UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION IV
612 EAST LAMAR BLVD, SUITE 400
ARLINGTON, TEXAS 76011-4125

February 4, 2009

Mr. Adam C. Heflin, Senior Vice
President and Chief Nuclear Officer
Union Electric Company
P.O. Box 620
Fulton, MO 65251

Dear Mr. Heflin:

Subject: CALLAWAY PLANT - NRC INTEGRATED INSPECTION REPORT 05000483/2008005

Dear Mr. Heflin:

On December 31, 2008, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at your Callaway Plant. The enclosed integrated inspection report documents the inspection findings, which were discussed on December 30, 2008, with Mr. F. Diya, Vice President, Nuclear, and other members of your staff.

The inspections 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. The inspectors reviewed selected procedures and records, observed activities, and interviewed personnel.

This report documents eight inspector-identified and self-revealing findings of very low safety significance (Green). Seven of these findings were determined to involve violations of NRC requirements. Additionally, three licensee-identified violations, which were determined to be of very low safety significance, are listed in this report. However, because of the very low safety significance and because they are entered into your corrective action program, the NRC is treating these findings as noncited violations, consistent with Section VI.A.1 of the NRC Enforcement Policy. If you contest the violations or the significance of the noncited violations, 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, D.C. 20555-0001, with copies to the Regional Administrator, U.S. Nuclear Regulatory Commission, Region IV, 612 E. Lamar Blvd, Suite 400, Arlington, Texas, 76011-4125; the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, Washington, D.C. 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, and its enclosure, will be 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).

Sincerely,

/RA/

Vincent G. Gaddy, Chief
Project Branch B
Division of Reactor Projects

Docket: 50-483
License: NPF-30

Enclosure:
NRC Inspection Report 05000483/2008005
w/Attachment: Supplemental Information

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File located: R:_Reactors\CW\2008\2008005RP-DED.doc ML 090350718

SUNSI Rev Compl.	☑ Yes ☐ No	ADAMS	☑ Yes ☐ No	Reviewer Initials	VGG
Publicly Avail	☑ Yes ☐ No	Sensitive	☐ Yes ☑ No	Sens. Type Initials	VGG

RIV:SRI:DRP/B	C:DRS/OG	C:DRS/PSB1	C:DRS/EB2	C:DRS/EB1
DDumbacher	RIantz	MShannon	NO'Keefe	TRFranholtz
/RA/Vgg for -E	/RA/	/RA/	/RA/	/RA/
1/20/09	1/15/09	1/15/09	1/15/09	1/15/09
C:DRS/PSB2	C:DRP/B			
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U.S. NUCLEAR REGULATORY COMMISSION

REGION IV

Docket:	050-483
License:	NPF-30
Report:	05000483/2008005
Licensee:	Union Electric Company
Facility:	Callaway Plant
Location:	Junction Highway CC and Highway O Fulton, MO
Dates:	September 25, through December 31, 2008
Inspectors:	D. Dumbacher, Senior Resident Inspector J. Groom, Resident Inspector R. Kopriva, Senior Reactor Inspector J. Adams, PhD., Reactor Inspector Larry Ricketson, P.E., Senior Health Physicist, Plant Support Branch Don Stearns, Health Physicist, Plant Support Branch
Approved By:	V. Gaddy, Chief, Project Branch B Division of Reactor Projects

SUMMARY OF FINDINGS

IR 05000483/2008005: 9/25-12/31/2008; Callaway Plant, Integrated Resident and Regional Report; Maintenance Risk Assessments and Emergent Work Control, Postmaintenance Testing, Refueling and Other Outage Activities, Access Control to Radiologically Significant Areas, and Event Follow-up.

The report covered a 3-month period of inspection by resident inspectors and announced baseline inspections by regional based inspectors. Seven Green noncited violations and one Green Finding were identified. The significance of most findings is indicated by their color (Green, White, Yellow, or Red) using Inspection Manual Chapter 0609, "Significance Determination Process." Findings for which the significance determination process does not apply may be Green or be assigned a severity level after NRC management review. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, "Reactor Oversight Process," Revision 4, dated December 2006.

A. NRC-Identified Findings and Self-Revealing Findings

Cornerstone: Initiating Events

- Green. The inspectors identified a self-revealing noncited violation of Technical Specification 5.4.1.a, "Procedures," after improper isolation of the main steam isolation valves by the Callaway control room operators resulted in a reactor trip signal and auxiliary feedwater actuation on October 11, 2008. Procedure OTG-ZZ-00006, "Plant Cooldown Hot Standby to Cold Shutdown," allowed premature main steam isolation valve closures just after entering Mode 4. The operator then decided to reopen main steam isolation Valve A and atmospheric Steam Dump A. This created a significant increase in steam flow from the steam generator which caused the steam generator level to swell up to the P-14 steam generator high level feedwater isolation setpoint. The steam generator levels all decreased to the steam generator narrow range low-low setpoint generating the need for auxiliary feedwater actuation.

  This finding was greater than minor because it was associated with the Initiating Events cornerstone attribute of procedural quality and it affected the objective to limit the likelihood of those events that upset plant stability and challenge critical safety functions during shutdown as well as power operations. Using Manual Chapter 0609.04, "Phase 1 - Initial Screening and Characterization of Findings," this finding is determined to be of very low safety significance since this finding did not affect the Technical Specification limit for reactor coolant system leakage, did not contribute to both the likelihood of a reactor trip and mitigation equipment or functions not being available, and did not increase the likelihood of a fire or internal/external flooding. This finding had a crosscutting aspect in the area of human performance associated with the decision making component because the licensee failed to communicate, in a timely manner, decisions to personnel who have a need to know the information in order to perform work safely [H.1(c)] (Section 1R20).

- Green. The inspectors identified a self-revealing finding for failure of the engineering department to perform a material equivalency evaluation to ensure replacement components do not adversely affect plant operations. On November 11, 2008, Callaway Plant experienced a trip of main feedwater

Pump B due to low lube oil pressure. Since the reactor was at greater than 80 percent power, the plant operators inserted a manual reactor trip. Following the reactor trip, maintenance personnel discovered two pieces of o-ring foreign material within main feedwater Pump B bearing oil supply pressure regulating Valve FCV0970. The foreign material was found wrapped around the regulating spring which inhibited valve movement and caused the lube oil low pressure condition. The licensee determined that the ethylene propylene diene M-class type o-ring became pliable when exposed to lube oil and was allowed to fall and be introduced into the system as foreign material. The ethylene propylene diene M-class o-rings had been approved as an equivalent replacement in July 1999, for the vendor recommended Buna-N type o-rings without performing an engineering material equivalency evaluation. Buna-N material is approved for use in petroleum based systems while ethylene propylene diene M-class is not.

This finding is greater than minor because it is associated with the design control attribute of the Initiating Events cornerstone and affects the cornerstone objective to limit the likelihood of those events that upset plant stability and challenge critical safety functions during shutdown and power operations. Using Manual Chapter 0609.04, "Phase 1 – Initial Screening and Characterization of Findings," the finding is determined to be potentially risk significant because it contributed to both the likelihood of a reactor trip and the likelihood that mitigation equipment or functions will not be available. When evaluated per Manual Chapter 0609 Appendix A, "Determining the Significance of Reactor Inspection Findings for At-Power Situations," and the Callaway Plant Phase 2 pre-solved table item “Failure to Reestablish Main Feedwater,” the inspectors determined this finding to be of very low safety significance. This issue was entered into the licensee's corrective action program as Callaway Action Request 200811781. This finding was determined to not have a crosscutting aspect because the performance deficiency is not indicative of current licensee performance (Section 40A3).

Cornerstone: Mitigating Systems

- Green. The inspectors identified a noncited violation of 10 CFR 50.65(a)(4), for failure to adequately assess and manage shutdown risk associated with maintenance activities in the reactor building. Specifically, on October 15, 2008, the inspectors found foreign material exclusion covers installed on the Train B containment recirculation sump. The covers were installed on October 14, 2008, per the direction of the containment coordinator without notification to the control room. The covers were installed to prevent debris from entering the sump. Following discussions with operations personnel, the inspectors found that the Train B containment recirculation sump was inappropriately credited in the licensee’s shutdown safety assessment. An updated shutdown safety assessment was performed and it was determined that plant risk remained unchanged.

  This finding is greater than minor because the licensee’s risk assessment inappropriately credited risk-significant structures, systems and components that were unavailable during maintenance. This finding affected the Mitigating Systems cornerstone. Using Manual Chapter 0609, Appendix G, “Shutdown Operations Significance Determination Process,” the finding was found to be of very low safety significance because the licensee maintained two trains of decay heat removal operable and adequate equipment was available to support feed and bleed operations for at least 24 hours. This issue was entered into the licensee's corrective action program as Callaway Action Request 200810540. This finding had a crosscutting aspect in the area of human performance associated with the decision making component because the licensee failed to obtain interdisciplinary input on safety-significant or risk-significant decisions. Specifically, the containment coordinator made a decision affecting the availability of the containment recirculation sumps without consulting the control room to determine the impact on plant risk [H.1 (a)] (Section 1R13).

- Green. The inspectors identified a self-revealing noncited violation of 10 CFR Part 50, Appendix B, Criterion III, “Design Control,” after a trip of the residual heat removal Train A room cooler fan revealed that AmerenUE had not adequately selected and reviewed the suitability of the newly installed fan motor thermal overloads. Additionally, the inspectors identified that the postmaintenance testing prescribed for the modified fan motor breaker did not allow sufficient time to challenge the thermal overload settings. On October 8, 2008, residual heat removal Train A room cooler fan shut down after only 22 minutes of run time. The breaker replacement modification used a calculation originally performed for the initial design of the old breaker which did not account for the cooler fan motor being a 20 horsepower motor name-plated down to a 10 horsepower rating.

  This finding is greater than minor because it is similar to Manual Chapter 0612 "Examples of Minor Issues," Example 3j, in that the engineering calculation error resulted in a condition where there was a reasonable doubt on the operability of the component and a significant programmatic deficiency associated with postmaintenance test requirements was identified that could lead to worse errors if uncorrected. The inspectors determined that the finding impacted the Mitigating Systems cornerstone. Using Manual Chapter 0609.04, "Phase 1 – Initial Screening and Characterization of Findings," the issue screened as very low safety significance because it was not a design or qualification deficiency that resulted in a loss of operability or functionality, did not create a loss of system safety function of a single train for greater than Technical Specification allowed outage time and did not affect seismic, flooding, or severe weather initiating events. This issue was entered into the licensee's corrective action program as Callaway Action Request 200810223. The inspectors determined that this finding had a crosscutting aspect in the area of problem identification and resolution associated with the corrective action component because the AmerenUE modification for certain motor control center breakers failed to have a low enough threshold to identify fan motor rating and thermal overload setting errors [P.1(a)] (Section 1R19).

- Green. The inspectors identified a self-revealing noncited violation of Technical Specification 5.4.1.a, “Procedures,” after improper restoration of the essential service water supply to the emergency diesel generator Train A lubricating oil cooler resulted in significant water flow into the emergency diesel room on October 22, 2008. Two restoration evolutions associated with the essential service water and the emergency diesel generator systems had been proceeding in parallel. The reactor operator restoring the emergency diesel generator assumed the essential service water supply was to remain isolated to the emergency diesel generator and thus changed the already approved worker protection assurance Clearance 71899 to leave the oil cooler drain valve open with no tag. Starting the essential service water pump pressurized the drain valve and produced significant water spray flow into the emergency diesel generator room until noticed by a diesel vendor representative about 30 minutes later.

This finding was greater than minor because if left uncorrected the deficiencies could become a more significant safety concern. The finding affected the Mitigating Systems cornerstone. Using Manual Chapter 0609.04, "Phase 1 - Initial Screening and Characterization of Findings," this finding is determined to be of very low safety significance since this finding was not a design or qualification deficiency, did not represent a loss of system or train safety function and did not screen as potentially risk significant due to a flooding initiating event using the criteria on the characterization worksheet. This finding had a crosscutting aspect in the area of human performance associated with the work practices component because the licensee's pre-job briefing, self- and peer-checking, and proper documentation of activity were inadequate to overcome worker protection assurance clearance process problems and an inexperienced operating supervisor. These less than adequate worker practices resulted in personnel proceeding in the face of uncertainty [H.4(a)] (Section 1R20).

Cornerstone: Barrier Integrity

- Green. The inspectors identified a self-revealing noncited violation of Technical Specification 5.4.1a, "Procedures," for inadequate procedural guidance that resulted in the failure of the residual heat removal Train A pump mechanical seal. On October 22, 2008, the licensee discovered a solid stream of water issuing from the residual heat removal Train A pump mechanical seal. The failure occurred because of installation difficulties encountered on October 8, 2008, when the seal sleeve was installed with the seal locking collar engaged. This configuration resulted in increased loading on the seal seating surfaces that resulted in surface chipping and led to seal failure after approximately 48 hours of shutdown cooling operation. Mechanical seal replacement Procedure MPM-EJ-QP001, "Residual Heat Removal Pump Overhaul," did not specify that the seal sleeve needed to be installed prior to installing the seal-locking collar. Additionally, the installation procedure did not specify any post-installation acceptance criteria to ensure the seal is properly seated. An analysis of the seal failure determined that leakage would not exceed the 2 gallon per minute Technical Specification limit but would exceed the 1 gallon per minute administrative limit for emergency core cooling system leakage outside containment.

  This finding is more than minor because it was associated with the Barrier Integrity cornerstone attribute of procedural quality and affects the associated cornerstone objective to provide reasonable assurance that physical design barriers protect the public from radionuclide releases caused by accidents or releases. Using Manual Chapter 0609, Appendix H, "Containment Integrity Significance Determination Process," this finding was determined to be a Type B finding since it was related to a degraded condition that has potentially important implications for the integrity of the containment, without affecting the likelihood of core damage. This finding was found to be of very low safety significance since the 2 gallon per minute limit assumed in the post accident dose calculation was preserved and therefore the degraded condition would have no impact on large early release frequency. This issue was entered into the licensee's corrective action program as Callaway Action Request 200810933. This finding did not have a crosscutting aspect since it was not a performance deficiency indicative of current licensee performance (Section 1R19).

