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

May 6, 2005

Charles D. Naslund, Senior Vice
  President and Chief Nuclear Officer
Union Electric Company
P.O. Box 620
Fulton, MO 65251

SUBJECT: CALLAWAY PLANT - NRC INTEGRATED INSPECTION REPORT 05000483/2005002

Dear Mr. Naslund:

On March 24, 2005, the NRC completed an inspection at your Callaway Plant. The enclosed report documents the inspection findings which were discussed on March 31, 2005, 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 four findings that were evaluated under the risk significance determination process as having very low safety significance (Green). The NRC has 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. Additionally, one licensee-identified violation which was determined to be of very low safety significance is listed in Section 4OA7 of this report. These NCVs are described in the subject inspection report. If you contest these 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 (if any) 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

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NRC Inspection Report
05000483/2005002
w/attachment: Supplemental Information

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ENCLOSURE

U.S. NUCLEAR REGULATORY COMMISSION
REGION IV

Docket:	50-483
License:	NPF-30
Report:	05000483/2005002
Licensee:	Union Electric Company
Facility:	Callaway Plant
Location:	Junction Highway CC and Highway O Fulton, Missouri
Dates:	January 1 through March 24, 2005
Inspectors:	M. S. Peck, Senior Resident Inspector D. E. Dumbacher, Resident Inspector F. L. Brush, Senior Resident Inspector E. L. Crowe, Resident Inspector B. W. Tindell, Reactor Inspector
Approved By:	D. N. Graves, Chief, Project Branch B

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SUMMARY OF FINDINGS

IR 05000483/2005002; 01/01 - 03/24/2005; Callaway Plant: Equipment Alignment, Fire Protection, Personnel Performance During Nonroutine Plant Evolutions, Operability Evaluations

This report covered a 3-month inspection by resident inspectors and a region-based reactor inspector. Three Green noncited violations, one Green finding, and one licensee-identified violation 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. <u>NRC-Identified and Self-Revealing Findings</u>

Cornerstone: Initiating Events

- Green. A self-revealing finding was identified after an unplanned reactor trip on January 19, 2005, resulted from the licensee's ineffective use of industry operating experience. The plant tripped from low steam generator level after a feedwater regulating valve closed. The regulating valve closed after a control power supply shorted during a maintenance activity. The power supply shorted because the maintenance workers had used an inadequate work instruction. A similar event occurred at the Beaver Valley Nuclear Plant during June 2003. The licensee failed to effectively use the operating experience when planning and performing the maintenance activity. The licensee's failure to properly revise an incorrect work package before proceeding with the work activity, a poor prejob brief, and organizational time pressures also contributed to the event. Additionally, the licensee’s evaluation of the event identified contributing causes as root causes and did not take into account the programmatic issues to include operating experience reviews into work instruction development procedures. This finding had crosscutting aspects regarding human performance (inadequate procedure) and problem identification and resolution in that the evaluation of root versus contributing causes was deficient.

  This finding is greater than minor because the procedural adequacy attribute of the initiating events cornerstone objective is affected. The inspectors concluded the reactor trip is a transient initiator, affecting the initiating events cornerstone. The inspectors determined this finding to be of very low safety significance because the condition did not contribute to both the likelihood of a reactor trip and the unavailability of mitigating equipment functions. This issue was documented in the licensee’s corrective action program as Callaway Action Requests 200500322 and 200500354 (Section 1R14).

Cornerstone: Mitigating System

- Green. The inspectors identified a noncited violation of 10 CFR Part 50, Appendix B, Criterion XVI, after the licensee's cause determination and corrective actions were ineffective to prevent recurrence of safety injection pump discharge pipe voiding. Plant Technical Specifications required the licensee to verify that the emergency core cooling system piping was full of water every 31 days. The licensee established a 20 percent maximum void fraction as the acceptance limit for the safety injection pump hot leg injection discharge piping. On seven occasions during the past 2 years the surveillance acceptance criteria was not met. This finding had crosscutting aspects regarding problem identification and resolution in that the licensee's actions to determine the cause of the repeated surveillance failures and to implement corrective actions were not effective in preventing recurrence of the condition.

  This finding is greater than minor because voiding in emergency core cooling system piping affected the reactor mitigating systems cornerstone and the equipment performance attribute to ensure availability of systems that respond to prevent core damage. This finding is only of very low safety significance because the condition was not a design or qualification deficiency confirmed to result in loss of function per Generic Letter 91-18; did not result in an actual loss of safety function of a system; did not increase the likelihood of a fire; and did not screen as potentially risk significant due to a seismic, flooding, or severe weather initiating event. This issue was documented in the licensee’s corrective action program as Callaway Action Request 200501092 (Section 1R04).

- Green. A self-revealing, noncited violation of Technical Specification 5.4.1.d, "Fire Protection Program," was identified on March 3, after the licensee inadvertently isolated all plant fire water suppression from the reactor, auxiliary, control, and turbine buildings during surveillance testing. The isolation occurred due to an inadequate surveillance testing procedure. The licensee identified the isolation of the fire loops after about 15 minutes. The licensee reestablished the fire water suppression system after about 1.5 hours. This finding had crosscutting aspects regarding human performance in that the procedure used was inadequate.

  The finding is greater than minor because the unplanned isolation of fire water was associated with the "Protection Against External Factors" attribute of the mitigating systems cornerstone and affected the cornerstone objective to ensure availability of systems designed to respond to initiating events. The inspectors used Manual Chapter 0609, Appendix F, "Fire Protection Significance Determination Process," to analyze this finding because the condition had an adverse affect on fire defense-in-depth strategies. The senior reactor analyst evaluated the finding based on a bounding calculation for each fire area affected by the loss of fire water in the plant. The analyst concluded a plant-wide fire mitigation probability of 4.3×10^{-6} over the 2-hour exposure period. The analyst assumed that the maximum conditional core damage probability for any fire area was bounded by probability used to assess fires requiring control room evacuation (CCDP=0.1). The maximum resulting core damage probability from internal fires over the 2-hour period was the product of the plant-wide fire mitigation probability and 0.1. This bounded the risk of the finding resulting in no greater increase in core damage frequency than 4.3×10^{-7}. The analyst concluded that a systematic search and assessment effort was beyond the intended scope of the fire protection significance determination process. Therefore, in accordance with NRC Inspection Manual Chapter 0612, "Power Reactor Inspection Reports," Section 05.04.c, regional management reviewed this finding and determined that it was of very low risk significance. This issue was documented in the licensee's corrective action program as Callaway Action Request 2000501378 (Section 1R05).

- Green. An NRC identified, noncited violation of Technical Specification 5.4.1.d, "Fire Protection Program," was identified after the licensee failed to maintain the minimum number of fire brigade members on site. The licensee was required to maintain at least five fire brigade members on site at all times. Between January 24 and February 9, the outside equipment operator was assigned to the fire brigade 68 percent of the time. However, the outside equipment operator spent about 80 percent of the shift outside of the protected area, including attending equipment at the river pumping station, located eight miles from the site. The inspectors concluded that full fire brigade staffing would have been delayed about 20 to 30 minutes if the activation occurred while the equipment operator was performing outside duties. This finding had crosscutting aspects regarding human performance in that full fire brigade staffing was not ensured. This finding also had crosscutting aspects regarding problem identification and resolution in that the issue was not properly evaluated following documentation in the corrective action program two times.

