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

February 14, 2006

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 05000483/2005005

Dear Mr. Naslund:

On December 31, 2005, the NRC completed an inspection at your Callaway Plant. The enclosed report documents the inspection findings which were discussed on January 6, 2006, 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.

Based on the results of this inspection, the NRC has determined that one Severity Level IV violation of NRC requirements occurred. The NRC has also identified six additional issues that were evaluated under the risk significance determination process as having very low safety significance (Green). The NRC has determined that there are four violations associated with the significance determination process issues. In addition, licensee-identified violations which were determined to be of very low safety significance are listed in the report. All of the violations are being treated as noncited violations (NCVs), consistent with Section VI.A of the Enforcement Policy. The 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/

William B. Jones, Chief
Project Branch B
Division of Reactor Projects

Docket: 50-483
License: NPF-30

Enclosure:
NRC Inspection Report
05000483/2005005
w/attachment: Supplemental Information

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ENCLOSURE

U.S. NUCLEAR REGULATORY COMMISSION
REGION IV

Docket:	50-483
License:	NPF-30
Report No.:	05000483/2005005
Licensee:	Union Electric Company
Facility:	Callaway Plant
Location:	Junction Highway CC and Highway O Fulton, Missouri
Dates:	September 24 through December 31, 2005
Inspectors:	M. S. Peck, Senior Resident Inspector D. E. Dumbacher, Resident Inspector R. W. Deese, Senior Resident Inspector B. D. Baca, Health Physicist P. J. Elkmann, Emergency Preparedness Inspector T. F. Stetka, Senior Operations Engineer M. E. Murphy, Senior Operations Engineer J. F. Drake, Operations Engineer
Approved By:	W. B. Jones, Chief, Project Branch B

SUMMARY OF FINDINGS

IR 05000483/2005005; 09/24 - 12/31/2005; Callaway Plant: Equipment Alignment, Fire Protection, Personnel Performance During Nonroutine Plant Evolutions, Permanent Plant Mods, Refueling & Outage Activities, Licensed Operator Requal Program, and Emergency Plan & Emergency Action Level Change.

This report covered a 3-month inspection by region based reactor inspectors and resident inspectors. One Severity Level IV noncited violation, four Green noncited violations, and two Green findings 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 assigned a severity level after NRC management review. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG 1649, "Reactor Oversight Process," Revision 3, dated July 2000.

A. Inspector-Identified and Self-Revealing Findings

Cornerstone: Initiating Events

- Green. The inspectors determined that the failure to adhere to ANSI/ANS 3.5-1998, as endorsed by Regulatory Guide 1.149, "Nuclear Power Plant Simulation Facilities for Use in Operator Training and License Examinations," Revision 3, October 2001, as committed to in the Callaway Plant Simulation certification dated March 13, 2000, was a finding. Specifically, the simulator performance testing did not meet the standards specified in ANSI/ANS 3.5-1998 in that: (1) all required parameters during the simulator test were not recorded; and (2) simulator to baseline data comparisons were unavailable.

  The failure to evaluate and document simulator performance testing is more than minor because it affected the Operator Requalification attribute of the Mitigating Systems and Initiating Events cornerstone of reactor safety and is inconsistent with the requirements of 10 CFR 55.46 in that simulator fidelity issues may not be identified which have the potential of causing negative training. The finding was considered to be of very low safety significance because the discrepancies have not yet impacted operator actions in the plant such that safety-related equipment was made inoperable or that operators failed to properly respond to plant transients. This issue is documented in the facility licensee's corrective action program as Callaway Action Request 200503956 (Section 1R11).

Cornerstone: Mitigating Systems

- Green. The inspectors identified a noncited violation of 10 CFR Part 50, Appendix B, Criterion V, for the failure to adequately implement work order instructions and a procedure for the inspection of the containment recirculation sump enclosure. The licensee's inspections failed to identify a 1.5-inch hole in the sump cover, which could provide a path for foreign material to enter the containment sump. AmerenUE completed a detailed inspection of the sump on April 27, 2004, in response to NRC

Bulletin 2003-01, “Potential Impact of Debris Blockage on Emergency Sump Recirculation at Pressurized-Water Reactors,” but failed to identify the 1.5-inch hole. A subsequent inspection was performed on November 8, 2005, during Refueling Outage RF 14 that also did not identify the hole in the containment sump enclosure. This issue was entered into the corrective action program as Callaway Action Request 200509189.

This finding is greater than minor because it is associated with the mitigating systems cornerstone attribute of equipment performance and affects the associated cornerstone objective to ensure availability and reliability of the containment recirculation sump emergency core cooling system containment safety function. This finding is of very low safety significance because the condition was a qualification deficiency confirmed not to result in loss of function per Part 9900, Technical Assessment, “Operability Determination Process for Operability and Functional Assessment.” The cause of this finding is related to the crosscutting element of human performance in that personnel failed to adequately implement a work instruction and procedure in inspecting the containment sump configuration (Section 1R04).

- Green. The inspectors identified a noncited violation of Technical Specification 5.4.1.d, “Fire Protection Program Implementation,” associated with seven examples of inadequately performed continuous fire watches. In September 2005, AmerenUE provided verbal guidance to fire watch personnel that continuous fire watches may be met by a 15-minute roving fire patrol. The roving patrol did not ensure adequate compensatory action for fire areas with degraded detection or suppression capability. As a result, fire watch personnel were not available to promptly detect, report, and extinguish a fire while still in the incipient stage. AmerenUE did not evaluate this change to ensure no adverse affect on the ability to achieve and maintain safe shutdown in the event of a fire. This condition was entered into the corrective action program as Callaway Action Request 200510325.

  This finding is greater than minor because inadequate fire watches are associated with the reactor safety mitigating systems cornerstone attribute to provide protection against external factors and affect the associated cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. This finding is of very low safety significance because the condition had an adverse affect on the “Fixed Fire Protection Systems” element of fire watches posted as a compensatory measure for outages or degradations. A low degradation rating was assigned to this finding as the provision affected by this finding is expected to display nearly the same level of effectiveness and reliability. The cause of this finding is related to the crosscutting element of human performance in that the guidance provided was not adequate to ensure continuous fire watches were appropriately conducted (Section 1R05).

- Green. The inspectors identified a noncited violation of 10 CFR Part 50, Appendix B, Criteria V, “Instructions, Procedures, and Drawings,” associated with an inadequate engineering procedure used for the verification of design calculations. The inadequate procedure resulted in a nonqualified, nonsafety-related engineering calculation used to demonstrate that the safety-related containment recirculation sump valves were capable of performing the safety function described in the design bases. The performance deficiency associated with this finding involved the failure of engineering personnel to only use qualified calculations for safety-related applications. This finding was entered into the Corrective Action Program as Callaway Action Request 200509849.

This finding is greater than minor because, if left uncorrected, this finding would become a more significant safety concern. This finding is determined to have very low safety significance because this finding involves a design deficiency confirmed not to result in loss of operability per Part 9900, Technical Guidance, "Operability Determination Process for Operability and Functional Assessment." The cause of this finding is related to the crosscutting element of human performance in that the procedure did not ensure the calculations were qualified to support a design basis function of a safety-related component (Section 1R17).

Cornerstone: Barrier Integrity

- Green. The inspectors identified a noncited violation of Technical Specification 5.4.1.a, "Procedures," after AmerenUE Operations personnel failed to maintain the reactor coolant system heatup and cooldown temperature limits on two occasions. On November 7, 2005, plant operators decreased the reactor coolant system pressurizer surge line temperature 260EF in a one-hour period. The operators conducted the rapid cooldown after several containment lead shield blanket polyvinylchloride covers located on the pressurizer surge line melted. On November 8, 2005, plant operators increased the surge line temperature about 175EF in a one-hour period. Plant Technical Specification 3.4.3, "RCS [reactor coolant system] Pressure and Temperature (P/T) Limits," and plant procedures required reactor coolant system component temperature changes (except the pressurizer) be limited to 100EF in one hour. This finding was placed in the Corrective Action Program as Callaway Action Requests 200509487 and 200509143.

  This finding was greater than minor because it is associated with the reactor safety barrier integrity cornerstone attribute of equipment performance and affects the associated cornerstone objective to ensure reasonable assurance that the reactor coolant system piping barrier will protect the public from radionuclide releases caused by accidents or events. This finding is determined to have very low safety significance because an engineering evaluation concluded that the temperature transient did not significantly increase the likelihood of a loss of reactor coolant system inventory or degrade the ability to terminate a leak path. The cause of this finding is related to the crosscutting element of human performance in that the reactor coolant system pressurizer surge line heatup and cooldown limits were exceeded (Section 1R14).

Cornerstone: Emergency Preparedness

- Severity Level IV. The inspectors identified a violation of 10 CFR 50.54(q) for implementing a change to emergency action levels which decreased the effectiveness of the emergency plan. Emergency Implementing Plan Procedure EIP-ZZ-00101, "Classifying the Emergency," Revision 33, limited application of emergency action

Level 3E, “Fire within Protected Area Boundary NOT Extinguished with 15 minutes of Verification,” so that fires in some plant areas which would be classified under the previous revision may no longer be classifiable.

Implementation of changes to emergency action levels which decreased the effectiveness of the emergency plan was a performance deficiency. The finding is more than minor because removal of a classifiable condition from licensee emergency action levels has the potential to impact safety, and licensee implementation of a change to their emergency plan, which decreases the effectiveness of the plan without prior NRC approval, impacts the regulatory process. This finding is a violation of 10 CFR 50.54(q). The licensee has entered this issue into their corrective action system as Corrective Action Report 200510162 (Section 1EP4).

Cornerstone: Miscellaneous

- Green. The inspectors identified a finding after AmerenUE implemented less than adequate risk management controls of the spent fuel pool water inventory. On September 29, 2005, the core had been off-loaded to the spent fuel pool and the transfer canal weir wall removed. The inspectors identified that the shutdown safety plan did not establish specific controls for reactor refueling canal transfer tube Valve ECV-995, which isolated the fuel transfer canal from the containment cavity or provided for installation of the associated fuel transfer canal flange. Valve ECV-995 was closed but was not identified in the shutdown risk management system and did not have administrative controls established through the shutdown risk plan. NRC Information Notice 2005-16, “Outage Planning and Scheduling - Impacts on Risk,” emphasized that most spent fuel pool events had a common thread of human error and involved equipment misalignment. This finding was entered into the Corrective Action Program as Callaway Action Requests 200507593 and 200507693.

  This finding is greater than minor because, if left uncorrected, it would have become a more significant safety concern. Because Manual Chapter 0609, “Significance Determination Process,” does not specifically address findings related to the spent fuel pool inventory, this finding is determined to have very low safety significance based on NRC management review with input from a senior reactor analyst. The review considered that the procedure used to manipulate the valve was not in use during this period and that borated water makeup capabilities were available to the spent fuel pool. No violation of regulatory requirements occurred (Section 1R20).

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 of this report.

REPORT DETAILS

Summary of Plant Status

The Callaway Plant was shut down for Refueling Outage 14 at the beginning of the inspection period. Outage work included steam generator replacement and a major turbine overhaul. AmerenUE completed the refueling outage and synchronized the generator to the grid on November 19, 2005. The licensee returned to full power operations on November 23, 2005. AmerenUE operated the plant at full power for the remainder of the inspection period.

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

1R01 Adverse Weather Protection (71111.01)

a. Inspection Scope

Readiness for Seasonal Susceptibilities

The inspectors completed a review of the licensee's readiness of seasonal susceptibilities involving extreme low temperatures. The inspectors: (1) reviewed plant procedures, the Final Safety Analysis Report (FSAR), and Technical Specifications (TS) to ensure that operator actions defined in adverse weather procedures maintained the readiness of essential systems; (2) walked down portions of the two systems listed below to ensure that adverse weather protection features (heat tracing, space heaters, weatherized enclosures, temporary chillers, etc.) were sufficient to support operability, including the ability to perform safe shutdown functions; (3) evaluated operator staffing levels to ensure the licensee could maintain the readiness of essential systems required by plant procedures; and (4) reviewed the corrective action program to determine if the licensee identified and corrected problems related to adverse weather conditions.

- November 17, 2005: Essential service water pump house, Trains A and B

Documents reviewed by the inspectors included:

- Procedure OTS-ZZ-00007, Plant Cold Weather, Revision 10
- Procedure OTN-QJ-00003, Plant Freeze Protection Heat Tracing Procedure, Revision 3

The inspectors completed one sample.

b. Findings

No findings of significance were identified.

