UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION IV
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ARLINGTON, TEXAS 76011-4005

May 4, 2004

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 05000483/2004002

Dear Mr. Randolph:

On March 24, 2004, the NRC completed an inspection at your Callaway Plant. The enclosed report documents the inspection findings which were discussed on March 26, 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 NRC-identified and three self-revealing findings of very low safety significance. Three of these findings were determined to involve violations of NRC requirements. Because of the very low safety significance and because they are entered into your corrective action program, these violations are being treated as noncited violations (NCVs), consistent with Section VI.A of the Enforcement Policy. Additionally, 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 violations 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 2.390 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
05000483/2004002
w/attachment: Supplemental Information

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ENCLOSURE

U.S. NUCLEAR REGULATORY COMMISSION
REGION IV

Docket:	50-483
License:	NPF-30
Report:	05000483/2004002
Licensee:	Union Electric Company
Facility:	Callaway Plant
Location:	Junction Highway CC and Highway O Fulton, Missouri
Dates:	January 1 through March 24, 2004
Inspectors:	M. S. Peck, Senior Resident Inspector J. D. Hanna, Resident Inspector F. L. Brush, Senior Resident Inspector E. L. Crowe, Resident Inspector R. E. Lantz, Senior Emergency Preparedness Inspector N. F. O'Keefe, Senior Reactor Inspector
Approved By:	D. N. Graves, Chief, Project Branch B

CONTENTS

SUMMARY OF FINDINGS

IR 05000483/2004002; 01/01 - 03/24/2004; Callaway Plant. Fire Protection, Personnel Performance During Nonroutine Plant Evolutions and Other.

This report covered a 3-month inspection by resident inspectors and announced inspections by Regional emergency preparedness and reactor inspectors. Three Green noncited violations and one Green finding 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: Mitigating Systems

- Green. The alarm response procedure for responding to smoke in the control room outside supply duct was inadequate because it did not direct operators to isolate outside air makeup upon receipt of the alarm. This alarm would not cause an automatic isolation of the control room, so operators must recognize the condition and take manual action to prevent losing control room habitability. Failure to have a procedure, required by Technical Specification 5.4.1.a and Regulatory Guide 1.33, that provided appropriate response actions for abnormal or alarm conditions was a violation. This issue was entered into the licensee's corrective action program under Callaway Action Request 200306977.

  This issue was more than minor because failure to isolate the control room ventilation could lead to unnecessary evacuation, which would result in a plant transient and disabling much of the mitigation equipment that would otherwise be available. This issue was of very low safety significance because the frequency of the specific fire scenario necessary to cause an unnecessary control room evacuation was determined to be very small (Section 4OA5).

Cornerstone: Initiating Events

- Green. A self-revealing finding and noncited violation of Technical Specification 5.4.1, "Procedures," was identified after an operator error resulted in an unplanned reactor trip. The operator's action to open the main feedwater regulating valves before the plant was stable and at the prescribed power level resulted in a reactor trip on low steam generator water level.

  This finding is greater than minor because the reactor trip was a transient initiator affecting the initiating events cornerstone. The operator's failure to follow the procedure was a performance deficiency which affected the human performance attribute of the initiating events cornerstone. The inspectors determined this finding to be of very low safety significance (Green) 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 licensee placed this issue into the corrective action program as Callaway Action Request 200401167 (Section 1R14).

- Green. A self-revealing finding was identified after the unplanned loss of the turbine-driven auxiliary feedwater pump during a plant transient. After a reactor trip, an operator improperly secured the turbine-driven auxiliary feedwater pump which led to an overspeed trip.

  This finding was greater than minor because the loss of the turbine-driven auxiliary feedwater pump affected the availability/reliability objective of the mitigating system equipment performance cornerstone. The inspectors concluded that this finding was only of very low safety significance because: it was not a design or qualification deficiency, it did not represent the actual loss of the safety function of a system, it did not represent the actual loss of the safety function of a single train for greater than its Technical Specification allowed outage time, it did not represent the loss of a non-Technical Specification related train (designated as risk significant per 10 CFR 50.65 a(4)) for greater than 24 hours, and it did not screen as potentially risk significant due to a seismic, fire, flooding, or severe weather initiating event. The licensee placed the issue into the corrective action program as Callaway Action Request 200401167 (Section 1R14).

- Green. A self-revealing finding and noncited violation of Technical Specification 5.4.1, “Procedures,” was identified after an operator error resulted in an unplanned safety injection and main steamline isolation. The operator failed to place pressurizer pressure control in automatic during plant heatup operations. Pressurizer pressure exceeded the Permissive P-11 setpoint while the main steamline pressure was still below the safety injection setpoint.

  This finding is greater than minor because the safety injection was a transient initiator contributor affecting the initiating events cornerstone. The operator’s failure to follow the procedure was a performance deficiency which affected the human performance attribute of the initiating events cornerstone. The inspectors concluded that this finding was 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 (Section 1R14).

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. The reactor tripped from full power on January 27, 2004, following the failure of a main generator protection relay. The licensee restarted the plant on January 28. The reactor tripped again, from full power, on February 3, following the failure of a second main generator protection relay. The licensee restarted the plant on February 14. The reactor tripped a third time on February 15, due to an operator error. The licensee restarted the plant on February 17 and operated the facility 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 one detailed review of the station's adverse weather procedures affecting the essential service water (ESW) system during the week of January 26. The inspectors selected the ESW system due to its high importance to safety. The inspectors performed walkdowns to verify that the licensee's adverse weather preparations were adequate to protect the ESW system from freezing that might affect system accident mitigation capability or damage water filled piping. The inspectors also 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 three partial system walkdowns during the inspection period. On January 13, the inspectors walked down components of the motor-driven auxiliary feedwater (AFW) system located in the auxiliary building, condensate storage tank enclosure, and control building, while the redundant turbine-driven auxiliary feedwater (TDAFW) pump was out of service for planned maintenance. On January 12, the inspectors completed an auxiliary building and control building walkdown of component cooling water system Train A while the redundant train was out of service for planned maintenance. On February 19 and 20, the inspectors walked down the auxiliary building components of containment hydrogen control system Train A while the redundant train was out of service for emergent work. 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.