  This finding is greater than minor because the reactor safety mitigating systems cornerstone objective attribute to provide protection against external factors was affected. Manual Chapter 0609, Appendix F, "Fire Protection Significance Determination Process," does not address fire brigade performance deficiencies. Regional management review concluded this finding was of very low safety significance because it affected the fire prevention and administrative controls category and represented only a short duration degradation in fire brigade staffing. This issue was documented in the licensee's corrective action program as Callaway Action Request 200501985 (Section 1R05).

B. Licensee-Identified Violations

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

- A noncited violation of Technical Specification 5.4.1.d, "Fire Protection Program," was identified by the licensee for failing to maintain five of the six Halon gaseous fire suppression systems in the qualified configuration for an extended duration.

REPORT DETAILS

Summary of Plant Status: At the beginning of the inspection period, the Callaway Plant was operating at full power. An unplanned reactor trip occurred on January 19 following the failure of a control system power supply. The licensee restarted the plant on January 20 and operated at full power for the remainder of the inspection period.

6. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity

1R01 Adverse Weather Protection (71111.01)

a. Inspection Scope

The inspectors reviewed the licensee's site preparation for an anticipated severe weather condition (one inspection sample). The inspectors performed a detailed review of the station's adverse weather procedure, Emergency Implementing Plan Procedure EIP-ZZ-00231, "Response to Severe Thunderstorm/High Winds/Tornado Watches and Warnings," Revision 17. The inspectors also performed a walkdown of the outside areas near the ultimate heat sink pond to verify that the licensee properly implemented required administrative controls. The inspectors performed the walkdown on March 22 while thunderstorm conditions were forecast for the area.

b. Findings

No findings of significance were identified.

1R04 Equipment Alignment (71111.04)

a. Inspection Scope

Partial System Walkdowns. The inspectors completed three partial system walkdowns during the inspection period (three inspection samples). The inspectors performed the walkdowns to verify component alignment and subsystem operability. The inspectors used the Final Safety Analysis Report (FSAR), Technical Specifications (TSs), and the procedures and drawings listed in the attachment as the bases for acceptability. The following systems were included in the scope of this inspection:

- Safety injection (SI) system, Train B while the redundant train was out of service for scheduled maintenance. The inspectors walked down components located in the auxiliary and control buildings on January 4.

- Containment spray system, Train A while the redundant train was out of service for scheduled maintenance. The inspectors walked down components located in the auxiliary and control buildings on January 26.

- Essential service water, Train B while the redundant train was out of service for scheduled maintenance. The inspectors walked down components located in the auxiliary and control buildings on March 8.

Complete System Walkdown. The inspectors conducted a detailed review of the alignment and condition of the SI system on February 18 and 22 (one inspection sample). The inspectors completed a walkdown of system components located in the control and auxiliary buildings. The inspectors used the FSAR, TSs, and procedures and drawings listed in the attachment to verify proper system alignment. The inspectors also performed a review of the system health reports to determine whether the licensee had identified any significant maintenance problems with the system.

b. Findings

Ineffective Corrective Actions Associated with Emergency Core Cooling System (ECCS) Pipe Voiding

Introduction. The inspectors identified a Green noncited violation (NCV) of 10 CFR Part 50, Appendix B, Criterion XVI, after the licensee's cause determination and corrective actions were ineffective to prevent recurrence of SI pump discharge pipe voiding.

Description. TS Surveillance Requirement (SR) 3.5.2.3 required the licensee to verify the ECCS piping was full of water every 31 days. The licensee established a 20 percent maximum void fraction as the acceptance limit for the SI pump hot leg injection discharge piping. The acceptance criteria was based on established system operability limits to prevent pipe water hammer and potential transfer of noncondensable gases into the reactor vessel. The licensee determined that SI Train A discharge pipe voiding exceeded the acceptance limit on the following occasions:

- February 4, 2003 (Callaway Action Request (CAR) 200300943)
- March 4, 2003 (CAR 200301933)
- May 3, 2003 (CAR 200303454)
- November 5, 2004 (CAR 200408383)
- January 28, 2005 (CAR 200500584)
- February 4, 2005 (CAR 200500756)
- March 16, 2005 (CAR 200501654)

The licensee implemented the ECCS surveillance requirement by timing the duration noncondensable gases were vented at the outboard SI containment isolation vent valve. The licensee correlated the vent duration to the amount of voiding that was present in the piping. The licensee also used ultrasonic technology to measure the voiding internal to the discharge pipe. The licensee removed all noncondensable gases from the vent path at the conclusion of each surveillance. Noncondensable gases would accumulate in the discharge piping during each surveillance interval.

The inspectors concluded the cause of SI pipe voiding was inadequate system fill and vent following maintenance activities. The licensee dynamically filled the SI discharge piping during past refueling outages. The dynamic fill was performed by using the SI pumps to discharge fluid to the reactor vessel hot legs. Maintenance activities performed subsequent to the dynamic fill, such as replacement of the pump discharge relief valve during May 2004, left the discharge pipe voided. The licensee's effort to statically fill the piping following maintenance was unsuccessful, leaving sections of noncondensable gases trapped in long horizontal pipe sections. The inspectors concluded that these noncondensable gases would migrate over the surveillance interval from the horizontal pipe sections and accumulate at the surveillance vent point. The licensee's actions to determine the cause of the repeated surveillance failures and to implement corrective actions were not effective to identify the cause and prevent recurrence of the condition.

Analysis. The licensee's failure to identify the cause and implement corrective actions to prevent repetitive SI discharge pipe voiding was a performance deficiency. The inspectors used the at-power significance determination process to analyze this finding. This finding is greater than minor because gas accumulation in ECCS piping affected the reactor mitigating systems cornerstone and the equipment performance attribute to ensure availability of systems that respond to prevent core damage. This finding is only of very low safety significance because the condition was not a design or qualification deficiency confirmed to result in loss of function per Generic Letter 91-18; did not result in an actual loss of safety function of a system; did not increase the likelihood of a fire; and did not screen as potentially risk significant due to a seismic, flooding, or severe weather initiating event. This finding had problem identification and resolution crosscutting aspects because licensee personnel did not adequately identify the negative trend of failed surveillances or properly evaluate the cause of the repetitive condition.

Enforcement. Title 10 CFR, Part 50, Appendix B, Criterion XVI, "Corrective Action," required the licensee to determine the cause of significant conditions adverse to quality and to implement corrective actions to preclude repetition. Contrary to the above, the licensee failed to determine the cause of the SI pump voiding, a significant condition adverse to quality, and to implement corrective actions to preclude repetition. Because of the very low safety significance of this finding, and the licensee’s action to place this issue in their corrective action program as CAR 200501092, this violation is being treated as an NCV in accordance with Section VI.A.1 of the Enforcement Policy (NCV 05000483/2005002-01).