1R04 Equipment Alignment (71111.04)

Partial Walkdowns

a. Inspection Scope

The inspectors: (1) walked down portions of three risk important systems and reviewed plant procedures and documents to verify that critical portions of the selected systems were correctly aligned; and (2) compared deficiencies identified during the walkdown to the licensee's FSAR and corrective action program to ensure problems were being identified and corrected.

- October 17, 2005, Emergency diesel generator (EDG), Train A
- November 8, 2005, Containment recirculation sump, Train A
- December 21, 2005, Centrifugal charging pump, Train A

Documents reviewed by the inspectors are listed in the attachment.

The inspectors completed three samples.

b. Finding - Inadequate Inspection of the Containment Recirculation Sump

Introduction: The NRC identified a Green noncited violation (NCV) of 10 CFR Part 50, Appendix B, Criterion V, for the failure to adequately implement work order instructions and a procedure for inspection of the containment recirculation sump enclosure. The licensee's inspections failed to identify a 1.5-inch hole in the sump cover which could provide a path for foreign material to enter into the containment sump emergency core cooling system (ECCS) containment recirculation sump.

Description: On November 8, 2005, the inspectors identified a 1.5-inch hole penetrating the containment recirculation Sump A ceiling. FSAR Section 6.2.2.1.2.2 stated that "the recirculation sumps are covered with the concrete pads supporting the accumulator tanks; thus, debris cannot fall directly upon the screening structure." FSAR Table 6.2.2-1 established a maximum ⅛-inch gap for the sump screen. The screen prevents the introduction of foreign material and debris that could degrade long-term core cooling during an ECCS recirculation mode of operation. NRC Bulletin 2003-01, "Potential Impact of Debris Blockage on Emergency Sump Recirculation at Pressurized-Water Reactors," alerted the licensee to the susceptibility of recirculation sump failures. AmerenUE's August 8, 2003, response to the bulletin included a commitment to inspect the containment sumps and verify screen gap tolerances. The Callaway quality control technicians' detailed inspection on April 27, 2004 (Work Package W229952), did not identify the 1.5-inch hole. During Refueling Outage RF-14, AmerenUE performed Procedure OSP-EJ-00003, "Containment Recirculation Sump Inspection," Revision 5, that required quality control and operations personnel to verify that all sump penetrations were sealed prior to reactor startup. This inspection performed on November 8, 2005, did not identify the hole in the containment sump cover.

Analysis: The performance deficiency associated with this finding involved licensee personnel failure to effectively inspect the containment sump to assure any opening or gaps in the sump cover were in accordance with the design basis. This finding was greater than minor because it is associated with the mitigating systems cornerstone attribute of equipment performance and affects the associated cornerstone objective to ensure availability and reliability of the containment recirculation sump ECCS safety function. Using the Manual Chapter 0609, “Significance Determination Process,” Phase 1 Worksheet, this finding is determined to have very low safety significance because the condition is a qualification deficiency confirmed not to result in loss of operability per Part 9900, Technical Guidance, “Operability Determination Process for Operability and Functional Assessment.” The cause of this finding is related to the crosscutting element of human performance in that personnel failed to adequately implement a work instruction and procedure for inspecting the containment sump configuration.

Enforcement: The inspectors identified an NCV of 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," because AmerenUE did not properly implement work instructions and a test procedure for inspecting the ECCS containment sump. Contrary to **the above, inspections of the containment sump conducted on April 27, 2004, to verify sump design and licensing basis and the subsequent November 8, 2005, sump inspection did not** verify conformance of containment Sump A. The corrective actions to restore compliance included repair of the hole and actions taken to improve inspection techniques. Because of the very low safety significance and the licensee’s action to place this issue in their corrective action program as Callaway Action Request (CAR) 200509189, this violation is being treated as an NCV in accordance with Section VI.A.1 of the Enforcement Policy (NCV 05000483/2005005-01).

1R05 Fire Protection (71111.05)

a. Inspection Scope

Quarterly Inspection

The inspectors walked down the nine listed plant areas to assess the material condition of active and passive fire protection features and their operational lineup and readiness. The inspectors: (1) verified that transient combustibles and hot work activities were controlled in accordance with plant procedures; (2) observed the condition of fire detection devices to verify they remained functional; (3) observed fire suppression systems to verify they remained functional and that access to manual actuators was unobstructed; (4) verified that fire extinguishers and hose stations were provided at their designated locations and that they were in a satisfactory condition; (5) verified that passive fire protection features (electrical raceway barriers, fire doors, fire dampers, steel fire proofing, penetration seals, and oil collection systems) were in a satisfactory material condition; (6) verified that adequate compensatory measures were established for degraded or inoperable fire protection features and that the compensatory measures were commensurate with the significance of the deficiency; and (7) reviewed the FSAR to determine if the licensee identified and corrected fire protection problems.

- September 25, 2005, Fire Area RB, Reactor building
- November 10, 2005, Fire Area RB, Reactor building
- November 22, 2005, Fire Area A-2, ECCS, Train A
- November 22, 2005, Fire Area A-4, ECCS Rooms, Train A
- November 22, 2005, Fire Area A-9, Residual heat removal (RHR) heat exchanger room, Train A
- November 22, 2005, Fire Area A-10, RHR heat exchanger room, Train B
- November 30, 2005, Fire Area C-9, Switchgear room, Train A
- November 30, 2005, Fire Area C-10, Switchgear room, Train B
- November 30, 2005, Fire Area D-1, Diesel generator, Train A

Documents reviewed by the inspectors are listed in the attachment.

The inspectors completed nine samples.

b. Finding - Failure to Adequately Implement Continuous Compensatory Fire Watches

Introduction: The inspectors identified a noncited violation of TS 5.4.1.d, "Fire Protection Program Implementation," associated with seven examples of inadequately performed continuous fire watches.

Description: Procedure APA-ZZ-0703, "Fire Protection Operability Criteria and Surveillance Requirements," required AmerenUE to establish compensatory continuous watches in specified fire areas as a result of degraded fire detection or suppression capability. The continuous fire watch is an uninterrupted observation post within a single fire area. The physical presence of fire watch personnel provides reasonable assurance that a fire would be prevented through prompt recognition and disposition of fire hazards. If a fire occurred, despite these efforts, fire watch personnel would promptly detect, report, and extinguish the fire while still in the incipient stage.
Procedure SDP-KC-00001, "Requirements for and Duties of Compensatory Fire Watches," Revision 5, required fire watches to maintain watch over the entire assigned space with a minimum of patrolling.

In September 2005, AmerenUE provided verbal guidance to fire watch personnel that continuous watch requirements may be met by a 15-minute roving fire patrol. Callaway Facility Operating License, Amendment 169 (5) (d), required that changes that adversely affect the ability to achieve and maintain safe shutdown in the event of a fire receive prior NRC approval. The inspectors concluded that reducing continuous watch requirements to a 15-minute roving patrol adversely affected the ability to achieve and maintain safe shutdown in the event of a fire. The inspections identified seven examples of compensatory continuous fire watches where one fire watch person was assigned simultaneously to multiple fire areas and building levels:

Date	Fire Impairment Number	Continuous Fire Areas Concurrently Watched by a Single Individual
September 5	12260	A-1, A-8, A11, A12, A-24, A-25
September 6	12260	A-1, A-8, A11, A12, A-24, A-25
September 7	12260	A-1, A-8, A11, A12, A-24, A-25
September 8	12260	A-1, A-8, A11, A12, A-24, A-25
September 25	12269	A-1, A-8, A11, A12, A-24, A-25
September 26	12269	A-1, A-8, A11, A12, A-24, A-25
September 30	12244	A-1, A-13, A-14, A-15

Analysis: The performance deficiency associated with this finding involved the failure of AmerenUE to establish adequate continuous fire watches. This finding is greater than minor because this finding was associated with the reactor safety mitigating systems cornerstone attribute to provide protection against external factors and affects the associated cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. 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 the "Fixed Fire Protection Systems" element of fire watches posted as a compensatory measure for outages or degradations. A low degradation rating was assigned to this finding as the provision affected by this finding is expected to display nearly the same level of effectiveness and reliability. Using Manual Chapter 0609, Appendix F, this finding is determined to have very low safety significance. The inspectors concluded that the new guidance created situations which resulted in inadequate compensatory fire watch coverage. The cause of this finding is related to the crosscutting element of human performance in that the guidance was not adequate to ensure continuous fire watches were appropriately implemented.

Enforcement: Callaway Plant Technical Specification 5.4.1.d, "Fire Protection Program Implementation," required that the Fire Prevention Program be implemented and maintained per written procedures. The Fire Prevention Program requirements for fire watches were implemented by Procedure SDP-KC-00001, "Requirements for and Duties of Compensatory Fire Watches," Revision 5. Procedure SDP-KC-00001 established a requirement for compensatory continuous watches within specified fire areas as a result of degraded fire detection or suppression capability. Contrary to Procedure SDP-KC-00001 and the fire program, AmerenUE failed to perform compensatory continuous watches within certain specified fire areas with degraded fire detection or suppression capability between September 5 and 30, 2005. Because this finding is of very low safety significance and was entered into the licensee's corrective action program (CAR 200510325), it is being treated as an NCV, consistent with Section VI.A of the NRC Enforcement Policy (NCV 05000483/2005005-02).

1R07 Heat Sink Performance (71111.07)

a. Inspection Scope

The inspectors reviewed licensee programs, verified performance against industry standards, and reviewed critical operating parameters and maintenance records for the containment cooler heat exchangers. The inspectors verified that: (1) performance tests were satisfactorily conducted for heat exchangers/heat sinks and reviewed for problems or errors; (2) the licensee utilized the periodic maintenance method outlined in Electric Power Research Institute NP-7552, "Heat Exchanger Performance Monitoring Guidelines;" (3) the licensee properly utilized biofouling controls; (4) the licensee's heat exchanger inspections adequately assessed the state of cleanliness of their tubes, and (5) the heat exchanger was correctly categorized under the maintenance rule.

Documents reviewed by the inspectors are listed in the attachment.

The inspectors completed one sample.

b. Findings - Indeterminate Containment Cooler Operability and Heat Removal Capability

Introduction: An unresolved item was identified for containment cooler heat removal capability. AmerenUE will provide the inspectors additional testing results to complete the inspection. This issue will remain unresolved pending additional review by the inspectors. No analysis or enforcement reviews were performed for this unresolved item.

Description: The inspectors reviewed available containment cooler testing data but were not able to confirm that the heat exchangers were capable of the design bases heat removal duty. FSAR Section 6.2.1.3, "Mass and Energy Release Analyses for Postulated Loss-of-Coolant Accidents," and Section 6.2.1.4, "Mass and Energy Release Analysis for Postulated Secondary Pipe Ruptures Inside Containment," stated that a containment cooler duty of 141 million British Thermal Units per hour, at 277EF was used in the accident analysis. TS Surveillance Bases 3.6.6.7, "Containment Spray and Cooling Systems," stated that the heat removal capability of each cooler train was verified on an 18-month frequency. TS bases, Figure B.3.6.6-1, "Containment Cooler Heat Removal Minimum Cooling Flow Rate," established the minimum heat removal capability as a function of essential service water (ESW) flow, assuming no fouling, to meet design bases requirements. Plant engineering monitored ESW flow but not heat removal capability. AmerenUE committed, by letter, "Response to Generic Letter 89-13, Service Water System Problems Affecting Safety Related Equipment," January 29, 1990, to verify the heat transfer capability of all safety-related heat exchangers cooled by ESW. In addition, AmerenUE also committed to trend and compare the containment cooler heat removal rates to the design requirements to promote identification of degraded cooling equipment. Title 10 of the Code of Federal Regulations, Part 50, Appendix B, "Test Control," required AmerenUE to establish a test program to assure that the containment cooler's performance satisfactorily met acceptance limits established in applicable design documents. Based on the information provided by AmerenUE, the inspectors were not able to conclude that the containment coolers were capable of removing design basis heat loads.

AmerenUE identified high differential pressure across the ESW side of containment Cooler SGN01A on May 17, 2004 (Refueling Outage 14 Work Document P701990). The high differential pressure was indicative of heat exchanger degradation due to macrofouling. AmerenUE restarted and operated the plant until September 17, 2005, without adequately assessing the affect of fouling on heat exchanger performance. AmerenUE cleaned the heat exchanger during Refueling Outage 14. AmerenUE did not perform testing prior to the cleaning to determine if any additional degradation had occurred during the 18-month operating cycle. The inspectors were not able to verify, based on the documentation reviewed, that the heat exchanger was capable of performing the design bases function during Cycle 14. This issue is considered unresolved pending additional NRC review of AmerenUE containment cooler testing (Unresolved Item 05000483/2005005-03).