Complete System Walkdown. The inspectors conducted a detailed review of the alignment and condition of the Train A emergency diesel generator on March 5. The inspectors completed a system walkdown of components located in the emergency diesel generator and control buildings. The inspectors used the FSAR, Section 8.3.1.1.3, “Standby Power Supply,” Third Quarter 2003 System Health Report, and the procedures and drawings listed in the attachment to verify proper system alignment.

b. Findings

No findings of significance were identified.

1R05 Fire Protection (71111.05)

a. Inspection Scope

The inspectors performed seven 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 Area A-3, boric acid tank Rooms 1116, 1117, and 1407 on January 9
- Fire Area A-21, control room air conditioning and filtration Room 1501, on January 9
- Fire Area C-21, lower cable spreading Room 3501, on January 11
- Ultimate heat sink cooling towers cooling tower area, north cell, on February 2
- Ultimate heat sink cooling towers cooling tower area, south cell, on February 2
- Fire Area C-13, access control and electrical equipment air conditioning Room 1, on January 26
- Fire Area A-1H, auxiliary building pipe chase, Rooms 1206 and 1207, on January 29

b. Findings

No findings of significance were identified.

1R06 Flood Protection Measures (71111.06)

a. Inspection Scope

The inspectors completed one flood protection walkdown of the containment spray and residual heat removal rooms on February 24. The inspectors conducted the walkdown to verify that the licensee had implemented adequate protection for equipment below the postulated flood-line, including electrical conduits, holes, and wall penetrations. The inspection included a walkdown of the common drains, sumps, sump pumps, level alarms, and control circuits. The inspectors used Request for Resolution 16409, “Watertight Door Matrix,” November 14, 1996; and the FSAR, Section 3.4, “Water Level Flood Design,” as the bases for acceptability of the plant configuration.

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 one licensed operator simulator training exercise and postscenario critique. The inspectors observed the exercise 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 5, "Main Seam Line Break While in Mode 3," on January 30.

b. Findings

No findings of significance were identified.

1R12 Maintenance Effectiveness (71111.12Q)

a. Inspection Scope

The inspectors reviewed two samples of equipment maintenance issues. 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 Maintenance Rule (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 5, during the review.

The inspectors performed an in-office review of the following Maintenance Rule (a)(1) evaluations:

- CAR 200307247, Unexpected entry into Technical Specification Action Statement for Component Cooling Water (CCW) Train B

- CAR 200307304, Oil leak from a gasket on an outboard motor bearing of the normal charging pump

b. Findings

No findings of significance were identified.

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

a. Inspection Scope

The inspectors reviewed seven 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:

- Reactor trip Breaker B trip actuating device operational test, Surveillance S716356, from the control room on January 2. The inspectors also evaluated associated CAR 200400013, failure to recognize Technical Specification 3.4.11.B for trip actuating device operational testing.

- Removal of the TDAFW pump from service on January 12, for planned maintenance. The inspectors observed the licensee’s risk continency activities from the control room and auxiliary building.

- Unplanned inoperability of the TDAFW pump on February 3 and 4, and associated CAR 200400798. The inspectors observed the licensee’s risk continency activities from the control room and auxiliary building.

- Planned outage of essential 480 volt Bus NG01A on February 10. The inspectors observed the licensee’s compensatory actions from the control room and the essential switchgear room.

- Planned performance of Surveillance S719675, removing CCW Train B from service on February 26, and the CAR 200401481 task sheet failed to identify that the CCW system was not functional.

- Unplanned emergency diesel Generator B failure on March 11 and CAR 200401869, failure of emergency diesel Generator B. The inspectors observed the licensee's risk continency activities from the control building, switchyard, and diesel generator building.

- Unplanned high pressure coolant injection valve failure on February 3 and CAR 200400789. The inspectors reviewed the licensee's risk assessment for the auxiliary contactor failure.

b. Findings

No findings of significance were identified.

1R14 Personnel Performance During Nonroutine Plant Evolutions (71111.14)

a. Inspection Scope

The inspectors reviewed four nonroutine plant evolutions, events, and/or transient operations for personnel performance. The inspectors also considered 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 events:

- CAR 200400629, Reactor trip due to the failure of the main generator distance protection relay on January 27

- CAR 200400791, Reactor trip due to the failure of a main generator dead machine protection relay on February 3

- CAR 200401076, Safety injection (SI) due to operator error on February 11

- CAR 200401167, Reactor trip due to operator error on February 15

b. Findings

.1 Unplanned Reactor Trip Due to Operator Error

Introduction. A self-revealing Green finding and noncited violation (NCV) of Technical Specification 5.4.1, "Procedures," was identified after an operator error resulted in an unplanned reactor trip.

Description. On February 15, an operator error resulted in an unplanned reactor trip during power ascension. The reactor tripped from low SG water levels 28 minutes after operations personnel synchronized the main turbine-generator to the grid. Steam generator level oscillations began immediately after the generator output breakers were closed. The level oscillations were caused by a combination of SG shrinkage induced by decreasing feedwater temperatures and the high rate of load increase.

The licensee had not aligned extraction steam to the feedwater heaters before beginning the power ascension. Feedwater temperature dropped from 327°F to 240°F during the first 15 minutes of power ascension due to the lack of extraction heating steam. Plant procedures allowed plant operation below 25 percent generator load without feedwater heating in service. The high rate of load increase contributed to the level oscillations. The operator raised plant load about 120 MWe during the 17-minute transient. This power rate increase was equivalent to about 35 percent load change per hour. The operator controlled SG levels with the bypass feedwater regulating valves during power ascension.

The operator opened the main feedwater regulating valves about 15 minutes after the SG oscillations began in an attempt to dampen the level oscillations. The magnitude of the level oscillations increased dramatically after the main regulating valves were opened. SG levels reached the high level trip setpoint about 4 minutes later, resulting in a feedwater isolation and main turbine trip. SG levels quickly dropped to the low level reactor trip setpoint. The operator’s action to open the main feedwater regulating valves before the plant was stable was the direct cause of the reactor trip.

