regulatory Commission, Region IV, 611 Ryan Plaza Drive, Suite 400, Arlington, Texas 76011; the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001; and the NRC Resident Inspector at the Callaway Plant facility.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter, its enclosure, and your response will be made available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records 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).

Should you have any questions concerning this inspection, we will be pleased to discuss them with you.

Sincerely,

/RA/

David N. Graves, Chief
Project Branch B
Division of Reactor Projects

Docket: 50-483
License: NPF-30

Enclosure:
NRC Inspection Report
05000483/2003006
w/attachment: Supplemental Information

cc w/enclosure:
Professional Nuclear Consulting, Inc.
19041 Raines Drive
Derwood, MD 20855

John O'Neill, Esq.
Shaw, Pittman, Potts & Trowbridge
2300 N. Street, N.W.
Washington, DC 20037

Mark A. Reidmeyer, Regional
Regulatory Affairs Supervisor
Regulatory Affairs
AmerenUE
P.O. Box 620
Fulton, MO 65251

Manager - Electric Department
Missouri Public Service Commission
301 W. High
P.O. Box 360
Jefferson City, MO 65102

Ronald A. Kucera, Deputy Director
for Public Policy
Department of Natural Resources
P.O. Box 176
Jefferson City, MO 65102

Rick A. Muench, President and
Chief Executive Officer
Wolf Creek Nuclear Operating Corporation
P.O. Box 411
Burlington, KS 66839

Dan I. Bolef, President
Kay Drey, Representative
Board of Directors Coalition
for the Environment
6267 Delmar Boulevard
University City, MO 63130

Chris R. Younie, Manager
Quality Assurance
AmerenUE
P.O. Box 620
Fulton, MO 65251

Jerry Uhlmann, Director
State Emergency Management Agency
P.O. Box 116
Jefferson City, MO 65102-0116

Scott Clardy, Director
Section for Environmental Public Health
P.O. Box 570
Jefferson City, MO 65102-0570

Keith D. Young, Manager
Regulatory Affairs
AmerenUE
P.O. Box 620
Fulton, MO 65251

David E. Shafer
Superintendent, Licensing
Regulatory Affairs
AmerenUE
P.O. Box 66149, MC 470
St. Louis, MO 63166-6149

Chief Technological Services Branch
National Preparedness Division
Department of Homeland Security
Emergency Preparedness & Response Directorate
FEMA Region VII
2323 Grand Boulevard, Suite 900
Kansas City, MO 64108-2670

Electronic distribution by RIV:
Regional Administrator **(BSM1)**
DRP Director **(ATH)**
DRS Director **(DDC)**
Senior Resident Inspector **(MSP)**
Branch Chief, DRP/B **(DNG)**
Senior Project Engineer, DRP/B **(RAK1)**
Staff Chief, DRP/TSS **(PHH)**
RITS Coordinator **(NBH)**
D. Jackson, OEDO RIV Coordinator **(DAJ1)**
CWY Site Secretary **(DVY)**
W. A. Maier, RSLO **(WAM)**
Dale Thatcher **(DFT)**
G. F. Sanborn, D:ACES **(GFS)**
K. D. Smith, RC **(KDS1)**
F. J. Congel, OE **(FJC)**
OE:EA File **(RidsOeMailCenter)**

ADAMS: ✖ Yes ☐ No Initials: __dng___
✖ Publicly Available ☐ Non-Publicly Available ☐ Sensitive ✖ Non-Sensitive

R:_CW\2003\CW2003-06RP-MSP.wpd ML040360357

RIV:RI:DRP/B	PE:DRP/D	SRI:DRP/B	SRI:DRP/B	C:DRS/OB
JDHanna	DEDumbacher	FLBrush	MSPeck	ATGody
T - DNGraves	**/RA/**	**E - DNGraves**	**E - DNGraves**	**/RA/**
1/29/04	1/29/04	1/30/04	1/29/04	1/ /04

DRS/EB	C:DRS/PSB	C:DRP/B		
BWHenderson	TWPruett	DNGraves		
E - DNGraves	**MPShannon for**	**/RA/**		
2/3/04	2/3/04	2/5/04		

OFFICIAL RECORD COPY T=Telephone E=E-mail F=Fax

ENCLOSURE

U.S. NUCLEAR REGULATORY COMMISSION
REGION IV

Docket:	50-483
License:	NPF-30
Report:	05000483/2003006
Licensee:	Union Electric Company
Facility:	Callaway Plant
Location:	Junction Highway CC and Highway O Fulton, Missouri
Dates:	September 21 through December 31, 2003
Inspectors:	M. S. Peck, Senior Resident Inspector J. D. Hanna, Resident Inspector D. E. Dumbacher, Project Engineer F. L. Brush, Senior Resident Inspector, Wolf Creek Generating Station R. E. Lantz, Senior Emergency Preparedness Inspector L. T. Ricketson, P.E., Senior Health Physicist B. W. Henderson, Reactor Inspector
Approved By:	D. N. Graves, Chief, Project Branch B

SUMMARY OF FINDINGS

IR 05000483/2003006; 09/21 - 12/31/03; Callaway Plant. Personnel Performance During Nonroutine Plant Evolutions, As Low As Reasonably Achievable Planning and Controls, and Event Followup.

This report covered a 3-month inspection by resident inspectors and announced inspections by Regional emergency preparedness, health physics, and reactor inspectors. Three Green noncited violations, one Green finding, and one unresolved item with potential safety significance greater than Green were identified. The significance of most findings is indicated by their color (Green, White, Yellow, Red) using Inspection Manual Chapter 0609, "Significance Determination Process." Findings for which the significance determination process does not apply may be Green or be assigned a severity level after NRC management review. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG 1649, "Reactor Oversight Process," Revision 3, dated July 2000.

A. Inspector-Identified and Self-Revealing Findings

Cornerstone: Initiating Events

- Green. The inspectors identified a finding associated with an unplanned plant transient resulting from the failure of an operator to follow a written procedure. The transient occurred after the unexpected loss of all plant service cooling water and all but one of the condenser circulating water pumps. Cooling water was lost after an operator inadvertently opened the feeder breaker supplying power to the pumps.

  This finding is greater than minor because the operator error affected the human performance attribute of the initiating events cornerstone. The inspectors determined that the finding did not contribute to the likelihood of a primary or secondary system loss of coolant accident initiator, did not contribute to a loss of mitigation equipment functions, and did not increase the likelihood of a fire or internal/external flood. The finding was similar to Example 4.b in MC 0612, Appendix E, and was entered into the licensee's corrective action program as Callaway Action Request CAR 200308178 (Section 1R14).