1R11 Licensed Operator Requalification Program (71111.11Q and 71111.11B)

.1 Quarterly Inspections

a. Inspection Scope

The inspectors observed testing and training of senior reactor operators and reactor operators to identify deficiencies and discrepancies in the training, to assess operator performance, and to assess the postexercise critique. The inspectors observed a “Just In-Time Reactor Startup” training scenario conducted on November 13, 2005.

Documents reviewed by the inspectors are listed in the attachment.

The inspectors completed one sample.

b. Findings

No findings of significance were identified.

.2 Biennial Inspection

a. Inspection Scope

To assess the performance effectiveness of the licensed operator requalification program, the inspectors conducted both on-site and in-office reviews involving personnel interviews, operating and written examinations, and operating examination activities.

During the on-site review, the inspectors interviewed five licensee personnel, consisting of three instructors, one operator and a training supervisor, to determine their understanding of the policies and practices for administering requalification examinations. The inspectors also reviewed operator performance on the written and operating examinations. These reviews included observations of portions of the operating examination by the inspectors. The operating examinations observed included job performance measures and four scenarios that were used in the current biennial requalification cycle. These observations allowed the inspectors to assess the licensee's effectiveness in conducting the operating test to ensure operator mastery of the training program content.

The results of these examinations were reviewed to determine the effectiveness of the licensee's appraisal of operator performance and to determine if feedback of performance analysis into the requalification training program was being accomplished. The inspectors interviewed members of the training department and reviewed minutes of training review group meetings to assess the responsiveness of the licensed operator requalification program in incorporating the lessons learned from both plant and industry events. Examination results were also assessed to determine if they were consistent with the guidance contained in NUREG 1021, "Operator Licensing Examination Standards for Power Reactors," Revision 9, and NRC Manual Chapter 0609, Appendix I, "Operator Requalification Human Performance Significance Determination Process."

Additionally, the inspectors assessed the Callaway Plant-referenced simulator for compliance with 10 CFR 55.46, "Simulator Facilities." This assessment included the adequacy of the licensee's simulation facility for use in operator licensing examinations and for satisfying experience requirements as prescribed by 10 CFR 55.46. In addition, the inspectors reviewed selected applicant personnel qualitative statements (NRC Form 398) to verify their accuracy. During the Form 398 reviews, the inspectors noted that several applicants were given credit for reactivity and control manipulations on the simulator instead of on the actual plant. While this simulator usage is permitted by 10 CFR 55.46, the simulator must meet the standards of fidelity as required by 10 CFR 55.46(c)(2). Based on this observation and the requirements of 10 CFR 55.46, the inspectors expanded their review of the simulator testing. This review expansion included a review of the simulator annual performance test book. The inspectors reviewed a sample of simulator performance test records (transient tests, surveillance tests, and malfunction tests), simulator deficiency report records, and processes for ensuring simulator fidelity commensurate with 10 CFR 55.46. The inspectors reviewed selected simulator deficiency reports generated by the licensee that did not result in changes to the configuration of the simulator to assess the responsiveness of the licensee's simulator configuration management program. The inspectors also interviewed members of the licensee's simulator configuration control group as part of this review.

During the in-office review, the inspectors evaluated whether the written examination was developed and administered in accordance with the standards described in NUREG 1021 and evaluated any issues identified in accordance with NRC Manual Chapter 0609, Appendix I. The written examination review was focused on quality aspects of the examination, such as discrimination validity, examination question psychometric quality, and examination integrity.

b. Findings

1. Evaluation of the Written Examination

As a result of the review of the written requalification examinations, the inspectors identified that the quality of the examinations developed by the licensee appeared to not meet the guidance set forth in NUREG 1021, ES-602, Attachment 1, Section B, "Open-Reference Guidelines." The term "open reference" means that the candidates are allowed to use any reference to assist them when taking the examination.

Since the operators are allowed to use examination question references while taking the examination, test questions should be developed that do more than test for mere recall and/or memorization. Open-reference questions should have the operators demonstrate an understanding of an issue by using their knowledge to address real-life situations and problems. The NUREG further states with regard to direct look up questions that removing from the stem of the question any information that cues the operator to the answer's location does not make the question acceptable.

With regard to the open-reference questions, the NUREG also addresses "Direct Lookup" questions. Direct lookup questions only test memory because the information is readily available. This is a less valid means of testing candidate knowledge and only demonstrates that a candidate knows where to find information. Therefore, the discrimination validity of the question is critical to differentiate the safe operator from the unsafe operator.

Additionally, other than demonstrating that a candidate knows where to find information, the licensee's biennial requalification examinations appeared to not test the understanding or analysis of the information that would be applied on the job. These issues will be reviewed as Unresolved Item (URI) 05000483/2005005-04, Adequacy of the Biennial Requalification Written Examination (CAR 200600528).

2. Simulation Facility Performance

Introduction: During a review of the simulator annual performance test book, the inspectors identified a Green finding for the failure to conduct simulator performance testing in accordance with ANSI/ANS 3.5, "Nuclear Power Plant Simulators for use in Operator Training and Examination," 1998.

Description: A review of the Steady State and Normal Evolution tests contained in the annual performance test book for the simulator revealed that the licensee did not compare all of the required parameters listed in ANSI 3.5-1998 to actual plant data; specifically, Thot, Tcold, core megawatt thermal, steam flow, feed flow, letdown flow, charging flow, and turbine first stage pressure. In lieu of this comparison, the licensee utilized an "expert panel review" to determine if the simulator operation mimics the actual plant. When the inspectors requested the baseline data to support the analysis documentation, the licensee was unable to provide the data. The licensee stated that the analysis was done by a panel of experts and that the signature on the meeting minutes constituted the required analysis and baseline data. The 1998 version of

ANSI/ANS 3.5, requires that the annual simulator performance tests be conducted such that the key parameters listed in Appendix B of this standard are recorded and that these records be compared to actual or reference plant data (if available) or engineering data from the FSAR. If such engineering data is not available in the FSAR, the standard permits the use of data from subject matter expert estimates to determine acceptability of the test.

Analysis: The inspectors determined that the failure to adhere to ANSI/ANS 3.5-1998, as endorsed by Regulatory Guide 1.149, "Nuclear Power Plant Simulation Facilities for Use in Operator Training and License Examinations," Revision 3, October 2001, as committed to in the Callaway Plant Simulation certification dated March 13, 2000, was a performance deficiency. Specifically, the simulator performance testing did not meet the standards specified in ANSI/ANS 3.5-1998 in that: (1) all required parameters during the simulator test were not recorded; and (2) simulator to baseline data comparisons were unavailable.

The NRC has determined that traditional enforcement does not apply because the issue did not have any actual safety consequence or potential for affecting the NRC's regulatory function and did not result in any willful violation of NRC requirements or licensee procedures. The performance deficiency is more than minor because it affected the ability of the simulator transient tests to detect fidelity issues with the simulator and affects the Human Performance (Human Error) attribute of the Initiating Events and Mitigating Systems cornerstones.

Enforcement: No violation of regulatory requirements occurred. The examiners determined that the finding did not represent a noncompliance because Callaway Plant performed some testing even though the testing was not sufficient in scope and because no actual events have occurred that could be attributed to a lack of simulator fidelity testing: Finding (FIN) 05000483/2005005-05, Failure to Conduct Simulator Testing in Accordance with ANSI/ANS 3.5-1998 (CAR 200600527).

3. Adequacy of Plant-Referenced Simulator to Conform with Simulator Requirements for Reactivity and Control Manipulation Credits

As the result of reviewing NRC Form 398, the inspectors noted that the licensee used the simulator to meet reactivity and control manipulation experience requirements for initial operator and senior operator license applicants in accordance with 10 CFR 55.46(c)(2)(ii). For the manipulations, the licensee used a single page "sign-off" sheet for documentation. To use the simulator for reactivity and control manipulation credit, the regulation requires that significant control manipulations are completed without procedural exceptions, simulator performance exceptions, or deviation from the approved training scenario sequence. Furthermore, the ANSI standard requires that these items be performed without offsets in the simulator and without time-compression techniques that expected alarms are generated as required in real time with no unexpected alarms generated during the scenario sequence. The documentation provided could not be used to verify each of the requirements as specified in the regulations and standards.

The safety significance of this issue could be more than minor due to the apparent failure to meet the requirements of 10 CFR 55.46(c)(2)(ii) with regard to assuring maintenance of the plant referenced simulator fidelity. Accordingly, a URI was opened pending further review of the simulator in subsequent inspections. The licensee entered this issue into their corrective action program as CAR 200600529: URI 05000483/200505-06, Adequacy of Plant-Referenced Simulator to Conform with Simulator Requirements for Reactivity and Control Manipulation Credits.

1R12 Maintenance Effectiveness (71111.12Q)

a. Inspection Scope

The inspectors reviewed the two listed maintenance activities to: (1) verify the appropriate handling of structures, systems, and components (SSC) performance or condition problems; (2) verify the appropriate handling of degraded SSC functional performance; (3) evaluate the role of work practices and common cause problems; and (4) evaluate the handling of SSC issues reviewed under the requirements of the maintenance rule, 10 CFR Part 50, Appendix B, and the TSs.

- September 30, 2005, CAR 200507636, Missing spring in ventilation door solenoid lock assembly

- August 2, 2005, CAR 200505344, Fuel building roll-up door

Documents reviewed by the inspectors included:

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

- Maintenance Rule Program

The inspectors completed two samples.

b. Findings

No findings of significance were identified.

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

a. Inspection Scope

Risk Assessment and Management of Risk

The inspectors reviewed the three listed assessment activities to verify:
(1) performance of risk assessments when required by 10 CFR 50.65 (a)(4) and licensee procedures prior to changes in plant configuration for maintenance activities and plant operations; (2) the accuracy, adequacy, and completeness of the information considered in the risk assessment; (3) that the licensee recognizes, and/or enters as

applicable, the appropriate licensee-established risk category according to the risk assessment results and licensee procedures; and (4) the licensee identified and corrected problems related to maintenance risk assessments.

- October 17, 2005, Essential power, Train B, planned outage, in-office review

- October 31, 2005, Spent fuel pool time-to-boil method, in-office review

- November 21, 2005, Unplanned emergent maintenance on ESW inlet isolation Valve EFHV52, in-office review

Documents reviewed by the inspectors included:

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

- Procedure EDP-ZZ-01129, Callaway Plant Risk Assessment, Revision 8

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

The inspectors completed three samples.

Emergent Work Control

The inspectors: (1) verified that the licensee performed actions to minimize the probability of initiating events and maintained the functional capability of mitigating systems and barrier integrity systems; (2) verified that emergent work-related activities such as troubleshooting, work planning/scheduling, establishing plant conditions, aligning equipment, tagging, temporary modifications (TMs), and equipment restoration did not place the plant in an unacceptable configuration; and (3) reviewed the FSAR to determine if the licensee identified and corrected risk assessment and emergent work control problems.

- October 17, 2005, Essential power, Train B, planned outage. The inspectors observed compensatory risk mitigation actions from the control building and completed an in-office review.

- November 21, 2005, ESW inlet isolation Valve EFHV52. The inspectors observed compensatory risk mitigation actions from the control building and completed an in-office review.

Documents reviewed by the inspectors included:

- Nuclear Management and Resource Council 93-01, Industry Guidelines for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants, Revision 3

- Procedure EDP-ZZ-01129, Callaway Plant Risk Assessment, Revision 9

The inspectors completed two samples.

b. Findings

No findings of significance were identified.

1R14 Personnel Performance During Nonroutine Plant Evolutions (71111.14)

a. Inspection Scope

The inspectors: (1) reviewed operator logs, plant computer data, and/or strip charts for the below listed evolutions to evaluate operator performance in coping with nonroutine events and transients; (2) verified that operator actions were in accordance with the response required by plant procedures and training; (3) attended and/or reviewed postevent critic meetings; and (4) verified that the licensee has identified and implemented appropriate corrective actions associated with personnel performance problems that occurred during the nonroutine evolutions sampled.