General Operating Procedure OTG-ZZ-00003, "Plant Startup Hot Zero Power to 30% Power," Revision 27, required the operator to stabilize turbine load at greater than 240 megawatts before transferring feedwater control to the main regulating valves. The turbine load was unstable at about 140 megawatts when the operator opened the main feedwater regulating valves.

Analysis. This finding is greater than minor because the reactor trip was a transient initiator affecting the initiating events cornerstone. The operator’s failure to follow the procedure was a performance deficiency which affected the human performance attribute of the initiating events cornerstone. The inspectors determined this finding to be of very low safety significance (Green) using the significance determination process for reactor inspection findings for at-power situations. This 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 equipment functions, and did not increase the likelihood of a fire or internal/external flood. This finding is similar to Example 4.b in Manual Chapter 0612, Appendix E.

Enforcement. Technical Specification 5.4.1.a required the licensee to implement the applicable procedures listed in Regulatory Guide 1.33, Revision 2, Appendix A. Regulatory Guide 1.33, Appendix A, Section 2.f, included procedures for changing load. General Operating Procedure OTG-ZZ-00003, "Plant Startup Hot Zero Power to 30% Power," Revision 27, was used by the licensee for load changes under 30 percent power. Procedure OTG-ZZ-0003, Section 6.4.26, required the operator to stabilize turbine and reactor power at a load greater than 240 megawatts before transferring feedwater control from the bypass valves to the main feedwater regulating valves.

Contrary to the above, neither turbine nor reactor power was stable and load was not greater than 240 megawatts when the operator transferred feedwater control from the bypass to the main feedwater valves. Because of the very low safety significance and the licensee's action to place this issue in their corrective action program as CAR 200401167, this violation is being treated as an NCV in accordance with Section VI.A.1 of the Enforcement Policy (NCV 50-483/2004002-01). The human performance crosscutting aspects of this NCV are discussed in Section 4OA4.

.2 Loss of the TDAFW Pump During a Transient

Introduction. A self-revealing Green finding was identified after an unplanned loss of the TDAFW pump during a plant transient.

Description. On February 15, an operator error resulted in an unplanned SG level transient and reactor trip. The operator manually started the TDAFW pump to restore SG levels prior to the reactor trip. The reactor subsequently tripped on low SG level. The low SG level also generated an automatic TDAFW pump start signal. After the reactor trip, the operator observed a reactor coolant system cooldown. The operator closed both auxiliary feedwater turbine steam supply valves and the trip and throttle valve to limit the cooldown rate by reducing steam loads. All three valves immediately reopened when the hand switches were released because the TDAFW pump automatic initiation signal was still present. The TDAFW turbine tripped on overspeed. The overspeed condition resulted because the turbine was still rotating at about 95 rpm when the valves reopened. The operator had been trained to take manual control of the turbine and lower the speed demand signal to reduce the steam load and the reactor cooldown rate instead of closing the turbine steam supply valves. This action would have preserved the availability of the TDAFW pump. The licensee placed this issue into the corrective action program as CAR 200401167.

Analysis. This finding was greater than minor because the loss of the TDAFW pump affected the availability/reliability objective of the mitigating system equipment performance cornerstone. The inspectors concluded that the method used to secure the TDAFW pump was a performance deficiency. Because this finding involved the loss of availability of a mitigating system, the inspectors evaluated this finding using the significance determination process for reactor inspection findings for at-power situations. The inspectors concluded that this finding was only of very low safety significance because:

- it was not a design or qualification deficiency,
- it did not represent the actual loss of the safety function of a system,
- it did not represent the actual loss of the safety function of a single train for greater than its Technical Specification allowed outage time,

- it did not represent the loss of a non-Technical Specification related train (designated as risk significant per 10 CFR 50.65 a(4)) for greater than 24 hours, and

- it did not screen as potentially risk significant due to a seismic, fire, flooding, or severe weather initiating event.

Enforcement. No violation of regulatory requirements occurred. The inspectors determined that this finding did not represent a noncompliance because the operator's actions were consistent with plant procedures. Normal Operating Procedure OTN-AL-00001, "Auxiliary Feedwater System," Revision 13, allowed the operator to shutdown the TDAFW pump by closure of the trip and throttle valve (FIN 50-483/2004002-02). The human performance crosscutting aspects of this NCV are discussed in Section 4OA4.

.3 SI Due to Operator Error

Introduction. A self-revealing Green finding and NCV of Technical Specification 5.4.1, "Procedures," was identified after an operator error resulted in an unplanned SI during plant heatup operations.

Description. On February 11, an operator error resulted in an unplanned SI and main steamline (MSL) isolation during a reactor heatup. The SI was generated from the combination of reactor pressure above the Permissive P-11 setpoint (1,970 psig) and MSL pressure less than 615 psig. The plant heatup procedure required the operator to place the pressurizer pressure control in automatic, controlling at 1,900 psig, until the reactor coolant system temperature was increased to greater than 500°F. This action would have insured that the MSL pressure would have been above the 615 psig SI reactor protection setpoint prior to the automatic reset of Permissive P-11. The operator failed to place the pressurizer pressure control in automatic prior to proceeding with the reactor heatup.

Analysis. This finding is greater than minor because the SI was a transient initiator contributor affecting the initiating events cornerstone. The operator's failure to follow the procedure was a performance deficiency which affected the human performance attribute of the initiating events cornerstone. The inspectors evaluated this finding using the significance determination process for reactor inspection findings for at-power situations. The inspectors concluded that this 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.

Enforcement. Technical Specification 5.4.1.a required the licensee to implement the applicable procedures recommended in Regulatory Guide 1.33, Revision 2, Appendix A. Regulatory Guide 1.33, Appendix A, Section 2.a, included procedures used to heatup the plant from cold shutdown to hot standby. General Operating

Procedure OTG-ZZ-00001, "Plant Heatup Cold Shutdown to Hot Standby," Revision 33, was used by the licensee for plant heatup. Procedure OTG-ZZ-0001, Section 6.4.13 required the operator to place the pressurizer pressure control in automatic at 1,900 psig until the reactor coolant system temperature was increased to greater than 500°F. Contrary to the above, the operator did not place the pressurizer pressure control in automatic at 1,900 psig until the reactor coolant system temperature was increased to greater than 500°F. Because of the very low safety significance and the licensee’s action to place this issue in their corrective action program (CAR 200401076), this violation is being treated as an NCV in accordance with Section VI.A.1 of the Enforcement Policy (NCV 50-483/2004002-03). The human performance crosscutting aspects of this NCV are discussed in Section 4OA4.