Cornerstone: Occupational Radiation Safety

- Green. The inspectors identified a noncited violation of Technical Specification 5.7.1 because the licensee failed to barricade a high radiation area to prevent inadvertent entry. Specifically, on October 21, 2003, while performing independent radiation measurements, the inspectors identified a high radiation area on the 2031-foot elevation of the radwaste building that was not enclosed by a barricade. Radiation dose rates around a demineralizer sample panel drain tank were as high as 140 millirems per hour at 30 centimeters from the surface penetrated by the radiation. This finding is in the licensee's corrective action program as Callaway Action Request 200307676.

  This finding is greater than minor because inadequate controls of high radiation areas affected the licensee's ability to ensure adequate protection of worker health and safety from exposure to radiation and affected the cornerstone attribute/exposure control. Because the finding involved the potential for workers to receive significant unplanned, unintended dose as a result of conditions contrary to Technical Specification requirements, the inspectors used the Occupational Radiation Safety Significance Determination Process described in Manual Chapter 0609, Appendix C, to analyze the significance of the finding. The inspectors determined that a substantial potential for overexposure did not exist; therefore, the finding had very low significance (Section 2SO2).

Cornerstone: Barrier Integrity

- TBD. A finding was identified concerning a sequencing error in Emergency Operating Procedure E-3, “Steam Generator Tube Rupture,” that could result in increased postaccident public radiation dose. The sequence error delayed termination of safety injection during simulator exercises. The prolonged accident recovery time increased the postulated radiological source term released during the accident. The failure to maintain the emergency operating procedure consistent with the accident analysis was an apparent violation of Technical Specification 5.4, “Procedures.” This issue was entered into the licensee’s corrective action program as Callaway Action Request 200304922.

  This finding is unresolved pending completion of a significance determination. This issue was more than minor because the emergency operating procedure quality attribute of the barrier integrity cornerstone is affected by the procedural error (Section 4OA3).

- Green. The inspectors identified a noncited violation of 10 CFR Part 50, Appendix B, Criterion XI, “Test Control.” This finding is related to inadequate testing of the pressurizer power-operated relief valve block valve following modifications to the actuator circuit. The testing failed to detect that the modification was installed such that valve actuator failure occurred.

  This finding is greater than minor because the block valve failure affected the reactor coolant system equipment and barrier performance attribute of the barrier integrity cornerstone. The inspectors evaluated the condition with the Phase 2 worksheet of Manual Chapter 0609, “Significance Determination Process,” because the finding involved the reactor coolant system barrier. The finding was only of very low safety significance because the block valve failure did not significantly contribute to an increase in core damage frequency. The licensee placed this issue in their corrective action program as Callaway Action Request 200306563 (Section 4OA3).

- Green. The inspectors identified a noncited violation of 10 CFR Part 50, Appendix B, Criterion III, “Design Control.” This finding is related to inadequate translation of design information into the work instructions for modifications to a pressurizer power-operated relief valve block valve actuator circuit. The inadequate work instructions resulted in the failure of the valve actuator following return to service.

This finding is greater than minor because the block valve failure affected the reactor coolant system equipment and barrier performance attribute of the barrier integrity cornerstone. The inspectors evaluated the condition with the Phase 2 worksheet of Manual Chapter 0609, “Significance Determination Process,” because the finding involved the reactor coolant system barrier. This finding is only of very low safety significance because the block valve inoperability did not significantly contribute to an increase in core damage frequency. The licensee placed this issue in their corrective action program as Callaway Action Request 200306563 (Section 4OA3).

B. Licensee-Identified Violations

Violations of very low 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.

REPORT DETAILS

Summary of Plant Status: The Callaway Plant was operating at full power at the beginning of the inspection period. On October 21, 2003, the licensee completed a Technical Specification required shutdown following the failure of one of the station's uninterruptible power supply inverters. On October 24 the licensee restarted the plant following completion of repairs. On November 12, 2003, an operator error resulted in a loss of service water that necessitated a power reduction to approximately 65 percent power. Reactor power was restored and the licensee operated the plant at full power for the remainder of the inspection period.

1. REACTOR SAFETY
Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity

1R01 Adverse Weather Protection (71111.01)

a. Inspection Scope

The inspectors performed a detailed review of the station's adverse weather procedures affecting the essential service water (ESW) ultimate heat sink and refueling water storage tank on November 21, 2003. The inspectors selected these two systems due to their high importance to safety. The inspectors also performed walkdowns to verify that the licensee's adverse weather preparations were adequate to protect these two systems from cold weather. The inspectors discussed adverse weather precautions with the licensee and reviewed Special Operating Procedure OTS-ZZ-00007, "Plant Cold Weather," Revision 7.

b. Findings

No findings of significance were identified.

1R04 Equipment Alignment (71111.04)

a. Inspection Scope

Partial System Walkdowns. The inspectors performed two partial system walkdowns during the inspection period. On October 21, 2003, the inspectors walked down the 125 volt dc buses (NK02, NK03, and NK04) and vital inverters (NN02, NN03, and NN04), while the redundant bus (NK01) and vital inverter (NN11) were out of service for corrective maintenance. On November 4, the inspectors walked down Train A of the ESW system while the redundant train was out of service for planned maintenance. In each case, the inspectors checked for correct component alignment and evaluated operability by comparing the selected equipment to the applicable Final Safety Analysis Report (FSAR) sections and the procedures and drawings listed in the attachment.

b. Findings

No findings of significance were identified.

1R05 Fire Protection (71111.05)

a. Inspection Scope

The inspectors performed nine walkdowns of the accessible portions of the fire areas described below. These walkdowns were performed to assess the licensee's control of transient combustible materials, ignition sources, fire detection and suppression capabilities, fire barriers, and related compensatory measures. The inspectors also reviewed commitments described in the FSAR, Section 9.5.1, "Fire Protection System," and Appendix 9.5B, "Fire Hazard Analysis," to determine requirements for fire protection design features, fire area boundaries, and combustible loading requirements for each fire area. The inspectors walked down:

- Fire Areas A-17 and A-18, the north and south electrical penetration rooms (Rooms 1409 and 1410), on October 5, 2003
- Fire Areas A-21 and A-22, the control room air conditioning and filtration units (Rooms 1501 and 1512), on October 5, 2003
- Fire Areas ESW Train A, ESW Train B, and the ESW pumphouse on October 6, 2003
- Fire Areas C-10 and D-2, Train B of the essential safety feature switchgear and emergency diesel generator (EDG) rooms, on October 17, 2003
- Fire Areas C-9 and D-1, Train A of the essential safety feature switchgear and EDG rooms, on November 2, 2003
- Fire Areas A-29, auxiliary feedwater pump Room B and feedwater pump valve Compartment 2, on December 17, 2003
- Fire Area A-23B, main steam isolation valve Rooms 1 and 2, on October 19, 2003
- Fire Area A-20, personnel hatch and component cooling water surge tank area, on December 29, 2003
- Fire Area A-27, reactor trip switchgear room, on December 29, 2003

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification Activities Review by Resident Staff (71111.11Q)

a. Inspection Scope

The inspectors observed two licensed operator simulator training exercises and postscenario critiques. The inspectors observed the exercises to assess operator performance during high-risk operator actions associated with the emergency plan, lessons learned items, and plant operational experiences. The inspectors observed Licensed Operator Continued Training Simulator Scenario 4, NRC Bulletin 2003-01, "Potential Impact of Debris Blockage on Emergency Sump Recirculation," on November 7 and simulator Scenario 2, "Large Break Loss of Coolant Accident," on December 3, 2003.

b. Findings

No findings of significance were identified.