- November 7, 2005, CAR 200509143, Rapid pressurizer surge line cooldown due to melting lead blankets

- November 8, 2005, CAR 200509191, Pressurizer surge line heatup rate exceeded

- November 14, 2005, CAR 200509345, Unplanned securing of the steam dumps and subsequent reactor coolant system (RCS) heatup with initiating RCS temperature at 340EF

- November 15, 2005, Plant cooldown to remove a shim on Steam Generator D

Documents reviewed by the inspectors included:

- Procedure OTG-ZZ-00001, Plant Heatup, Cold Shutdown to Hot Standby, Revision 46

- Procedure APA-ZZ-00500, Corrective Action Program, Revision 38

- Procedure OSP-BB-00007, RCS Heatup and Cooldown Limitations, Revision 9

The inspectors completed four samples.

b. Finding

Failure to Follow Procedures Resulted in Violation of RCS Cooldown and Heatup Rate Limits

Introduction. The inspectors identified a Green NCV of TS 5.4.1.a, “Procedures,” after AmerenUE operations personnel failed to maintain the RCS temperature limits on two occasions.

Description. On November 7, 2005, plant operators terminated a plant heatup and decreased the RCS pressurizer surge line temperature 260EF in one hour. The operators initiated the rapid cooldown by isolating pressurizer auxiliary spray, resulting in an in-surge of cooler RCS water. The operators conducted the rapid cooldown after several containment lead shield blanket polyvinylchloride covers in containment unexpectedly melted. The shield blankets had not been removed from the uninsulated pressurizer surge line prior to plant heatup due to a work scheduling error. The licensee identified a second example of excessive surge line temperature on November 8, 2005. Plant operators increased the surge line temperature about 175EF in one hour during a plant heatup.

TS 3.4.3, “RCS Pressure and Temperature (P/T) Limits,” required temperature changes of all RCS components (except the pressurizer) be limited to 100EF in one hour. The TS Bases defined the surge line as part of the RCS. General Operating Procedure OTG-ZZ-00001, “Plant Heatup, Cold Shutdown to Hot Standby,” required operating personnel maintain greater than 5 gpm auxiliary spray and a pressurizer outsurge. Procedure OSP-BB-00007, “RCS Heatup and Cooldown Limitations,” required that RCS temperature changes not exceed 100EF in one hour during cooldown/heatup evolutions. The inspectors identified that operations personnel failed to recognize the applicability of the TS and apply the appropriate TS action statement.

Analysis: The performance deficiency associated with this finding involved failure of operations personnel to follow established procedures and recognize the appropriate TS action. This finding was greater than minor because it is associated with the reactor safety barrier integrity cornerstone attribute of equipment performance and affects the associated cornerstone objective to ensure reasonable assurance that the RCS piping barrier will protect the public from radionuclide releases caused by accidents or events. Using Manual Chapter 0609, “Significance Determination Process,” Appendix G, “Shutdown Operations,” this finding was determined to have very low safety significance because, based on the engineering evaluation of RCS thermal stress resulting from the temperature transients, the condition did not significantly increase the likelihood of a loss of RCS inventory and did not degrade the licensee’s ability to terminate a leak path. The cause of this finding is related to the crosscutting element of human performance because of personnel failure to follow procedures.

Enforcement: TS 5.4.1.a, “Procedures,” required that written procedures be established, implemented, and maintained covering the activities specified in Appendix A, “Typical Procedures for Pressurized Water Reactors,” of Regulatory Guide 1.33, “Quality Assurance Program Requirements (Operation),” February 1978.

Regulatory Guide 1.33, Appendix A, Section 2a, required general plant operating procedures for cold shutdown to hot standby to be implemented. Entry into TS 3.4.3, "RCS Pressure and Temperature (P/T) Limits," action was required when an RCS component temperature transient exceeded 100EF cooldown and or heatup limit within a one-hour period. Callaway Procedure OSP-BB-00007, "RCS Heatup and Cooldown Limitations," required that RCS temperature changes shall not exceed 100EF in one hour during cooldown or during heatup evolutions. Contrary to these requirements, on November 7 and 8, 2005, operations personnel did not maintain the RCS temperature rate less than 100EF within one hour. Because of the very low safety significance and the licensee's action to place this issue in their corrective action program as CARs 200509487 and 200509143, this violation is being treated as an NCV in accordance with Section VI.A.1 of the Enforcement Policy (NCV 50-483/2005005-07).

1R15 Operability Evaluations (71111.15)

a. Inspection Scope

The inspectors: (1) reviewed plant status documents such as operator shift logs, emergent work documentation, deferred modifications, and standing orders to determine if an operability evaluation was warranted for degraded components; (2) referred to the FSAR and design basis documents to review the technical adequacy of licensee operability evaluations; (3) evaluated compensatory measures associated with operability evaluations; (4) determined degraded component impact on any TSs; (5) used the significance determination process to evaluate the risk significance of degraded or inoperable equipment; and (6) verified that the licensee has identified and implemented appropriate corrective actions associated with degraded components.

- Operability Determination 200509277, Overpressurization of the turbine-driven auxiliary feedwater pump (TDAFP) during the backleakage test of its discharge check valve

- Operability Determination 200509374, Pressurizer power-operated relief valve stroke time basis

- Operability Determination 200509368, Excessive stroke time of feedwater isolation Valve AEFV0040

- Operability Determination 200505062, Insufficient time to transfer ECCS and containment spray to cold leg recirculation

- Operability Determination 2005003773, Degraded containment cooler heat removal capability

The inspectors completed five samples.

b. Findings

No findings of significance were identified.

1R16 Operator Workarounds (71111.16)

a. Inspection Scope

Selected Operator Workarounds

The inspectors reviewed the two listed operator workarounds to: (1) determine if the functional capability of the system or human reliability in responding to an initiating event is affected; (2) evaluate the effect of the operator workaround on the operator's ability to implement abnormal or emergency operating procedures; and (3) verify that the licensee has identified and implemented appropriate corrective actions associated with operator workarounds.

- November 23, 2005, In-office review of the degradation of main steam line Monitor 16
- November 23, 2005, Maintenance repair of Bistable SB069 and permissive indicating panel

Documents reviewed by the inspectors included:

- December 2005, Operator Work Around and Burdens list
- Procedure APA-ZZ-00018, Conduct of Operations - Quality Control, Revision 7
- Procedure ODP-ZZ-00001, Operations Department - Code of Conduct, Revision 25

The inspectors completed two samples.

Cumulative Review of the Effects of Operator Workarounds

The inspectors reviewed the cumulative effects of operator workarounds to determine: (1) the reliability, availability, and potential for misoperation of a system; (2) if multiple mitigating systems could be affected; (3) the ability of operators to respond in a correct and timely manner to plant transients and accidents; and (4) if the licensee has identified and implemented appropriate corrective actions associated with operator workarounds.

The inspectors reviewed the Operator Workaround and Burdens List.

The inspectors completed one sample.

b. Findings

No findings of significance were identified.

1R17 Permanent Plant Modifications (71111.17)

a. Inspection Scope

Annual Review

The inspectors reviewed key affected parameters associated with energy needs, materials/replacement components, timing, heat removal, control signals, equipment protection from hazards, operations, flowpaths, pressure boundary, ventilation boundary, structural, process medium properties, licensing basis, and failure modes for the modification listed below. The inspectors verified that: (1) modification preparation, staging, and implementation did not impair emergency/abnormal operating procedure actions, key safety functions, or operator response to loss of key safety functions; (2) postmodification testing maintained the plant in a safe configuration during testing by verifying that unintended system interactions will not occur, SSC performance characteristics still meet the design basis, the appropriateness of modification design assumptions, and the modification test acceptance criteria has been met; and (3) the licensee has identified and implemented appropriate corrective actions associated with permanent plant modifications.

- November 1, 2005, Modification MP 05-3051, Containment Sump Valves EJHV8811A and EJHV8811B. The inspectors performed an in-office review and performed a walkdown of the affected equipment in the auxiliary building.

Documents reviewed by the inspectors are listed in the attachment.

The inspectors completed one sample.

b. Findings - Use of a Nonqualified Calculation in a Safety Related Modification

Introduction. The NRC identified a Green NCV of 10 CFR Part 50, Appendix B, Criteria V, “Instructions, Procedures, and Drawings,” associated with an inadequate engineering procedure used to verify calculations. The inadequate procedure resulted in the use of a nonqualified, nonsafety-related engineering calculation to demonstrate the safety function of the containment recirculation sump valves following a modification.

Description: AmerenUE failed to ensure a nonsafety-related vendor supplied calculation was qualified before use to demonstrate the design bases function of safety-related components after a modification. AmerenUE identified that maximum postaccident differential pressure assumed between the containment recirculation sump and RHR system was incorrect. Based on industry operational experience (OE), engineering determined the maximum design differential pressure the containment sump valve operators would have to open against increased from 53 pounds per square inch differential (psid) to 468 psid. The Engineering Department generated Modification MP 05-3051, Containment Sump Valves EJHV8811A and EJHV8811B, to increase valve operator opening torque. To support the modification, AmerenUE purchased nonsafety-related Calculation KCI 330-001-DC1, Revision 0, October 18, 2005, from a vendor. The calculation used a new, realistic approach to establish the maximum valve operator torque. Plant engineering used the operator torque developed from this calculation to ensure the sump valves would open against the higher differential pressure after modification. Engineering personnel used Procedure EDP-ZZ-04023, "Calculations," Revision 17, to qualify the vendor supplied calculation before approved use in the safety-related application. Procedure EDP-ZZ-04023 provided insufficient detail to enable engineering personnel to verify the design by either an alternate method or suitable test program to qualify the nonsafety-related calculation.

Analysis: The performance deficiency associated with this finding involved the failure of engineering personnel to only use qualified calculations for safety-related modifications. This finding is greater than minor because, if left uncorrected, this finding would become a more significant safety concern affecting other safety-related modifications. This finding affected the mitigating systems cornerstone. Using the Manual Chapter 0609, "Significance Determination Process," Phase 1 Worksheet, this finding is determined to have very low safety significance because this finding involves a design deficiency confirmed not to result in loss of operability per Part 9900, Technical Guidance, "Operability Determination Process for Operability and Functional Assessment." The cause of this finding is related to the crosscutting element of human performance in that the procedure did not ensure the calculations were qualified to support a design basis function of a safety-related component.

Enforcement: Title 10 of the Code of Federal Regulations, Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," required that activities affecting quality be prescribed by documented instructions or procedures appropriate to the circumstances. Contrary to this, Procedure EDP-ZZ-04023, required for an activity affecting quality, was not appropriate to the circumstances. Specifically, on October 28, 2005, Procedure EDP-ZZ-04023 was not adequate to ensure the qualification of nonsafety-related Calculation KCI 330-001-DC1, Revision 0, before use in Calculation EJ-42, Revision 0, an activity affecting quality. Because this finding is of very low safety significance and was entered into AmerenUE's Corrective Action Program (CAR 200509849), this violation is being treated as an NCV, consistent with Section VI.A of the NRC Enforcement Policy (NCV 05000483/2005005-08).

1R19 Postmaintenance Testing (71111.19)

a. Inspection Scope

The inspectors selected the six listed postmaintenance test (PMT) activities of risk significant systems or components. For each item, the inspectors: (1) reviewed the applicable licensing-basis and/or design-basis documents to determine the safety functions; (2) evaluated the safety functions that may have been affected by the maintenance activity; and (3) reviewed the test procedure to ensure it adequately tested the safety function that may have been affected. The inspectors either witnessed or reviewed test data to verify that acceptance criteria were met, plant impacts were evaluated, test equipment was calibrated, procedures were followed, jumpers were properly controlled, the test data results were complete and accurate, the test equipment was removed, the system was properly re-aligned, and deficiencies during testing were documented. The inspectors also reviewed the FSAR to determine if the licensee identified and corrected problems related to postmaintenance testing.

- September 29, 2005, PMT W236012/920, Containment cooler train. The inspectors observed the PMT from the reactor building and the control room and performed an in-office review.

- October 12, 13, and 14, 2005, PMTs 222071/912, and W715936/900, ESW Train A, motor and pump replacement. The inspectors observed the PMT from the ESW pump room and the control room and performed an in-office review.

- October 12, 2005, PMTs W236513/900, W236509/940, and P711090/900, EDG Train A, major overhaul. The inspectors observed the PMT from the EDG room and the control room and performed an in-office review.

- November 22, 2005, PMTs 05110929/200 and 05110929/910, TDAFP discharge Check Valve ALHV0054. The inspectors observed the PMT from the auxiliary building and the control room and performed an in-office review.

- November 1, 2005, PMT P721574/910, Overhaul of NN Inverter 14. The inspectors performed an in-office review.

- December 28, 2005, PMT 05112449/900, Ultimate heat sink electrical room fan. The inspectors performed an in-office review.

Documents reviewed by the inspectors are listed in the attachment.

The inspectors completed six samples.

b. Findings

No findings of significance were identified.