1R05 Fire Protection (71111.05)

Routine Fire Inspection Walkdowns

a. Inspection Scope

The inspectors performed eight fire zone walkdowns to verify that the licensee maintained fire areas in accordance with the Fire Hazards Analysis Report (eight inspection samples). The fire zones were chosen based on their risk significance as described in the individual plant examination of external events. The walkdowns focused on control of combustible materials and ignition sources, operability and material condition of fire detection and suppression systems, and the material condition of passive fire protection features. The following fire zones were inspected:

- Fire Area A-13, Auxiliary feedwater pump room on January 9
- Fire Area A-14, Auxiliary feedwater pump room on January 9
- Fire Area A-15, Auxiliary feedwater pump room on January 12
- Fire Area C-1, Control building elevation 1974 piping space on January 11
- Fire Area A-28, Auxiliary shutdown panel room on January 11
- Fire Areas C-13 and 14, Class 1E air conditioning rooms on January 12
- Fire Area A-2, Safety-related pump area rooms on February 18
- Fire Area A-4, Safety-related pump area, Rooms 1107-1110 on February 23

b. Findings

.1 Unplanned Loss of Water Fire Suppression Due to Inadequate Testing Procedure

Introduction. A Green, self-revealing NCV of TS 5.4.1.d, "Fire Protection Program," was identified after the licensee inadvertently isolated the plant fire water suppression from the reactor, auxiliary, control, and turbine buildings during surveillance testing.

Description. On March 3, licensee personnel inadvertently isolated both fire water loop supply headers during surveillance testing. The isolation resulted in the unplanned loss of all fire water to the reactor, auxiliary, control, and turbine buildings. The isolation occurred during the performance of Engineering Surveillance Procedure ESP-KC-03003, "Fire Main Flow Test," Revision 9. The surveillance implemented the triennial flow test of the system loops as required by FSAR 9.5.1.6, Table 9.5.12 "Fire Protection System Requirements." At Callaway, either of the two fire loops can provide water suppression to the plant buildings. Procedure ESP-KC-03003, Section 6.2, removed Loop 1 from service, tested, and returned it to service. Section 6.3 repeated the evolution for Loop 2. While not explicitly addressed in the procedure, the licensee intended to restore flow in Loop 1 before removing Loop 2 from service. However, due to poor procedural placekeeping, the licensee removed Loop 2 from service before restoring Loop 1. The licensee discovered that Loop 1 was still isolated about 15 minutes after isolating Loop 2. Although the system could have been restored immediately if required, the licensee systematically restored Loop 1 to service to prevent inadvertent actuation of suppression systems or inducing a water hammer. This activity took about 1.5 hours.

The inspectors concluded that Procedure ESP-KC-03003 was less than adequate. The procedure was in an outdated format, did not specify the performance sequence, and did not alert the performer that an incorrect sequence could result in the disruption of station water fire suppression or require continuous use or placekeeping. Administrative Procedure APA-ZZ-00101, "Preparation, Review, and Approval of Written Instructions," Revision 40, required Procedure ESP-KC-03003 to have been upgraded to the correct format during previous revisions. The licensee revised Procedure ESP-KC-03003 in October 2004 and February 2005 without performing the upgrade. Also, Procedure APA-ZZ-00101 required procedures that could either affect safety functions or impact plant operations to be designated as "continuous use." Administrative Procedure APA-ZZ-00100, "Use and Adherence to Procedure and Written Instructions, Revision 17, Section 4.6, required placekeeping to be used on all "continuous use" procedures. The licensee had not incorporated the required placekeeping measures into Procedure ESP-KC-03003.

Analysis: In accordance with NRC Inspection Manual Chapter 0612, Appendix B, "Issue Screening," the inspectors determined that the failure to properly realign one subsystem of the fire water system prior to removing the redundant subsystem from service was a licensee performance deficiency because the system was required to remain in service. The issue was more than minor because the finding was associated with the "Protection Against External Factors," attribute of the mitigating systems cornerstone and affected the cornerstone objective to ensure availability of systems designed to respond to initiating events, because the complete fire water system function was made unavailable. This finding had crosscutting aspects regarding human performance in that the procedure utilized was inadequate.

In reviewing this finding, a senior reactor analyst performed the following bounding calculation for each fire area in the plant:

- The fire ignition frequency (or likelihood) per year calculated for the specified fire compartment at Callaway was taken from Table 4.3.3.1-3, "Fire Ignition Frequency Results for Callaway Plant Fire Compartments," on pages 4-60 of the Individual Plant Examination of External Events submitted by Union Electric/Callaway to the U.S. Nuclear Regulatory Commission on June 30, 1995.

- The analyst selected a bounding factor of 0.9 to represent the probability that a given fire would either be self-limiting or be extinguished by manual suppression prior to damaging additional risk-significant equipment in the compartment. Therefore, the analyst assumed that 10 percent of power plant fires would cause damage to equipment contained in the compartment.

- For each fire area, the analyst calculated the product of the fire ignition frequency and the nonsuppression factor discussed above. This product represented a bounding assessment of the fire mitigation frequency per year.

- For each fire area, the analyst then divided the fire mitigation frequency per year by 8760 hours/year, to obtain the fire mitigation frequency per hour.

- Finally, each fire mitigation frequency per hour was multiplied by the 2-hour exposure time for the deficiency. This resulted in the fire mitigation probability during the exposure time.

Using the data calculated above for each fire area, the analyst then performed the following calculations:

- The analyst calculated the sum of the fire mitigation probabilities for all fire areas to obtain a plant-wide fire mitigation probability of 4.3×10^{-6} over the 2-hour exposure period.

- The analyst assumed that the maximum conditional core damage probability (CCDP) for any fire area was the bounding value used to assess fires requiring control room evacuation (CCDP=0.1). As such, the maximum core damage probability possible from internal fires over a 2-hour period would be the product of the plant-wide fire mitigation probability and 0.1. This bounds the risk of the finding. Therefore, if a thorough evaluation of the subject finding were performed, the resulting change in core damage frequency could be no higher than 4.3×10^{-7}.

Additionally, in accordance with the NRC Inspection Manual Chapter 0609, Appendix F, "Fire Protection Significance Determination Process," assumptions and limitations, the significance determination process is intended to support the assessment of known issues only in the context of an individual fire area. A systematic search and assessment effort was considered beyond the intended scope of the fire protection significance determination process. Therefore, in accordance with NRC Inspection Manual Chapter 0612, "Power Reactor Inspection Reports," Section 05.04.c, regional management reviewed this finding and determined that it was of very low risk significance (Green).

Enforcement. TS 5.4.1.d, “Fire Protection Program,” required that the licensee establish, implement, and maintain written procedures for the fire protection program. FSAR 9.5.1.6, Table 9.5.12, "Fire Protection System Requirements," Item 2, “Fire Suppression Water System,” required a triennial flow test of the system fire loops. Contrary to the above, the licensee failed to maintain an adequate written procedure for the flow testing of the system loops. Because this finding is of very low safety significance and was entered into the licensee's corrective action program as CAR 200501378, this violation is being treated as an NCV, consistent with Section VI.A of the NRC Enforcement Policy (NCV 05000483/2005002-02).

.2 Fire Drill Observation

a. Inspection Scope

The inspectors observed an announced fire drill on February 14 (one inspection sample). The inspectors observed fire drill Scenario 05A0A, “Fire in the Demineralized and Potable Water Building,” to evaluate fire brigade performance. The inspectors evaluated fire brigade members as they donned protective clothing, entered the fire area, and utilized fire preplan strategies. The inspectors also evaluated fire brigade and control room communications, whether sufficient firefighting equipment was available, and the postdrill critique.

b. Findings

Introduction. A Green, NRC-identified, NCV of TS 5.4.1.d, "Fire Protection Program," was identified after the licensee failed to maintain the minimum number of fire brigade members on site.