1R12 Maintenance Effectiveness (71111.12Q)

a. Inspection Scope

The inspectors reviewed three samples of equipment maintenance problems. The inspectors performed the review to verify that the licensee's maintenance efforts met 10 CFR 50.65, "Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants." The inspectors focused on maintenance rule characterization of failed components, risk significance, determination of the (a)(1) classification, corrective actions, and the appropriateness of established performance goals and monitoring criteria. The inspectors also evaluated emergent equipment issues to determine if problems were identified at the appropriate level and entered into the corrective action program. The inspectors used Administrative Procedure EDP-ZZ-01128, "Maintenance Rule Program," Revision 4, during the review. The inspectors performed an in-office review of the following Maintenance Rule (a)(1) evaluations:

- Callaway Action Request (CARs) 200304482 and 200304484, Steam Generator B level channel transmitter drifted out of tolerance
- CAR 200307851, Vital Inverter NN11 exceeded the Maintenance Rule availability goal
- CAR 200307344, Steam generator atmospheric dump valve failures

b. Findings

No findings of significance were identified.

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

a. Inspection Scope

The inspectors reviewed six maintenance risk assessments. The inspectors compared the licensee's risk assessment and risk management activities against the requirements of 10 CFR 50.65(a)(4); the recommendations of Nuclear Management and Resource Council 93-01, "Industry Guidelines for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," Revision 3; and Engineering Department Procedure EDP-ZZ-01129, "Callaway Plant Risk Assessment," Revision 2. The inspectors also reviewed the effectiveness of the licensee's contingency actions to mitigate increased risk resulting from degraded equipment. The inspectors evaluated the following risk assessments by in-office review and control room walkdowns:

- ESW Train B outage on November 4 and 5, 2003
- EDG B outage on November 4 and 5, 2003
- ESW Train A outage on September 23 and 24, 2003
- EDG Train A outage on September 23 and 24, 2003
- Component Cooling Water Train A outage on December 9 and 10, 2003
- Unplanned inoperability of EDG B and one off-site power supply on December 30, 2003

b. Findings

No findings of significance were identified.

1R14 Personnel Performance During Nonroutine Plant Evolutions (71111.14)

The inspectors reviewed personnel performance during nonroutine plant evolutions, events, and transient operations. The inspectors also selected licensee event reports (LERs) where personnel performance issues were identified as a causal factor to the event or condition. The inspectors' review included operator response following reactor trips which required more than routine expected operator actions or involved operator errors. The inspectors selected the following seven events:

- Reactor coolant system leakage following restoration from a letdown outage on October 16, 2003, as described in CAR 200307546

- Technical Specification required reactor shutdown on October 21 following a failed vital inverter, as described in CAR 200307636 and Surveillance Report SP03-016, “Failure of Inverter NN11 Forced Outage Activities,” on October 30, 2003

- Reactor shutdown on March 22 due to excessive leakage of a pressurizer safety valve, as described in CAR 200301460

- Reactor protection system actuation while performing testing, as described in LER 50-483/02-013-00

- Failure of the containment spray pumps due to gas binding, as described in LER 50-483/03-005-00

- Unplanned power reduction on November 12, as described in CAR 200308178

- Failure to meet licensing bases operator response times during performance of emergency procedures on December 3, 2003, as described in CARs 200308666 and 200308667. The inspectors also observed plant simulator validation of the main steamline break, chemical volume control system malfunction, reactor coolant system instrument line leak, and small break loss of coolant accidents on December 16 and 17.

b. Findings

.1 Operator Error Resulted in an Unplanned Reactor Power Transient

Introduction. The inspectors identified a Green finding associated with an unplanned plant transient which resulted from the failure of an operator to follow a written procedure.

Description. On November 12, 2003, an operator error resulted in an unplanned reactor power transient. The transient occurred following the unexpected loss of all plant service cooling water and all but one of the condenser circulating water pumps. Cooling water was lost after an operator inadvertently opened the breaker supplying the pumps. The error occurred while the operator was restoring the normal alignment to a service and condenser circulating water pumphouse electrical bus following maintenance. Normal Operating Procedure OTN-PB-0001, "Non-Class 1E 4.16 KV Electrical System," required the operator to hold syndication Switch 2201/2102 closed when closing bus feeder Breaker PB122. The operator mistakenly held the incorrect syndication switch closed, resulting in the loss of electrical power. Plant operators quickly reduced reactor power from 100 to 66 percent to avoid a reactor trip.

Analysis. This finding is greater than minor because the operator error and initiation of a power transit affected the human performance attribute of the initiating events cornerstone. The failure of the operator to follow Normal Operating

Procedure OTN-PB-0001 was a performance deficiency and resulted in a perturbation of plant stability. The inspectors determined the finding to be of very low safety significance (Green) using the Significance Determination Process for Reactor Inspection Findings for At-Power Situations. The inspectors determined that the finding is of very low safety significance because the condition did not contribute to the likelihood of a primary or secondary system loss of coolant accident initiator, did not contribute to a loss of mitigation of equipment functions, and did not increase the likelihood of a fire or internal/external flood. The finding is similar to Example 4.b in MC 0612, Appendix E, and was entered into the licensee's corrective action program as CAR 200308178 (FIN 50-483/0306-01).

Enforcement. No violation of regulatory requirements occurred. The inspectors determined that the finding did not represent a noncompliance because it occurred on nonsafety-related electrical equipment.

.2 Critical Operator Emergency Operating Procedure (EOP) Response Times Exceeded

The licensee identified an error in EOP E-3, "Steam Generator Tube Rupture," which resulted in an increase in postulated postaccident public radiation dose (LER 50-483/03-006-00, described in Section 4OA3 of this report). The error resulted in critical operator action taking longer than assumed in the accident analysis. The licensee determined that the condition may apply to other EOPs during the extent of condition review. In December, the licensee performed a validation of operator response times credited in the accident analysis. The evaluation identified the following variations from the accident analysis:

- Station Blackout Event

  The accident analysis allowed 30 minutes from event initiation for operators to open control room cabinets containing instruments and controls used to mitigate the event. The licensee's validation process revealed that 34 minutes were needed. As compensatory action, the licensee established a night order to maintain the affected cabinets unlocked. The operator response time was reduced to 24 minutes with the cabinets unlocked.