1R15 Operability Evaluations (71111.15)

a. Inspection Scope

The inspectors reviewed six 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:

- CAR 200400798, Operability determination of elevated TDAFW turbine control oil temperature on February 7
- CAR 200401546, Operability determination of pressure binding and duel indication in fire protection Velan parallel slide gate Valve KCHV0253 on March 1
- CAR 200400641, Operability determination of Channel 12 of loose parts monitoring system out of service on January 27
- CAR 200400798, Operability determination of the TDAFW pump with casing joint steam leak on February 6
- CAR 200400717, Operability determination of degraded fire barriers on January 30
- CAR 200401780, Operability determination of water intrusion into lubricating oil of the TDAFW pump discovered on March 9

b. Findings

No findings of significance were identified.

1R16 Operator Workarounds (71111.16)

a. Inspection Scope

The inspectors completed one evaluation of the effect of one operator workaround during the inspection. The inspectors also reviewed the first quarter 2004 Operator Workaround List and the affect of the workarounds on the ability of operators to implement plant emergency operating procedures (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 workaround on the failure of compressor air system service air supply pressure control Valve KAPV0011 on February 24. The inspectors also attended a plant monthly operator workaround meeting on March 23.

b. Findings

No findings of significance were identified.

1R19 Postmaintenance Testing (71111.19)

a. Inspection Scope

The inspectors reviewed nine postmaintenance tests 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:

- Retest R720365A, performed on February 14, following corrective maintenance on main feedwater pump Valve FCHV0312. The inspectors performed an in-office review of the completed test package.

- Retests R71977B and R71977C, performed on February 14, following corrective maintenance on the TDAFW turbine control system. The inspectors observed portions of the test from the TDAFW pump room and the control room. The inspectors also performed an in-office review of the completed test packages.

- Retest R719821B, TDAFW pump flow to SG B, performed on February 14, following corrective maintenance on the TDAFW pump turbine control system and repair of the turbine governor valve. The inspectors performed an in-office review of the completed test package.

- Retest R547596A, performed on January 13, following preventive maintenance on 480 volt feeder Breaker NG03CCF4. The inspectors performed an in-office review of the completed test package.

- Retest R231288A, performed on January 13, following preventive maintenance on auxiliary feedwater level control Valve ALHV00008. The inspectors performed an in-office review of the completed test package.

- Retest R645003A and R645003A, performed on January 30, following corrective maintenance on ESW emergency makeup Valve EFHV0044. The inspectors performed an in-office review of the completed test package and walked down the affected components in the auxiliary building.

- Retest 227581A, performed on January 28, following modifications to containment spray Pump B. The inspectors performed an in-office review of the completed test package and walked down the affected components in the auxiliary building.

- Retest 232705B, performed on February 7, following corrective maintenance on safety injection header Valve EPHV8880. The inspectors performed an in-office review of the completed test package and walked down the affected components in the auxiliary building.

- Retests of the TDAFW pump performed on February 13 and 14, following an overspeed trip event. The inspectors performed an in-office review of the completed test package.

b. Findings

No findings of significance were identified.

1R20 Refueling and Outage Activities (71111.20)

a. Inspection Scope

The inspectors evaluated three forced outages to verify that the licensee considered shutdown risk in developing outage schedules, adhered to administrative risk reduction methodologies to control plant configuration, developed mitigation strategies for losses of key safety functions, and adhered to operating license and Technical Specification requirements that ensured defense-in-depth. The inspectors observed portions of the reactor cooldown and heatup processes to verify that Technical Specification restrictions were followed by the licensee. The inspectors also observed portions of plant startups and outage control of equipment. Licensee activities during the forced outages began January 27, February 3, and February 15.

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 systems listed 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 S717273 and Operations Surveillance Procedure OSP-EG-0001B, "CCW Valve Alignment Surveillance - Train B," Revision 2. The inspectors completed an in-office review on January 4 and an auxiliary building walkdown on January 12.

- Surveillance S717888 and Operations Surveillance Procedure OSP-EG-V0002A, "CCW Train 'A' Containment Isolation Valves Inservice Test," Revision 5. The inspectors completed an in-office review on January 4.

- Surveillance P7156761, Inspection/Surveillance of high radiation areas. The inspectors observed the licensee perform the surveillance in the auxiliary and radioactive waste buildings on January 9.

- Surveillance S716356, Reactor trip breaker test, CAR 200400013, and Operations Surveillance Procedure OSP-SB-0001B, "Reactor Trip Breaker "B" Trip Actuating Device Operational Test," Revision 12. The inspectors completed an in-office review and observed portions of the test from the control room on January 24.

- Operations Surveillance Procedure OSP-NE-0001A, "Standby Diesel Generator 'A' Periodic Test," Revision 13. The inspectors observed the testing in the emergency diesel generator building and control room on January 15.

b. Findings

No findings of significance were identified.

1R23 Temporary Plant Modifications (71111.23)

a. Inspection Scope

The inspectors sampled two 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 modifications 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," Revision14.

- Temporary Modification of the emergency preparedness sirens affected by the Federal Signal electronics failure. The inspectors completed an in-office review of the modification on February 9.

- Temporary Modification TPM 04-0005, disablement of the ESW ultimate heat sink heat tracing control circuit. The inspectors walked down the affected plant equipment in the ultimate heat sink cooling tower on February 2. The inspectors also reviewed CAR 200400753, ESW Train B inoperable due to a failed heat trace on ultimate heat sink riser.

b. Findings

No findings of significance were identified.