Description. The inspectors identified that the licensee did not maintain minimum fire brigade staffing over an extended period. FSAR Section 16.12.1, "Administrative Controls, Unit Staff," required the licensee to maintain at least five fire brigade members on site at all times. The inspectors reviewed operations shift manning rosters between January 24 and February 9. During this period, the outside equipment operator (EO) was assigned to the fire brigade 68 percent of the time. However, the outside EO spent about 80 percent of the shift outside of the protected area, including attending equipment at the river pumping station, located 8 miles from the site. The inspectors concluded that full fire brigade staffing would have been delayed about 20 to 30 minutes if the activation occurred while the EO was performing duties outside of the plant site area. FSAR 2.1.1.2.1.1 defines the Plant Site Area and FSAR Figure 2.1-2 does not show the road to the river pumping station as part of the Plant Site Area.

On January 3, 2004, the outside EO was assigned to the fire brigade. The licensee initiated an unannounced fire drill. The shift supervisor determined that the outside EO, who was outside the plant protected area, would be delayed in responding to the drill. The shift supervisor reassigned the primary EO (the designated safe shutdown operator) to respond to the fire drill. This was documented in the fire drill critique as CAR 200400065. Reliance on the designated safe shutdown operator for fire brigade duties was inconsistent with plant minimum staffing requirements. FSAR 16.12.1, "Administrative Controls," excluded members of the minimum shift crew necessary for safe shutdown from fire brigade duties. Off-normal Operating Procedure OTO-ZZ-00001, “Control Room Inaccessibility,” established the primary EO as an individual responsible for performing designated safe shutdown activities. The inspectors identified a separate recognition by the licensee of potential decreased fire brigade effectiveness in the postevent critique following a September 18, 2004, fire (CAR 200408626). On the day of the fire, an extra EO was on the shift and assigned to the fire brigade rather than the outside EO. During the critique, fire brigade members raised the concern that, had the outside EO been assigned to the fire brigade, the individual would not have been available to respond to the fire in a timely manner. No action was taken by the licensee to address this concern. NRC Information Notice 95-33, “Switchgear Fire and Partial Loss of Offsite Power at Waterford Generating Station, Unit 3,” highlighted the importance of ensuring personnel are not assigned potentially conflicting duties.

Analysis. The licensee's failure to maintain minimum fire brigade staffing on site was a performance deficiency. This finding is greater than minor because the reactor safety mitigating systems cornerstone objective attribute to provide protection against external factors was affected. The inspectors referred to Manual Chapter 0609, Appendix F, "Fire Protection Significance Determination Process," to analyze this finding because the condition had an adverse affect on fire brigade effectiveness related to defense-in-depth strategies. Appendix F states that it does not apply to fire brigade performance deficiencies. Regional management review of this issue concluded this finding was of very low safety significance because it represented only a short duration staffing deficiency in the fire brigade. This finding, which was related to inadequate resources to ensure minimum fire brigade staffing, is associated with the crosscutting area of human performance. This finding, which was also related to a less than adequate condition evaluation, is also associated with the crosscutting area of problem identification and resolution.

Enforcement. TS 5.4.1.d required that the licensee maintain a fire protection program. FSAR 9.5.1.6, "Callaway Plant Fire Protection Program," and FSAR 9.5.1.8, "Callaway Plant Fire Brigade," established the fire protection program requirements for maintaining the fire brigade. FSAR 16.12.1, "Administrative Controls, Unit Staff," required the licensee to maintain five available fire brigade members on site at all times. Contrary to the above, the licensee failed to maintain five available fire brigade members on site at all times between January 24 and February 9. Because this finding is of very low safety significance and was entered into the licensee's corrective action program (CAR 200501985) this violation is being treated as an NCV, consistent with Section VI.A of the NRC Enforcement Policy (NCV 05000483/2005002-03).

1R06 Flood Protection Measures (71111.06)

a. Inspection Scope

The inspectors completed a flood protection walkdown of the essential service water (ESW) system, SI pump rooms, and charging pump rooms on March 21 (one inspection sample). 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, watertight doors, sumps, sump pumps, level alarms, and control circuits. The inspectors used FSAR, Section 3.4, "Water Level Flood Design," and Appendix 3B, "Hazard Analysis," as the bases for acceptability of the plant configuration.

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification Program (71111.11Q)

a. Inspection Scope

The inspectors observed one licensed operator continuing training simulator exercise, "Just In-Time Reactor Startup," and postscenario critique (one inspection sample) on January 20. The inspectors observed the exercise to assess operator performance during high-risk operator actions associated with reactor startup, lessons learned items, and plant operating experiences.

b. Findings

No findings of significance were identified.

1R12 Maintenance Effectiveness (71111.12Q)

.1 Routine Maintenance Effectiveness Inspection (71111.12Q)

a. Inspection Scope

The inspectors reviewed two equipment performance issues to assess the licensee's implementation of the maintenance rule program (two inspection samples). The inspectors conducted the review to verify that component performance problems were properly included in the scope of the licensee's maintenance rule program and that the appropriate performance criteria were established. The inspectors used the requirements outlined in 10 CFR 50.65 and Engineering Department Procedure EDP-ZZ-01128, "Maintenance Rule Program," as the bases of acceptance. The inspectors reviewed the following equipment performance problems:

- CAR 200409592, Maintenance preventable functional failure of the reactor coolant pump thermal barrier return Valve BBHV0016
- CAR 200500416, Maintenance preventable functional failure of the plant computer

b. Findings

No findings of significance were identified.

.2 Biennial Maintenance Rule Implementation (71111.12B)

a. Inspection Scope

Periodic Evaluation Reviews

The inspectors reviewed the licensee's overall implementation of the Maintenance Rule, 10 CFR 50.65, "Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants." The inspectors reviewed the licensee's Maintenance Rule periodic assessment for Cycle 13, which covered the licensee's Maintenance Rule Program activities for the period from November 26, 2002, through June 13, 2004. The resulting adjustments to the balance of equipment reliability and availability were also evaluated.

The inspectors reviewed systems and functions that had suffered some degraded performance or condition to assess the licensee's periodic evaluation activities. The inspectors selected the following six systems and functions for a detailed review:

- Main Generation
- Diesel Generators
- Freeze Protection
- Essential Service Water
- Containment Isolation
- 125 Volt dc

For these systems, the inspectors reviewed the use of performance history and operating experience in adjusting preventive maintenance, (a)(1) goals, and (a)(2) performance criteria. The inspectors also reviewed adjustments to the scope of the Maintenance Rule Program and adjustments to the definitions of availability hours and required available hours.

b. Findings

No findings of significance were identified.

.3 Identification and Resolution of Problems

a. Inspection Scope

The inspectors evaluated the use of the Corrective Action Program within the Maintenance Rule Program. The review was accomplished by the examination of a sample of corrective action documents and work orders. The purpose of the review was to determine that the identification of problems and implementation of corrective actions were acceptable.

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 risk assessments for planned or emergent maintenance activities to verify that the licensee met the requirements of 10 CFR 50.65(a)(4) for assessing and managing increases in plant risk (six inspection samples). The inspectors compared the licensee's risk assessments and risk management actions 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." The inspectors reviewed the following risk assessments:

- Unplanned emergency diesel generator (EDG) Train B outage on January 27. The inspectors performed an in-office review of the licensee's risk assessment and observed compensatory risk mitigation actions from the control room.