- Main Steamline Break Event

  The accident analysis allowed 10 minutes from event initiation for operators to terminate charging flow into the reactor coolant system. The licensee's validation process revealed that up to 41 minutes were needed. The licensee completed an operability determination, concluding that the excessive operator action time was acceptable.

- Chemical Volume Control System (CVCS) Line Break Outside Containment Event

  The accident analysis allowed 30 minutes for operators to isolate a CVCS leak. This time was not met in the first simulator crew validation. Due to simulator fidelity issues, the first crew did not recognize the event. The simulated area radiation monitor readings were not consistent with expected plant response. Two subsequent crews met the allowed 30-minute acceptance criteria.

The licensee was not able to retrieve previous EOP validation records. This issue was considered unresolved pending an evaluation of the safety consequences of critical operator response times not being met and a review of the operability determination associated with the main steamline break accident (URI 50-483/0306-02).

1R15 Operability Evaluations (71111.15)

a. Inspection Scope

The inspectors reviewed three operability determinations to verify that the licensee properly evaluated the operability of plant components and systems. The inspectors compared the technical adequacy of the evaluations to requirements stated in the Technical Specifications, the FSAR, and associated design-bases documents.

- CARs 20030653 and 200306618, Operability determination of the power-operated relief valve (PORV) block valve with Annunciator 64A "Cold O/P Block Valve Not Open" actuated, September 7, 2003
- CAR 200306563, Operability determination of the PORV block valve stem following overstress due to the actuator malfunction, September 6, 2003
- CAR 200307007, Operability determination of defective low voltage General Electric AKR-30 circuit breakers, September 26, 2003

b. Findings

No findings of significance were identified.

1R16 Operator Workarounds (71111.16)

a. Inspection Scope

The inspectors completed an evaluation of the cumulative effects of two operator workarounds during the inspection. The inspectors reviewed the November 2003 Operator Workaround List and the affect of the workarounds on the ability of operators to implement plant EOPs. The inspectors completed the review to verify that the cumulative effect of workarounds did not challenge the operators' capability to respond to plant transients and events. The inspectors completed an in-office review and control room walkdown of the following workarounds on November 18.

- Erratic indication of containment recirculation sump level Indicator EJLI008
- ESW Train B manual isolation Valve EFHV42, seat leakage

b. Findings

No findings of significance were identified.

1R19 Postmaintenance Testing (71111.19)

a. Inspection Scope

The inspectors reviewed four postmaintenance retests that could potentially affect risk significant systems or components. The inspectors completed an in-office review to verify that each test adequately demonstrated system operability and capability. The inspectors used Technical Specifications, the FSAR, and ASME Section XI to determine system and component requirements. The inspectors' review included the following postmaintenance retests:

- Retests R680010A and R221610A following preventive maintenance on Train B of the ESW system. The inspectors observed portions of the test from the control room on November 5, 2003, and subsequently performed an in-office review of the completed test package.

- Retests R68863A and R68001A following preventive maintenance on Train B of the EDG system. The inspectors observed portions of the test from the control room and EDG room on November 5, 2003, and subsequently performed an in-office review of the completed test package.

- Retest R660047A, Overhaul Train B of the diesel-driven fire-pump engine on November 11, 2003. The inspectors completed an in-office review of the testing documentation.

- Retest R688638A, EDG Train B and essential 4 kV bus and breaker preventive maintenance completed on November 5, 2003. The inspectors completed an in-office review of the testing documentation.

b. Findings

No findings of significance were identified.

1R22 Surveillance Testing (71111.22)

a. Inspection Scope

The inspectors observed and/or reviewed five surveillance tests to verify that the systems tested were capable of performing their safety function and to assess their operational readiness. The inspectors compared the following surveillance tests against requirements in plant Technical Specifications, ASME Code Section XI, the FSAR, and licensee procedural requirements:

- Surveillance S682464, safety injection (SI) relief valve, performed on September 18, 2003. The inspectors completed an in-office review and a plant walkdown of the SI pump discharge piping.

- Surveillances S71119, residual heat removal Pump A in-service test and S711119, valve in-service test, performed on September 25. The inspectors completed an in-office review.

- Surveillance S714360, emergency core cooling system pump venting and flow path verification performed on October 14. The inspectors completed an in-office review.

- Surveillance S712318, ESW valve stroke test, performed on October 25. The inspectors completed an in-office review.

- Surveillance S712320, ESW Pump A in-service test, performed on October 25. The inspectors completed an in-office review.

b. Findings

No findings of significance were identified.

1R23 Temporary Plant Modifications (71111.23)

a. Inspection Scope

The inspectors sampled three temporary plant modifications by in-office review and walked down affected plant equipment to verify that the installation was consistent with the modification documents. The inspectors reviewed the configuration control of the modification to verify that the plant documents, such as drawings and procedures, were updated, including applicable operating and maintenance procedures. The inspectors reviewed postinstallation test results to confirm that the actual impact of the temporary modifications on the permanent systems and interfacing systems were satisfactory. The inspectors compared temporary modification documentation against the requirements established in Administrative Procedure APA-ZZ-00605, "Temporary System Modifications," Revision 14, and system requirements contained in FSAR Section 9.1.3.

- Temporary Plant Modification TPM03-0008, reactor water transfer pumps with alternate suction from demineralized water system. The inspectors walked down the affected plant equipment in the auxiliary building, control building, and reactor water storage tank on October 1, 2003

- Temporary Plant Modification TPM 03-011, temporary patch on ESW Piping, EF-031-HBC-30, on December 11, 2003

- Modification CMP-03-1026, sealant injection for Handhole 9C on Steam Generator C, on October 23, 2003

The inspectors also reviewed Surveillance Report SP03-014, "Quality Assurance Surveillance of Reactor Makeup Water Storage Tank Work Activities," October 31, 2003.

b. Findings

No findings of significance were identified.

Cornerstone: Emergency Preparedness

1EP4 Emergency Action Level and Emergency Plan Changes (71114.04)

a. Inspection Scope

The inspectors conducted an in-office review of Revision 31 to Emergency Plan Implementing Procedure EIP-ZZ-00100, "Classification of Emergencies," submitted November 24, 2003. The inspectors compared this revision to the previous revision, the requirements of 10 CFR 50.54(q), and Appendix E to determine if the revision decreased the effectiveness of the emergency plan. Revision 31 was an administrative change to incorporate the correct terminology for the emergency response organization call-out system, which had been previously revised. This revision is subject to future inspections to ensure that the impact of the changes continues to meet NRC regulations. The inspectors completed one inspection requirement sample.

b. Findings

No findings of significance were identified.