Cornerstone: Emergency Preparedness

1EP2 Alert Notification System Testing (71114.02)

a. Inspection Scope

The inspectors evaluated the adequacy of licensee methods for testing the alert and notification system in accordance with 10 CFR Part 50, Appendix E. The alert and notification system testing program was evaluated against the criteria contained in NUREG-0654, “Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants,” Revision 1, Federal Emergency Management Agency Report REP-10, “Guide for the Evaluation of Alert and Notification Systems for Nuclear Power Plants,” and the licensee’s current Federal Emergency Management Agency-approved alert and notification system design report. The inspectors also reviewed the documents described in the attachment to this report. The inspectors completed one sample during this inspection.

b. Findings

No findings of significance were identified.

1EP3 Emergency Response Organization Augmentation Testing (71114.03)

a. Inspection Scope

The inspectors reviewed the results of an October 7, 2003, unannounced off-hours call-in augmentation drill. The inspectors also interviewed members of the emergency planning staff responsible for training and testing of the emergency response organization. The inspectors evaluated drill performance and training implementation against emergency plan implementation procedures and other documents related to the emergency response organization augmentation system to determine the licensee personnel's ability to staff emergency response facilities in accordance with their emergency plan and the requirements of 10 CFR Part 50, Appendix E. The inspectors completed one sample during this inspection.

b. Findings

No findings of significance were identified.

1EP5 Correction of Emergency Preparedness Weaknesses and Deficiencies (71114.05)

a. Inspection Scope

The inspectors reviewed a summary of all CAR system action requests associated with emergency preparedness generated between October 2002 and March 2004, to determine the licensee's ability to identify and correct problems in accordance with the requirements of 10 CFR 50.47(b)(14) and 10 CFR Part 50, Appendix E. The inspectors also reviewed four drill reports, two self-assessments, two quality assurance audits, 22 specific action requests, and other documents listed in the attachment to this report. Corrective actions were evaluated against the requirements of Procedure APA-ZZ-00500, “Corrective Action Program,” Revision 34. The inspectors completed one sample during this inspection.

b. Findings

No findings of significance were identified.

1EP6 Drill Evaluation (71114.06)

a. Inspection Scope

The inspectors observed one licensee emergency drill to evaluate the adequacy of the drill and to verify proper emergency action level classification and protective action recommendations. The inspectors observed the rapid responder drill from the control room simulator and Technical Support Center on February 25. 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 31; and Emergency Plan Implementing Procedure EIP-ZZ-00201, "Notifications," Revision 38, to evaluate licensee performance.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA1 PI Verification (71151)

.1 Emergency Preparedness Cornerstone:

a. Inspection Scope

The inspectors sampled submittals for the PIs listed below for the period from April 1 through December 31, 2003. The definitions and guidance of Nuclear Engineering Institute 99-02, “Regulatory Assessment Indicator Guideline,” were used to verify the licensee’s basis for reporting each data element in order to verify the accuracy of PI data reported during the assessment period.

- Drill and exercise performance
- Emergency response organization participation
- Alert and notification system reliability

The inspectors reviewed a 100 percent sample of drill and exercise scenarios, licensed operator simulator training sessions, notification forms, and attendance and critique records associated with training sessions, drills, and exercises conducted during the verification period. The inspectors reviewed the qualification, training, and drill participation records for a sample of 10 emergency responders. The inspectors reviewed alert and notification system maintenance records and procedures, and a 100 percent sample of siren test results. The inspectors also interviewed licensee personnel that were responsible for collecting and evaluating the performance indicator data. The inspectors completed three samples during this inspection.

b. Observations

The inspectors reviewed the drill records for the unannounced off-hours call-in drill conducted on October 7, 2003. This drill required the augmenting emergency response organization to respond to the site after being notified of a simulated event in progress at the Callaway Plant. The drill was evaluated for contribution to the drill and exercise performance and the emergency response organization emergency preparedness PIs. The inspectors noted that the drill scenario did not involve development of protective action recommendations, since there were no radiological consequences onsite or offsite and the scenario only escalated to a Site Area Emergency declaration.

The inspectors asked how drill participation credit was evaluated for the dose assessment coordinator and protective measures coordinator. Both of these positions are designated as key emergency response organization members in the emergency operations facility, whose primary function is related to formulation and review of protective action recommendations. Nuclear Engineering Institute 99-02, “Regulatory Assessment Performance Indicator Guidelines,” Revision 2, Section 2.4, states that drill participation credit can only be given to key emergency response organization members when the drill contributes to the drill and exercise PI and is a performance enhancing experience related to the key member's primary function, which in this case was evaluation of protective action recommendations. The licensee stated that a specific evaluation was not done for each drill to verify that the Nuclear Engineering Institute participation guidance was met. The inspectors characterized this as an area for improvement in the PI program at the Callaway Plant. The licensee wrote CAR 200402033, "Evaluation of DEP PI for participation," to address this observation. The licensee also verified and the inspectors agreed that the reported emergency response organization PI would not have been affected.

.2 Reactor Safety Cornerstone

a. Inspection Scope

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

- Safety system unavailability - emergency ac power system
- Reactor coolant system specific activity
- Reactor coolant system leak rate

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 emergency ac power system. 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)

1. Annual Sample Review

a. Inspection Scope

The inspectors reviewed performance and facility problems documented in the licensee's corrective action program for calendar years 2003 and 2004. The inspectors selected 22 action requests for detailed review based on their impact on risk significant planning standards, emergency worker protection, and the ability to staff and maintain emergency response facilities. The selected action requests were reviewed to ensure that the full extent of the issues were identified, an appropriate evaluation was performed, appropriate corrective actions were identified and prioritized, and effective corrective actions were completed. The inspectors evaluated the action requests against the requirements of Procedure APA-ZZ-00500, "Corrective Action Program," Revision 34.

b. Findings and Observations

No findings of significance were identified.

2. Routine Review of Identification and Resolution of Problems

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 were identified, the licensee performed appropriate evaluations, and corrective actions were specified and prioritized. The inspectors evaluated the reports against the requirements of Administrative Procedure APA-ZZ-00500, "Corrective Action Program," Revision 34, and 10 CFR Part 50, Appendix B. The inspectors also attended the corrective action Management Oversight Committee Review Meeting on March 25. The inspectors reviewed the following two samples:

- CAR 200401174, failure to lock the containment hatch on a high radiation door on February 16

- CARs 200203882 and 200400789, failure of safety-related auxiliary contacts on February 10

b. Findings

Introduction. A licensee identified Green finding and NCV of Technical Specification 5.7.2 was associated with the failure to maintain a locked high radiation area.