1R20 Refueling and Outage Activities (71111.20)

a. Inspection Scope

The inspectors reviewed the following risk significant refueling items or outage activities to verify defense-in-depth commensurate with the outage risk control plan, compliance with the TSs, and adherence to commitments in response to Generic Letter 88-17, “Loss of Decay Heat Removal”: (1) the risk control plan; (2) tagging/clearance activities; (3) RCS instrumentation; (4) electrical power; (5) decay heat removal; (6) spent fuel pool cooling; (7) inventory control; (8) reactivity control; (9) containment closure; (10) reduced inventory or midloop conditions; (11) refueling activities; (12) heatup and cooldown activities; (13) restart activities; and (14) licensee identification and implementation of appropriate corrective actions associated with refueling and outage activities. The inspectors' containment inspections included observations of the containment sump for damage and debris, and supports, braces, and snubbers for evidence of excessive stress, water hammer, or aging.

- October 29, 2005, Precore alterations verifications
- October 30, 2005, ECCS full flow test, from control room
- October 31, 2005, Fuel handling from the reactor building and control room
- October 31, 2005, Spent fuel pool time-to-boil method, in-office review
- November 13, 2005, Containment closure walkdown
- November 17, 2005, Reactor startup from the control room and the outage control center

Documents reviewed by the inspectors are listed in the attachment.

The inspectors completed one sample.

b. Findings - Less Than Adequate Spent Fuel Pool Water Inventory Risk Controls

Introduction

The inspectors identified a Green finding after AmerenUE implemented less than adequate risk management controls of the spent fuel pool water inventory following reactor core offload.

Description: The inspectors identified that AmerenUE had not implemented shutdown risk administrative controls on the fuel transfer tube gate valve and the associated flange during the period the fuel was offloaded to the spent fuel pool. On September 29, 2005, the core had been off-loaded to the spent fuel pool and the transfer canal weir gate was removed. In this configuration, the fuel transfer tube valve, if opened, would provide a drain path from the spent fuel pool through an open weir wall. Valve ECV-995 was closed but not identified in the shutdown risk management system and did not have administrative controls to protect against misalignment. The licensee provided the inspectors a calculation during the inspection that demonstrated that Valve ECV-995 could be opened during the period of concern. AmerenUE's risk guidelines, specified in Procedure APA-ZZ-00150, Appendix H, "Project Risk Management Guidelines," provided for measures to be in place to avoid risk. This finding was entered into the Corrective Action Program as CARs 200507593 and 200507693.

Analysis: The performance deficiency associated with this finding involved failure of the licensee to identify and implement inventory risk controls associated with the spent fuel pool. This finding is greater than minor because, if left uncorrected, this condition could become a more significant safety concern. NRC Information Notice 2005-16, "Outage

Planning and Scheduling - Impacts on Risk," described operating experience related to refueling risk management. Information Notice 2005-16 emphasized that most spent fuel pool events had a common thread of human error and involved equipment misalignment. NRC Manual Chapter 0609, "Significance Determination Process," does not specifically address findings related to the spent fuel pool inventory. Therefore, this issue was evaluated by NRC management with input from a senior reactor analyst. This finding was determined to be of very low safety significance based on the fact that the procedure used to manipulate the valve was not in use during this period and that borated water makeup capabilities were available to the spent fuel pool.

Enforcement: No violation of regulatory requirements occurred. The inspectors determined that this finding did not represent a noncompliance because it did not involve a safety-related or TS required procedure (FIN 05000483/2005005-09).

1R22 Surveillance Testing (71111.22)

a. Inspection Scope

The inspectors reviewed the FSAR, procedure requirements, and TSs to ensure that the seven listed surveillance activities demonstrated that the SSCs tested were capable of performing their intended safety functions. The inspectors either witnessed or reviewed test data to verify that the following significant surveillance test attributes were adequate: (1) preconditioning; (2) evaluation of testing impact on the plant; (3) acceptance criteria; (4) test equipment; (5) procedures; (6) jumper/lifted lead controls; (7) test data; (8) testing frequency and method demonstrated TS operability; (9) test equipment removal; (10) restoration of plant systems; (11) fulfillment of American Society of Mechanical Engineers code requirements; (12) updating of performance indicator data; (13) engineering evaluations, root causes, and bases for returning tested SSCs not meeting the test acceptance criteria were correct; (14) reference setting data; and (15) annunciators and alarms setpoints. The inspectors also verified that the licensee identified and implemented any needed corrective actions associated with the surveillance testing.

- September 28, 2005, Surveillance S724682, Boric acid walkdown. The inspectors observed portions of the walkdown in the reactor building and completed an in-office review of the completed test documentation.

- October 30, 2005, Surveillance S72279, ECCS check valve flow test. The inspectors observed portions of the test from the reactor building and the control room and completed an in-office review of the completed surveillance test package.

- November 1, 2005, Surveillance S05514649, RCS flow test. The inspectors observed portions of the test from the control room and completed an in-office review of the completed test documentation.

- November 16, 2005, Surveillance 05513671/500, TDAFP inservice test. The inspectors observed portions of the testing in the auxiliary building and completed an in-office review of the test documentation.

- November 17, 2005, Surveillance 05511199, Estimated critical rod position. The inspectors observed portions of the testing from the control room and completed an in-office review of the test documentation.

- November 17, 2005, Surveillance 726457, Low power physics test program with dynamic rod worth measurement. The inspectors observed portions of the testing from the control room and completed an in-office review of the test documentation.

- November 25, 2005, Surveillance 05101397, Feedwater isolation valve tests. The inspectors observed portions of the testing from the control room and auxiliary building and completed an in-office review of the test documentation.

Documents reviewed by the inspectors are listed in the attachment.

The inspectors completed seven samples.

b. Findings

No findings of significance were identified.

1R23 Temporary Plant Modifications (71111.23)

a. Inspection Scope

The inspectors reviewed the FSAR, plant drawings, procedure requirements, and TSs to ensure that the three below listed TMs were properly implemented. The inspectors: (1) verified that the modifications did not have an affect on system operability/availability; (2) verified that the installation was consistent with modification documents; (3) ensured that the postinstallation test results were satisfactory and that the impact of the temporary modifications on permanently installed SSCs were supported by the test; (4) verified that the modifications were identified on control room drawings and that appropriate identification tags were placed on the affected drawings; and (5) verified that appropriate safety evaluations were completed. The inspectors verified that licensee identified and implemented any needed corrective actions associated with temporary modifications.

- November 15, 16, and 17, 2005, TM 05-0021, Reactor coolant pump vibration circuit. The inspectors walked down portions of the TM in the control building and completed an in-office review.

- November 15, 16, and 17, 2005, TM ETP-SE-ST003, Reactivity computer for low power physics testing. The inspectors walked down portions of the TM located in the control building and completed an in-office review.

- November 15, 16, and 17, 2005, TM ET-SE-ST003, Nuclear instrument channel trip setpoints. The inspectors walked down portions of the TM located in the control building and completed an in-office review.

Documents reviewed by the inspectors included:

- Procedure ETP-SE-ST003, Precritical alignment/hookup of advanced digital reactivity computer, Revision 6

- Administrative Procedure APA-ZZ-00605, Temporary system modifications, Revision 18

The inspectors completed three samples.

b. Findings

No findings of significance were identified.

Cornerstone: Emergency Preparedness

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

a. Inspection Scope

The inspectors performed in-office reviews of Revision 27 to the Callaway Plant Radiological Emergency Response Plan, and Revision 33 to Procedure EIP-ZZ-00101, "Classification of Emergencies." These revisions were compared to their previous revisions, to the criteria of NUREG-0654, "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants," Revision 1; to NEI 99-01, "Methodology for Development of Emergency Action Levels," Revision 2; and to the requirements of 10 CFR 50.47(b)(4) and 50.54(q) to determine if the revisions decreased the effectiveness of the plan. These revisions:

- Made minor administrative updates and corrections and updated titles

- Clarified steam generator leakage terminology for a loss of containment in EAL 2, Indicator 3b

- Clarified verification of an earthquake in EAL 3H, Indicator 1c

- Clarified the timeliness of classification with regard to validation of alarms

- Clarified the definitions of steam generator leakage and faulted steam generator as applied to fission product barriers

- Revised the reactor coolant temperature threshold for a potential loss of containment in EAL 2, Indicator 7b, based on updated engineering calculations

- Revised the reactor vessel level threshold for potential loss of fuel cladding in EAL 2, Indicator 6b, based on revised emergency operating procedures

- Revised the description of telephone systems used in emergency response facilities based on replacement of some phones

- Added shelter as an option for recommendations of protective actions for the general public

- Added five special needs facilities in the emergency planning zone

- Added descriptions of a safety significance fire to EAL 3E, and defined the time a fire is out

The inspectors completed two samples during this inspection.

b. Findings

Introduction: A violation of 10 CFR 50.54(q) was identified for implementation of a decrease of effectiveness in the licensee's emergency plan. The licensee implemented a change to EAL 3E (Notification of Unusual Event) which defined a fire as having safety significance only when it was located within 50 feet of vital areas, unless the smoke or water stream from fighting the fire directly impacted listed safety-related equipment.

Description: The NRC identified that on June 8, 2005, the licensee implemented a change to its EAL bases, which was an apparent decrease in effectiveness of the licensee's emergency plan, because it restricted applicability of EAL 3E, "Fire within Protected Area Boundary NOT Extinguished with 15 minutes of Verification." Specifically, the revised bases clearly limited a plant fire adjacent to a vital area as one that is within 50 feet of a vital area, except in cases where smoke or water from fighting the fire directly affected safety-related equipment. The inspector determined that fires in some plant areas, such as areas of the turbine building, which were classifiable under EIP-ZZ-00101, Revision 32, may not have been classifiable using the revised EAL bases.

Analysis: Implementation of changes to emergency action levels which decreased the effectiveness of the emergency plan, was a performance deficiency. The finding had a credible impact on the emergency preparedness cornerstone objective because a licensee is less capable of implementing adequate measures to protect the health and safety of the public during a radiological emergency if initiating conditions are removed from licensee emergency action levels. This finding is more than minor because: (1) restricting or limiting a classifiable condition in the licensee EALs has the potential to impact safety; and (2) licensee implementation of a change to their emergency plan which decreases the effectiveness of the plan without prior NRC approval impacts the regulatory process. The finding also involves a violation of NRC requirements, subject to enforcement action under the terms of the NRC Enforcement Policy.

Enforcement: Licensee implementation, without prior NRC approval, of an EAL change which decreases the effectiveness of the emergency plan is a violation of 10 CFR 50.54(q), which states, in part, "A licensee authorized to possess and operate a nuclear power reactor shall follow and maintain in effect emergency plans that meet the standards in §50.47(b) and the requirements in Appendix E of this part . . . The nuclear power reactor licensee may make changes to these plans without Commission approval only if the changes do not decrease the effectiveness of the plans and the plans, as changed, continue to meet the standards of §50.47(b) and the requirements of Appendix E to this part."

In accordance with Manual Chapter 0609, Appendix B, §2.2(e) and §4.4, the inspector evaluated the significance of the finding using the "General Statement of Policy and Procedure for NRC Enforcement Actions" (Enforcement Policy), Section IV, "Significance of Violations." The finding was determined to be a Severity Level IV violation because: (1) a single EAL at the Notification of Unusual Event classification level was affected, and (2) the violation was determined not to be a licensee failure to meet or implement one emergency planning standard involving assessment or notification.

Because this performance deficiency is of very low safety significance and has been entered into the licensee's corrective action system (CAR 200510162), this violation is being treated as an NCV, consistent with Section VI.A of the NRC Enforcement Policy: NCV 05000483/2005005-10 (Change in Emergency Action Level 3E decreased the effectiveness of the Emergency Plan).

2. RADIATION SAFETY

Cornerstone: Occupational Radiation Safety

2OS2 ALARA Planning and Controls (71121.02)

a. Inspection Scope

The inspector assessed licensee performance with respect to maintaining individual and collective radiation exposures as low as is reasonably achievable (ALARA). The inspector used the requirements in 10 CFR Part 20 and the licensee's procedures required by Technical Specifications as criteria for determining compliance. The inspector interviewed licensee personnel and reviewed:

- Current 3-year rolling average collective exposure

- Eight outage work activities scheduled during the inspection period and associated work activity exposure estimates which were likely to result in the highest personnel collective exposures.