- Green. The inspectors identified a self-revealing noncited violation of Technical Specification 5.4.1a, "Procedures," for the failure to close Valve BNV0002 during a fill of the spent fuel pool resulting in approximately 2000 gallons of water being inadvertently transferred from the spent fuel pool to the refueling water storage tank. On November 7, 2008, Procedure OTN-EC-00001 was performed to add makeup water to the spent fuel pool. Prior to performing the evolution, operations briefed that the refueling water storage tank was on recirculation and that this alignment needed to be secured prior to performing a fill of the spent fuel pool. Following termination of the refueling water storage tank recirculation lineup and after a fill of the spent fuel pool was initiated, the control room received annunciator "RWST Lev HILO." The crew recognized that an inadvertent transfer of spent fuel pool water to the refueling water storage tank was in progress and directed that Valves ECV0076 and BNV0002 be closed. It was later discovered that poor communication between operators on the status of Valve BNV0002 resulted in the refueling water storage tank remaining on recirculation during the fill operation.

  This finding is more than minor because it was associated with the Barrier Integrity cornerstone attribute of human performance and affects the associated cornerstone objective to provide reasonable assurance that physical design barriers protect the public from radionuclide releases caused by accidents or releases. Using Manual Chapter 0609.04, "Phase 1 - Initial Screening and Characterization of Findings," this finding was determined to be of very low safety significance because it only represents a degradation of the radiological barrier function provided by the spent fuel pool. This issue was entered into the licensee's corrective action program as Callaway Action Request 200811692. This finding had a crosscutting aspect in the area of human performance associated with the work control component because operations personnel failed to effectively communicate work status to the control room [H.3(b)] (Section 4OA3).

Cornerstone: Occupational Radiation Safety

- Green. The inspectors reviewed a self-revealing, noncited violation of Technical Specification 5.7.1, which resulted from a failure of three individuals to comply with high radiation area entry requirements. Specifically, on October 20, 2008, three engineers touring the reactor building entered a posted high radiation area without signing in on a radiation work permit which allowed entry into a high radiation area, and did not receive a briefing on dose rates in the high radiation area. Shortly after entering the high radiation area, one of the engineers received an electronic dosimeter rate alarm when dose rates in the area exceeded the 50 millirem per hour setpoint. The licensee entered this event into their corrective action program and conducted an Event Review Team meeting to determine the probable causes that led to the event and recommend corrective actions to prevent the event from happening in the future.

Failure to comply with high radiation area entry requirements is a performance deficiency. This finding is greater than minor because it was associated with the cornerstone attribute of exposure control and affected the cornerstone objective, in that, the failure to meet high radiation area entry requirements increases the potential for increased radiation dose. This finding involved an individual workers' unplanned, unintended dose or potential of such dose (resulting from actions or conditions contrary to Technical Specifications) which could have been significantly greater as a result of a single minor, reasonable alteration of the circumstances. Using the Occupational Radiation Safety Significance Determination Process, the inspectors determined the finding to have very low safety significance because (1) it was not associated with ALARA planning or work controls, (2) there was no overexposure, (3) there was no substantial potential for an overexposure, and (4) the ability to assess dose was not compromised. Additionally, the finding had a crosscutting aspect in the area of human performance, work practices component, because the workers failed to use error prevention tools such as self- and peer-checking [H.4(a)] (Section 2OS1).

B. Licensee-Identified Violations

Violations of very low safety significance, which were identified by the licensee, have been reviewed by the inspectors. Corrective actions taken or planned by the licensee have been entered into the licensee's corrective action program. These violations and corrective action tracking numbers (Callaway Action Requests) are listed in Section 4OA7.

REPORT DETAILS

Summary of Plant Status

AmerenUE operated the Callaway Plant near 100 percent until October 11, 2008, when the plant was shut down for Refueling Outage 16. The plant was returned to service at 97 percent power on November 11, 2008, when a manual reactor trip necessitated by a main feedwater Pump B trip occurred. The plant was restarted on November 12, 2008, and returned to near 100 percent power on November 14, 2008. On December 11, 2008, an automatic reactor trip occurred from 100 percent power after condensate Pump C experienced an electrical motor fault. The plant was restarted December 12, 2008, and returned to near 98 percent power when on December 14, 2008, operators performed a manual reactor trip after recognizing that condensate Pump B had experienced an electrical fault trip. The plant was restarted on December 22, 2008, and returned to 100 percent power on December 25, 2008. The plant was maintained at full power the remainder of the inspection period.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity, and Emergency Preparedness

1R04 Equipment Alignments (71111.04)

.1 Partial Walkdown

a. Inspection Scope

The inspectors performed partial system walkdowns of the following risk-significant systems:

- October 8, 2008, charging system, pressurizer auxiliary spray line
- October 12, 2008, cold overpressure mitigation system
- October 28, 2008, instrument Busses NN01 and NN03 during an unplanned loss of Bus NN04

The inspectors selected these systems based on their risk significance relative to the reactor safety cornerstones at the time they were inspected. The inspectors attempted to identify any discrepancies that could affect the function of the system, and, therefore, potentially increase risk. The inspectors reviewed applicable operating procedures, system diagrams, Final Safety Analysis Report, Technical Specification requirements, administrative Technical Specifications, outstanding work orders, condition reports, and the impact of ongoing work activities on redundant trains of equipment in order to identify conditions that could have rendered the systems incapable of performing their intended functions. The inspectors also walked down accessible portions of the systems to verify system components and support equipment were aligned correctly and operable. The inspectors examined the material condition of the components and observed operating parameters of equipment to verify that there were no obvious deficiencies. The inspectors also verified that the licensee had properly identified and resolved equipment alignment problems that could cause initiating events or impact the capability of mitigating systems or barriers and entered them into the corrective action program with the appropriate significance characterization. Specific documents reviewed during this inspection are listed in the attachment.

These activities constitute completion of three partial system walkdown samples as defined in Inspection Procedure 71111.04-05.

b. Findings

No findings of significance were identified.

.2 Complete Walkdown

a. Inspection Scope

The inspectors performed two complete system alignment inspections to verify the functional capability of the systems. The inspectors selected these systems because they were either safety-significant or risk-significant in the licensee's probabilistic risk assessment. The inspectors walked down the systems to review mechanical and electrical equipment line ups, electrical power availability, system pressure and temperature indications, as appropriate, component labeling, component lubrication, component and equipment cooling, hangers and supports, operability of support systems, and to ensure that ancillary equipment or debris did not interfere with equipment operation. The inspectors reviewed a sample of past and outstanding work orders to determine whether any deficiencies significantly affected the system function. In addition, the inspectors reviewed the corrective action program database to ensure that system equipment-alignment problems were being identified and appropriately resolved. Specific documents reviewed during this inspection are listed in the attachment.

- October 24, 2008, Trains A and B spent fuel pool cooling system with a recent full core offload
- December 4, 2008, normal service water system when it was the only source of water available to essential service water Train A components

These activities constitute completion of two complete system walkdown samples as defined in Inspection Procedure 71111.04-05.

b. Findings

No findings of significance were identified.

1R05 Fire Protection (71111.05)

Quarterly Fire Inspection Tours

a. Inspection Scope

The inspectors conducted fire protection walkdowns that were focused on availability, accessibility, and the condition of firefighting equipment in the following risk-significant plant areas:

- October 14, 2008, reactor building (all elevations)

- October 27, 2008, turbine building (all elevations)

- November 2, 2008, fire Area A-4, Rooms 1107 and 1108, centrifugal charging pump Train B and safety injection pump rooms

- November 4, 2008, fire Areas A-13 and A-14, auxiliary feedwater pump rooms

- November 7, 2008, fire Area A-3, Room 1407, boric acid batch tank addition room

- December 8, 2008, fire Area C-1, Room 3301, essential service water piping modification

The inspectors reviewed areas to assess if licensee personnel had implemented a fire protection program that adequately controlled combustibles and ignition sources within the plant; effectively maintained fire detection and suppression capability; maintained passive fire protection features in good material condition; and had implemented adequate compensatory measures for out of service, degraded or inoperable fire protection equipment, systems, or features, in accordance with the licensee's fire plan. The inspectors selected fire areas based on their overall contribution to internal fire risk as documented in the plant's Individual Plant Examination of External Events with later additional insights, their potential to affect equipment that could initiate or mitigate a plant transient, or their impact on the plant's ability to respond to a security event. Using the documents listed in the attachment, the inspectors verified that fire hoses and extinguishers were in their designated locations and available for immediate use; that fire detectors and sprinklers were unobstructed, that transient material loading was within the analyzed limits; and fire doors, dampers, and penetration seals appeared to be in satisfactory condition. The inspectors also verified that minor issues identified during the inspection were entered into the licensee's corrective action program. Specific documents reviewed during this inspection are listed in the attachment.

These activities constitute completion of six quarterly fire-protection inspection samples as defined in Inspection Procedure 71111.05.

b. Findings

No findings of significance were identified.

1R06 Flood Protection Measures (71111.06)

a. Inspection Scope

The inspectors reviewed the Final Safety Analysis Report, the flooding analysis, and plant procedures to assess seasonal susceptibilities involving internal flooding; reviewed the Final Safety Analysis Report and corrective action program to determine if licensee personnel identified and corrected flooding problems; inspected underground bunkers/manholes to verify the adequacy of sump pumps, level alarm circuits, cable splices subject to submergence, and drainage for bunkers/manholes; verified that operator actions for coping with flooding can reasonably achieve the desired outcomes; and walked down the area listed below to verify the adequacy of equipment seals located below the flood line, floor and wall penetration seals, watertight door seals, common drain lines and sumps, sump pumps, level alarms, and control circuits, and temporary or removable flood barriers. Specific documents reviewed during this inspection are listed in the attachment.

- October 27, 2008, review of October 22, 2008, water intrusion in emergency diesel generator Train A, Room 5210

These activities constitute completion of one flood protection measures inspection sample as defined in Inspection Procedure 71111.06-05.

b. Findings

No findings of significance were identified.

1R08 In-service Inspection Activities (71111.08)

Completion of Sections .1 through .5, below, constitutes completion of one sample as defined in Inspection Procedure 71111.08-05.

.1 Inspection Activities Other Than Steam Generator Tube Inspection, Pressurized Water Reactor Vessel Upper Head Penetration Inspections, and Boric Acid Corrosion Control (71111.08-02.01)

a. Inspection Scope

The inspection procedure requires review of two or three types of nondestructive examination activities and, if performed, one to three welds on the reactor coolant system pressure boundary. Also review one or two examinations with recordable indications that have been accepted by the licensee for continued service.

The inspectors directly observed the following nondestructive examinations:

System	Identification	Exam Type	Result
Reactor Coolant System	Report Number BOP-PT-08-350J Job # 08006823 Pressurizer Auxiliary Spray Piping Socket Welds	PT	No Indications
Reactor Coolant System	Report Number BOP-PT-08-5358 Job # 08006823 Pressurizer Auxiliary Spray Piping Socket Welds	PT	No Indications
Reactor Coolant System	Report Number BOP-PT-08-359 Job # 08006823 Pressurizer Auxiliary Spray Piping Socket Welds	PT	No Indications

Reactor Coolant System	Report Number BOP-PT-08-360 Job # 08006823 Pressurizer Auxiliary Spray Piping Socket Welds	PT	No Indications
Reactor Coolant System	UT-08-021 Calibration/Examination – Pressurizer Spray Line Valve PCV-455C to 4 in. Pipe	UT	No Indications
Reactor Coolant System	UT-08-029 and UT-08-031 Calibration/Examination – Steam Generator A, Vessel to Tube Sheet	UT	No Indications
Containment	Report # 16-34-01 ASME Section XI IWE Containment Pressure Boundary Inspection - Emergency Personnel Hatch (Auxiliary Access Hatch)	VT	Oxidation found on conduit hanger connection point. Acceptance criteria not met. Job 08007879 written to clean/repair/ inspect. CAR 200811011
Containment	Report # 16-33-01 ASME Section XI IWE Containment Pressure Boundary Inspection - Personnel Hatch	VT	Some surface oxidation identified. Acceptable per Procedure ESP-ZZ-016. CAR 200811011
Component Cooling Water	Report # 5042-08-114 Component ID: 2-EG-03-R011 Heat Exchanger Piping Support/Restraint	VT	No Indications

The inspectors reviewed records for the following nondestructive examinations:

System	Identification	Exam Type	Result
Essential Service Water	Report Number BOP-MT-08-019 Job # 07008489 Weld Prep Area – ¼ in. Hole Repair	MT	No indications
Essential Service Water	Report Number BOP-MT-08-020 Job # 07008491 Weld Prep Area ¼ in. Hole Repair	MT	No indications
Essential Service Water	Report Number BOP-MT-08-021 Job # 07008489 Base Metal Repair of the drilled hole in line EF-036-HBC-8 in.	MT	No indications