1EP6 Drill Evaluation (71114.06)

a. Inspection Scope

The inspectors observed two licensee emergency drills to evaluate the adequacy of the drill conduct and to verify proper emergency action level classification and protective action recommendations. The inspectors observed the October 7, 2003, unannounced off-hours shift augmentation call-in drill from the control room and technical support center and Rapid Responder Drill SRO-REP-2 from the control room simulator on December 3, 2003. The inspectors compared drill observations against Operations Procedure ODP-ZZ-0025, "EOP Usage," Revision 5; Emergency Plan Implementing Procedure EIP-ZZ-00101, "Classification of Events," Revision 30; and Emergency Plan Implementing Procedure EIP-ZZ-00201, "Notifications," Revision 37, to evaluate licensee performance. The inspectors also reviewed the licensee's postdrill corrective actions, as documented in CARs 200307348, 200307371, 200307372, 200307377, 200307380, and 200307384.

b. Findings

No findings of significance were identified.

2. RADIATION SAFETY

Cornerstone: Occupational Radiation Safety

2OS2 As Low as is Reasonably Achievable (ALARA) Planning and Controls (71121.02)

a. Inspection Scope

To assess the licensee's program to maintain occupational exposures ALARA, the inspectors reviewed work activities conducted during two forced outages. The inspectors assessed the licensee's performance in implementing physical and administrative controls for airborne radioactivity areas, radiation areas, and high radiation areas; radiation worker practices; and work activity dose results against procedural and regulatory requirements. (No high exposure work activities in high radiation or airborne areas were performed during the inspection. Therefore, this aspect could not be evaluated).

The inspectors interviewed radiation protection staff members and other radiation workers to determine the level of planning, communication, ALARA practices, and supervisory oversight integrated into work planning and work activities. The inspectors reviewed initial and emergent work scopes and estimated person-hours provided to the radiation protection group for accuracy. In addition, the following items were reviewed and compared with procedural and regulatory requirements:

- Plant collective exposure history for the past 3 years, current exposure trends, source term measurements, and 3-year rolling average dose information
- Summer reliability forced outage ALARA report
- ALARA program procedures

- Processes, methodology, and bases used to estimate, justify, adjust, track, and evaluate exposures

- Three ALARA prejob, in-progress, and postjob reviews and associated radiation work permit packages from a forced outage, which resulted in the highest personnel collective exposures

- The use and result of administrative and engineering controls to achieve dose reductions

- Summary of corrective action documents written since the last inspection and selected documents relating to exposure tracking, higher than planned exposure levels, radiation worker practices, and repetitive or significant individual deficiencies.

The inspector completed 10 of the required samples.

Section 2OS2 evaluated the effectiveness of the licensee's problem identification and resolution processes regarding exposure tracking, higher than planned exposure levels, and radiation worker practices. No findings of significance were identified.

b. Findings

Introduction. The inspector identified a Green finding and noncited violation (NCV) of Technical Specification 5.7.1 because a high radiation area was not barricaded.

Description. While performing independent area radiation surveys on the 2031-foot elevation of the radwaste building, the inspector identified dose rates exceeding 100 millirems per hour at 30 centimeters from a collection tank. The tank, which collected liquid draining from a demineralizer sample panel, was partially shielded with lead blankets. A radiation protection technician confirmed the inspector's finding and measured dose rates as high as 140 millirems per hour at 30 centimeters from the side of the tank.

Analysis. This finding was greater than minor because inadequate controls of high radiation areas affected the licensee's ability to ensure adequate protection of worker health and safety from exposure to radiation and affected the cornerstone attribute/exposure control. Because the finding involved the potential for workers to receive significant, unplanned, unintended dose as a result of conditions contrary to Technical Specification requirements, the inspector used the Occupational Radiation Safety Significance Determination Process described in Manual Chapter 0609, Appendix C, to analyze the significance of the finding. The inspector determined that a substantial potential for overexposure did not exist; therefore, the finding had very low significance.

Enforcement. 10 CFR 20.1003 defines a high radiation area as an area, accessible to individuals, in which radiation levels from radiation sources external to the body could result in an individual receiving a dose equivalent in excess of 0.1 rem (100 millirem) in 1 hour at 30 centimeters from the radiation source or 30 centimeters from any surface that the radiation penetrates.

Technical Specification 5.7.1 requires that each high radiation area be barricaded and conspicuously posted as a high radiation area. However, the requirement was violated when the dose rate outside the collection tank exceeded 100 millirems per hour at 30 centimeters from the surface that radiation penetrated. Because the finding was of very low safety significance and was entered into the corrective action program as CAR 200307676, this violation was treated as an NCV, consistent with Section VI.A of the NRC Enforcement Policy (NCV 50-483/0306-03).

4. OTHER ACTIVITIES

4OA1 Performance Indicator (PI) Verification (71151)

a. Inspection Scope

The inspectors sampled licensee submittals for the two PIs listed below for the period from October 2002 through September 2003. The inspectors used the definitions and guidance contained in Nuclear Energy Institute (NEI) 99-02, "Regulatory Assessment Indicator Guideline," Revision 2, to verify the accuracy of the PI data reported by the licensee.

Reactor Safety Cornerstone

- Auxiliary feedwater system
- Residual heat removal system

The inspectors reviewed a selection of LERs, portions of operator log entries, daily morning reports, the monthly operating reports, and PI data sheets to determine whether the licensee adequately identified the number of unavailable hours for the auxiliary feedwater and residual heat removal systems. This number was compared to the number reported for the PI during the current quarter. In addition, the inspectors also interviewed licensee personnel associated with PI data collection, evaluation, and distribution.

b. Findings

No findings of significance were identified.

4OA2 Identification and Resolution of Problems (71152)

a. Inspection Scope

The inspectors performed detailed in-office reviews and walkdowns of plant equipment related to two significant conditions adverse to quality. The inspectors reviewed licencee CAR reports to verify that the full extent of the issues was identified, that the licensee performed appropriate evaluations, and that corrective actions were specified and prioritized. The inspectors evaluated the reports against the requirements of Administrative Procedure APA-ZZ-00500, "Corrective Action Program," Revision 21, and 10 CFR Part 50, Appendix B. The inspectors also attended the corrective action Management Oversight Committee Review Meeting on November 21, 2003. The inspectors reviewed the following two samples:

- CAR 200307636, Vital Inverter NN11 failure on October 20, 2003
- CAR 200306563, Motor failure of a PORV block valve on September 4, 2003

b. Findings

No findings of significance were identified.

4OA3 Event Followup (71153)

1. (Closed) LER 50-483/03-001-00: Improper administrative controls resulted in a Technical Specification violation.

On January 8, 2003, the licensee declared primary containment isolation Valve EGHV00061 inoperable following failure of a stroke-time test. The operating authority manually closed the valve and removed power as required by plant Technical Specification 3.6.3, Action A.1. The licencee continued to use the containment penetration by opening a parallel flow path through containment isolation Valve EGHV00133. However, Valve EGHV00133 was not designed to automatically close on a containment isolation signal as required by General Design Criteria 56. Also, the containment penetration no longer met single failure criteria because Valve EGHV00133 was powered from the same source as the inboard containment isolation valve. Use of Valve EGHV00133 in Mode 1 was a violation of Technical Specification 3.6.3. This issue was dispositioned as a finding and NCV of very low safety significance (50-483/0303-01) in Callaway Plant Integrated NRC Inspection Report 05000483/200303. The inspectors did not identify any additional findings. This LER is closed.