- Site-specific trends in collective exposures, plant historical data, and source-term measurements

- Site-specific ALARA procedures
- Eight work activities of highest exposure significance completed during the last outage.
- ALARA work activity evaluations, exposure estimates, and exposure mitigation requirements
- Intended versus actual work activity doses and the reasons for any inconsistencies
- Interfaces between operations, radiation protection, maintenance, maintenance planning, scheduling, and engineering groups
- Person-hour estimates provided by maintenance planning and other groups to the radiation protection group with the actual work activity time requirements
- Dose rate reduction activities in work planning
- Assumptions and basis for the current annual collective exposure estimate, the methodology for estimating work activity exposures, the intended dose outcome, and the accuracy of dose rate and man-hour estimates
- Method for adjusting exposure estimates, or replanning work, when unexpected changes in scope or emergent work were encountered
- Use of engineering controls to achieve dose reductions and dose reduction benefits afforded by shielding
- Exposures of individuals from selected work groups
- Source-term control strategy
- Declared pregnant workers during the current assessment period, monitoring controls, and the exposure results
- Self-assessments, audits, and special reports related to the ALARA program since the last inspection
- Corrective action documents related to the ALARA program and follow-up activities such as initial problem identification, characterization, and tracking
- Effectiveness of self-assessment activities with respect to identifying and addressing repetitive deficiencies or significant individual deficiencies

Either because the conditions did not exist or an event had not occurred, no opportunities were available to review the following items:

- Records detailing the historical trends and current status of tracked plant source terms and contingency plans for expected changes in the source term due to changes in plant fuel performance issues or changes in plant primary chemistry

- Radiation worker and radiation protection technician performance during work activities in radiation areas, airborne radioactivity areas, or high radiation areas

The inspector completed 15 of the required 15 samples and 6 of the optional samples.

b. Findings

No findings of significance were identified.

4OA2 Identification and Resolution of Problems (71152)

.1 Routine Review of Identification and Resolution of Problems

The inspectors performed a daily screening of items entered into the licensee's corrective action program. This assessment was accomplished by reviewing the daily CAR Screening Report, Control Room Logs, and attending selected Corrective Action Review Board and work control meetings. The inspectors: (1) verified that equipment, human performance, and program issues were being identified by the licensee at an appropriate threshold and that the issues were entered into the corrective action program; (2) verified that corrective actions were commensurate with the significance of the issue; and (3) identified conditions that might warrant additional follow-up through other baseline inspection procedures.

.2 Selected Issue Follow-up Inspection

In addition to the routine review, the inspectors selected the two below listed issues for a more in-depth review. The inspectors considered the following during the review of the licensee's actions: (1) complete and accurate identification of the problem in a timely manner; (2) evaluation and disposition of operability/reportability issues; (3) consideration of extent of condition, generic implications, common cause, and previous occurrences; (4) classification and prioritization of the resolution of the problem; (5) identification of root and contributing causes of the problem; (6) identification of corrective actions; and (7) completion of corrective actions in a timely manner.

- October 19, 2005, CAR 200508393, Tin whiskers: untimely corrective actions for 10 CFR Part 21

- November 11, 2005, CAR 200509277, Unplanned pressurization and failure of the TDAFP lube oil cooler

The inspectors completed two samples.

.3 Exposure Tracking, Higher than Planned Exposure Levels, and Radiation Worker Practices

Section 2OS2 evaluated the effectiveness of the licensee's problem identification and resolution processes regarding exposure tracking, higher than planned exposure levels, and radiation worker practices. The inspectors reviewed the corrective action documents listed in the attachment against the licensee's problem identification and resolution program requirements.

.4 Semiannual Trend Review

a. Inspection Scope

The inspectors completed a semiannual trend review of repetitive or closely related issues that were documented in plant trend reports, problem lists, performance indicators, system health reports, QA audit reports, corrective action documents, and corrective maintenance documents to identify trends that might indicate the existence of more safety significant issues. The inspectors' review consisted of the 6-month period of July through December 2005. When warranted, some of the samples expanded beyond those dates to fully assess the issue. The inspectors also reviewed corrective action program items listed in the attachment. The inspectors compared and contrasted their results with the results contained in the licensee's quarterly trend reports. Corrective actions associated with a sample of the issues identified in the licensee's trend report were reviewed for adequacy.

Documents reviewed by the inspectors are listed in the attachment.

b. Findings and Observations

1. Adverse Trend in Human Performance

The NRC identified an adverse human performance trend in December 2004 (Inspection Report 05000483/2004005). The NRC subsequently identified a substantive crosscutting issue in the area of human performance during the 2004 end-of-cycle assessment. The substantive crosscutting issue was based on seven NRC findings specifically related to personnel errors that occurred during 2004 and affected the initiating events, mitigating systems, and barrier integrity cornerstones. AmerenUE completed a stream analysis of the human performance events to identify commonality and root causes in April 2005. In June 2005, the NRC and AmerenUE concluded that the adverse trend continued during the first two calendar quarters in 2005 (NRC Inspection Report 05000483/2005003). AmerenUE implemented the following corrective actions in August 2005 to address the root causes of poor human performance (CAR 200501425):

- Established the Event Prevention Steering Committee
- Enhanced the plant observation process by establishing metrics and accountability

- Identified and addressed deficiencies in the station root cause analysis processes

- Implemented a station focus of defense-in-depth error prevention tool/activities

The inspectors concluded that the adverse human performance trend continued during the third and fourth quarters 2005. On November 28, 2005, the Callaway Event Prevention Steering Committee also identified an adverse trend associated with station noncompliance with written instructions (CAR 200509697). Examples used by the licensee to identify the trend included:

- CAR 200507092, September 20, 2005, Valve repositioned without the appropriate procedure

- CAR 200507699, October 2, 2005, 28 wire strand jack cables dropped from the containment polar crane to the cavity deck

- CAR 200508510, October 22, 2005, Failure to re-terminate 480 volt energized leads

- CAR 200508753, October 28, 2005, Adverse trend of falling objects

- CAR 200509404, November 15, 2005, Partial reactor trip due to failure to follow lock and tag notes

2. Adverse Trend in Corrective Action

The inspectors identified an adverse trend associated with ineffective corrective actions. The inspectors considered the following examples of corrective actions that failed to prevent recurrence of previously identified problems. The inspectors screened the examples using Manual Chapter 0612, Appendix B, "Issue Screening," and concluded each example had only minor safety significance:

- CAR 200509345, Unplanned main steam dump closure during reactor trip breaker testing

- CAR 200509474, Removal of the reactivity computer test leads out of sequence caused a false pressurizer low level signal and charging system flow reduction

- CAR 2005007860, Condensate storage tank wiper seal repeat cracking

- CAR 200207808, Inadequate procedure resulted in the overpressurization of the TDAFP suction piping and lube oil cooler

On December 1, 2005, AmerenUE also identified an adverse trend in corrective actions resulting in a "Red" corrective action program system health indicator.

4OA5 Other Activities

.1 Temporary Instruction 2515/160, "Pressurizer Penetration Nozzles and Steam Space Piping Connections in U.S. Pressurized Water Reactors (NRC Bulletin 2004-01)

a. Inspection Scope

Industry OE has demonstrated that Alloy 82/182/600 materials exposed to primary coolant water (or steam) at the normal operating conditions of pressurized water reactor plants have cracked due to primary water stress corrosion cracking. The NRC issued Bulletin 2004-01, "Inspection of Alloy 82/182/600 Materials Used in the Fabrication of Pressurizer Penetrations and Steam Space Piping Connections at Pressurized-water Reactors," was issued to alert licensee's to the susceptibility of Alloy 82/182/600 materials to cracking. The Callaway RCS has five pressurizer connections that were applicable to the vulnerabilities described in NRC Bulletin 2004-01. The inspectors compared the AmerenUE examinations of these five Alloy 82/182/600 pressurizer piping connections with the licensee's commitments documented in ULNRC-05031, "Response to NRC Bulletin 2004-01, Inspection of Alloy 82/182/600 Materials used in the Fabrication of Pressurizer Penetrations and Steam Space Piping Connections at Pressurized-Water Reactors," July 27, 2004. The inspectors performed this comparison to verify that the examinations were consistent with the AmerenUE response to the bulletin.

The inspectors reviewed records and examination procedures for visual examinations (listed in the attachment) conducted during Refueling Outage 13 (Spring 2004) and Refueling Outage 14 (Fall 2005). The inspectors performed this review to verify that the bare metal examinations were adequate to detect the presence of boric acid crystals. The inspectors used the guidance in Inspection Procedure 57050, "Visual Testing Examination," as acceptance criteria for this review. The inspectors reviewed volumetric examinations conducted during 1992 and 1996. The inspectors used the guidance in Inspection Procedure 57080, "Ultrasonic Testing Examination," as acceptance criteria for this review. The inspectors also reviewed the qualifications and certifications of the personnel performing the examination and assessed the techniques used to detect small boric acid deposits on the subject locations.

b. Findings

No findings of significance were identified. The inspectors concluded that the inspections conducted by AmerenUE were consistent with the licensee's response to NRC Bulletin 2004-01. The inspectors concluded:

- Personnel performing the examination were qualified, knowledgeable, and certified as visual examination Level 2 inspectors. Each inspector also received additional training for identification of boric acid deposits.

- The examinations were performed in accordance with station procedures and were capable of identifying leakage in pressurizer penetration nozzle or steam space piping components, as discussed in NRC Bulletin 2004-01.

- The inspectors reviewed the photographic record of the examination and verified that the physical condition of the penetration nozzles and steam space piping components were good, without debris, insulation, dirt, or boron from other sources during the visual examination.

- The visual examination covered a 360° circumference of all the affected nozzles.

- The examination was sufficient to identify and characterize small boron deposits, as described in NRC Bulletin 2004-01.

AmerenUE did not identify any material deficiencies, cracks, or corrosion. No indications of boric acid leaks from pressure-retaining components were identified during the examinations and volumetric or surface examination techniques were not used to augment the inspections.

.2 (Closed Unresolved Item 05000483/2005004-03) Potential Failure of the RHR Containment Recirculation Sump Valves During Certain Design Bases Events

AmerenUE evaluated the containment recirculation sump valve operator torque needed to open against the maximum calculated differential pressure that could be experienced across the valve. AmerenUE determined that the valves were required to operate against a 53 psid (Calculation RFR 05353, Revision F, October 31, 1989). AmerenUE evaluated OE from the Catawba and McGuire plants (CAR 200504370) during June 2005. This OE alerted the industry to the potential of higher than previously considered differential pressure across the RHR sump valves. In response to the OE, AmerenUE operated the RHR pumps for 30 minutes in the minimum flow configuration and observed 189 psid across the sump valve. AmerenUE concluded that the maximum differential pressure the valve actuator would be required to open against was 189 psid. Engineering personnel concluded valve operability based on a linear extrapolation of the actuator torque to the new conditions.

Subsequently, AmerenUE evaluated additional OE from the Wolf Creek plant on September 21, 2005 (CAR 200507150). This OE alerted the industry to the potential of additional differential pressure that could develop across the RHR sump valves while in the minimum flow mode. AmerenUE reevaluated the RHR valves and determined that the maximum differential pressure the valves had to open against could be 468 psid. AmerenUE verified past sump valve operability using actual valve factors and a realistic lock-rotor valve operator torque. AmerenUE modified the operators to provide higher opening torque to ensure future RHR valve operability. The failure of Ameren to ensure suitability of the RHR containment suction valves' function to open under all safety-related design bases conditions was a licensee-identified violation of 10 CFR Part 50, Appendix B, Criteria III, "Design Control." The enforcement aspects of this violation are discussed in Section 4OA7 of this report.

.3 (Closed Apparent Violation 05000483/2005004-01) Failure to Maintain Cold Overpressure Mitigation Measures as Required by TSs

a. Inspection Scope

A senior reactor analyst performed a Phase 3 significance determination of apparent violation 05000483/2005004-01. The inspectors evaluated this finding using the guidance in Manual Chapter 0612, Power Reactor Inspection Reports, dated September 30, 2005, for determining whether a violation is licensee-identified because this finding had not been closed prior to the revised guidance being issued. This apparent violation is closed as a licensee-identified violation of very low safety significance. The violation is documented in Section 4OA7 of this report.

b. Findings

Introduction: The senior reactor analysts completed the significance determination of the apparent violation documented in NRC Inspection Report 05000483/2005004. The apparent violation involved the failure of AmerenUE operations personnel to ensure no more than one centrifugal charging pump was capable of injecting into the reactor vessel while in Mode 5, as required by TS 3.4.12.