Essential Service Water	Report Number BOP-MT-08-023 Job # 07008165 Welds Spool # 1 to Existing Line 056-HBC-16 in.	MT	No indications
Essential Service Water	Report Number BOP-MT-08-024 Job # 07008165 Welds Spool # 2 to Existing Line 056-HBC-16 in.	MT	No indications
Containment Spray	Report Number BOP-PT-08-294 Job # 08004424 ¾ in. Valve to Pipe	PT	No indications
Containment Spray	Report Number BOP-PT-08-297 Job # 08004424 Fillet welds from ¾ in. SA312 TP304 pipe to ¾ in. SA182 F304 Elbow	PT	No indications
Nuclear Sampling	Report Number BOP-PT-08-310 Job # 06121606 The reducing coupling to valve safe-end on the upstream side of valve SJHV0015	PT	No indications
Nuclear Sampling	Report Number BOP-PT-08-311 Job # 06121606 The Tubing adapter to reducing weld on the Up-stream side of valve SJHV0015	PT	No indications
Chemical and Volume Control	Report Number BOP-PT-08-312 Job # 08501938 Saddle Weld to Pipe	PT	No indications
Auxiliary Feedwater System	Report Number BOP-PT-08-317 Job # W231106 Turbine-Driven Aux Feed Pump	PT	No indications
Essential Service Water	Report Number BOP-PT-08-319 Job # 07008163 No indications Weld Spool # 2	PT	No indications
Essential Service Water	Report Number BOP-PT-08-320 Job # 07008163 Welds Spool # 1	PT	No indications

Essential Service Water	Report Number BOP-PT-08-335 Job # 07008489 Welds 6 in. Schedule. 40 Stainless Steel 90 degree Elbow to 6 in. Schedule. 40 Carbon Steel Pipe (Spool # 3 to Existing Line)	PT	No indications
Essential Service Water	Report Number BOP-PT-08-336 Job # 07008492 Welds 6 in. Schedule. 40 Stainless steel 90 degree ell to 6 in. Schedule. 40 Carbon Steel Pipe. Weld Spool # 1 to Existing Line 065 – HVC-6 in.	PT	No indications
Chemical and Volume Control	Report Number-BOP-PT-08-337 Job # 08006823 2 in. Schedule 160 pipe to 2 in. 6000# SW Coupling	PT	No indications
Essential Service Water	Report Number BOP-PT-08-338 Job # 07008491 Feedwater-14 Welds 8 in. Schedule. 40 pipe to 8 in. Schedule. 40 pipe	PT	No indications
Essential Service Water	Report Number BOP-PT-08-339 Job # 07008489 Welds 8 in. Schedule 40 pipe to 8 in. Schedule 40 Weld Spool #2 to Spool #3	PT	No indications
Residual Heat Removal System	UT-08-006 Calibration/Examination – Circumferential Weld, U8 in. Pipe to 8 in. Elbow	UT	No indications
Residual Heat Removal System	UT-08-007 Calibration/Examination – Circumferential Weld, 8 in. Elbow to 8 in. Pipe	UT	No indications
Residual Heat Removal System	UT-08-008 Calibration/Examination – Circumferential Weld, 10 in. Tee to 10 in. Pipe	UT	No indications

Residual Heat Removal System	UT-08-009 Calibration/Examination – 10 in. Pipe to 10 in X 8 in. Reducer	UT	No indications
Residual Heat Removal System	UT-08-010 Calibration/Examination – Circumferential Weld, 10 in. Pipe to 10 in. Tee	UT	No indications
Residual Heat Removal System	UT-08-011 Calibration/Examination – Circumferential Weld, 10 in. tee to 10 in. Pipe	UT	No indications
Reactor Coolant System	UT-08-014 Calibration/Examination – 3 inch Pipe to Elbow	UT	No indications
Reactor Coolant System	UT-08-015 Calibration/Examination – Weld Overlay of Surge Nozzle to Safe-End Weld and Pipe to Safe-End Weld	UT	No indications
Reactor Coolant System	UT-08-016 Calibration/Examination – Weld Overlay of Spray Nozzle to Safe-End Weld and Pipe to Safe-End Weld	UT	No indications
Reactor Coolant System	UT-08-017 Calibration/Examination –Weld Overlay of Relief Nozzle to Safe-End Weld and Pipe to Safe-End Weld	UT	No indications
Reactor Coolant System	UT-08-018 Calibration/Examination – Weld Overlay of Safety Nozzle to Safe-End Weld and Pipe to Safe-End Weld	UT	No indications
Reactor Coolant System	UT-08-019 Calibration/Examination – Weld Overlay of Safety Nozzle to Safe-End Weld and Pipe to Safe-End Weld	UT	No indications
Reactor Coolant System	UT-08-020 Calibration/Examination – Weld Overlay of Safety Nozzle to Safe-End Weld and Pipe to Safe-End Weld	UT	No indications

During the review and observation of each examination, the inspectors verified that activities were performed in accordance with ASME Boiler and Pressure Vessel Code requirements and applicable procedures. Indications were compared with previous examinations and dispositioned in accordance with ASME Code and approved procedures. The qualifications of all nondestructive examination technicians performing the inspections were verified to be current.

One nondestructive examination with a relevant indication was accepted by the licensee for continued service.

System	Identification	Exam Type	Result
Reactor Coolant System - Pressurizer	Component ID: 2-TBB03-CIRCUM-1-W, Shell to Upper Head	UT	Previously identified indications acceptable in accordance with ASME Section XI, 1998 Edition, 2000 Addenda

The inspectors verified, by review, that the welding procedure and the welders' qualifications for the repair of the pressurizer auxiliary spray line had been properly qualified in accordance with ASME Code, Section IX, requirements. The inspectors also verified, through observation and record review, that essential variables for the gas tungsten arc welding process (machine) process were identified, recorded in the procedure qualification record, and formed the bases for qualification of the welding procedure specifications.

b. Findings

No findings of significance were identified.

.2 Vessel Upper Head Penetration Inspection Activities (71111.08-02-02)

a. Inspection Scope

The inspection procedure requires observation or review of the reactor vessel head bare metal visual examinations, or review of the post examination videotape and examination procedures. In particular, review licensee criteria for confirming visual examination quality and instructions resolving interference or masking issues. Also, if the licensee is performing non-visual nondestructive examination of the reactor vessel head, review a sample of these examinations.

The licensee was not required to perform any volumetric nondestructive examination of the reactor vessel upper head penetrations during Refueling Outage 16 per the licensee's nondestructive examination inspection plan. In Attachment II to ULNRC 4630, "Response to NRC Bulletin 2002-01," dated April 1, 2002, AmerenUE states that the Callaway Plant is considered to have low susceptibility to circumferential cracking of the reactor pressure vessel head penetration nozzles. Per revised NRC Order EA-03-009, dated February 20, 2004, the inspections applicable to plants in the low category are defined in Paragraph IV.C.(3): a bare metal visual examination of 100 percent of the reactor pressure vessel head surface per IV.C.(5)(a) shall be performed at least every third refueling outage or every 5 years, whichever occurs first. The licensee last performed a full reactor pressure vessel head surface examination during Refueling Outage 15 in the spring of 2007.

Paragraph IV.D of NRC Order EA-03-009 requires that visual inspections shall be performed to identify potential boric acid leaks from pressure retaining components above the reactor pressure vessel head. These inspections are to be performed each refueling outage. While performing this boric acid inspection, a small amount of boric acid was identified coming from one of the control rod drive mechanism canopy seals. The licensee elected to install a mechanical nozzle seal assembly clamp to repair the flawed canopy seal. The canopy seal is used to form a secondary seal to the threaded connection of the control rod drive mechanism, and the threaded connection is used as the pressure retaining boundary. ASME Code Section XI is not applicable in this application because the mechanical nozzle seal assembly clamp is not being used as the primary pressure boundary retaining device. Since the purpose of the control rod drive mechanism mechanical nozzle clamp assembly is to form the secondary seal, the clamp materials must meet the requirements of ASME Section III, Part NB-3671.7, "Sleeve Coupled and Other Patented Joints." The inspectors reviewed the clamp drawings, installation procedure, and the video of the clamp installation. No problems or concerns were identified.

b. Findings

No findings of significance were identified.

.3 Boric Acid Corrosion Control Inspection Activities (71111.08-02.03)

a. Inspection Scope

The inspectors reviewed the licensee's boric acid corrosion control program and inspection activities, and verified that visual inspections emphasized locations where boric acid leaks could cause degradation of safety significant components.

The inspectors reviewed seven instances where boric acid deposits were found on reactor coolant system piping components:

Component Number	Description	Callaway Action Request
BBV0104	RCS Pressurizer BBLT0460 vent valve	200810457
BBV0130	RCP A Seal Water supply drain	200810578
BGHV8147	CVCS Regenerative Heat Exchanger to Loop 4 Cold Leg Isolation	200810457
BGV0003	CVCS Letdown Orifice A Out Throttle Valve	200810457
EMV0134	SI Pump A Discharge to Hot Leg Loop 3 Test Connection Downstream Isolation	200810457
EPFO0003	Boron Flow Orifice	200810457, and 200810613
EPV0013	SI Accumulator Tank B EPPT0962 and EPLT0592 Variable Root	200810457

The condition of all the components was appropriately entered into the Callaway Action Request system, and corrective actions taken were consistent with ASME code requirements. The inspectors reviewed nine engineering evaluations performed as required by the Callaway Action Requests. The evaluations were conducted and the affected components were either cleaned, replaced, tightened, or had their packing adjusted. No evidence of corrosion wastage, or significant damage was found. Of the nine evaluations reviewed, all were found acceptable based on ASME Code Case N-566-2.

b. Findings

No findings of significance were identified.

.4 Steam Generator Tube Inspection Activities (71111.08-02.04)

a. Inspection Scope

The steam generators were not required to be inspected during Refueling Outage 16. For Callaway Plant, Technical Specification 5.5.9, "The Steam Generator Program," establishes the criteria for inspection of the steam generators. Per Technical Specification 5.5.9.d.2, "The program shall be established and implemented to ensure that steam generator tube integrity is maintained. In addition, the program shall include the following provisions: Inspect 100 percent of the tubes at sequential periods of 144, 108, 72, and, thereafter, 60 effective full power months. The first sequential period shall be considered to begin after the first in-service inspections of the steam generators." The steam generators were replaced in the fall of 2005, Refueling Outage 14. The licensee completed a full, 100 percent inspection of the steam generators in the spring of 2007, Refueling Outage 15. Per their program, the next interval for any steam generator inspection will be during the fall of 2011, Refueling Outage 18.

b. Findings

No findings of significance were identified.

.5 Identification and Resolution of Problems (71111.08-02)

a. Inspection scope.

The inspection procedure requires review of a sample of problems associated with in-service inspections documented by the licensee in the corrective action program for appropriateness of the corrective actions.

The inspectors reviewed 16 corrective action reports which dealt with in-service inspection activities and found the corrective actions were appropriate. Action requests reviewed are listed in the documents reviewed section. From this review the inspectors concluded that the licensee has an appropriate threshold for entering issues into the corrective action program and has procedures that direct a root cause evaluation when necessary. The licensee also has an effective program for applying industry operating experience.

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification Program (71111.11)

a. Inspection Scope

On December 10, 2008, the inspectors observed a crew of licensed operators in the plant's simulator during licensed operator requalification examinations to verify that operator performance was adequate, evaluators were identifying and documenting crew performance problems, and training was being conducted in accordance with licensee procedures. The inspectors evaluated the following areas:

- Licensed operator performance
- Crew's clarity and formality of communications
- Crew's ability to take timely actions in the conservative direction
- Crew's prioritization, interpretation, and verification of annunciator alarms
- Crew's correct use and implementation of abnormal and emergency procedures
- Control board manipulations
- Oversight and direction from supervisors
- Crew's ability to identify and implement appropriate Technical Specification actions and emergency plan actions and notifications

The inspectors compared the crew's performance in these areas to pre-established operator action expectations and successful critical task completion requirements. Specific documents reviewed during this inspection are listed in the attachment.

These activities constitute completion of one quarterly licensed-operator requalification program sample as defined in Inspection Procedure 71111.11.

b. Findings

No findings of significance were identified.

1R12 Maintenance Effectiveness (71111.12)

a. Inspection Scope

The inspectors evaluated degraded performance issues involving the following risk significant systems:

- December 8, 2008, Callaway Action Requests 200708186 and 200810287, volume control system relief Valve BG8117 transients

- December 9, 2008, Safety injection Trains A and B unavailability for operating Cycle 16
- December 9, 2008, Callaway Action Request (CAR) 200804010, control rod drive mechanism Fans A and B failures

The inspectors independently verified the licensee's actions to address system performance or condition problems in terms of the following:

- Implementing appropriate work practices
- Identifying and addressing common cause failures
- Scoping of systems in accordance with 10 CFR 50.65(b)
- Characterizing system reliability issues for performance
- Charging unavailability for performance
- Trending key parameters for condition monitoring
- Ensuring proper classification in accordance with 10 CFR 50.65(a)(1) or (a)(2)
- Verifying appropriate performance criteria for structures, systems, and components classified as having an adequate demonstration of performance through preventive maintenance, as described in 10 CFR 50.65(a)(2), or as requiring the establishment of appropriate and adequate goals and corrective actions for systems classified as not having adequate performance, as described in 10 CFR 50.65(a)(1)

The inspectors assessed performance issues with respect to the reliability, availability, and condition monitoring of the system. In addition, the inspectors verified maintenance effectiveness issues were entered into the corrective action program with the appropriate significance characterization. Specific documents reviewed during this inspection are listed in the attachment.

These activities constitute completion of three quarterly maintenance effectiveness samples as defined in Inspection Procedure 71111.12-05.

b. Findings

No findings of significance were identified.