- Planned unavailability of the centrifugal charging pump, Train A, due to surveillance testing on January 6. The inspectors performed an in-office review of the licensee's risk assessment and CAR 200500108, Incorrect restoration of the centrifugal charging pump, and observed compensatory risk mitigation actions from the control room.

- Planned unavailability of the turbine driven auxiliary feedwater pump due to valve operability surveillance testing on February 14. The inspectors performed an in-office review of the licensee's risk assessment and observed compensatory risk mitigation actions from the control room.

- Unplanned inoperability of ESW Train B and EDG Train B due to pinhole leaks in the ESW piping on February 22-23. The inspectors performed an in-office review of the licensee's risk assessment and observed compensatory risk mitigation actions from the control room.

- Planned unavailability of ESW Train A and EDG Train A due to maintenance on March 8. The inspectors performed an in-office review of the licensee's risk assessment and observed compensatory risk mitigation actions from the control room.

- Unplanned unavailability of ESW Train B and EDG Train B due to ESW pump discharge pipe pinhole leaks and wall thinning on March 23. The inspectors performed an in-office review of the licensee's risk assessment and observed compensatory risk mitigation actions from the control room.

b. Findings

No findings of significance were identified.

1R14 Personnel Performance During Nonroutine Plant Evolutions (71111.14)

a. Inspection Scope

The inspectors reviewed three non-routine plant events for personnel performance (three inspection samples). The inspectors reviewed each event to verify proper operator response. The inspectors used operator logs, plant computer data, charts and CARs to determine what occurred, how the operators responded, and whether the response was in accordance with plant procedures.

- CAR 200500354, Reactor trip during replacement of a primary power supply on January 19 due to inadequate procedure.
- Reactor startup on January 20-21. The inspectors observed startup activities from the control room.
- CAR 200501378, Unplanned isolation of the firewater suppression system on March 3 due to the failure to follow procedure.

b. Findings

Unplanned Reactor Trip Due to Ineffective Use of Industry Operating Experience During a Maintenance Activity

Introduction. A self-revealing Green finding was identified after an unplanned reactor trip resulted from the licensee's ineffective use of industry operating experience (OE) during a corrective maintenance activity.

Description. An unplanned reactor trip from full power occurred on January 19 due to an inadequately planned corrective maintenance activity. The Callaway Plant design includes primary and secondary control power supplies for important non-safety related systems. The secondary supply will automatically power control system loads following failure of the primary supply. A primary supply failed on January 18. The licensee prepared a corrective maintenance work instruction to replace the failed power supply. The primary power supply was bolted to an instrument rack directly above the secondary supply. A two-inch gap existed between the two power supplies. Technicians inserted a foreign material exclusion tool into the gap to help support the weight of the primary power supply after unbolting it from the rack. The weight of the primary power supply on the foreign material exclusion tool forced the lower power supply upper cage downward, contacting and shorting an energized internal heat sink. The short circuit resulted in a momentary interruption of the secondary power supply output. This interruption caused a main feedwater regulating valve to close. The reactor trip followed due to low steam generator level.

A similar event occurred at the Beaver Valley Nuclear Plant during June 2003 (OE 16445). The OE recommended that technicians place a tool between the power supplies to distribute the weight of the upper power supply to the lower power supply frame, protecting the top cover from pressure over the energized heat sink. The licensee OE coordinator had distributed the OE information to system engineering, instrument and controls department, and maintenance training. The inspectors determined that the work planner did not consider the Beaver Valley OE during preparation of the corrective maintenance work package. The work planner stated that the station had not established clear expectations for ensuring all applicable OE was considered during work package preparation. The work planner also stated that industry OE was difficult to access at the Callaway Plant. Administrative Procedure APA-ZZ-00322, Appendix C, "Work Document Planning," Revision 1, Section 4.2.1, stated that a review of OE must be performed and included in the work document. However, station management did not provide guidance for the scope of OE reviews or controls to ensure that all applicable OE was reviewed by work planners and incorporated into work documents.

The work package instructions were written for a protection set replacement rather than for a control power supply replacement. The protection set power supplies did not have the two-inch gap between the upper and lower power supplies. The work planner had not walked down the job site prior to releasing the work package.

Procedure APA-ZZ-00322, Appendix C, Section 4.6.1 stated that work areas should be walked down as necessary. A walkdown of the job location would have alerted the work planner that the instructions were for different equipment. Station management had not provided guidance or controls to establish when work planner walkdowns were required. The inspectors determined that the failure to properly revise the incorrect work package, a poor pre-job brief, and organizational time pressures all contributed to the event.

The inspectors concluded that the licensee's root cause analysis of the reactor trip was less than adequate. The licensee concluded three root causes of the event:

- The issue of placing pressure on the secondary power supply cover was not clearly understood.
- The work instructions did not adequately address the differences in removal techniques between the control and protection cabinets.
- The appropriate tools were not available for removal of the primary power supply.

The licensee did not determine the conditions which caused these three circumstances. Administrative Procedure APA-ZZ-00500, “Corrective Action Program,” defined an apparent cause as the reason for an adverse condition arrived at through a limited investigation of circumstances by someone knowledgeable in a related area. Procedure APA-ZZ-00500 defined a root cause as the most probable reason for a problem which, if corrected, is most likely to prevent recurrence of an event. The licensee's root causes more closely resembled apparent causes. The licensee's root cause analysis did not include a systematic evaluation of the casual factors, including latent organizational weakness to prevent recurrence of the event. The inspectors had previously identified a deficient root cause analysis associated with a February 15, 2004 reactor trip as discussed in NRC Supplemental Inspection Report 05000483/2004009.

Analysis. The licensee's use of an inadequate work instruction to replace the power supply was a performance deficiency. This finding is greater than minor because the procedural adequacy attribute of the initiating events cornerstone objective is affected. The inspectors used the significance determination process for at-power reactor inspection findings to analyze this finding because the reactor trip was a transient initiator, affecting the initiating events cornerstone. The inspectors determined this finding to be of very low safety significance (Green) because the condition did not contribute to both the likelihood of a reactor trip and the unavailability of mitigating equipment functions. This finding, which is related to inadequate work instructions to support a maintenance task, is associated with the crosscutting area of human performance. Additionally, this finding had crosscutting aspects related to problem identification and resolution in that the evaluation of root versus contributing causes was deficient.

Enforcement. No violation of regulatory requirements occurred. The inspectors determined that this finding did not represent a noncompliance because it occurred on nonsafety-related secondary plant equipment (FIN 05000483/2005002-04).