Analysis: The performance deficiency associated with this finding involved the licensee's failure to establish and follow adequate procedures. This finding is greater than minor because it would have become more significant, if left uncorrected, in that inadvertent starting of the charging pump could have challenged the piping integrity of the RCS system. The inspectors used Appendix G, "Shutdown Operations Significance Determination Process," of Manual Chapter 0609, "Significance Determination Process," to determine the significance of this finding. Unplanned entry into cold overpressurization conditions represented additional risk incurred above the planned outage risk. The additional risk associated with the ability of the centrifugal charging pump to inject into the RCS constituted additional risk above the planned outage risk. Phase 1 screening of this finding was performed using Appendix G and the Attachment 1 checklists. Management review determined that significance determination process Phase 3 analysis was needed for this finding.

The senior reactor analysts' review of the Callaway cold overpressure mitigation (COMS) precursor involved having both centrifugal charging pumps capable of RCS injection. This condition lasted approximately 20 minutes.

The following conditions existed at the time of the event:

- Pressurizer level was at 5 percent
- There was a high pressurizer level alarm at 90 percent
- There was an alarm at 5 percent level above program increase
- RCS level was not being changed at the time of the event

- No testing was being performed on or in systems connected to the RCS that could perturb RCS level

- No work was being performed on RCS level indication other than adding an additional, alternate reactor vessel level indication with separate tap locations

- The safety injection pumps were in pull-to-lock

- The accumulators were isolated and vented

- Each RHR train had a suction relief valve with a lift setpoint of 450 pounds per square inch gauge (psig) (986 gpm discharge capacity)

- Both trains of RHR were aligned to the RCS with one train providing decay heat removal. Therefore, both RHR suction relief valves were available to relieve a postulated cold overpressure challenge

- Two power-operated relief valves were available for COMS, the low power-operated relief valve setpoint was at 500 psig, the high power-operated relief valve setpoint was at 525 psig

- The pressurizer was vented to atmosphere (via a ¾-inch manual vent valve)

To assess the risk of the event required an estimate of the likelihood that the operators would have initiated RCS injection, resulting in a solid RCS. Based on the above information, it appears that no plant operations were being performed at the time that had the potential to trigger the operators to initiate RCS injection. Additionally, the alarm indicating 5 percent above program level provides additional assurance that the likelihood of overfilling the RCS during the 20-minute time period was small.

If a postulated RCS pressure challenge were to occur, the dominant core damage scenario involves both RHR suction relief valves failing to reseat after an RCS pressure challenge. In this design, the RHR suction relief valves have a lower relief setpoint than the pressure-operated relief valves. Should one RHR relief valve fail to reseat, the operators could isolate the valve and use the alternate train of RHR for decay heat removal. If both relief valves were to fail to reseat, the operators would be directed to increase charging and isolate the leak. In this plant condition, steam generator cooling is not anticipated to match decay heat; therefore, the RCS may re-pressurize until steam generator cooling can remove decay heat. For this situation, both pressure-operated relief valves would be available should RCS pressure increase to the COMS setpoint.

In summary, combining the small likelihood of having an RCS pressure challenge during the 20-minute period, the likelihood of having both RHR relief valves stick open after a challenge, and the failure of both pressure-operated relief valves to relieve pressure, the core damage frequency delta for this finding is estimated to be less than 1E-6. Therefore, this finding can be characterized in the significance determination process as

Green. It is important to note that the licensee's robust COMS mitigation capability (the availability of both RHR suction relief valves and the pressure-operated relief valves) was significant in reducing the risk of this finding.

The review of the licensee's analysis only considered the likelihood of the COMS system failing to provide RCS pressure relief following a demand. The licensee did not consider that an RCS pressure demand may result in the RHR suction relief valve lifting and not reseating. This scenario results in a loss of coolant accident in the RHR system as described above.

This finding affected the barrier integrity cornerstone and the configuration control, procedure quality, and human performance attributes of maintaining functionality of the RCS. The senior reactor analyst determined that this finding is only of very low significance.

Enforcement: The enforcement aspects of this finding are discussed in Section 4OA7 of this report.

4OA6 Management Meetings

Exit Meeting Summary

On December 14, 2005, the health physics inspector presented the ALARA inspection results to Mr. A. Heflin, Vice President, and other members of his staff who acknowledged the findings.

On January 6, 2006, 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 emergency preparedness inspector conducted a telephonic exit interview on January 12, 2006, to present the inspection results to Mr. M. Reidmeyer, Supervisor, Regional Regulatory Affairs, and other members of his staff who acknowledged the findings.

The operations branch inspectors conducted an exit meeting on June 9, 2005, regarding the on-site portion of the inspection with Mr. R. Roselius and other members of the licensee's staff. On December 15, 2005, the inspectors discussed biennial written requalification examination issues with the licensee. After NRC management review of the biennial written requalification examination observations, the inspectors again discussed the unresolved item identified during the review of the written biennial requalification exams with the licensee during a teleconference on January 23, 2006.

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

4OA7 Licensee-Identified Violations

The following violations 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.

- Title 10 CFR Part 50, Appendix B, Criteria III, “Design Control,” required that measures be established for the selection and suitability of application of equipment essential to the safety-related functions of the SSCs. Contrary to this, on October 31, 1989, and October 5, 2005, the selection and suitability of application for the RHR containment sump valve operators was inadequate to ensure all safety-related functions. AmerenUE had established an insufficient maximum differential pressure design that the sump valves would have to open against during certain design bases events. This was identified in the licensee’s corrective action program as CAR 200504370. This finding is of very low safety significance because it does not represent a design or qualification deficiency confirmed not to result in loss of operability per Part 9900, Technical Guidance, “Operability Determination Process for Operability and Functional Assessment.”

- Title 10 CFR 55.49 requires examination integrity to be maintained. The regulation further defines an examination compromise as any activity, regardless of intent, that affected or could have affected the equitable and consistent administration of an examination.

  During a review of CARs, the inspectors noted that two events occurred that had the potential to effect the integrity of the requalification examinations. The first event occurred on May 26, 2005, and involved leaving data on the simulator's "white board" from the previous scenario training crew. The data displayed provided information that could be used by the oncoming training crew to assist them with the scenario (since the same scenario was to be run). This compromise was identified by the licensee's oncoming training crew. As a result, the oncoming crew was given a different scenario.

  The second event occurred on June 8, 2005, and involved the accidental observation of some pages out of a written examination by a candidate assigned to the training staff. This candidate was scheduled to take the same specific examination. When the licensee identified this compromise, the candidate was rescheduled to take a different written examination.

  These findings are greater than minor because a compromise of the integrity of the annual requalification examinations could lead to operators (who would normally have failed the examination) with deficient knowledge and skills to remain on shift. Allowing operators with deficient knowledge and skills to remain on shift increases the likelihood that a human performance error could initiate a reactor safety event or inhibit the appropriate mitigating response to such an event. Contrary to the above, the licensee failed to adequately assure that examination security was maintained during the administration of examinations. The finding is of very low safety significance because the potential for examination compromise was extremely low. These findings have been entered into the corrective action program as CARs 200503988 and 200503985, respectively.

- TS 5.4.1.a, “Procedures,” and Regulatory Guide 1.33, Appendix A, required procedures for shutdown to be implemented. Procedure OSP-BG-00002, “Verify One Centrifugal Charging Pump Incapable of Injection into RCS,” required the licensee to ensure only one centrifugal charging pump was capable of injecting to the RCS during Mode 5 operations with limited RCS vent path. Contrary to the above, on September 20, 2005, the licensee failed to ensure only one centrifugal charging pump was capable of injecting to the RCS. This finding is greater than minor because it would have become more significant, if left uncorrected, in that inadvertent starting of the charging pump could have challenged the piping integrity of the RCS system. This finding was determined to be very low significance after completion of a Phase 3 SDP by the senior reactor analyst as documented in Section 40A5 of this report. This finding was identified in the licensee’s corrective action program as CAR 200507092.

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee

W. Arbour, Senior Operations Training Supervisor
S. Aufdemberge, Operating Supervisor
K. Bruckerhoff, Supervisor, Emergency Preparedness
F. Diya, Manager, Engineering Services
R. Farnam, General Supervisor, Radiation Protection
S. Ganz, Operating Supervisor
J. Geyer, Health Physicist, Radiation Protection
K. Gilliam, ALARA Coordinator, Radiation Protection
S. Halverson, General Supervisor, Simulator
A. Heflin, Site Vice President
T. Herrmann, Vice President, Engineering
J. Hiller, Regulatory Affairs, Engineer
G. Hurla, Supervisor, Radiation Protection
M. Jennings, Operating Supervisor
L. Kanuckel, Manager, Quality Assurance
S. Kochert, Operating Supervisor
V. Miller, ALARA Specialist, Radiation Protection
R. Moody, Operating Supervisor
T. Moser, Manager, Plant Engineering
C. Naslund, Senior Vice President and Chief Nuclear Officer
R. Nelson, Shift Supervisor
D. Neterer, Manager, Operations
M. Reidmeyer, Supervisor, Regional Regulatory Affairs
R. Roselius, Superintendent, Training
K. Young, Manager, Regulatory Affairs

LIST OF ITEMS OPENED AND CLOSED

Opened

05000483/2005005-03	URI	Indeterminate Containment Cooler Operability and Heat Removal Capability (Section 1R07)
05000483/2005005-04	URI	Adequacy of the Biennial Requalification Written Examination (Section 1R11)
05000483/2005005-06	URI	Adequacy of Plant-Referenced Simulator to Conform with Simulator Requirements for Reactivity and Control Manipulation Credits (Section 1R11)

Opened and Closed

05000483/2005005-01	NCV	Minimum gap size exceeded for containment recirculation sump (Section 1R04)
05000483/2005005-02	NCV	Seven examples of inadequately performed continuous fire watches (Section 1R05)
05000483/2005005-05	FIN	Failure to Conduct Simulator Testing in Accordance with ANSI/ANS 3.5-1998 (Section 1R11)
05000483/2005005-07	NCV	Failure to Follow Procedures Resulted in Violation of RCS Cooldown and Heatup Rate Limits (Section 1R14)
05000483/2005005-08	NCV	Use of a Nonqualified Calculation in a Safety-Related Modification (Section 1R17)
05000483/2005005-09	FIN	Less Than Adequate Spent Fuel Pool Water Inventory Risk Controls (Section 1R20)
05000483/2005005-10	NCV	Change in EAL 3E decreased the effectiveness of the Emergency Plan (Section 1EP4)

Closed

05000483/2005004-03	URI	Potential Failure of the RHR Containment Suction Valves During Certain Design Bases Events (Section 4OA5)
05000483/2005004-01	AV	Failure to Maintain Cold Overpressure Mitigation Measures as Required by TSs (Section 4OA5)

DOCUMENTS REVIEWED

Section 1R04: Equipment Alignment

Drawings

E-23KJ01A, Revision 14, Diesel General KKJ01 Engine Control (Start / Stop) Circuit
M22-BG03, Chemical and Volume Control System
M22-BG05, Chemical and Volume Control System
M22-EJ01, Residual Heat Removal System

Miscellaneous

Callaway Action Request 200509189

Procedure OSP-EJ-00003, Containment Recirculation Sump Inspection, Revision 5

FSAR Table 6.2.2-1, Comparison of the Recirculation Sump Design with each of the Positions of Regulatory 1.82

NRC Bulletin 2003-01, Potential Impact of Debris Blockage on Emergency Sump Recirculation at Pressurized-Water Reactors

Work package W229952, Recirculation Sump Inspection

ULNRC-04966, Callaway Plant, Union Electric Co. Supplement to Response to NRC Bulletin 2003-01, Potential Impact of Debris Blockage on Emergency Sump Recirculation at Pressurized Water Reactors

Section 1R05: Fire Protection

Procedures

APA-ZZ-00743, Fire Team Organization and Duties, Revision 18
EIP-ZZ-00226, Fire Response Procedure for Callaway Plant, Revision 11
SDP-KC-00001, Requirements for and Duties of Compensatory Fire Watches, Revision 5

Requests for Resolution

RFR 15704, Electrical Safety Equipment Lockers, Revisions A and B
RFR 18572, Allowed Storage of D/G Tool Boxes and Barring Device, Revision A
RFR 3487, Breaker Test Area in NB01 Switchgear Room 3301, Revision B

Miscellaneous

Information Notice 97-48, Inadequate or Inappropriate Interim Fire Protection Compensatory Measures

Section 1R07: Heat Sink Performance

Callaway Action Requests

CAR 200503773, Containment cooler heat removal surveillance requirements
CAR 200502534, Incorrect component cooling water heat exchanger indication

Drawings

M-22EF02, Essential Service Water System
M-22EF08, Essential Service Water Containment Air Coolers