2. (Closed) LER 50-483/03-006-00: Incorrect sequencing of procedure steps could have resulted in delayed recovery from a steam generator tube rupture.

a. Inspection Scope

The inspectors reviewed the LER and associated condition adverse to quality report to verify that the licensee adequately addressed the causes of the condition and performed appropriate corrective actions. Licensed operators were not able to demonstrate completion of key accident mitigation steps within the time frame assumed in the licensing bases during simulator exercises. Section 1R14, “Personnel Performance During Non-Routine Plant Evolutions,” of this report includes a discussion of the inspectors' and licensee’s followup of the extent of condition review generated as a result of this event. This LER is closed.

b. Findings

Introduction. A finding was identified for the failure of the licensee to maintain EOP E-3 consistent with the accident analysis. Incorrect operator actions in EOP E-3 resulted increased postulated radiological dose to the public due to prolonged accident recovery time. This is an unresolved item pending completion of the significance determination process.

Description. On June 3, 2003, an error in EOP E-3, “Steam Generator Tube Rupture,” was identified during development of licensed operator training. EOP E-3 required the operator to arm the pressurizer PORVs to provide reactor cold overpressurization (COP) protection prior to SI termination. The COP circuit opened the PORVs and challenged reactor subcooling margin during simulator exercises. The loss of subcooling margin delayed the termination of SI. The delay in SI termination prolonged accident recovery time and increased the influx of reactor coolant to the faulted steam generator. The increase of reactor coolant to the faulted steam generator increased the postulated source term release from the atmospheric PORV. The increased source term resulted in increased postaccident public radiation dose.

The licensee added a step to arm the PORVs to EOP E-3, in 1988, based on the Westinghouse Emergency Response Guides. The plant response was affected after Refueling 11 when the licensee modified the COP control circuit. The modification removed a pressurizer PORV interlock, allowing COP operation prior to SI termination. The licencee first observed the condition on the simulator in 1999 after a computer modeling upgrade.

Analysis. This finding is unresolved pending review of additional information to be provided by the licensee. This finding is more than minor because the EOP quality attribute of the barrier integrity cornerstone is affected by the procedural error. The significance of exceeding postulated postaccident dose limits established in the safety evaluation report could not be determined based on the information available. Additional analysis by the licensee is needed to evaluate this finding for determining safety significance and enforcement.

Enforcement. The failure to maintain EOP E-3 consistent with the accident analysis was an apparent violation of Technical Specification 5.4, “Procedures.” This issue was not an immediate safety concern because the licensee completed revisions to EOP E-3, which permitted accident recovery within the time frames specified within the accident analysis. The inspectors will also consider the licensee’s EOP validation methodology and review past EOP operator validations. This finding is identified as an unresolved item (URI 50-483/0306-04). This issue was being tracked in the licensee’s corrective action program under CAR 200304922.

3. (Closed) Unresolved Item 50-483/0305-01: Licensee request for Notice of Enforcement Discretion following the failure of a pressurizer PORV block valve

a. Inspection Scope

On September 4, 2003, the actuator for pressurizer PORV block Valve BBHV8000B failed. The actuator failed after the valve was returned to service following completion of control circuit modifications. On September 6, the licensee requested the NRC to exercise enforcement discretion for the requirements of Technical Specification Limiting Conditions for Operation 3.4.11, "Pressurizer Power Operated Relief Valves." The licensee sought enforcement discretion to permit noncompliance with Limiting Conditions for Operation 3.4.11 for an additional 48 hours. The licensee requested the Limiting Conditions for Operation extension to permit repair, testing, and restoration of the block valve before a plant shutdown would be required. The NRC verbally approved the licensee’s verbal request for enforcement discretion on September 6. The licensee completed block valve repairs and testing and exited Technical Specification 3.4.11 and the Notice of Enforcement Discretion on September 7. The inspectors reviewed the block valve modification design, installation, and postmodification testing (PMT) as followup action to the enforcement discretion.

b. Findings

.1 Inadequate PMT of a pressurizer PORV block valve

Introduction. The inspectors identified a Green finding and an NCV of 10 CFR Part 50, Appendix B, Criterion XI, “Test Control.” This violation was related to inadequate testing of a PORV block valve following modifications to the actuator circuit.

Description. On September 4, 2003, pressurizer block Valve BBHV8000B failed. The actuator failed after the valve was returned to service following completion of control circuit modifications. The licensee had modified the actuator circuit to seal-in the closed handswitch position. The licensee completed a PMT (Retest R685881A) after the modification was installed. The licensee declared the block valve operable after successfully stroking the valve. Approximately 40 minutes later, a control room operator identified that the open indicating light for the valve handswitch was not illuminated. The licensee subsequently determined that the block valve actuator motor had remained energized and that the thermal overloads and actuator motor had failed. The PMT scope and acceptance criteria were not adequate to identify that the modification was improper.

Analysis. The licensee's failure to implement an adequate PMT scope was a performance deficiency. This finding is greater than minor because the block valve failure affected the reactor coolant system equipment and barrier performance attribute of the barrier integrity cornerstone. The inspectors evaluated the condition with the significance determination process Phase 2 worksheet because the finding involved the reactor coolant system barrier. The "Stuck Open PORV" was the dominant core damage sequence in the Phase 2 analysis. The inspectors assumed that the operator lost the ability to isolate a small loss of coolant accident through the PORV. The inspectors did not assume any recovery credit and an initiating event likelihood of less than 3 days. The finding was only of very low safety significance because the block valve inoperability did not significantly contribute to an increase in core damage frequency.

Enforcement. Title 10 of the Code of Federal Regulations, Part 50, Appendix B, Criterion XI, "Test Control," required that a test program demonstrate that structures, systems, and components will perform satisfactorily in service. Contrary to the above, the scope of Retest R685881A was not adequate to demonstrate that the pressurizer PORV block valve actuator would perform satisfactorily in service. Because of the very low safety significance and the licensee's action to place the issue in their corrective action program (CAR 200306563), this violation is being treated as an NCV in accordance with Section VI.A.1 of the Enforcement Policy (EA-04-013; NCV 50-483/0306-05).

.2 Inadequate work instructions resulted in an improperly installed modification

Introduction. The inspectors identified a Green finding and NCV of 10 CFR Part 50, Appendix B, Criterion III, "Design Control." This violation was related to inadequate incorporation of design information into the work instructions for modifications to the pressurizer PORV block valve actuator circuit.