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

a. Inspection Scope

The inspectors reviewed licensee personnel's evaluation and management of plant risk for the maintenance and emergent work activities affecting risk-significant and safety-related equipment listed below to verify that the appropriate risk assessments were performed prior to removing equipment for work:

- October 11, 2008, routine risk associated with vehicles and temporary diesels in the switchyard

- October 13, 2008, routine shutdown risk assessment associated with reactor coolant system drain-down

- October 15, 2008, routine shutdown risk assessment associated with identification of temporary foreign material exclusion covers on the Train B containment recirculation sump

- October 26, 2008, emergent risk assessment required by Technical Specification 3.0.4 for transition to Mode 6 with Train B control room air conditioning system inoperable

- November 5-6, 2008, emergent risk assessment for failed turbine-driven auxiliary feedwater pump governor servo while the plant was in Mode 3

- December 2, 2008, planned routine risk associated with the draining of essential service water system Train A tie-in of the new high density polyethylene piping

The inspectors selected these activities based on potential risk significance relative to the reactor safety cornerstones. As applicable for each activity, the inspectors verified that licensee personnel performed risk assessments as required by 10 CFR 50.65(a)(4) and that the assessments were accurate and complete. When licensee personnel performed emergent work, the inspectors verified that the licensee personnel promptly assessed and managed plant risk. The inspectors reviewed the scope of maintenance work, discussed the results of the assessment with the licensee's probabilistic risk analyst or shift technical advisor, and verified plant conditions were consistent with the risk assessment. The inspectors also reviewed the Technical Specification requirements and inspected portions of redundant safety systems, when applicable, to verify risk analysis assumptions were valid and applicable requirements were met. Specific documents reviewed during this inspection are listed in the attachment.

These activities constitute completion of six maintenance risk assessments and emergent work control inspection samples as defined in Inspection Procedure 71111.13-05.

b. Findings

Introduction. The inspectors identified a Green noncited violation of 10 CFR 50.65(a)(4), because the licensee failed to adequately assess and manage shutdown risk associated with maintenance activities in the reactor building.

Description. On October 14, 2008, the licensee installed tarpaulin foreign material exclusion covers on the Train B containment recirculation sump to prevent debris from entering the sump. The covers were installed per the direction of the containment coordinator and without notification to the control room. At the time the sump was covered, the plant was in Mode 6 with the reactor vessel head de-tensioned and reactor coolant system inventory approximately 10 inches below the reactor vessel flange. The licensee’s shutdown safety assessment conducted per Procedure EDP-ZZ-01129, "Callaway Plant Risk Assessment," determined plant risk was yellow based on actual reactor coolant system level being maintained less than 94.1 inches. Yellow was the licensee's second highest of four risk categories. Because the covers were installed without notifying the control room, the shutdown safety assessment crediting containment recirculation capability in the reactor coolant system inventory section was incorrect.

Resident inspectors noted the installed foreign material exclusion covers on containment recirculation sump Train B during a routine walkdown of containment on October 15, 2008. The inspectors questioned the operations staff if the foreign material exclusion covers were appropriate since recirculation capability was credited in the current shutdown safety assessment. The licensee confirmed that the containment recirculation sumps were inappropriately credited in the current shutdown safety assessment. The licensee performed an updated shutdown safety assessment and determined that plant risk remained unchanged.

Analysis. The inspectors determined that the licensee's failure to adequately assess and manage the shutdown risk associated with maintenance activities in the reactor building was a performance deficiency. The inspectors determined that the finding impacted the Mitigating Systems cornerstone. The finding was determined to be more than minor because the licensee's risk assessment inappropriately credited risk-significant structures, systems and components that were unavailable during maintenance. Specifically, the unavailability of containment recirculation sumps was not accurately captured in the shutdown risk assessment for October 15, 2008. The inspectors assessed the finding using Manual Chapter 0609, Appendix G, "Shutdown Operations Significance Determination Process." Using Attachment 1, "Phase 1 Operational Checklists for Both PWRs and BWRs," the inspectors determined the finding to be of very low safety significance because the licensee maintained two trains of decay heat removal operable and adequate equipment was available to support feed and bleed operations for at least 24 hours. This finding had a crosscutting aspect in the area of human performance associated with the decision making component because the licensee failed to obtain interdisciplinary input on safety-significant or risk-significant decisions. Specifically, the containment coordinator made a decision affecting the availability of the containment recirculation sumps without consulting the control room to determine the impact on plant risk [H.1(a)].

Enforcement. Title 10 of the Code of Federal Regulations, Part 50.65(a)(4), requires, in part, that licensees assess and manage the increase in risk that may result from proposed maintenance activities. Contrary to the above, on October 15, 2008, the licensee failed to adequately assess and manage the increased risk of maintenance in the reactor building that made containment recirculation sump Train B unavailable. Because this issue was of very low safety significance and was entered into the corrective action program as CAR 200810540, this violation is being treated as a noncited violation consistent with Section VI.A.1 of the NRC Enforcement Policy: Noncited Violation (NCV) 05000483/2008005-01, "Inadequate Shutdown Risk Assessment for Maintenance Activities in the Reactor Building."

1R15 Operability Evaluations (71111.15)

a. Inspection Scope

The inspectors reviewed the following issues:

- September 27, 2008, pressurizer auxiliary spray line leak, CAR 200809886

- October 9, 2008, residual heat removal Train A pump mechanical seal leak, CAR 2008010241

- October 9, 2008, fuel oil return line leakage on emergency diesel generator Train A, CAR 200810222

- October 22, 2008, component cooling water Train A heat exchanger divider plate leak-by, CAR 200810719

- October 26, 2008, Battery NK12, Cell 34 low individual cell voltage, CAR 200811097

- November 16, 2008, containment Cooler SGN01A leaks caused by essential service water column separation during engineered safeguards testing, CAR 200810348

- November 3, 2008, degraded liner discovered in the normal containment sump, CAR 200811479

The inspectors selected these potential operability issues based on the risk-significance of the associated components and systems. The inspectors evaluated the technical adequacy of the evaluations to ensure that Technical Specification operability was properly justified and the subject component or system remained available such that no unrecognized increase in risk occurred. The inspectors compared the operability and design criteria in the appropriate sections of the Technical Specifications and Final Safety Analysis Report to the licensee's evaluations, to determine whether the components or systems were operable. Where compensatory measures were required to maintain operability, the inspectors determined whether the measures in place would function as intended and were properly controlled. The inspectors determined, where appropriate, compliance with bounding limitations associated with the evaluations. Additionally, the inspectors also reviewed a sampling of corrective action documents to verify that the licensee was identifying and correcting any deficiencies associated with operability evaluations. Specific documents reviewed during this inspection are listed in the attachment.

These activities constitute completion of seven operability evaluations inspection samples as defined in Inspection Procedure 71111.15-05

b. Findings

No findings of significance were identified.

1R18 Plant Modifications (71111.18)

.1 Temporary Modifications

a. Inspection Scope

The inspectors reviewed temporary modifications and the associated safety evaluation screening against the system design bases documentation, including the Final Safety Analysis Report and the Technical Specifications, and verified that the modifications did not adversely affect the system operability/availability. The inspectors also verified that installation and restoration were consistent with the modification documents and that configuration control was adequate. Additionally, the inspectors verified that the temporary modifications were identified on control room drawings, appropriate tags were placed on the affected equipment, and licensee personnel evaluated the combined effects on mitigating systems and the integrity of radiological barriers.

- October 6, 2008, Temporary Modification 06-006 to adjust gain on steam dump control circuitry for plant Tavg coast down

- November 4, 2008, Temporary Modification 08-0010 to blank flange off - one bundle of cooling coils for containment Cooler SGN01B during Cycle 17

- November 12, 2008, Temporary Modification 08-0004 to provide backup emergency diesel generators to support essential service water underground pipe replacement

These activities constitute completion of three samples for temporary plant modifications as defined in Inspection Procedure 71111.18-05

.2 Permanent Modifications

a. Inspection Scope

The inspectors reviewed key affected parameters associated with energy needs, materials/replacement components, timing, heat removal, control signals, equipment protection from hazards, operations, flow paths, pressure boundary, ventilation boundary, structural, process medium properties, licensing basis, and failure modes for the modification listed below. The inspectors verified that 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; postmodification testing will maintain the plant in a safe configuration during testing by verifying that unintended system interactions will not occur, systems, structures and components' performance characteristics still meet the design basis, the appropriateness of modification design assumptions, and the modification test acceptance criteria will be met; and licensee personnel identified and implemented appropriate corrective actions associated with permanent plant modifications. Specific documents reviewed during this inspection are listed in the attachment.

- October 17, 2008, permanent modification of residual heat removal system suction relief discharge line to the pressurizer relief tank

These activities constitute completion of one sample for permanent plant modifications as defined in Inspection Procedure 71111.18-05

b. Findings

No findings of significance were identified.

1R19 Postmaintenance Testing (71111.19)

a. Inspection Scope

The inspectors reviewed the following postmaintenance activities to verify that procedures and test activities were adequate to ensure system operability and functional capability:

- October 8, 2008, residual heat removal Train A room cooler breaker replacement
- October 8, 2008, residual heat removal Train A pump mechanical seal
- October 21, 2008, emergency diesel generator Train A following fuel injector overhauls and engine balance
- October 23, 2008, Valve EJHV8804A following Limitorque torque switch adjustment
- October 30, 2008, pressurizer relief tank modification visual inspection (VT-2)
- October 29, 2008, essential service water Train A high density polyethylene piping hydrostatic tests
- November 6, 2008, turbine-driven auxiliary feedwater Pump KFC02 remote servo replacement
- November 12, 2008, main feedwater Pump Trains A and B testing following lube oil gasket replacement

The inspectors selected these activities based upon the structures, systems, or component's ability to affect risk. The inspectors evaluated these activities for the following (as applicable):

- The effect of testing on the plant had been adequately addressed; testing was adequate for the maintenance performed
- Acceptance criteria were clear and demonstrated operational readiness; test instrumentation was appropriate

The inspectors evaluated the activities against the Technical Specifications, the Final Safety Analysis Report, 10 CFR Part 50 requirements, licensee procedures, and various NRC generic communications to ensure that the test results adequately ensured that the equipment met the licensing basis and design requirements. In addition, the inspectors reviewed corrective action documents associated with postmaintenance tests to determine whether the licensee was identifying problems and entering them in the corrective action program and that the problems were being corrected commensurate with their importance to safety. Specific documents reviewed during this inspection are listed in the attachment.

These activities constitute completion of eight postmaintenance testing inspection samples as defined in Inspection Procedure 71111.19-05.

b. Findings

1. Introduction. A self-revealing Green noncited violation of 10 CFR Part 50, Appendix B, Criterion III, “Design Control,” was identified after a trip of residual heat removal Train A room cooler fan revealed that AmerenUE had not adequately selected and reviewed the suitability of the fan motor thermal overloads. Additionally, the inspectors identified that the postmaintenance testing prescribed for the newly installed fan motor breaker did not allow sufficient time to challenge the thermal overload settings.

Description. On October 8, 2008, residual heat removal Train A room cooler fan shut down after only 22 minutes of run time. The electrical maintenance department had completed the postmaintenance test of the new fan motor breaker. Troubleshooting determined that motor fan Breaker NG01ACF3 tripped at 32 amps of current draw and that the overload heaters were only sized for a continuous 29 amps. The overload heaters were installed as part of the corrective action Modification MP01-1003 breaker bucket replacement project to replace faulty auxiliary contacts. The modification used Calculation NG-23, originally performed for the initial design of the old Breaker NG01ACF3, which did not account for the cooler fan motor being a 20 horsepower motor name-plated down to a 10 horsepower rating. This motor had been installed as such during initial construction of the plant to ensure the fan would reach rated speed within a design requirement of 5 seconds. The as-listed nameplate maximum current draw for the motor was 23 amps. Actual current draw is dependent on the motor horsepower and the flow rate or load on the fan motor.

The inspectors confirmed that this was the only fan that had a newly installed breaker. The thermal overloads are not part of the control circuitry for motor-operated valves. The inspectors’ review of the postmaintenance test prescribed for the modification determined that a runtime of 30 minutes would have been needed to challenge the overload heater's trip open setpoint. The licensee restored the fan motor using the old breaker in time to not exceed the supported residual heat removal Train A pump Technical Specification 3.5.2 allowed outage time.

Analysis. The performance deficiency associated with this finding involved the licensee’s failure to ensure the design of the residual heat removal fan cooler motor control circuitry was suitable for all plant conditions. The inspectors determined that the finding impacted the Mitigating Systems cornerstone. This finding is greater than minor because it is similar to Manual Chapter 0612 "Examples of Minor Issues," Example 3j, in that the engineering calculation error resulted in a condition where there was a reasonable doubt on the operability of the component and a significant programmatic deficiency associated with postmaintenance test requirements was identified that could lead to worse errors if uncorrected. Using Manual Chapter 0609.04, "Phase 1 – Initial Screening and Characterization of Findings," the issue screened as very low safety significance because it was not a design or qualification deficiency that resulted in a loss of operability or functionality, did not create a loss of system safety function of a single train for greater than Technical Specification allowed outage time and did not affect seismic, flooding, or severe weather initiating events. The inspectors determined that this finding has a crosscutting aspect in the area of problem identification and resolution associated with the corrective action component because the AmerenUE modification for certain motor control center breakers failed to have a low enough threshold to identify fan motor rating and thermal overload setting errors [P.1(a)].

Enforcement. Title 10 of the Code of Federal Regulations Part 50, Appendix B, Criterion III, "Design Control," requires, in part, that measures be established for the selection and review for suitability of application of materials, parts, equipment, and processes that are essential to the safety related functions of structures, systems, and components. Contrary to the above, prior to October 8, 2008, AmerenUE failed to ensure that the residual heat removal Train A room cooler would be able to perform its safety related function due to a design deficiency associated with the fan motor thermal overloads. Because this finding is of very low safety significance and has been entered into the licensee's corrective action program as CAR 200810223, this violation is being treated as a noncited violation consistent with Section VI.A of the NRC Enforcement Policy: NCV 05000483/2008005-02, "Failure to Ensure the Suitability of the Design of the Residual Heat Removal Train A Pump Room Cooler."

2. Introduction. A self-revealing Green noncited violation of Technical Specification 5.4.1a, "Procedures," was identified for inadequate procedural guidance that resulted in the failure of the residual heat removal Train A pump mechanical seal.