1R15 Operability Evaluations (71111.15)

a. Inspection Scope

The inspectors reviewed six operability determinations involving risk significant equipment during the inspection (six inspection samples). The inspectors reviewed the technical adequacy of the operability determinations to verify that operability was justified and compensatory measures were appropriate and controlled. The inspectors reviewed plant status documents such as operator shift logs, emergent work documentation, deferred modifications, and standing orders to determine if an operability determination was warranted for degraded components. The inspectors used the FSAR, TSs, and design basis documents as the bases to determine the technical adequacy of licensee prepared operability determinations. The inspectors reviewed the following equipment conditions and associated operability determinations:

- Operability Determination 200409580, ESW pump room damper found failed on December 29, 2004
- Operability Determination 200409571, Failure of main steam isolation Valve C four-way valve on December 29, 2004
- Operability Determination 200500371, Feedwater isolation Valve C leakage on January 19
- Operability Determination 200500865, Water-jacket leak on EDG Train A on February 9
- Operability Determination 200500543, Water-jacket leak on EDG Train B on January 27
- Operability Determination 200500756, Voiding in SI pump discharge piping on February 4

b. Findings

Main Steam Isolation Valve (MSIV) Operability Question

The inspectors identified that the licensee did not consider both actuation trains as required attendant equipment for MSIV operability. The Callaway Plant has one MSIV on each main steam line. Each MSIV has two actuation trains. Each train is capable of independently closing the MSIV. The electrical solenoids for the pneumatic/hydraulic power trains are energized from separate safety-related sources. Each MSIV position will fail "as is" if both trains of control power are lost. FSAR Section 10.3.1.1, "Safety Design Bases," stated component redundancy is provided so that MSIV safety functions can be performed, assuming a single active component failure coincident with the loss of off-site power (meeting General Design Criteria 34). FSAR Section 10.3.3, "Safety Evaluation," stated that no single failure will compromise the MSIV system safety functions. Both MSIV actuation trains were required to meet the Callaway accident analysis, as described in the FSAR. The Callaway accident analysis Calculation of Record M-YY-43, "Containment Pressure/Temperature Response to a Main Steam Line Break (MSLB)," included a case that required both MSIV actuation trains for accident mitigation. This accident assumed the loss of off-site power and the failure of one instrument logic train as the single failure. This assumed single active component failure would result in the unavailability of a diesel generator, one containment heat removal train, and one MSIV actuator train (FSAR Table 6.2.1-58, MSLB Case 16). Based on the FSAR commitment that each MSIV meet single failure criteria, and the accident analysis reliance on the availability of both MSIV actuating trains, the inspectors questioned if both MSIV actuation trains were required attendant equipment for MSIV operability.

TS 3.7.2, "MSIVs," required all four MSIVs to be operable in Modes 1, 2, and 3. The TS action required the licensee to either restore a failed MSIV to operable condition within 8 hours or be in Mode 2 in the next 6 hours. TS Table 3.3.2-1, "Engineering Safety Feature Actuation System Instrumentation," also required the licensee maintain two MSIV actuation trains operable in Modes 1, 2, and 3. Table 3.3.2-1 required the licensee to restore an inoperable train within 6 hours or be in Mode 3 in the following 12 hours. The licensee declared MSIV C four-way valve (ABHV0020V14) inoperable at 7:55 a.m. on December 29, 2004. The failure of the four-way valve rendered one MSIV actuation train inoperable. The licensee did not consider the failed actuation train as required attendant equipment and did not declare the MSIV inoperable. The licensee restored the four-way valve at 6:12 a.m. on December 31, about 46 hours later.

The licensee reviewed MSIV functionality with one MSIV actuator train inoperable. The accident analysis postulated that peak containment pressure and temperature (FSAR Table 6.2.1-58, MSLB Case 16) was dependent on the heat removal capability of the containment coolers. TS 3.6.6, "Containment Spray and Cooling Systems," required that the containment coolers be capable of removing 141.4 million British thermal units per hour (MBTU/hr) during accident conditions. The licensee replaced the containment cooler coils during Refueling Outage 12. The new coils were capable of removing 155 MBTU/hr during accident conditions. The licensee reevaluated the postaccident containment response with one MSIV actuation train unavailable and the new containment cooler capability. The licensee demonstrated successful accident mitigation with the additional containment cooler capacity. The licensee proposed incorporating these new accident analysis assumptions into the Calculation of Record and FSAR. However, the licensee's 10 CFR 50.59, "Changes, Tests and Experiments," evaluation concluded that the proposed change resulted in more than a minimal increase in the consequences than previously evaluated in the FSAR. As a result, the licensee determined that an NRC approved license amendment was required before these changes could be incorporated in the plant licensing bases. This issue is considered unresolved (Unresolved Item 05000483/2005002-05) pending a determination by the NRC if both MSIV actuator trains are required attendant equipment for MSIV operability.

1R16 Operator Workarounds (71111.16)

a. Inspection Scope

The inspectors performed an evaluation of the cumulative effect of operator workarounds and a detailed review of the workaround to verify 4.16 kV Magne-Blast breaker control circuit continuity (two inspection samples). The inspectors reviewed the February 2005 operator workaround and burden lists and assessed the effect of the workarounds on the ability of operators to implement plant emergency procedures. The inspectors completed the review to verify that the workarounds did not challenge the operators' capability to respond to plant transients and events. The inspectors also completed an in-office review and control building walkdown of the vital 4.16 kV Magne-Blast breaker workaround on February 24.

b. Findings

No findings of significance were identified.

1R19 Postmaintenance Testing (71111.19)

a. Inspection Scope

The inspectors reviewed and/or observed nine risk significant postmaintenance tests (PMTs) to verify that the licensee adequately demonstrated the safety function of components affected by maintenance activities (nine inspection samples). The inspectors verified that testing procedures were properly reviewed and approved and incorporated appropriate acceptance criteria. The inspectors used information in the TSs, the FSAR, and Section XI of the American Society of Mechanical Engineers Code, as the bases for acceptability of sampled PMTs. The inspectors completed an in-office review of the completed work packages. The sample included the following PMTs:

- PMT P710322, Following preventive maintenance on the 125 volt dc vital battery Charger 3 (NK23) on February 1 and CAR 200500689, Failure of NK23 during restoration following maintenance.
- PMT P685529, Following preventive maintenance of 480 volt safety-related circuit Breaker NK23 on February 1.
- PMT 712979/900, Following preventive maintenance of the containment spray Train B pump motor on January 26.
- PMT 041089/910, Following corrective maintenance of the reactor coolant Pump D thermal barrier outlet isolation Valve BBHV0016 on December 30, 2004.
- PMT W239612/910, Following corrective maintenance of the control building ventilation system on February 2. The inspectors completed a control building walkdown of the completed work on March 3.
- PMT P729302, Following preventive maintenance on the EDG Train A starting air compressor on February 7. The inspectors completed an EDG building walkdown of the completed work on March 3.
- PMT 05102427, Following corrective maintenance on the NK11 125 volt dc vital battery on February 22. The inspectors completed a walkdown of the completed work on February 24.
- PMTs 05101706/905, 05101706/910, and 05101706/920, Following corrective maintenance of EDG Train B on January 27.

- PMTs P663878/900, Following preventive maintenance of containment spray Train B suction valve power supply breaker on January 26.

b. Findings

No findings of significance were identified.

1R20 Refueling and Outage Activities (71111.20)

a. Inspection Scope

The inspectors evaluated licensee activities during the forced outage (one inspection sample) beginning January 19 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 the operating license and TS requirements that ensured defense-in-depth. The inspectors observed portions of plant startups and reviewed the licensee's control of equipment.

b. Findings

No findings of significance were identified.