Description. The licencee completed modifications to the pressurizer PORV block Valve BBHV8000B actuator circuit on September 4, 2003. The modification design required removal of an existing jumper in the valve actuator breaker cubicle. The modification work instructions omitted steps for removal of this jumper connection from the control circuit. The jumper electrically bypassed the valve actuator open limit switch and torque switch after the modification was completed. The jumper caused the actuator motor to remain energized after the valve reached the full open position. The valve actuator motor and thermal overloads failed after the valve was returned to service.

Analysis. The licensee's failure to adequately translate the modification design into work instructions was a performance deficiency. This finding is greater than minor because the block valve failure affected the reactor coolant system equipment and barrier performance attribute of the barrier integrity cornerstone. The inspectors evaluated the condition with the Phase 2 worksheet because the finding involved the reactor coolant system barrier. The "Stuck Open PORV" was the dominant core damage sequence in the Phase 2 analysis. The inspectors assumed that the operator lost the ability to isolate a small loss of coolant accident through the PORV. The inspectors did not assume any recovery credit and an initiating event likelihood of less than 3 days. The finding was only of very low safety significance because the block valve inoperability did not significantly contribute to an increase in core damage frequency.

Enforcement. Title 10 of the Code of Federal Regulations, Part 50, Appendix B, Criterion III, "Design Control," required that the licensee establish measures to assure applicable design bases were correctly translated into specifications and instructions. Contrary to the above, the licensee did not correctly translate the applicable design bases into modification RFR 20411B. The work instructions did not include the design specification to remove an electrical jumper. Because of the very low safety significance and the licensee's action to place the issue in their corrective action program (CAR 200306563), this violation is being treated as an NCV in accordance with Section VI.A.1 of the Enforcement Policy (EA-04-012; NCV 50-483/0306-06).

4. (Closed) LER 50-483/03-007-00: Failure to maintain a control room ventilation boundary resulted in an unanalyzed condition.

On July 17, 2003 the licensee identified that plant configuration compromised the integrity of the control room envelope. The licensee determined that air entering the control building from normally open pressure boundary Door 32201 resulted in higher postulated postaccident operator thyroid dose than approved in the accident analysis. The licensee identified the condition during a plant configuration review in preparation for responding to NRC Generic Letter 2003-01, "Control Room Habitability." The licensee's immediate corrective actions included maintaining Door 32201 in the closed position. The finding was greater than minor because, if left uncorrected, the finding would become a more significant safety concern. The finding was associated with the barrier integrity cornerstone. The inspectors concluded that the finding was only of very low safety significance because the finding only represented a degradation of the radiological barrier function provided for the control room. The condition was a violation of Technical Specifications 3.7.10, "Control Room Ventilation System." This issue was entered into the licensee's corrective action program as CAR 200305274. The enforcement aspects of the violation are discussed in Section 4OA7. This LER is closed.

5. <u>(Closed) LER 50-483/02-011-00</u>: In-service inspection results for Steam Generator A classified as C-3

   During the refueling Outage 12 in-service inspection, the licensee identified that greater than one percent of tubes in Steam Generator A were defective. The licensee classified the inspection results as C-3 in accordance with Technical Specification 5.5.9 and performed the required NRC reporting per Table 5.5.9-2. The licensee plugged all defective tubes prior to returning the steam generator to service. The inspection results were described in NRC Inspection Report 50-483/02-06, Section 1R08.2. No new findings were identified in the inspectors' review. The licensee documented the issue in CAR 200207229. This LER is closed.

6. <u>(Closed) LER 50-483/03-04-00</u>: Boron dilution mitigation system BDMS blocked in Mode 3.

   Technical Specification 3.3.9 states, in part, that two trains of the BDMS shall be operable or immediately suspend operations involving positive reactivity additions. On three different occasions (November 24, 2002, December 17, 2002, and April 2, 2003), the licensee had blocked the BDMS while withdrawing control rod shutdown banks for a reactor startup. This action placed the unit in an unanalyzed condition as specified by FSAR accident analysis Section 15.4.6.2, where BDMS is credited for automatically terminating a reactor coolant dilution event. This event is described in the licensee's corrective action program as CAR 200302704. Using the Reactor Safety Significance Determination Process, the inspectors determined that the finding is of very low safety significance and is being treated as an NCV because the finding was similar to Example 2.g of Appendix E of Inspection Manual Chapter 0612 (a mode change being made without all required equipment being operable). The enforcement aspects of the violation are discussed in Section 4OA7. This LER is closed.

4OA4 <u>Crosscutting Aspects of Findings</u>

Section 1R14 of this report documents a human performance error that resulted in a plant transient while operating the nonsafety-related electrical system.

Section 4OA3 of this report documents two human performance errors. One resulted in an inadequate modification package that incorporated into the control circuit a pressurizer power-operated valve block valve. The second error, an inadequate postmaintenance test, did not identify the inadequacy of the installed modification nor the subsequent failure of a pressurizer power-operated valve block valve prior to returning it to service.

4OA6 Management Meetings

Exit Meeting Summary

On January 6, 2004, the resident inspectors presented their inspection results to Mr. G. Randolph, Senior Vice President-Generation and Chief Nuclear Officer, and other members of his staff who acknowledged the findings.

On December 3, 2003, the inspectors presented the inspection results to Mr. L. Graessle, Superintendent, Protective Services during a phone conversation. The licensee acknowledged the findings presented.

On October 23, 2003, the inspectors presented the inspection results to Mr. R. Affolter, Vice President, Nuclear, and other members of his staff who acknowledged the findings.

The inspectors verified that no proprietary information was reviewed during the inspection.

4OA7 Licensee-Identified Violations

The following findings of very low safety significance (Green) were identified by the licensee and are violations of NRC requirements which meet the criteria of Section VI of the NRC Enforcement Policy, NUREG-1600, for being dispositioned as NCVs.

- Technical Specification 3.3.9 states, in part, that two trains of the BDMS shall be operable. Contrary to this, on three different occasions the licensee had blocked the BDMS while withdrawing control rod shutdown banks for a reactor startup. This event was identified in the licensee’s corrective action program as CAR 200302704. This finding is of very low safety significance because it was similar to Example 2.g of Appendix E of Inspection Manual Chapter 0612 (a mode change being made without all required equipment being operable).

- Technical Specification 3.7.10 required the licensee to maintain integrity of the control room ventilation envelope. Contrary to the above, the licensee had failed to maintain the integrity of the control room envelope for greater than 3 years. This was identified in the licensee’s corrective action program as CAR 200305274 and was reported as LER 50-483/03-007-00. This finding was of very low safety significance because the finding only represented a degradation of the radiological barrier function provided for the control room.