Description. On October 22, 2008, AmerenUE staff identified a solid stream of water issuing from the residual heat removal Train A pump mechanical seal. The seal had a leakage acceptance criterion that requires less than sixty drops per minute total leakage. The failure occurred because of installation difficulties encountered on October 7, 2008, when the seal sleeve was installed with the seal locking collar engaged. This configuration resulted in increased loading on the seal seating surfaces that resulted in surface chipping and led to seal failure after approximately 48 hours of shutdown cooling operation. Mechanical seal replacement Procedure MPM-EJ-QP001, Residual Heat Removal Pump Overhaul," did not specify that the seal sleeve needed to be installed prior to installing the seal-locking collar. Additionally, the installation procedure did not specify any post-installation acceptance criteria to ensure the seal is properly seated.

Leakage from the residual heat removal Train A pump mechanical seal is an input into the 2-gallon per minute limit specified in Technical Specification 5.5.2, "Primary Coolant Sources Outside Containment," and Final Safety Analysis Report Table 15.6-6, "Parameters Used in Evaluating the Radiological Consequences of a Loss-of-coolant-accident." The licensee administratively controls total emergency core cooling system leakage outside containment to less than 1-gallon per minute to meet the Technical Specification and Final Safety Analysis Report requirements. While the leakage observed on October 22, 2008, was not quantified, the pump shaft is equipped with a disaster bushing used to limit mechanical seal leakage in the event of a catastrophic seal failure. The design of the disaster bushing is such that there is reasonable doubt the 2 gallon per minute assumption used in the post accident dose analysis would be preserved. The licensee determined that reliance on the seal disaster bushing in a post-accident environment results in exceeding the administrative limit for emergency core cooling system leakage outside containment but the Technical Specification limit would be preserved.

The licensee, in consultation with the pump seal vendor, addressed the installation difficulties encountered during the October 7, 2008, replacement.
Procedure MPM-EJ-QP001 was modified to allow the seal sleeve to be installed prior to installing the locking collar and to check the seal housing distance post-installation to ensure the seal is properly seated.

Analysis. The performance deficiency associated with this finding involved the failure of the licensee to provide adequate procedural guidance for residual heat removal pump seal installation. This finding is more than minor because it was associated with the barrier integrity cornerstone attribute of procedural quality and affects the associated cornerstone objective to provide reasonable assurance that physical design barriers protect the public from radionuclide releases caused by accidents or releases. Using Manual Chapter 0609, Appendix H, "Containment Integrity Significance Determination Process," this finding was determined to be a Type B finding since it was related to a degraded condition that has potentially important implications for the integrity of the containment, without affecting the likelihood of core damage. This finding was found to be of very low safety significance since the 2-gallon per minute limit assumed in the post accident dose calculation was preserved and therefore the degraded condition would have no impact on large early release frequency. This finding did not have a crosscutting aspect since it was not a performance deficiency indicative of current licensee performance.

Enforcement. Technical Specification 5.4.1(a) required written procedures be established, implemented, and maintained as recommended by Regulatory Guide 1.33, Revision 2, Appendix A, February 1978. Appendix A recommends procedures for maintenance that can affect the performance of safety-related systems. Contrary to this requirement, Procedure MPM-EJ-QP001, Revision 13, did not contain enough detail to successfully replace the residual heat removal pump seal on October 7, 2008. Because of the very low safety significance of this finding and because the issue was entered into the licensee's corrective action program as CAR 200810933, it is being treated as a noncited violation, consistent with Section VI.A.1 of the Enforcement Policy: NCV 05000483/2008005-03, "Inadequate Maintenance Procedure Results in Residual Heat Removal Mechanical Seal Failure."

1R20 Refueling and Other Outage Activities (71111.20)

.1 Callaway Plant Refueling Outage 16

a. Inspection Scope

The inspectors reviewed the outage safety plan and contingency plans for the Callaway Plant Refueling Outage 16, conducted from October 11, 2008, through November 7, 2008, to confirm that licensee personnel had appropriately considered risk, industry experience, and previous site-specific problems in developing and implementing a plan that assured maintenance of defense-in-depth. During the refueling outage, the inspectors observed portions of the shutdown and cooldown processes and monitored licensee controls over the outage activities listed below.

- Configuration management, including maintenance of defense-in-depth, is commensurate with the outage safety plan for key safety functions and compliance with the applicable Technical Specifications when taking equipment out of service
- Clearance activities, including confirmation that tags were properly hung and equipment appropriately configured to safely support the work or testing
- Installation and configuration of reactor coolant pressure, level, and temperature instruments to provide accurate indication, accounting for instrument error

- Status and configuration of electrical systems to ensure that Technical Specifications and outage safety plan requirements were met, and controls over switchyard activities

- Monitoring of decay heat removal processes, systems, and components

- Verification that outage work was not impacting the ability of the operators to operate the spent fuel pool cooling system

- Reactor water inventory controls, including flow paths, configurations, and alternative means for inventory addition, and controls to prevent inventory loss

- Controls over activities that could affect reactivity

- Maintenance of secondary containment as required by the Technical Specifications

- Refueling activities, including fuel handling and sipping to detect fuel assembly leakage

- Startup and ascension to full power operation, tracking of startup prerequisites, walkdown of the containment to verify that debris had not been left which could block emergency core cooling system suction strainers, and reactor physics testing

- Licensee identification and resolution of problems related to refueling outage activities

Specific documents reviewed during this inspection are listed in the attachment.

These activities constitute completion of one refueling outage inspection sample as defined in Inspection Procedure 71111.20-05.

b. Findings

1. Introduction. A self-revealing Green noncited violation of Technical Specification 5.4.1.a, “Procedures,” was identified after Callaway control room operators' improper isolation of the main steam isolation valves resulted in a reactor trip signal and auxiliary feedwater actuation.

Description. On October 11, 2008, during a normal plant shutdown and cooldown with the reactor plant in Mode 4 at 340 degrees Fahrenheit, the operators performed Procedure OTG-ZZ-00006, "Plant Cooldown Hot Standby to Cold Shutdown," steps to close the main steam isolation valves. The steps allowed these actions prematurely as the wording was, ". . . close prior to any reactor coolant system cold leg temperature decreasing below 300 degrees Fahrenheit." The effect of closing the main steam isolation valves with the condenser steam dumps in service was to stop the cooldown and resulted in an increasing reactor coolant system temperature. The balance of plant reactor operator then reopened main steam isolation Valve A. Thinking that the main steam isolation valve was not responding to the open signal the operator also opened atmospheric Steam Dump A. Shortly afterward, main steam isolation Valve A did open. This created a significant increase in steam flow from the steam generator that caused the steam generator level to swell up to the P-14 steam generator high level feedwater isolation setpoint. Without a main feedwater supply the steam generator levels all decreased to the steam generator narrow range low-low setpoint which resulted in a reactor trip signal and auxiliary feedwater actuation. The steam generator level control was subsequently restored and the plant shutdown continued by establishing a residual heat removal lineup.

Several crews had reviewed this evolution in just-in-time training. Similar problems with the main steam isolation valve closure steps on the simulator resulted in the undesired reactor coolant system heatup. This prompted feedback from another crew to the outage planning group to delay the main steam isolation valve closure step until the reactor coolant system was cooled down to exactly 300 degrees Fahrenheit. This feedback, although incorporated into the outage schedule, did not get communicated to the crew performing the step on October 11, 2008. A contributing cause of the inadequate procedural guidance and the inconsistent training was the recent change in the main steam isolation valve design. The main steam isolation valves rely on the steam pressure at the valves to act through ported lines to the valve disc to hold the valves open. With decreasing steam pressures the main steam isolation valves will drift closed just after reactor coolant system temperature drops to less than 300 degrees Fahrenheit. The main steam isolation valves had been installed in the previous refueling outage without sufficient procedure review or training on the impact of the design change during a plant shutdown.

<u>Analysis</u>. The performance deficiency associated with this finding involved the licensee's failure to ensure the procedures impacted by the main steam isolation valve design change was correct and clear. This finding was greater than minor because it was associated with the Initiating Events cornerstone attribute of procedural quality and it affected the objective to limit the likelihood of those events that upset plant stability and challenge critical safety functions during shutdown as well as power operations. Using Manual Chapter 0609.04, "Phase 1 - Initial Screening and Characterization of Findings," this finding is determined to be of very low safety significance since it did not affect the Technical Specification limit for reactor coolant system leakage or mitigation systems safety function, did not contribute to both the likelihood of a reactor trip and mitigation equipment or functions not being available, and did not increase the likelihood of a fire or internal/external flooding. This finding had a crosscutting aspect in the area of human performance associated with the decision making component because the licensee failed to communicate, in a timely manner, decisions to personnel who have a need to know the information in order to perform work safely [H.1(c)].

<u>Enforcement</u>. Technical Specification 5.4.1.a, "Procedures," required that written procedures be established and implemented covering 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 2J, required operating procedures for hot standby to cold shutdown. Procedure OTG-ZZ-00006 provided operator guidance for implementing plant cooldown from hot standby to cold shutdown. Contrary to the above, on October 11, 2008, Procedure OTG-ZZ-00006 was not adequate to ensure plant cooldown actions were performed in a controlled manner. Because of the very low safety significance of this finding and because the licensee has entered this issue into their corrective action program as CAR 200810293, this violation is being treated as a noncited violation in

accordance with Section VI.A.1 of the Enforcement Policy: NCV 05000483/2008005-04, "Failure to Maintain an Adequate Plant Shutdown Procedure."

2. Introduction. A self-revealing Green noncited violation of Technical Specification 5.4.1.a, “Procedures,” was identified after Callaway's control room operator’s improper restoration of the essential service water supply to the emergency diesel generator Train A lubricating oil cooler resulted in significant water flow into the emergency diesel room. The improper restoration was due to an altered, inadequate worker protection assurance clearance order.

   Description. On October 22, 2008, during a refueling outage, operating department efforts to return emergency diesel generator Train A to service for a maintenance run resulted in the essential service water system flowing water through an open drain valve into the diesel room. Two restoration evolutions associated with the essential service water and the emergency diesel generator systems had been proceeding in parallel. The reactor operator restoring the emergency diesel generator assumed the essential service water system was to remain isolated to the emergency diesel generator components and thus changed the already approved worker protection assurance Clearance 71899 to leave the oil cooler drain valve open with no tag. The original worker protection assurance had pre-established that the 2-inch drain valve was to be restored in the closed position. Callaway clearance Procedure ODP-ZZ-00310, "WPA and Caution Tagging," did require that an operating supervisor verify that restoration positions for all components were correct for the current plant conditions. However, the only valves isolating the open drain valve were designated as "restore to open" on worker protection assurance Clearance 71899.

   Several tagout process issues were revealed by this event:

   - There were over 30 designated administrators with electronic approval authority to allow an operating supervisor permission to approve tagouts. It was apparent that this was not well known as several administrators and operating supervisors that had left the company were still in the system.

   - Operating supervisors with approval authority were not required to have current formal license training to demonstrate integrated systems knowledge. Some were normally part of the work control or training department staffs.

   - The worker protection assurance restoration lineups were sometimes being electronically altered after the operating supervisor had "authorized" the clearance without his re-approval or knowledge of the change.

   - Worker protection assurance restoration approvers (operating supervisors) were not always actually reviewing the final prepared restoration lineups.

   - The quality and breadth of training on positioning and sequencing tagout restorations was inadequate.

   - Restoration of systems relied more heavily on the experience of the operating supervisor controlling the evolution pre-job brief than on the approval process.

   - The review that a restoration lineup was correct involved the preparing reactor operator and the approving operating supervisor.

Despite the incorrect tagout, another opportunity to prevent the flooding was the supervisor's pre-job brief with the operating field technicians who questioned the drain valves prescribed open position. However, the supervisor assumed the emergency diesel generator would not need cooling and thus informed the operating technicians that the prescribed positions were correct. A peer-check or additional review of worker protection assurance Clearance 71899 after being altered by the reactor operator may have discovered the inadequate restoration positions.

Starting the essential service water system and pump per Procedure OTN-EF-000001 "ESW Train A – Fill and Vent," pressurized the drain valve and produced significant water flow into the emergency diesel generator room. The water spray went undetected for about 30 minutes until a diesel vendor representative noticed it and informed the operations department. Floor drains in the room and an open door to the outside prevented the room from flooding.

Analysis. The performance deficiencies associated with this finding involved the licensee's failure to ensure the system restoration associated with the worker protection assurance clearance process was correct, programmatic control issues associated with the tagout process, an inadequate response to peer-checking, and an inadequate walkdown of the essential service water system components prior to pressurizing the system. This finding was greater than minor because, if left uncorrected the deficiencies could become a more significant safety concern. This finding affected the Mitigating Systems cornerstone. Using Manual Chapter 0609.04, "Phase 1 - Initial Screening and Characterization of Findings," this finding is determined to be of very low safety significance since this finding was not a design or qualification deficiency, did not represent a loss of system or train safety function and did not screen as potentially risk significant due to a flooding initiating event using the criteria on the characterization worksheet. This finding had a crosscutting aspect in the area of human performance associated with the work practices component because the licensee's pre-job briefing, self- and peer-checking, and proper documentation of the activity were inadequate to overcome worker protection assurance clearance process problems and an inexperienced operating supervisor. These less than adequate worker practices resulted in personnel proceeding in the face of uncertainty [H.4(a)].