Description. On February 16, a security officer identified that the containment hatch was unlocked. The licensee maintained the containment hatch as a locked high radiation entryway. Personnel had exited through the airlock about 5 hours prior to the discovery of the unlocked entryway. Access control documentation indicated that a heath physics technician had locked the door after egress and that a second technician had verified that the door was locked. The radiation dose rates inside containment exceeded 1.0 rem/hour at 30 centimeters from the source.

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. The failure of the two heath physics technicians to ensure the door was locked was a performance deficiency and affected the exposure/contamination control and monitoring attribute of the radiation safety cornerstone. The inspectors used the occupational radiation safety determination to analyze the significance of the finding because this issue involved the potential for workers to receive significant unplanned and unintended dose. This finding only had very low safety significance because the substantial potential for an overexposure did not exist and the licensee's ability to assess dose was not compromised. This finding is similar to Example 2.b in Manual Chapter 0612, Appendix E, and was entered into the licensee's corrective action program as CAR 200401174.

Enforcement. Technical Specification 5.7.2 requires that entryways to each high radiation area with dose rates greater than 1.0 rem/hour at 30 centimeters from the radiation source be locked or continuously guarded. The enforcement aspects of this NCV are discussed in Section 4OA7. The human performance crosscutting aspects of this NCV are discussed in Section 4OA4.

4OA3 Event Followup (71153)

1. (Closed) LER 50-483/2004-001-00: Manual initiation of the ESW system following the loss of normal service water.

   On November 12, 2003, an operator error resulted in an unplanned reactor power transient. The transient occurred following an 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 electrical alignment to an electrical bus following maintenance. Normal Operating Procedure OTN-PB-0001, "Non-Class 1E 4.16 KV Electrical System," required the operator to hold synchronization Switch 2201/2102 closed when closing the bus feeder Breaker PB122. The operator mistakenly held the incorrect synchronization switch closed, resulting in the loss of power. Plant operators quickly reduced reactor power from 100 to 66 percent to avoid a reactor trip.

   This issue was dispositioned as a finding of very low safety significance (FIN 50-483/2003006-01) in Callaway Plant Integrated Inspection Report 05000483/2003006 and was entered into the licensee's corrective action program as CAR 200308178. The inspectors reviewed the LER and no additional findings of significance were identified.

2. (Closed) LER 50-483/2003-009-00: Failure of an electrical inverter resulted in a Technical Specification required shutdown.

   On October 21, 2003, the licensee completed a Technical Specification required shutdown from 100 percent power following the failure of vital power Inverter NN11. The licensee determined that the inverter failed due to a faulted static transfer switch circuit board. The licensee completed repairs and restarted the unit on October 24. The inspectors reviewed the licensee's corrective actions and operating experience associated with inverter reliability. The inspectors reviewed the LER and no findings of significance were identified. The licensee documented the failed equipment in CAR 200307636.

3. (Closed) LER 50-483/2003-008-00: Technical Specification violation due to valve control circuit modification.

On September 4, 2003, the licencee completed modifications to a pressurizer power-operated relief block valve actuator circuit. The modification design required removal of an existing jumper from the valve breaker cubicle. The modification work instructions omitted the steps necessary for removal of this jumper connection. The jumper electrically bypassed the valve actuator open limit and torque switches after the valve was returned to service. The jumper resulted in the failure of the actuator because the motor remained energized after the valve reached the full open position. The licensee's failure to ensure the modification design was correctly translated into work instructions was a violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control."

This issue was dispositioned as a NCV and finding of very low safety significance (NCV 50-483/2003006-06) in Callaway Plant Integrated Inspection Report 05000483/2003006. The inspectors reviewed the LER and did not identify any additional findings.

4. (Closed) Unresolved Item URI 50-483/2003006-04: Failure of the licensee to maintain EOP E-3 consistent with the accident analysis.

A finding was identified for the failure of the licensee to maintain EOP E-3 consistent with the accident analysis. This finding was unresolved pending NRC completion of the significance determination process. This finding was more than minor because the EOP quality attribute of the barrier integrity cornerstone is affected by the procedural error.

Findings

Introduction. A licensee identified Green finding and NCV of Technical Specification 5.4, "Procedures," was identified following the failure to maintain EOP E-3 consistent with the accident analysis. Incorrect operator actions in EOP E-3 resulted in an increased postulated radiological dose to the public due to prolonged accident recovery time.

Description. On June 3, 2003, an error in EOP E-3, "Steam Generator Tube Rupture," was identified during development of licensed operator training. Emergency Operating Procedure E-3 required the operator to arm the pressurizer power-operated relief valves (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 operator's ability to terminate 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 PORVs. The increased source term resulted in increased postaccident public radiation dose.

During 1988, the licensee added the step to EOP E-3 to arm the PORVs based on the Westinghouse Emergency Response Guides. The plant response was affected after Refueling Outage 11 when the licensee modified the COP control circuit. The modification removed the pressurizer PORV interlock, which allowed 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 more than minor because the EOP quality attribute of the barrier integrity cornerstone is affected by the procedural error. The inspectors evaluated the condition with the significance determination process Phase 2 worksheet, because this finding involved the reactor coolant system barrier. The SG tube rupture was the dominant core damage sequence in the Phase 2 analysis. The inspectors assumed that the operator was able to depressurize the reactor coolant system to less than the SG relief valve setpoint using at least one of the pressurizer PORVs and two of the steam generator PORVs. The inspectors assumed an initiating event likelihood of between 100 and 1,000 years. This finding was only of very low safety significance because the primary system depressurization delay did not significantly contribute to an increase in core damage frequency.

Enforcement. The failure to maintain EOP E-3 consistent with the accident analysis was a violation of Technical Specification 5.4, “Procedures.” The enforcement aspects of the violation are discussed in Section 4OA7. This issue was entered into the licensee’s corrective action program as CAR 200304922.