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee

R. Affolter, Vice President - Nuclear
M. Evans, Manager, Nuclear Engineering
R. Farnum, Supervisor, Health Physics Operations
K. Gilliam, Supervisor, Health Physics
L. Graessle, Superintendent, Protective Services
M. Hale, Superintendent, Health Physics
D. Neterer, Superintendent, Operations
M. Reidmeyer, Supervisor, Regional Regulatory Affairs
W. Witt, Plant Manager
K. Young, Manager, Regulatory Affairs

LIST OF ITEMS OPENED AND CLOSED

Opened

50-483/0306-01	FIN	Operator error resulted in an unplanned reactor power transient (Section 1R14).
50-483/0306-02	URI	Critical operator EOP response times exceeded (Section 1R14).
50-483/0306-03	NCV	Failure to barricade a high radiation area (Section 2OS2).
50-483/0306-04	URI	Unresolved item pending completion of the significance determination process of the failure of the licensee to maintain EOP E-3 consistent with the accident analysis (Section 4OA3).
50-483/0306-05	NCV	Inadequate postmodification test of a pressurizer power-operated relief block valve (Section 4OA3).
50-483/0306-06	NCV	Inadequate incorporation of design information into work instructions lead to the failure of a pressurizer block valve (Section 4OA3).

Closed

50-483/0306-01	FIN	Operator error resulted in an unplanned reactor power transient (Section 1R14).
50-483/0306-03	NCV	Failure to barricade a high radiation area (Section 2OS2).

50-483/2003-001-00	LER	Improper administrative controls resulted in a Technical Specification violation (Section 4OA3).
50-483/2003-006-00	LER	Incorrect sequencing of procedure steps could have resulted in delayed recovery from a steam generator tube rupture (Section 4OA3).
50-483/0305-01	URI	Licensee request for Notice of Enforcement Discretion following the failure of a pressurizer PORV block valve (Section 4OA3).
50-483/0306-05	NCV	Inadequate postmodification test of a pressurizer power-operated relief block valve (Section 4OA3)
50-483/0306-06	NCV	Inadequate incorporation of design information into work instructions lead to the failure of a pressurizer block valve (Section 4OA3).
50-483/2003-007-00	LER	Failure to maintain a control room ventilation boundary resulted in an unanalyzed condition (Section 4OA3).
50-483/2002-011-00	LER	In-service inspection results for Steam Generator A classified as C-3 (Section (4OA3).
50-483/2003-04-00	LER	Boron dilution mitigation system blocked in Mode 3 (Section 4OA3).

DOCUMENTS REVIEWED

Procedures

Administrative Procedure APA-ZZ-00151, Revision 0, Forced Outage Response Plan

Health Physics Department Procedure HDP-ZZ-06100 Reactor Building Access, Revision 1

Health Physics Technical Procedure HTP-ZZ01102, Pre-work ALARA Planning and Briefing, Revision 17

Health Physics Technical Procedure HTP-ZZ-01201, Preparation and Maintenance of General and Specific Radiation Work Permits, Revision 31

Health Physics Technical Procedure HTP-ZZ-01103, Post-Work ALARA Review, Revision 13

Operations Surveillance Procedure OSP-EF-00001, Essential Service Water Valve Alignment, Revision 5

Operations Surveillance Procedure OSP-EF-P001A, Essential Service Water Pump A In-service Test, Revision 40

Operations Surveillance Procedure OSP-EF-V001A, Essential Service Water Valve In-service Test, Revision 25

Operations Surveillance Procedure OSP-KC-0001, Fire Pump Starting Test and Fire Water Shortage Tank Inspection, Revision 11

Operations Surveillance Procedure OSP-NE-0001B, Standby Diesel Generator B Periodic Tests, Revision 13

Operations Surveillance Procedure OSP-SA-00003, Mode 1 - 3 ECCS Venting Trains A and B, Revision 17

Normal Operating Procedure OTN-QJ-00003, Borated Refueling Water Storage Tank Heat Tracing Lineup, Check-Off List Number 4, Revision 0

Normal Operating Procedure OTN-QJ-00003, Essential Service Water System Heat Tracing Lineup for Train A, Check-Off List Number 8, Revision 0

Normal Operating Procedure OTN-QJ-00003, Essential Service Water System Heat Tracing Lineup for Train B, Check-Off List Number 9, Revision 0

Normal Operating Procedure OTN-NK-0001, Class IE 125 VDC Electrical System, Revision 9

Normal Operating Procedure OTN-EF-00001, Essential Service Water System, Revision 25

Normal Operating Procedure OTN-NN-00001, 120 V Vital AC Instrument Power - Class 1E, Revision 10

Callaway Ameren UE Workman's Protection Assurance Tagout Control Sheet - WPA Number 48896

Drawing E-21NK01, Revision 6, Class 1E 125 VDC System

Drawing E-21NN01, Revision 8, Class 1E 120 VDC System

Electrical Schematic, Pressurizer Relief Isolation E23BB39, Revision 11

Electrical Schematic, Pressurizer Relief Isolation E23BB28, Revision 2

Raceway Plan, Diesel Generator Building, E-2R5112(Q), Revision 13

Piping and Instrumentation Drawing M-22AL01

Piping and Instrumentation Drawing M-22EF01

Piping and Instrumentation Drawing M-22EF02

WCAP-15983-P, "Technical Justification for Eliminating Pressurizer Surge Line Rupture as the Structural Design Basis for the Callaway Nuclear Plant," February 2003

WCAP-16019-P, “Technical Justification for Eliminating 10" Accumulator Lines Rupture as the Structural Design Basis for the Callaway Nuclear Plant,” February 2003

WCAP-16020-P, “Technical Justification for Eliminating 14" Residual Heat Removal Lines Rupture as the Structural Design Basis for the Callaway Nuclear Plant,” February 2003

<u>CARs</u>

200301827	200304063	200307203
200302598	200304605	200307230
200302994	200305638	20030734
200303390	200306260	200307369
200303427	200307101	200307381
200303469	200307107	200307433
200303775	200307111	200307804
200304008	200307113	

<u>Radiation Work Permits</u>

RWP W702164	Replace Valve B88010B
RWP S714763	Boric Acid Inspection in the Reactor Building, Including the Reactor Head, Bio-Shield and General Areas
RWP W715548	Torque 9C Handhole Cover Bolting
RWP C715568	Inject High Pressure Sealing Compound
RWP W228991	Repair Leak at Inlet Fitting on BB-FT-0434
RWP W706785	Repair Leak at Lower Flange EBG02

<u>Self-Assessment and Quality Verification</u>

Quality Assurance Department Audit Report AP03-002 (March 18, 2003)
Surveillance Report SP03-008, "Health Physics Postings"

LIST OF ACRONYMS

ALARA	as low as is reasonably achievable
ASME	American Society of Mechanical Engineers
BDMS	boron dilution mitigation system
CAR	Callaway Action Request
COP	cold overpressurization
EDG	emergency diesel generator
ESW	essential service water
EOP	emergency operating procedure
FSAR	Final Safety Analysis Report

LER	licensee event report
NCV	noncited violation
NEI	Nuclear Energy Institute
PI	performance indicator
PORV	power-operated relief valve
PMT	postmodification testing
SI	safety injection