Enforcement. Technical Specification 5.4.1.a, "Procedures," required that written procedures be established and implemented covering 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 1C, requires administrative procedures for equipment control is implemented correctly. Procedure ODP-ZZ-00310 provided operator guidance for implementing workman's protection assurance clearance of tagouts. Specifically the clearance of tagouts required the shift manager or designee to account for current plant conditions. Contrary to the above, on October 22, 2008, the shift manager designee did not adequately review worker protection assurance Clearance 71899 as required by Procedure ODP-ZZ-00310 to ensure restoration actions of the emergency service water to emergency diesel generator Train A accounted for the current plant conditions. Because of the very low safety significance of this finding and because the licensee has entered this issue into their corrective action program as CAR 200810902, this violation is being treated as a noncited violation in accordance with Section VI.A.1 of the Enforcement Policy: NCV 05000483/2008005-05, "Failure to Adequately Implement Plant Equipment Control Tagout Procedure."

.2 Other Outage Activities

a. Inspection Scope

The inspectors evaluated outage activities for two unscheduled outages that began on December 11, 2008 and continued through December 25, 2008. Two reactor trips were initiated due to electrical faults developed on the condensate Pumps B and C. The inspectors reviewed activities to ensure that the licensee considered risk in developing, planning, and implementing the outage schedule.

The inspectors observed or reviewed the reactor shutdown and cooldown, outage equipment configuration and risk management, electrical lineups, selected clearances, control and monitoring of decay heat removal, control of containment activities, startup and heatup activities, and identification and resolution of problems associated with the outage. Additionally, the inspectors reviewed the extent of condition and apparent cause for the electrical faults on the condensate Pumps B and C.

This inspection constitutes one other outage sample as defined in Inspection Procedure 71111.20-05

b. Findings

No findings of significance were identified.

1R22 Surveillance Testing (71111.22)

a. Inspection Scope

The inspectors reviewed the Final Safety Analysis Report, procedure requirements, and Technical Specifications to ensure that the nine surveillance activities listed below demonstrated that the systems, structures, and/or components tested were capable of performing their intended safety functions. The inspectors either witnessed or reviewed test data to verify that the significant surveillance test attributes were adequate to address the following:

- Preconditioning
- Evaluation of testing impact on the plant
- Acceptance criteria
- Test equipment
- Procedures
- Jumper/lifted lead controls
- Test data
- Testing frequency and method demonstrated Technical Specification operability
- Test equipment removal

- Restoration of plant systems
- Fulfillment of ASME Code requirements
- Updating of performance indicator data
- Engineering evaluations, root causes, and bases for returning tested systems, structures, and components not meeting the test acceptance criteria were correct
- Reference setting data
- Annunciator and alarm setpoints.

The inspectors also verified that licensee personnel identified and implemented any needed corrective actions associated with the surveillance testing.

- October 2, 2008, Job 07509412, Procedure OSP-NE-0024B, Standby Diesel Generator A, 24-hour run and hot restart
- October 11, 2008, Job 07504910, Procedure OSP-SA-2413A, Engineered safety features Train A actuation signal testing
- October 17, 2008, Job 07504731, Procedure OSP-EJ-PV04A/B, Residual heat removal Train A and reactor coolant system check valve in-service test
- October 27, 2008, Jobs 08511230 and 05516932, Procedures OSP-GT-00003 and ISL-SE-00N32, Verifying requirements to enter Mode 6
- November 2, 2008, Job 07506014, Procedure ESP-EF-0002A, Essential service water Train A flow verification test
- November 4, 2008, Jobs 08510067 and 08510068, Procedures OSP-AL-PV04B and OSP-AL-PV04A, Motor-driven auxiliary feedwater pump comprehensive pump and check valve in-service test
- November 6, 2008, Procedure OSP-SF-00005, Revision 17, Engineering estimated critical position calculation
- December 9, 2008, Cumulative review of containment leak rate testing during Refueling Outage 16 and testing specific to Valves EVHV0048 and EVHV0050
- December 10, 2008, Reactor coolant system leak rate surveillance

Specific documents reviewed during this inspection are listed in the attachment.

These activities constitute completion of six routine, two in-service test, and one reactor coolant system surveillance inspection samples as defined in Inspection Procedure 71111.22-05.

b. Findings

No findings of significance were identified.

1EP6 Drill Evaluation (71114.06)

Training Observations

a. Inspection Scope

The inspectors observed a simulator training evolution licensed operators on September 25, 2008 and December 10, 2008, which required emergency plan implementation by the operations crew. This evolution was planned to be evaluated and included in performance indicator data regarding drill and exercise performance. The inspectors observed event classification and notification activities performed by the crew. The inspectors also attended the post-evolution critique for the scenario. The focus of the inspectors' activities was to note any weaknesses and deficiencies in the crew's performance and ensure that the licensee evaluators noted the same issues and entered them into the corrective action program. As part of the inspection, the inspectors reviewed the scenario package and other documents listed in the attachment.

These activities constitute completion of two samples as defined in Inspection Plan 71114.06-05.

b. Findings

No findings of significance were identified.

2. RADIATION SAFETY

Cornerstone: Occupational Radiation Safety

2OS1 Access Control to Radiologically Significant Areas (71121.01)

a. Inspection Scope

This area was inspected to assess licensee personnel's performance in implementing physical and administrative controls for airborne radioactivity areas, radiation areas, high radiation areas, and worker adherence to these controls. The inspectors used the requirements in 10 CFR Part 20, the Technical Specifications, and the licensee's procedures required by Technical Specifications as criteria for determining compliance. During the inspection, the inspectors interviewed the radiation protection manager, radiation protection supervisors, and radiation workers. The inspectors performed independent radiation dose rate measurements and reviewed the following items:

- Performance indicator events and associated documentation packages reported by the licensee in the Occupational Radiation Safety Cornerstone
- Controls (surveys, posting, and barricades) of radiation, high radiation, or airborne radioactivity areas
- Radiation work permits, procedures, engineering controls, and air sampler locations

- Conformity of electronic personal dosimeter alarm set points with survey indications and plant policy; workers' knowledge of required actions when their electronic personnel dosimeter noticeably malfunctions or alarms
- Barrier integrity and performance of engineering controls in airborne radioactivity areas
- Adequacy of the licensee's internal dose assessment for any actual internal exposure greater than 50 millirem committed effective dose equivalent
- Physical and programmatic controls for highly activated or contaminated materials (non-fuel) stored within spent fuel and other storage pools
- Self-assessments, audits, licensee event reports, and special reports related to the access control program since the last inspection
- Corrective action documents related to access controls
- Licensee actions in cases of repetitive deficiencies or significant individual deficiencies
- Radiation work permit briefings and worker instructions
- Adequacy of radiological controls, such as required surveys, radiation protection job coverage, and contamination control during job performance
- Dosimetry placement in high radiation work areas with significant dose rate gradients
- Changes in licensee procedural controls of high dose rate - high radiation areas and very high radiation areas
- Controls for special areas that have the potential to become very high radiation areas during certain plant operations
- Posting and locking of entrances to all accessible high dose rate - high radiation areas and very high radiation areas
- Radiation worker and radiation protection technician performance with respect to radiation protection work requirements

Specific documents reviewed during this inspection are listed in the attachment.

These activities constitute completion of the required 21 samples as defined in Inspection Procedure 71121.01-05.

b. Findings

Introduction. The inspectors reviewed a self-revealing, noncited violation of Technical Specification 5.7.1 which resulted from a failure by three individuals to comply with high radiation area entry requirements. The violation had very low safety significance.

Description. On October 20, 2008, three engineers were touring the reactor building to familiarize themselves with some of the major pieces of equipment. The engineers did not plan to go inside the bioshield, did not sign in on a radiation work permit which allowed entry into a high radiation area, and did not receive a briefing on dose rates in high radiation areas. The three engineers met a fourth engineer who volunteered to guide them. The fourth engineer was signed on to a radiation work permit which allowed entry into high radiation areas (RWP 890401HRA). The fourth engineer asked what radiation work permit the others were using (RWP 890401NONHRA), but apparently did not understand it did not allow entry into high radiation areas. Subsequently, all four individuals entered the bioshield, a high radiation area. In approximately one minute, one of the engineers received an electronic dosimeter dose rate alarm when dose rates in the area exceeded the 50 millirem per hour setpoint. Dose rates in the immediate area were as high as 150 millirem per hour. The individuals left the area immediately and reported the alarm to radiation protection personnel. The licensee established an event review team and interviewed the workers involved to determine the facts described above. The licensee determined the apparent cause of the event was the failure to perform a self- and peer-check.

Analysis. Failure to comply with high radiation area entry requirements is a performance deficiency. This finding is greater than minor because it is associated with the cornerstone attribute of exposure control and affected the cornerstone objective, in that, failure to meet high radiation area entry requirements increases the potential for increased dose. This finding involved an individual worker's unplanned, unintended dose or potential of such a dose (resulting from actions or conditions contrary to Technical Specifications) which could have been significantly greater as a result of a single minor, reasonable alteration of the circumstances. Using the Occupational Radiation Safety Significance Determination Process, the inspectors determined the finding to have very low safety significance because (1) it was not associated with ALARA planning or work controls, (2) there was no overexposure, (3) there was no substantial potential for an overexposure, and (4) the ability to assess dose was not compromised. Additionally, the finding had a crosscutting aspect in the area of human performance, work practices component, because the workers failed to use error prevention tools such as self- and peer-checking [H.4(a)].

Enforcement. According to the definitions in 10 CFR 20.1003, high radiation area means an area, accessible to individuals, in which radiation levels could result in an individual receiving a dose equivalent in excess of 0.1 rem (1 mSv) in one hour at 30 centimeters from the radiation source or from any surface that the radiation penetrates. Technical Specification 5.7.1.b requires access to, and activities in, each such area shall be controlled by means of a radiation work permit that includes specification of radiation dose rates in the immediate work area and other appropriate radiation protection equipment and measures. Technical Specification 5.7.1.e requires entry into such areas shall be made only after dose rates in the area have been determined and entry personnel are knowledgeable of them. Contrary to the above, on October 20, 2008, three engineers entered the high radiation area within the bioshield and failed to sign onto a radiation work permit that specified radiation dose rates in the immediate work area and were not knowledgeable of the dose rates in the area. Because this failure to meet high radiation area entry requirements is of very low safety significance and has been entered into the licensee’s corrective action program as CAR 200810771, this violation is being treated as a noncited violation, consistent with Section VI.A of the NRC

Enforcement Policy: NCV 05000483/2008005-06, Failure to Comply with High Radiation Area Entry Requirements.

2OS2 ALARA Planning and Controls (71121.02)

a. Inspection Scope

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

- Current 3-year rolling average collective exposure
- Five (to ten) outage or on-line maintenance 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
- Three (to five) work activities of highest exposure significance completed during the last outage
- ALARA work activity evaluations, exposure estimates, and exposure mitigation requirements
- Integration of ALARA requirements into work procedure and radiation work permit (or radiation exposure permit) documents
- Shielding requests and dose/benefit analyses
- Dose rate reduction activities in work planning
- Postjob (work activity) reviews
- 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 re-planning work, when unexpected changes in scope or emergent work were encountered
- Exposure tracking system
- Use of engineering controls to achieve dose reductions and dose reduction benefits afforded by shielding
- Workers' use of the low dose waiting areas

- First-line job supervisors' contribution to ensuring work activities are conducted in a dose efficient manner
- 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
- Source-term control strategy or justifications for not pursuing such exposure reduction initiatives
- Specific sources identified by the licensee for exposure reduction actions, priorities established for these actions, and results achieved since the last refueling cycle
- Radiation worker and radiation protection technician performance during work activities in radiation areas, airborne radioactivity areas, or high radiation areas
- Resolution through the corrective action process of problems identified through postjob reviews and postoutage ALARA report critiques
- 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

Specific documents reviewed during this inspection are listed in the attachment.

These activities constitute completion of 11 of the required 15 samples and 10 of the optional samples as defined in Inspection Procedure 71121.02.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA1 Performance Indicator Verification (71151)

.1 Data Submission Issue

a. Inspection Scope

The inspectors performed a review of the data submitted by the licensee for the third quarter 2008 performance indicators for any obvious inconsistencies prior to its public release in accordance with IMC 0608, "Performance Indicator Program."

This review was performed as part of the inspectors' normal plant status activities and, as such, did not constitute a separate inspection sample.

b. Findings

No findings of significance were identified.

.2 Mitigating Systems Performance Index - Heat Removal System

a. Inspection Scope

The inspectors sampled licensee submittals for the Mitigating Systems Performance Index - Heat Removal System performance indicator for the period from the third quarter 2007 through the third quarter 2008. To determine the accuracy of the performance indicator data reported during those periods, performance indicator definitions and guidance contained in NEI Document 99-02, “Regulatory Assessment Performance Indicator Guideline,” Revision 5, was used. The inspectors reviewed the licensee’s operator narrative logs, issue reports, event reports, mitigating systems performance index derivation reports, and NRC integrated inspection reports for the period of July 2007 through September 2008 to validate the accuracy of the submittals. The inspectors reviewed the mitigating systems performance index component risk coefficient to determine if it had changed by more than 25 percent in value since the previous inspection, and if so, that the change was in accordance with applicable NEI guidance. The inspectors also reviewed the licensee’s issue report database to determine if any problems had been identified with the performance indicator data collected or transmitted for this indicator and none were identified. Specific documents reviewed are described in the attachment to this report.

These activities constitute completion of one mitigating systems performance index - heat removal system sample as defined in Inspection Procedure 71151-05.

b. Findings

No findings of significance were identified.

.3 Mitigating Systems Performance Index - Residual Heat Removal System

a. Inspection Scope

The inspectors sampled licensee submittals for the Mitigating Systems Performance Index - Residual Heat Removal System performance indicator for the period from the third quarter 2007 through the third quarter 2008. To determine the accuracy of the performance indicator data reported during those periods, performance indicator definitions and guidance contained in NEI Document 99-02, “Regulatory Assessment Performance Indicator Guideline,” Revision 5, was used. The inspectors reviewed the licensee’s operator narrative logs, issue reports, mitigating systems performance index derivation reports, event reports and NRC integrated inspection reports for the period of July 2007 through September 2008 to validate the accuracy of the submittals. The inspectors reviewed the mitigating systems performance index component risk coefficient to determine if it had changed by more than 25 percent in value since the previous inspection, and if so, that the change was in accordance with applicable NEI guidance. The inspectors also reviewed the licensee’s issue report database to determine if any problems had been identified with the performance indicator data collected or transmitted for this indicator and none were identified. Specific documents reviewed are described in the attachment to this report.