1R22 Surveillance Testing (71111.22)

a. Inspection Scope

The inspectors observed and/or reviewed seven risk significant surveillance tests to verify that the licensee adequately demonstrated component safety functions and to assess operational readiness (seven inspection samples). The inspectors verified that testing procedures were properly reviewed and approved with appropriately incorporated acceptance criteria. The inspectors used information in the TSs, the FSAR, Section XI of the American Society of Mechanical Engineers Code, and licensee procedural requirements as the bases for acceptability of sampled surveillance tests. The samples included the following surveillance tests:

- Surveillances S731680 and S731601, Inservice testing of centrifugal charging Train A cooling pump and refueling water storage tank suction Valve BNLCV0112D. The inspectors observed testing from the auxiliary building on January 6 and performed an in-office review of the completed surveillance package.

- Surveillance P714276, Containment spray Train B on January 26. Inspectors observed the test from the auxiliary building.

- Surveillances S05504978 and S05504980, NK12 and NK14 weekly vital battery inspection on January 28.

- Surveillance 05504346/500, ECCS vent and flow path verifications, on January 28 and CAR 200500584, Voiding in the safety injection pump discharge piping.

- Surveillance S732483, Safety injection Train B slave relay test, performed on January 27. The inspectors completed an in-office review of the completed surveillance test package.

- Surveillances 733325 and 733272, Turbine-driven auxiliary feedwater operability tests performed on February 14. The inspectors completed an in-office review of the completed surveillance test package.

- Surveillance P732105, Site fire protection walkdown on January 27. The inspectors completed an in-office review of the completed surveillance test package and also reviewed CAR 200500518, emergency lights not operable.

b. Findings

No findings of significance were identified.

1R23 Temporary Plant Modifications (71111.23)

a. Inspection Scope

The inspectors reviewed two temporary plant modifications during the inspection (two inspection samples). The inspectors evaluated the licensee's 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 performed an in-office review to verify that the installation was consistent with the modification documents and the plant design bases as described in the FSAR and selected radiological release permits. The inspectors compared the temporary modification (TM) documentation against the requirements established in Administrative Procedure APA-ZZ-00605, "Temporary System Modifications." The inspection samples included:

- TM 05-001, Modification of the auto-isolation feature of the radiological release isolation circuit. The inspectors walked down the installation of the temporary modification and reviewed batch liquid radioactive release Permit RP09-2005-L0007

- TMs 05-002 and 05-004, modification of the automatic rod control circuitry on February 3 and 23

b. Findings

No findings of significance were identified.

1EP6 Drill Evaluation (71114.06)

a. Inspection Scope

The inspectors observed one emergency drill during the inspection period (one inspection sample). The inspectors observed the exercise to evaluate drill adequacy and to verify that the licensee implemented proper emergency action level classification and protective action recommendations. The inspectors observed the Radiological Emergency Response Plan Cycle 05-1 Drill, “Security Events and ERO Response,” conducted on February 9. The inspectors observed the exercise from the Technical Support Center. The inspectors compared drill observations against Emergency Plan Implementing Procedure EIP-ZZ-00101, "Classification of Events," and Emergency Plan Implementing Procedure EIP-ZZ-00201, "Notifications,” to evaluate licensee performance.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA2 Identification and Resolution of Problems (71152)

.1 Daily Reviews

a. Inspection Scope

The inspectors performed a daily review of items entered into the licensee’s corrective action program. The inspectors performed the screening to identify any repetitive equipment failures or adverse human performance trends for followup. The inspectors attended selected conditions adverse to quality report screenings and daily plant status meetings. The inspectors also reviewed the January and February Corrective Action Health Reports for licensee identified adverse trends.

b. Findings

No findings of significance were identified.

.2 Annual Sample Review

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 (two inspection samples). The inspectors reviewed the licensee's CAR reports to verify that the full extent of the issues was identified, that the licensee performed appropriate evaluations, and that adequate corrective actions were specified and prioritized. The inspectors evaluated the reports against the requirements of Administrative Procedure APA-ZZ-00500, "Corrective Action Program," and 10 CFR Part 50, Appendix B. The inspectors reviewed the following two samples:

- CAR 200409052, Failure to follow Regulatory Guide 1.106, “Thermal Overload Protection for Electric Motors on Motor-Operated Valves,” Revision 1, for auxiliary feedwater flow control valves

- CAR 200501092, Adverse trend - recurrence of voiding in safety-related system

b. Findings

No findings of significance were identified.

.3 Cross-Reference to Problem Identification and Resolution Findings Documented Elsewhere

Section 1R04 described a finding related the licensee's failure to adequately identify and evaluate the cause of a negative trend of failed SI pump discharge pipe voiding surveillances. Consequently, the licensee did not implement corrective actions to prevent recurrence.

Section 1R05 described a finding related to the licensee’s failure to maintain the minimum number of fire brigade members on site. This condition was documented twice in the corrective action program. However, the licensee's condition evaluation was less than adequate and resulted in ineffective corrective actions.

Section 1R14 described a finding related to deficiencies in the licensee’s evaluation of root versus contributing causes regarding a plant trip that occurred during a control power supply replacement maintenance activity.

4OA3 Event Followup (71153)

1. (Closed) License Event Report (LER) 50-483/2005-001-00: Reactor Trip While Replacing Control Cabinet Power Supply RP043

A reactor trip occurred during the replacement of a failed balance-of-plant control power supply. This event, the licensee's followup actions, and the inspectors' findings were described in Section 1R14 of this report. The licensee documented the event in CARs 200500322 and 200500354. The inspectors reviewed the LER and no additional findings of significance were identified. This LER is closed.

4OA4 Crosscutting Aspects of Findings (71152)

Section 1R05 described a finding related to the licensee's inadvertent isolation of all the plant fire water suppression from the reactor, auxiliary, control, and turbine buildings during surveillance testing. This finding resulted from inadequate procedural resources to support the task.

Section 1R05 described a finding related to the licensee's failure to maintain the minimum number of fire brigade members on site. This finding resulted from the licensee's failure to ensure adequate resources for minimum fire brigade staffing.

Section 1R14 described a finding related to an unplanned reactor trip resulting from the licensee's ineffective use of industry OE during corrective maintenance. The licensee's ineffective use of OE resulted in inadequate resources to support the maintenance task.

Section 1R15 described a finding when the licensee failed to implement the required TS actions after exceeding the allowable out of service time for an MSIV. This finding was related to less than adequate procedural guidance and personnel knowledge of the plant licensing bases.

4OA6 Management Meetings

Exit Meeting Summary

On March 10, 2005, the inspectors presented the maintenance rule inspection results to Mr. T. Sharkey, Superintendent, Engineering Technical Support, and other members of his staff who acknowledged the findings.

On March 31, 2005, the resident inspectors presented their inspection results to Mr. C. Naslund, Senior Vice President 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 violation of very low safety significance (Green) was identified by the licensee and meets the criteria of Section VI of the NRC Enforcement Policy, NUREG 1600, for being dispositioned as a noncited violation.

A noncited violation of Technical Specification 5.4.1.d, "Fire Protection Program," was identified by the licensee for failing to maintain five of the six Halon gaseous fire suppression systems in the qualified configuration for an extended duration. Each of the Halon pneumatic actuators was equipped with Ports A and B. Because nonsymmetric actuator pistons were used in the pneumatic actuators, correct alignment of the ports was necessary to preserve system design and ensure reliable operation. At Callaway, the actuation piping to Ports A and B were reversed in five of the six Halon systems. The vendor did not plan to qualify this new configuration and the licensee promptly corrected the actuation piping. To determine the significance of this issue, the system vendor tested system actuation in the nonqualified configuration. The system actuated after only a short time delay. Using the criteria in Attachment 2 to Appendix F of Manual Chapter 0609, "Significance Determination Process," for findings against gaseous based suppression, the degradation rating was determined to be low and the finding screened to Green (very low safety significance). This issue was documented in the licensee's corrective action program as Callaway Action Request 200500217.The inspectors concluded this finding was of very low safety significance because of the low degradation level.