4OA4 Crosscutting Aspects of Findings (71152)

Section 1R14 of this report documents three human performance errors:

- Failure of an operator to follow procedures resulted in an unplanned reactor trip.
- An operator error resulted in the unplanned loss of the TDAFW pump during a plant transient.
- The failure of an operator to follow procedures resulted in an unplanned SI and MSL isolation.

Section 4OA2 of this report documents a human performance error resulting in an unattended and unlocked high radiation entryway.

4OA5 Other

(Closed) Unresolved Item 05000483/2003007-04: Inadequate smoke alarm response procedure for control room ventilation supply.

Introduction. A Green NCV was identified for an inadequate smoke alarm response procedure required by Technical Specification 5.4.1.a and Regulatory Guide 1.33.

Description. During the triennial fire protection inspection (NRC Inspection Report 05000483/2003007) the team identified that the alarm response procedure for responding to smoke in the control room outside supply duct was inadequate because it did not direct isolating outside air makeup upon receipt of the alarm. This alarm would not cause an automatic isolation of the control room, so operators must recognize the condition and take manual action to prevent losing control room habitability. Loss of control room habitability could cause operators to evacuate the control room, which would necessitate manually tripping the plant, establishing control of the plant at the alternate shutdown panel, and manually operating the plant with limited control and indication equipment. The significance of this finding had not been determined at the conclusion of the inspection.

Analysis. This issue was more than minor because failure to isolate the control room ventilation could lead to unnecessary evacuation, which would result in a plant transient and disabling much of the mitigation equipment that would otherwise be available. This affected the procedure quality attribute for the mitigating systems cornerstone objectives.

During the current inspection period, the Region IV Senior Reactor Analyst performed a Phase 3 analysis to determine the risk associated with this violation. This evaluation concluded that the violation was of very low safety significance. This was based on the following analysis:

- The probability of a fire in any given square mile of Missouri was 1.1E-3/yr, based on state-wide fire data reported in the Hannibal (Missouri) Courier Post article printed on November 2, 2003. The frequency of a fire within 2 miles of the plant in the 45 degree upwind sector was 1.8E-3/yr.

- The analyst assumed that 90 percent of the fires of concern would be reported to the control room by personnel outside who would observe the source in time for operators to take appropriate actions. Therefore the fraction of fires not reported would be 0.1.

- The combined generic error probability for diagnosing the problem (1E-2) and determining the appropriate action (1E-3) was determined to be 1.1E-2 using the draft SRAR-H Method (INEEL/EXT-02-10307). The diagnosis error probability was conservatively increased by a factor of five for the evaluation case due to the inadequate response procedure, resulting in a total error probability of 5.1E-2.

- In accordance with NRC Manual Chapter 0609, Appendix F, an evacuation of the control room results in a conditional core damage probability of 0.1.

Therefore the core damage frequency for the base case would be
(1.8E-3/yr)(0.1)(1.1E-2)(0.1) = 2.0E-7/yr.

The core damage frequency for the evaluated case would be
(1.8E-3/yr)(0.1)(5.1E-2)(0.1) = 9.2E-7/yr.

The change in core damage frequency associated with the performance issue is therefore 9.2E-7/yr - 2.0E-7/yr = 7.2E-7/yr.

The analyst considered these results to be bounding of the actual risk because of the use of conservative assumptions. The results indicated that this issue had very low (Green) significance.

Enforcement. Alarm Response Procedure OTA-KC-00008, Window 119/157, “Auxiliary Building Control Building Supply Air Supply Alarm,” Revision 9, a procedure required by Technical Specification 5.4.1.a and Regulatory Guide 1.33, was determined to be inadequate because it did not contain steps to secure outside makeup to ensure control room habitability when smoke was detected in this duct. Because this violation was of very low safety significance and has been entered into the licensee's corrective action program (CAR 200306977), this violation is being treated as a NCV, consistent with Section VI.A of the NRC Enforcement Policy: NCV 05000483/2004002-04, Inadequate Smoke Alarm Response Procedure for Control Room Supply.

4OA6 Management Meetings

Exit Meeting Summary

On March 18, the emergency preparedness inspector presented preliminary inspection results to Mr. R. Affolter, Vice-President, and members of his staff.

On March 26, 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.

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 5.4, “Procedures,” required the licensee to maintain the EOPs. Contrary to the above, the licensee failed to maintain EOP E-3 consistent with the accident analysis. Incorrect operator actions in EOP E-3 resulted in an increased postulated radiological dose to the public due to prolonged accident recovery time. This was identified in the licensee’s corrective action program as CAR 200304922 and was reported as LER 50-483/2003-006-00. This finding was of very low safety significance because the primary system depressurization delay did not significantly contribute to an increase in core damage frequency.

- Technical Specification 5.7.2 requires that entryways to each high radiation area, with dose rates greater that 1.0 rem/hour at 30 centimeters from the radiation source, be locked or continuously guarded. Contrary to the above, the licensee left the containment hatch, a high radiation area entryway, unlocked and not continuously guarded on February 16. This finding only had very low safety significance because the substantial potential for an overexposure did not exist and the licensee’s ability to assess dose was not compromised. This finding is similar to Example 2.b in Manual Chapter 0612, Appendix E, and was entered into the licensee’s corrective action program as CAR 200401174.