These activities constitute completion of one mitigating systems performance index - residual heat removal system sample as defined in Inspection Procedure 71151-05.

b. Findings

No findings of significance were identified.

.4 Reactor Coolant System Leakage

a. Inspection Scope

The inspectors sampled licensee submittals for the Reactor Coolant System Leakage performance indicator for the period from the third quarter 2007 through the third quarter 2008. To determine the accuracy of the performance indicator data reported during those periods, performance indicator definitions and guidance contained in NEI Document 99-02, "Regulatory Assessment Performance Indicator Guideline," Revision 5, was used. The inspectors reviewed the licensee's operator logs, reactor coolant system leakage tracking data, issue reports, event reports and NRC integrated inspection reports for the period of July 2007 through September 2008 to validate the accuracy of the submittals. The inspectors also reviewed the licensee's issue report database to determine if any problems had been identified with the performance indicator data collected or transmitted for this indicator and none were identified. Specific documents reviewed are described in the attachment to this report.

These activities constitute completion of one reactor coolant system leakage sample as defined in Inspection Procedure 71151-05.

b. Findings

No findings of significance were identified.

.5 Unplanned Transients per 7000 Critical Hours

a. Inspection Scope

The inspectors sampled licensee submittals for the unplanned transients per 7000 critical hour's performance indicator for the period from the third quarter 2007 through the second quarter 2008. To determine the accuracy of the performance indicator data reported during those periods, performance indicator definitions and guidance contained in Revision 5 of the NEI Document 99-02, "Regulatory Assessment Performance Indicator Guideline," were used. The inspectors reviewed the licensee's operator narrative logs, issue reports, event reports and NRC Inspection reports for the period of July 2007 to June 2008 to validate the accuracy of the submittals. The inspectors also reviewed the licensee's issue report database to determine if any problems had been identified with the performance indicator data collected or transmitted for this indicator and none were identified. Specific documents reviewed are described in the appendix to this report.

These activities constitute completion of one sample of unplanned transients per 7000 critical hours sample as defined by Inspection Procedure 71151-05.

b. Findings

No findings of significance were identified.

.6 Occupational Radiological Occurrences

a. Inspection Scope

The inspectors sampled licensee submittals for the Occupational Radiological Occurrences performance indicator for the period from April 1, to September 30, 2008. To determine the accuracy of the performance indicator data reported during those periods, performance indicator definitions and guidance contained in NEI Document 99-02, "Regulatory Assessment Performance Indicator Guideline," Revision 5, was used. The inspectors reviewed the licensee's assessment of the performance indicator for occupational radiation safety to determine if indicator related data was adequately assessed and reported. To assess the adequacy of the licensee's performance indicator data collection and analyses, the inspectors discussed with radiation protection staff, the scope and breadth of its data review, and the results of those reviews. The inspectors independently reviewed electronic dosimetry dose rate and accumulated dose alarm and dose reports and the dose assignments for any intakes that occurred during the time period reviewed to determine if there were potentially unrecognized occurrences. The inspectors also conducted walkdowns of numerous locked high and very high radiation area entrances to determine the adequacy of the controls in place for these areas.

These activities constitute completion of the occupational radiological occurrences sample as defined in Inspection Procedure 71151-05.

b. Findings

No findings of significance were identified.

.7 Radiological Effluent Technical Specifications/Offsite Dose Calculation Manual Radiological Effluent Occurrences

a. Inspection Scope

The inspectors sampled licensee submittals for the Radiological Effluent Technical Specifications/Offsite Dose Calculation Manual Radiological Effluent Occurrences performance indicator for the period from April 1, to September 30, 2008. To determine the accuracy of the performance indicator data reported during those periods, performance indicator definitions and guidance contained in NEI Document 99-02, "Regulatory Assessment Performance Indicator Guideline," Revision 5, was used. The inspectors reviewed the licensee's issue report database and selected individual reports generated since this indicator was last reviewed to identify any potential occurrences such as unmonitored, uncontrolled, or improperly calculated effluent releases that may have impacted offsite dose. The inspectors reviewed gaseous effluent summary data and the results of associated offsite dose calculations for selected dates between April 1, and September 30, 2008 to determine if indicator results were accurately reported. The inspectors also reviewed the licensee's methods for quantifying gaseous and liquid effluents and determining effluent dose. Additionally, the inspectors reviewed the licensee's historical 10 CFR 50.75(g) file and selectively reviewed the licensee's analysis for discharge pathways resulting from a spill, leak, or unexpected liquid discharge focusing on those incidents which occurred over the last few years.

These activities constitute completion of the radiological effluent Technical Specifications/offsite dose calculation manual radiological effluent occurrences sample as defined in Inspection Procedure 71151-05.

b. Findings

No findings of significance were identified.

4OA2 Identification and Resolution of Problems (71152)

Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity, Emergency Preparedness, Public Radiation Safety, Occupational Radiation Safety, and Physical Protection

.1 Routine Review of Identification and Resolution of Problems

a. Inspection Scope

As part of the various baseline inspection procedures discussed in previous sections of this report, the inspectors routinely reviewed issues during baseline inspection activities and plant status reviews to verify that they were being entered into the licensee's corrective action program at an appropriate threshold, that adequate attention was being given to timely corrective actions, and that adverse trends were identified and addressed. The inspectors reviewed attributes that included: the complete and accurate identification of the problem; the timely correction, commensurate with the safety significance; the evaluation and disposition of performance issues, generic implications, common causes, contributing factors, root causes, extent of condition reviews, and previous occurrences reviews; and the classification, prioritization, focus, and timeliness of corrective actions Minor issues entered into the licensee's corrective action program because of the inspectors' observations are included in the attached list of documents reviewed.

These routine reviews for the identification and resolution of problems did not constitute any additional inspection samples. Instead, by procedure, they were considered an integral part of the inspections performed during the quarter and documented in Section 1 of this report.

b. Findings

No findings of significance were identified.

.2 Daily Corrective Action Program Reviews

a. Inspection Scope

In order to assist with the identification of repetitive equipment failures and specific human performance issues for follow-up, the inspectors performed a daily screening of items entered into the licensee's corrective action program. The inspectors accomplished this through review of the station's daily corrective action documents.

The inspectors performed these daily reviews as part of their daily plant status monitoring activities and, as such, did not constitute any separate inspection samples.

b. Findings

No findings of significance were identified.

.3 Semi-Annual Trend Review

a. Inspection Scope

The inspectors performed a review of the Callaway Plant corrective action program and associated documents to identify trends that could indicate the existence of a more significant safety issue. The inspectors focused their review on repetitive equipment issues, but also considered the results of daily corrective action item screening discussed in Section 4OA2.2, above, licensee trending efforts, and licensee human performance results. The inspectors nominally considered the 6-month period of June 2008, through November 2008, although some examples expanded beyond those dates where the scope of the trend warranted.

The inspectors also included issues documented outside the normal corrective action program in major equipment problem lists, repetitive and/or rework maintenance lists, departmental problem/challenges lists, system health reports, quality assurance audit/surveillance reports, self-assessment reports, and Maintenance Rule assessments. The inspectors compared and contrasted their results with the results contained in the licensee's corrective action program trending reports. Corrective actions associated with a sample of the issues identified in the licensee's trending reports were reviewed for adequacy.

Licensee-identified Trends

The inspectors' review of the licensee quarterly trend reports noted and agreed with the conclusion that "Improvements Associated with Plant Status Control Performance," was needed.

Inspector-identified Trends

The inspectors review provided a more in-depth focused look at an increased negative trend of programmatic issues relating to the control of boric acid leakage. The following leakage related items were four significant inputs to the conclusion of a negative trend.

- Inspection finding NCV 05000483/2008004-01, Failure to implement boric acid corrosion control procedures
- CAR 200809886, Leak identified on pressurizer auxiliary spray Line BG-026-BCB-2"
- CAR 200810705, Untimely correction of boric acid leaks
- CAR 200810295, Reactor head canopy seal weld Number 24 leakage

The pressurizer auxiliary spray line boric acid leakage, which had likely existed for 18 months, was a large enough volume to require significant cleanup efforts on several levels below and inside the bioshield. The licensee had not identified this as a negative trend in the second or third quarterly trend reports but had listed the boric acid control program as an open area for improvement.

These activities constitute completion of one semi-annual trend inspection sample as defined in Inspection Procedure 71152-05.

b. Findings

No findings of significance were identified.

.4 Selected Issue Follow-up Inspection

a. Inspection Scope

During a review of items entered in the licensee's corrective action program, the inspectors focused on corrective actions associated with:

- The cause of a number of licensed operator entries into Technical Specification 3.4.1, "RCS Pressure, Temperature, and Flow Departure from Nucleate boiling (DNB) Limits," during the last operating cycle. Seventeen departure from nucleate boiling entries occurred from November 2005 to November 2008.
- November 28, 2008, the cumulative effects of operator workarounds.

These activities constitute completion of two in-depth problem identification and resolution samples, one of which was an operator workaround, as defined in Inspection Procedure 71152-05.

b. Findings and Observations

Callaway Plant operations department does not document operator actions taken to compensate for degraded or non-conforming conditions that complicate operation of plant equipment as 'operator workarounds.' Instead the AmerenUE staff only considers mitigating system workarounds that impact the operators' ability to implement abnormal and emergency operating procedures.

The inspectors noted on November 28, 2008, that back leakage from the safety injection Accumulator D fill line into both the safety injection pump discharge lines had required action to manually isolate the fill line air operated Valve EMHV 8878D. This is considered a degraded condition of a mitigating system. Isolating this fill line periodically requires containment entries to maintain the safety injection Accumulator D full as required by Technical Specification 3.5.1. NRC Inspection Procedure 71152, "Identification and Resolution of Problems," would consider this example as an operator workaround because it:

- Requires a change from longstanding practices
- Requires operation of a component in a manner dissimilar from other similar mitigating systems components
- Requires actions under potentially adverse environmental conditions

4OA3 Event Follow-up (71153)

.1 Unplanned Lifting of Letdown System Relief Valve BG8117

a. Inspection Scope

The inspectors, present in the control room during the plant shutdown on October 12, 2008, reviewed the cause, impact, and corrective actions associated with the unplanned lifting of letdown system relief Valve BG8117.

b. Findings

No findings of significance were identified.

.2 Inadvertent Loss of Spent Fuel Pool Inventory

a. Inspection Scope

The inspectors reviewed the unplanned transfer of approximately 2000 gallons of water which were inadvertently transferred from the spent fuel pool to the refueling water storage tank on November 7, 2008.

b. Findings

Introduction. A Green self-revealing noncited violation of Technical Specification 5.4.1.a was identified for the failure to close Valve BNV0002 during a fill of the spent fuel pool resulting in approximately 2000 gallons of water being inadvertently transferred from the spent fuel pool to the refueling water storage tank.

Description. On November 7, 2008, the licensee initiated Procedure OTN-EC-00001 to add makeup water to the spent fuel pool. Prior to performing the evolution, the operations crew was briefed that the refueling water storage tank was on recirculation through the fuel pool cleanup system and that this alignment needed to be secured prior to performing a fill of the spent fuel pool. Following termination of the refueling water storage tank recirculation lineup and approximately 5 minutes after opening Valve ECV0128 to initiate a fill of the spent fuel pool, the control room received annunciator "RWST Lev HILO." A review of plant computer data revealed that refueling water storage tank level had increased approximately 0.5 percent and spent fuel pool level had decreased approximately 2 inches. After recognizing that an inadvertent transfer of spent fuel pool water to the refueling water storage tank was in progress, the control room directed that Valves ECV0076 and BNV0002 be closed to terminate the transfer. It was later discovered that operations personnel did not adequately communicate the status of Valve BNV0002 resulting in the refueling water storage tank remaining on recirculation during the fill operation and approximately 2000 gallons of spent fuel pool water being transferred to the refueling water storage tank.

Analysis. The performance deficiency associated with this finding involved the failure to secure refueling water storage tank recirculation prior to filling the spent fuel pool. Specifically, the licensee failed to shut Valve BNV0002 prior to initiating a fill of the spent fuel pool. This finding is greater than minor because it was associated with the human performance attribute of the Barrier Integrity cornerstone and affects the associated cornerstone objective to provide reasonable assurance that physical design barriers protect the public from radionuclide releases caused by accidents or releases. Using Manual Chapter 0609.04, "Phase 1 - Initial Screening and Characterization of Findings," this finding was determined to be of very low safety significance because it only represents a degradation of the radiological barrier function provided by the spent fuel pool. This finding had a crosscutting aspect in the area of human performance associated with the work control component because the licensee failed to effectively communicate work status to the control room [H.3(b)].

Enforcement. Technical Specification 5.4.1.a, "Procedures," requires 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," February 1978. Appendix A, Item 3.h, required procedures for spent fuel pool cooling system operation. Procedure OTN-EC-00001, Addendum 6, "Filling the Spent Fuel Pool," Revision 3, Step 5.2.5.a, required the operator to verify the refueling water storage tank was not on recirculation prior to opening Valve ECV0128. Contrary to the above, on November 7, 2008, an operator failed to ensure Valve BNV0002 was shut prior to opening Valve ECV0128 resulting in approximately 2000 gallons of spent fuel pool water being transferred to the refueling water storage tank. Because this finding is of very low safety significance and was entered into the licensee's corrective action program as Callaway Action Request 200811692, this violation is being treated as a noncited violation, consistent with Section VI.A.1 of the NRC Enforcement Policy: NCV 05000483/2008005-07, "Failure to Terminate Refueling Water Storage Tank Recirculation Results in Inadvertent Loss of Spent Fuel Pool Inventory."