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee

T. Antweiler, Engineer, Reliability Engineering
G. Hughes, Supervising Engineer, Quality Assurance
T. Sharkey, Superintendent, Engineering Technical Support
T. Moser, Manager, Plant Engineering
R. Myatt, Supervisor, Reliability Engineering
C. Naslund, Senior Vice President and Chief Nuclear Officer
D. Neterer, Superintendent, Operations
J. Nurrenbern, Engineer, Reliability Engineering
E. Olson, Superintendent Performance Improvement
S. Petrel, Engineer, Regional Regulatory Affairs
M. Reidmeyer, Supervisor, Regional Regulatory Affairs
T. Sharkey, Superintendent, Engineering Technical Support
K. Young, Manager, Regional Regulatory Affairs
C. Younie, Manager, Operations

NRC

N. O'Keefe, Senior Reactor Inspector, Region IV

LIST OF ITEMS OPENED AND CLOSED

Opened

05000483/2005002-05	URI	Main Steam Isolation Valve Operability (Section 1R15)

Opened and Closed

05000483/2005002-01	NCV	Ineffective Cause Determination and Corrective Actions to Prevent Recurrence of ECCS Pipe Voiding (Section 1R04)
05000483/2005002-02	NCV	Unplanned Loss of Water Fire Suppression Due to an Inadequate Testing Procedure (Section 1R05)
05000483/2005002-03	NCV	Failure to Maintain the Minimum Number of Fire Brigade Members on Site (Section 1R05)
05000483/2005002-04	FIN	Unplanned Reactor Trip Due to Ineffective Use of Industry OE During a Maintenance Activity (Section 1R05)

Closed

05000483/2005-001-00	LER	Reactor Trip While Replacing Control Cabinet Power Supply RP043 (Section 4OA3)

DOCUMENTS REVIEWED

Procedures

APA-ZZ-00743, Fire Team Organization and Duties, Revision 18

APA-ZZ-00500, Corrective Action Program

APA-ZZ-00322, Revision 001, Appendix C, Work Document Planning

EDP-ZZ-01128, Maintenance Rule Program, Revision 6

EDP-ZZ-01129, Callaway Plant Risk Assessment, Revision 5

EIP-ZZ-00226, Fire Response Procedure for Callaway Plant

EIP-ZZ-00101, Classification of Events

ISF-EG-0F108, Fctnal-Flow; Cmpnt Cooling Wtr Flow, Revision 6

MPE-ZZ-QB004, Battery Charger Inspection, Revision 9

MSE-ZZ-QS002, GE AKR 30/50 Circuit Breaker Preventive Maintenance and Inspection, Revisions 18 and 19

MTE-ZZ-QN005, Electrical Scheme Checkout, Revision 5

ODP-ZZ-00001, Operations Department - Code of Conduct, Revision 23

OSP-BG-P005A, Centrifugal Charging Pump A In-Service Test, Revision 26

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

OSP-BN-V002A, BNLCV0112D In-Service Test, Revision 3

OSP-NE-0001B, Standby Diesel Generator B Periodic Tests, Revision 18

OSP-SA-0017B, Train B SIS-CSAS Slave Relay Test, Revision 15

OSP-EN-V001B, Train B Containment Spray Valve Operability, Revision 16

OTO-ZZ-00001, Control Room Inaccessibility, Revision 20

OSP-SA-00003, ECCS Venting

PM 0821393, Functional Test of Class 1E Breakers, Revision 0

Callaway Action Requests

199500001	200306053	200402672	200409592
200000113	200307394	200402877	200500108
200107278	200307587	200404269	200500140
200201932	200307589	200405778	200500193
200203337	200307630	200405859	200500245
200206583	200308178	200406390	200500371
200207763	200308561	200406683	200500532
200208352	200308871	200406982	200500543
200300003	200400065	200407453	200500564
200300731	200400565	200408114	200500689
200301722	200400629	200408342	200500855
200301765	200400710	200408626	200500865
200303009	200400753	200408649	200501006
200303106	200400791	200409052	200501097
200303935	200400883	200409206	200501114
200304760	200401820	200409295	200501378
200305970	200401869	200409557	
200305983	200401986	200409580	

Event Review Team Meeting Summary

AUCA 05-001, Halon Bottle Manual-Pneumatic Actuator Port Connection Error, January 13, 2005

AUCA 05-002, Cooling Tower Blowdown in NPDES Exceeded 190 ppb Total Residual Chlorine, January 14, 2005

AUCA 05-005, Valving Error Causes Fire Main to Depressurize

ALCA 05-003, Reactor Trip During Replacement of RP042 Primary Power Supply, January 19, 2005

Drawings

Piping and Instrumentation Diagram M-22EM01, Safety Injection System, Revision 30
Piping and Instrumentation Diagram M-22EM02, Safety Injection System, Revision 18
Piping and Instrumentation Diagram M-22EM03, Safety Injection System, Revision 12
Piping and Instrumentation Diagram M-22LF01, Floor and Equipment Drains System
Piping and Instrumentation Diagram M-22LF02, Floor and Equipment Drains System
Piping and Instrumentation Diagram M-22LF03, Floor and Equipment Drains System

Miscellaneous

Performance Monitoring Report - High Pressure Coolant Injection, February 17, 2005

Callaway Quarterly Performance Analysis Report, Fourth Quarter 2004

RP10-2005-L004;1, Batch Liquid Radioactive Release Permit Closure Summary

RP10-2005-L005;1, Batch Liquid Radioactive Release Permit Closure Summary

RP10-2005-L006;1, Batch Liquid Radioactive Release Permit Closure Summary

Fire Protection Impairment Permit 12242, Sprinkler System Inoperable, March 3

Request for Resolution 3863, Revision C, Fire loading in ECCS Pump Rooms, October 27, 1997

Request for Resolution 16265, Revisions A and B, Reinstall screens in LF system drains

Root Cause Manual, Revision 2

Maintenance Rule Periodic Assessment for Cycle 13

SA04-NE-F02, Self-Assessment for the Maintenance Rule Program, dated November 6, 2004

Quality Assurance Reports

AP04-014, Quality Assurance audit of security, January 10, 2005

AP04-015, Quality Assurance audit of access authorization and FFD, January 10, 2005

SP05-001, Quality Assurance surveillance of ongoing licensing and design related efforts for the Steam Generator Replacement Project, February 14, 2005

SP05-003, Quality Assurance surveillance of plant operation and organizational support following a plant trip, February 10, 2005

LIST OF ACRONYMS

CAR	Callaway Action Request
CCDP	conditional core damage probability
ECCS	emergency core cooling system
EDG	emergency diesel generator
EO	equipment operator
ESW	essential service water
FSAR	Final Safety Analysis Report
MBTU	million British thermal units
MSIV	main steam isolation valve
MSLB	main steam line break
NCV	noncited violation
OE	operating experience
PMTs	postmaintenance tests
SI	safety injection
SR	surveillance requirement
TSs	Technical Specifications