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee

R. Affolter, Vice President - Nuclear
K. Brukerhoff, Supervisor, Emergency Preparedness
S. Crawford, Emergency Response Coordinator
M. Evans, Manager, Nuclear Engineering
L. Graessle, Superintendent, Protective Services
J. Hiller, Engineer, Regional Regulatory Affairs
D. Lewis, Emergency Response Coordinator
D. Neterer, Superintendent, Operations
G. Pendergraff, Evaluator, Protective Services
M. Reidmeyer, Supervisor, Regional Regulatory Affairs
W. Witt, Plant Manager
K. Young, Manager, Regulatory Affairs

LIST OF ITEMS OPENED AND CLOSED

Opened

50-483/2004002-01	NCV	Reactor trip during power ascension (Section 1R14)
50-483/2004002-02	FIN	Loss of the TDAFW pump during a transient (Section 1R14)
50-483/2004002-03	NCV	Safety injection due to operator error (Section 1R14)
05-483/2004002-04	NCV	Inadequate smoke alarm response procedure for control room supply (Section 4OA5)

Closed

50-483/2004002-01	NCV	Reactor trip during power ascension (Section 1R14)
50-483/2004002-02	FIN	Loss of the TDAFW pump during a transient (Section 1R14)
50-483/2004002-03	NCV	Safety injection due to operator error (Section 1R14)
50-483/2004-001-00	LER	Manual initiation of the ESW system following the loss of normal service water (Section 4OA3)
50-483/2003-009-00	LER	Failure of an electrical inverter resulting in a Technical Specification required shutdown (Section 4OA3)

50-483/2003-008-00	LER	Technical Specification violation due to valve control circuit modification (Section 4OA3)
50-483/2003006-04	URI	Failure of the licensee to maintain EOP E-3 consistent with the accident analysis (Section 4OA3)
05-483/2003007-04	URI	Inadequate smoke alarm response procedure for control room supply (Section 4OA5)
05-483/2004002-04	NCV	Inadequate smoke alarm response procedure for control room supply (Section 4OA5)

DOCUMENTS REVIEWED

Procedures

EIP-ZZ-A0020, “Maintaining Emergency Preparedness,” Revision 22

EIP-ZZ-00200, “Augmentation of the Emergency Organization,” Revision 11

EIP-ZZ-A0001, “Emergency Response Organization,” Revision 8

EIP-ZZ-A0066, “RERP Training Program,” Revision 7

KDP-ZZ-0008, “Emergency On-site Siren Acceptance Test Plan,” Revision 0

KSP-ZZ-0001, “Alert and Notification Availability,” Revision 6

KDP-ZZ-00400, “Emergency Preparedness 10 CFR 50.54(Q) Evaluations,” Revision 10

KDP-ZZ-00410, “Radiological Emergency Response Plan (RERP) Change Notice/Revision Process,” Revision 9

EIP-ZZ-00101, “Classification of Emergencies,” Revision 31

KDP-ZZ-02001, “Drill and Exercise Program,” Revision 1

KDP-ZZ-02000, “Performance Indicator (PI) Data Collection,” Revision 1

EIP-ZZ-00201, “Notification,” Revision 38

EIP-ZZ-00212, “Protective Action Recommendations,” Revision 21

ISF-EG-000L2, "Functional Level for CCW Surge Tank "B" Lvl," Revision 9

MDP-ZZ-P0002, Live Load Packing, Revision 4

OTG-ZZ-0006, Plant Cool down from Hot Shutdown to Cold Shutdown, Revision 30

OSP-AL-P0002, “ Turbine Drive Auxiliary Feedwater Pump Operability In-Service Test,” Revision 40

OSP-EF-V001B, ESW Train B Valve Operability, Revision 27

OSP-EP-V001, Safety Injection Accumulator Valve Operability, Revision 7

OSP-EN-P001B, Containment Spray Inservice Test, Revision 25

OSP-FC-V0001, “Auxiliary Feedwater Pump turbine Valve Operability Test,” Revision 21

OSP-SA-02416, “ESFAS Turbine Drive Auxiliary Feedwater Pump Response Time Test,” Revision 5

OSP-SB-0001B, "Reactor Trip Breaker "B" Trip Actuating Device Operational Test," Revision 11

OTN-NE-001A, Standby Diesel Generator System Train A, Revision 13

OTN-RLL-RK055, “Annunciator Response Procedure for Windows 55A Through 55F”

ODP-ZZ-00023, “Work Screening and Processing,” Revision 2

Drawings

Piping and Instrumentation Diagram M-22GS01, Containment Hydrogen Control System, Revision 7

Emergency Diesel Generator Piping and Instrumentation Diagram Drawings M-22KJ01 through M-22KJ06 (Revisions 16, 17, 16, 15, 16, and 15, respectively)

Callaway Action Requests

200003122	200303666	200303094	200303695	200308079	200400471
200200019	200304145	200303103	200304619	200308465	200400562
200204064	200304522	200303148	200305194	200308749	200400789
200207705	200301633	200303224	200306198	200308750	200402033
200302347	200302600	200303568	200306929		

Work Requests

W228049

Self-Assessment and Quality Verification

Surveillance Report, SP02-061, "Operability/Habitability of the Emergency Facilities," January 27, 2003

SA04-EP-F01, January 9, 2004

Emergency Preparedness Drill Reports

Medical Emergency Response Team Drills 03M02 - August 14, 2003, and 03M03 - November 6, 2003

2003 Team 2 RERP Drill, August 20, 2003

Rapid Responder Drills: July, August, November, and December 2003

Quality Assurance Audits

AP02-016, December 20, 2002
AP03-020, January 9, 2004

Surveillance Reports

SP04-003, January 28-30, 2004, Surveillance of activities related to the January 27, 2004, Reactor Trip

SP04-006, February 3-17, 2004, Surveillance of turbine-driven auxiliary feedwater pump activities subsequent to the pump trip of February 3, 2004

Miscellaneous

Procedure Checkoff List for Containment Hydrogen Control System, dated April 26, 2002

Callaway Radiological Emergency Response Plan, Revision 25

Alert and Notification System Design Report, January 7, 2002

Emergency Surveillance Test ST-12096, "Test Emergency Siren Alerting System," test results for January 2003 through March 2004

Siren testing schedules and reports for first quarter 2003 through first quarter 2004

Federal Signal Corporation, Quality Issue Investigation and Root Cause Analysis of Model UV400, February 9, 2004

NRC Event Report 40462

LIST OF ACRONYMS

AFW	auxiliary feedwater
CAR	Callaway action request
CCW	component cooling water
COP	cold overpressurization
EOP	emergency operating procedure
ESW	essential service water
FIN	finding
FSAR	Final Safety Analysis Report
LER	licensee event report
MSL	main steamline
NCV	noncited violation
PI	performance indicator
PORV	power-operated relief valve
SG	steam generator
SI	safety injection
TDAFW	turbine-driven auxiliary feedwater
URI	unresolved item