Procedures

ESP-EF-002A, Essential Service Water Train A Flow Verification, Revision 0
OSP-EF-P001A, ESW Train A Inservice Test, Revision 43

Miscellaneous

SGN01A ETP-ZZ-03001, Heat Exchange Inspection Report, Revision 5, completed on September 23, 2005

Surveillance 05515092, Essential Service Water, performed on October 12, 2005

Work Package W 236012/920, Containment Cooler Unit A PMT

Section 1R11: Licensed Operator Requalification

Procedures

TDP-IS-00002, Simulator Configuration Management, Revision 4
TDP-IS-00001, Simulator Operation and Maintenance, Revision 3

JPMs

SRO-RER02C113J(TC), Emergency Event Classification, Revision 20040710

URO-SEG02C21J, Shift Non-essential CCW Supply Loops, Revision 20050604

URO-AEO05045J, Locally Close Valves for a CIS [Containment Isolation Signal]-B, Revision 20050508

URO-SGN02C27J, Secure D Containment Cooler Fan, Revision 20050421

URO-AEO15016J, Local Manual Start of NE02, Revision 20050422

URO-SBB04C67J(A), Pressurizer Level Channel Failure, Revision 20050421

SRO-RER02C143J(TC), Emergency Event Classification, Revision 20050413

URO-SAB04C61J(A), Place Steam Dumps in Steam Pressure Mode, Revision 20050413

URO-SEF02C03J, Manually Operate an ESW Train, Revision 20050413

EOP-SBG06014J, Shift and Vent CVCS Seal Water Injection Filters, Revision 20050314

URO-AEO05001j(A), Locally Start (NE01) Emergency Diesel, Revision 20050413

URO-SBG02C04J, Swap From B CCP to NCP, Revision 20050413

SRO-RER02C118(TC), Emergency Event Classification, Revision 20050323

URO-SGN04C71J(A), Start A Containment Cooler Fan, Revision 20050820

EOS-SNN03011J, Shift an Instrument Bus to Backup Power Supply, Revision 20050323

URO-Paralleling Diesel Generator A to XNB01, Revision 20050625

URO-SSF01C05J, Perform Control Rod Partial Movement Test, Revision 20050215

URO-AEO05PA023J, Locally Close Valves for a CIS-A, Revision 20050323

URO-SSP03C15J, Radiation Monitors Source Check, Revision 20050328

URO-AEO15029J, Locally Isolate a MSIV, Revision 20050314

URO-SBG02C01J, Placing Excess Letdown in Service, Revision 20050314

SRO-RER02C45J, Emergency Event Classification, Revision 20050414

URO-AEO01C151J(A), Emergency Boration Per /ES-0.1/Addendum 4, Revision 20050502

EOS-SNK01051J, Place NK22 in Service to Bus NK02, Revision 20050314

Scenarios

DS-07, Small Break LOCA With Failure of CPIS [Containment Purge Isolation System] and CCP [Component Cooling Pump]/Loss of NB01, Revision 20050520

DS-32, Faulted-Ruptured S/G, Revision 20050520

DS-14, Separate Faulted and Ruptured S/Gs, Revision 20050310

DS-24, Loss of Letdown, ATWS with Stuck Open Pressurizer Safety Valve, Revision 20050311

DS-15, Load Increase with Multiple Rod Drop/Pressurizer Steam Space Leak, Revision 20050507

DS-40, Faulted/Ruptured S/G, Revision 20050507

DS-04, Loss of Heat Sink without Bleed and Feed Required, Revision 20050514

DS-05, Faulted/Ruptured S/G, Revision 20050514

DS-01, ATWS, Revision 20050308

DS-26, Large LOCA and Transfer to Cold Leg Recirculation, Revision 20050310

DS-08, Feedline Break Inside Containment with CCP and SLIS Failures, Revision 20050414

DS-37, Station Blackout due to Seismic Conditions, Revision 20050329

DS-18, SGTR Without Pressurizer Pressure Control, Revision 20050422

DS-19, Turbine Trip Failure with Loss of Heat Sink, Revision 20050422

Written Examinations

T61.0810 8, LOCT Cycle 05-4 Biennial Exam, SRO Week 1

T61.0810 8, LOCT Cycle 05-4 Biennial Exam, URO Week 2

Miscellaneous

2003-2005 Continuing Sample Plan

Job-Duty-Task by Job for URO [Unit Reactor Operator] dated 3/17/05

Job-Duty-Task by Job for SRO dated 4/14/05

Written Summary of Simulator Testing Topic Public Meeting with Industry Focus Group (FG) on Operator Licensing Issues (DRAFT)

Response to April 7, 2004 Public Meeting Minutes Attachment 6

Callaway Plant Simulator White Paper showing how all parameters are demonstrated, June 8, 2005

Simulator Annual Performance Test Book

Simulator "Differences" List, May 16, 2005

Section 1R17: Permanent Plant Modifications

Calculations

330-001-DC1, Motor terminal voltage and nominal torque output, Revision 0

EJ-42, MOV sizing for EJHV8811A and EJHV8811B, Revision 0

Westinghouse Calculation SCP-05-69, Valve Factors for Valve Location 8811A and 8811B, October 28, 2005

Callaway Action Requests

200507150
200509849
200505194

Miscellaneous

Predictive Performance Report, E170.0197, CA 1527, May 10, 1990

Modification MP 05-3051

Section 1R19: Postmaintenance Testing

Procedures

OSP-SF-00005, Estimated Critical Rod Position Calculation ST-13002, Revision 16

ETP-ZZ-ST010, Low Power Physics Test Program with Dynamic Rod Worth Measurement, Revision 8

OSP-BG-0001A, Boron Injection Flowpaths, Revision 14

APA-ZZ-00500, Corrective Action Program, Revision 38

OSP-AL-P0002, Turbine-Driven Auxiliary Feedwater Pump Inservice Test, Revision 49

Miscellaneous

PM0826213, Overhaul of NN Inverter, PMB Charging-1-5.2-4, Revision 0

Section 1R20: Refueling and Outage Activities

Procedures

APA-ZZ-00150, Outage Preparation and Execution, Revision 12
EDP-ZZ-1129, Callaway Plant Risk Assessment, Revision 8
OSP-SF-00003, Pre-Core Alteration Verifications, Revision 12
OSP-SF-00003, Pre-Core Alterations Verifications, Revision 15
OSP-ZZ-00001, Control Room Shift and Daily Log Readings and Channel Checks, Revision 39
OTG-ZZ-00001, Plant Heatup Cold Shutdown to Hot Standby, Revision 45
OTG-ZZ-00006, Plant Cooldown Hot Standby to Cold Shutdown, Revision 6
OTO-KE-00001, Fuel Handling Accident, Revision 7
QCP-ZZ-05048, Boric Acid Walkdown for RCS Pressure Boundary, Revision 2

Miscellaneous

Curve Book, Figure 8-6, RCS Pressure-Temperature Limitations

Nuclear Utility Management and Resource Council 91-06, Guidelines for Industry Actions to Assess Shutdown Management

Quality Assurance Surveillance Reports

SP05-028, December 12, 2005, Assess lifting, removal and placement of the reactor vessel head and upper internals

SP05-047, December 9, 2005, Reduced inventory control, risk assessment, outage technical specifications

Callaway Action Requests

200002070	200501407	200504950
200202540	200501837	200505062
200302806	200501990	200505368
200307232	200502420	200505716
200307247	200502438	200506244
200307844	200502548	200507150
200402256	200503439	200507278
200500720	200503622	200508169
200500756	200503773	200507593
200501092	200504591	200507693

Section 1R22: Surveillance Testing

Procedures

OSP-EM-V0003, ECCS Check Valve Inservice Test IPTE, Revision 21
OSP-BB-00006, Reactor Coolant Circulation, Revision 7
OSP-BG-0001A “Boron Injection Flowpaths modes 4 through 8

Audits and Self-Assessments

Quality Assurance Surveillance Report SP05-027, November 6, 2005, Assess effectiveness of fuel movement, compliance to TSs and procedures applicable to fuel movement

Quality Assurance Surveillance Report SP05-037, November 18, 2005, Assess implementation of the steam generator replacement project

Section 71152: Identification and Resolution of Problems

Procedures

APA-ZZ-00500, Corrective Action Program, Revision 38

Callaway Action Requests

200507092
200507699
200508510
200508753
200509404

Miscellaneous

Callaway Plant Quarterly Performance Analysis Report Third Quarter

Event Review Team Meeting Summaries

AUCA 05-040, October 2, 2005, Strand wires were dropped from the TLD on the polar crane to the cavity deck

AUCA 05-047, October 15, 2005, Corrosion discovered on the new B low pressure turbine rotor

AUCA 05-049, October 17, 2005, Employee falls in containment while wearing fall protection

AUCA 05-050, October 19, 2005, Pit at VBS checkpoint lowered prematurely

AUCA 05-057, October 29, 2005, Leaking head gaskets during KKJ01B maintenance run

Surveillance Reports

SP05-034, September 24, 2005, Postmodification test planning for CMP 03-1014 - EP8818A-D valve replacements

SP05-045, September 29, 2005, Bottom mounted instrumentation inspection and cleaning

SP05-026, September 30, 2005, Assess various areas during plant shutdown

SP05-061, October 28, 2005, Refuel 14 worker practices

SP05-070, November 5, 2005, QA walkdowns to assure appropriate combustible loadings and housekeeping, and operable fire doors and halon systems

SP05-074, November 15, 2005, Assess interim compensatory actions in response to NRC Bulletin 20003-1 and Generic Letter 2004-2

SP05-044, November 23, 2005, Refuel 14 work activities on the TDAFP

SP05-056, November 29, 2005, Review of the tin whisker inspections

SP05-068, November 30, 2005, Assessment of Operating License Amendment 1248`

SP05-029, December 6, 2005, Assess effectiveness of control room personnel from Mode 3 ascending to Mode 1

SP05-058, December 14, 2005, QA assessment of Refueling Outage 14 mode change restraints

SP05-063, December 15, 2005, ESW strainer replacement activities

SP05-071, December 8, 2005, Review control logs and verify CARs were written when appropriate

SP05-078, November 30, 2005, Main feedwater regulation valve and bypass regulation valve testing in Refuel 14

Callaway Plant Quarterly Performance Analysis Report First Quarter

Callaway Plant Quarterly Performance Analysis Report Second Quarter

Quality Assurance Audits

AP05-010, October 5, 2005, Problem resolution, adverse trends, OQAM audit requirements/other commitments, review of self-assessments, organization, special nuclear material program, special nuclear material inventory, source control, and software management

Section 4OA5: Other Activities

Surveillances

S724682, Task 150, Inspection of pressurizer surge nozzle welds for boron
S724682, Leakage examination of the RCS, September 25, 2005
S714761, Leakage examination of the RCS, April 29, 2004

Procedures

QCP-ZZ-05048, Boric Acid Walkdown for RCS Pressure Boundary, Revision 2
QCP-ZZ-05048, Boric Acid Walkdown for RCS Pressure Boundary, Revision 1

Miscellaneous

Letter to the NRC from AmerenUE, ULNRC-05031, July 27, 2004, "Response to NRC Bulletin 2004-01, "Inspection of Alloy 82/182/600 Materials Used in the Fabrication of Pressurizer Penetrations and Steam Space Piping Connections at Pressurized Water Reactors"

UT Data Sheet 1021-02, Examination BB 2TBB03-1-w, April 18, 1992

UT Data Sheet 1021-01, Examination BB 2TBB03-2-w, April 17, 1992

UT Data Sheet 6276-95-01, Examination BB 2TBB03-3-A-w, October 30, 1996

UT Data Sheet 6276-95-02, Examination BB 2TBB03-3-B-w, October 30, 1996

UT Data Sheet 6276-001, Examination BB 2TBB03-3-C-w, October 27, 1996

UT Data Sheet 6276-002, Examination BB 2TBB03-4-w, October 27, 1996

CAR 200507515, Boric acid walkdown for Refuel 14

LIST OF ACRONYMS

ALARA	as low as is reasonably achievable
CAR	Callaway Action Request
COMS	cold overpressure mitigation
EAL	emergency action level
EDG	emergency diesel generator
ESW	essential service water
FIN	finding
FSAR	Final Safety Analysis Report
NCV	noncited violation
OE	operational experience
psid	pounds per square inch differential
psig	pounds per square inch gauge
PMT	postmaintenance test
RCS	reactor coolant system
RHR	residual heat removal
SSC	structures, systems, and components
TDAFP	turbine-driven auxiliary feedwater pump
TMs	temporary modifications
TSs	Technical Specifications
URI	unresolved item