.3 (Closed) Licensee Event Report (LER) 05000483/2008002-00. Void Found in Line EM-023-HCB – Residual Heat Removal Pump A to Safety Injection Pumps

On May 21, 2008, Callaway Plant personnel discovered a 6.6 cubic foot void of air within safety injection system common suction piping Line EM023-HCB – 6". The volume of air exceeded the allowable void fraction of 2.1 cubic feet required for operability. This voided piping, determined to have existed for over a year, was caused by relief valve maintenance on Valve EM8858A performed on May 7, 2007. The maintenance restoration failed to perform a fill and vent to ensure the suction pipe was full of water. The void was removed by venting the piping on May 21, 2008.

The inspectors reviewed the licensee's actions to address the degraded condition including identification of root and contributing causes and development and implementation of corrective actions. The inspectors determined that the licensee failed to restore compliance within a reasonable time to prevent void formation in the emergency core cooling system. The inspector's finding is documented in VIO 05000483/2008003-05, "Failure to Prevent Recurrence of Voids in Emergency Core Cooling System Cold Leg Recirculation Piping." Additionally, the inspectors identified an inadequate surveillance procedure that resulted in the licensee failing to maintain the emergency core cooling system full of water as required per Technical Specification Surveillance Requirement 3.5.2.3 which is documented in NCV 05000483/2008003-02, "Inadequate Surveillance Procedure Resulted in an Inoperable Emergency Core Cooling System."

The inspectors reviewed the licensee's response to VIO 05000483/2008003-05 which references EA-08-190. The inspectors found that the licensee's response adequately addressed the cause of the violation and corrective steps taken to restore compliance and prevent future violations. VIO 05000483/2008003-05 is closed.

Callaway Plant licensing staff performed a reportability evaluation and determined that the discovery of the void was not required to be reported to the NRC based on reasonable engineering judgment that the emergency core cooling system was still capable of performing its required safety function. The inspectors reviewed the licensee's reportability evaluation and determined that the licensee failed to consider the requirements for Technical Specification LCO 3.5.2, "ECCS – Operating," which requires two trains consisting of centrifugal charging subsystem, a safety injection subsystem, and a residual heat removal subsystem be operable. Since the void discovered in Line EM023-HCB – 6" had the ability to affect both trains of the centrifugal charging or safety injection subsystems simultaneously, the inspectors determined that the licensee failed to meet Technical Specification requirements for the emergency core cooling system and that the system was inoperable from May 7, 2007, until May 21, 2008. Consequently, the event resulted in a reportable event per the requirements of 10 CFR 50.73(a)(2)(i)(B), any operation or condition which was prohibited by the plant's Technical Specifications, and 10 CFR 50.73(a)(2)(vii), any event where a single cause or condition caused two independent trains or channels to become inoperable in a single system. Since the licensee failed to submit a required licensee event report within 60 days after discovery of an event requiring a report, the inspectors identified a Severity Level IV noncited violation of 10 CFR 50.73(a)(1) which is documented as NCV 05000483/2008004-02, "Failure to Submit a Licensee Event Report for a Condition Prohibited by the Plant's Technical Specifications."

The licensee submitted a Licensee Event Report for the void found in line EM-023-HCB - 6"on December 23, 2008. The inspectors reviewed the licensee's submittal and determined that the report adequately documented the summary of the event including the potential safety consequences, causes of the event and corrective actions required to address the performance deficiency. No additional findings were identified. This LER is closed.

.4 <u>(Closed) LER 05000483/2008003-00, Inadvertent P-14 Feedwater Isolation Signal Actuation Followed by a Reactor Trip Actuation Due to Steam Generator Low-Low Water Narrow Range Trip</u>

On October 11, 2008, during a plant shutdown for refueling, plant operators secured all main steam isolation valves and steam removal paths and then, a few minutes later, reestablished a steam flowpath through MSIV A and atmospheric Steam Dump A to control reactor coolant system temperature. This action created a significant increase in steam flow from Steam Generator A that caused the steam generator level to swell up to the P-14 high level feedwater isolation setpoint. Without a main feedwater supply the steam generator levels all decreased to the steam generator narrow range low-low setpoint which resulted in a reactor trip signal and auxiliary feedwater actuation. Actuation of the reactor protection system and auxiliary feedwater system were conditions reportable by 10 CFR 50.73(a)(2)(iv)(A). The licensee determined the cause of the actuations was that the shutdown procedure was inadequate in that it did not ensure another reactor coolant system heat sink existed prior to securing the main steam isolation valves and that the reactor operator took action outside the procedure without using human performance tools to verify reopening main steam isolation valves A was needed. Corrective actions included a revision to the specific plant operating procedures to maintain a reactor coolant system heat sink available during the cooldown evolution and improvements in operator license continued training. This self-revealing finding involved a violation of Technical Specification 5.4.1, "Procedures." The enforcement aspects of the violation are discussed in Section 1R20 of this report as NCV 05000483/2008005-04, "Failure to Maintain an Adequate Plant Shutdown Procedure." This LER is closed

.5 (Closed) LER 05000483/2008004-00, Failure to Maintain Containment Purge and Exhaust System In-service During Core Alterations with the Equipment Hatch Open

On October 17, 2008, the licensee identified that the Refueling Outage 16 core offload was recommenced with the containment equipment hatch open and the containment purge and exhaust system not in service. This was a condition prohibited by Technical Specification 3.9.4, "Containment Penetrations." The licensee determined the cause of this prohibited condition to be a failure to adequately implement Callaway Operating License Amendment 152 into plant procedures. Contributing causes were that the shift manager assumed that the administrative controls required by Technical Specification 3.9.4 were limited to closure of the equipment hatch and that the site procedural controls were adequate. Also the operating staff had a mindset to not consult the Technical Specification Bases unless the Technical Specification was viewed as unclear. Corrective actions included a revision to the specific plant operating procedures to address the controls for having the equipment hatch open and initiating operator training improvements regarding the containment equipment hatch. This licensee-identified finding involved a violation of Technical Specification 3.9.4, "Containment Penetrations." The enforcement aspects of the violation are discussed in Section 4OA7 of this report. This LER is closed.

.6 (Closed) LER 05000483/2008005-00, Reactor Manually Tripped Due to Main Feed Pump B Tripping on Low Lube Oil Pressure

a. Inspection Scope

On November 11, 2008, the licensee experienced a trip of the main feedwater Pump B turbine on low lube oil pressure. Since the plant was at greater than 80 percent power, the reactor was manually tripped per plant operating procedures. The inspectors responded to the plant and discussed the reactor trip with operations, engineering, and licensee management personnel to gain an understanding of the event and assess follow-up actions. The inspectors reviewed operator actions taken in accordance with licensee procedures and reviewed unit and system indications to verify that actions and system responses were as expected. The inspectors discussed the reactor trip with the licensee’s root cause analysis team and assessed the team’s actions to gather, review, and assess information leading up to and following the reactor trip. The inspectors reviewed the initial investigation report to assess the detail of review and adequacy of the root cause and proposed corrective actions prior to unit restart. The licensee’s investigation identified that the cause of the main feedwater pump trip was a low lube oil pressure condition that resulted from use of improper o-ring material. The inspectors also reviewed the initial licensee notification, EN44652, to verify that it met the requirements specified in NUREG-1022, "Event Reporting Guidelines." The inspectors reviewed this LER and documented the performance deficiency below. This LER is closed.

b. Findings

Introduction. A Green self-revealing finding was identified for the failure of engineering personnel to perform a material equivalencies evaluation to ensure replacement components do not adversely affect plant operations in accordance with licensee Procedure EDP-ZZ-04015, "Evaluating and Processing Requests for Resolution."

Description. On November 11, 2008, with the plant at 97 percent reactor power, Callaway Plant experienced a trip of main feedwater Pump B due to low lube oil pressure. Since the reactor was at greater than 80 percent power, the plant operators inserted a manual reactor trip in accordance with Procedure OTO-AE-00001, "Feedwater System Malfunction." Following the reactor trip, the licensee maintenance personnel discovered two pieces of o-ring foreign material within main feedwater Pump B bearing oil supply pressure regulating Valve FCV0970. The foreign material was found wrapped around the regulating spring which inhibited valve movement and caused the lube oil low pressure condition.

A review by the licensee determined that the o-ring foreign material originated from the upstream duplex basket Strainer FCBS0096. When the east side of the strainer was examined, it was noted that a new o-ring, installed in Refueling Outage 16, in November 2008, had expanded and fallen into the lube oil system when the basket was removed from the housing. A similar situation occurred when the west side strainer was examined. The o-rings were made of an ethylene propylene diene M-class (EPDM) rubber material which is incompatible with petroleum systems. This allowed the o-rings to become pliable when exposed to lube oil and prevented a secure fit within the basket strainer housing. The licensee determined that operational experience existed documenting a previous occurrence of the pliable EPDM type o-rings falling into the feedwater lube oil system when disassembling the strainer. The foreign material was then transported to Valve FCV0970, causing a trip of main feedwater Pump B and subsequent manual reactor trip.

The EPDM o-rings had been approved as an equivalent replacement for the vendor recommended Buna-N type o-rings. Buna-N material is approved for use in petroleum based systems. The use of EPDM o-rings was implemented in July 1999 without performing an engineering material equivalency evaluation to determine if the o-rings were compatible in the main feedwater pump lube oil system. Following discovery of the material incompatibility, the licensee removed the EPDM o-rings from service and replaced them with Buna-N type o-rings.

Analysis. The performance deficiency associated with this finding was the failure of the licensee to perform a material equivalency evaluation to ensure o-rings associated with the main feedwater pump lube oil basket strainers were compatible with petroleum based systems. This finding is greater than minor because it is associated with the design control attribute of the Initiating Events cornerstone and affects the cornerstone objective to limit the likelihood of those events that upset plant stability and challenge critical safety functions during shutdown and power operations. Using Manual Chapter 0609.04, "Phase 1 – Initial Screening and Characterization of Findings," the finding is determined to be potentially risk significant because it contributed to both the likelihood of a reactor trip and the likelihood that mitigation equipment or functions will not be available. When evaluated per Manual Chapter 0609 Appendix A, " Determining the Significance of Reactor Inspection Findings for At-Power Situations," and the

Callaway Plant Phase 2 pre-solved table item “Failure to Reestablish Main Feedwater,” the inspectors determined this finding to be of very low safety significance. This finding was determined to not have a crosscutting aspect because the performance deficiency is not indicative of current licensee performance.

Enforcement. Enforcement action does not apply because the performance deficiency did not involve a violation of regulatory requirements. The finding is of very low safety significance and the issue was entered into the licensee's corrective action program as CAR 200811781: FIN 05000483/2008005-08, "Failure to Evaluate Material Equivalencies Leads to a Manual Reactor Trip."

4OA5 Other Activities

.1 Quarterly Resident Inspector Observations of Security Personnel and Activities

a. Inspection Scope

During the inspection period, the inspectors performed observations of security force personnel and activities to ensure that the activities were consistent with Callaway Plant security procedures and regulatory requirements relating to nuclear plant security. These observations took place during both normal and off-normal plant working hours.

These quarterly resident inspector observations of security force personnel and activities did not constitute any additional inspection samples. Rather, they were considered an integral part of the inspectors’ normal plant status review and inspection activities.

b. Findings

No findings of significance were identified.

.2 Temporary Instruction 2515-172, “Reactor Coolant System Dissimilar Metal Butt Welds”

Temporary Instruction TI2515/172, “Reactor Coolant System Dissimilar Metal Butt Welds,” was performed at Callaway Plant during Refueling Outage 16 in October 2008.

a. Licensee’s Implementation of the MRP-139 Baseline Inspections (TI2515-172-03.01)

Licensee’s Implementation of the MRP-139 Baseline Inspections. Verify the following:

1. The licensee’s inspection program includes inspections of the pressurizer, hot leg and cold leg temperature dissimilar metal butt welds and that the schedules for these baseline inspections are consistent with the requirements stated in MRP-139. If any baseline inspection schedules deviate from MRP-139 guidelines, determine what deviations are planned and what the general basis for the deviation is.

   There are a total of 14 dissimilar metal butt welds in the Callaway plant, six on the pressurizer, four on the hot legs and four on the cold legs. The licensee did not perform qualified volumetric examinations of the pressurizer dissimilar metal butt welds prior to performing the weld overlays. The licensee program implementing the MRP-139 requirements includes inspection of all of these dissimilar metal butt welds and both the inspection plans and schedules are consistent with MRP-139.

2. The licensees have completed their MRP-139 baseline inspections of all pressurizer dissimilar metal butt welds by December 31, 2007.

   The licensee completed full structural weld overlays of all pressurizer dissimilar metal butt welds in April 2007 during Refueling Outage 15. Following installation of the weld overlays, the licensee performed ultrasonic examination of these welds both during Refueling Outage 15 and during the next refueling outage, Refueling Outage 16, which took place in the fall of 2008. There were no recordable indications detected during any of these examinations.

b. Volumetric Examinations (TI2515-172-03.02)

Licensees perform volumetric examinations as part of the inspection/mitigation activities as described in MRP-139. Perform the following inspections through either direct observation (preferred method) or records review. If no examinations are being performed during the current outage, perform a records review of an examination during the previous outage.

1. Observe or review at least one examination of a weld (for example, an examination of a weld that is categorized as not being mitigated, an examination of a weld prior to mitigation by either weld overlay or mechanical stress improvement, or an examination of a weld after mitigation by mechanical stress improvement). Verify that the inspection is performed in accordance with the guidelines in MRP-139, Section 5.1.

   The licensee did not perform and has no current plans to perform any mechanical stress improvements. However, the licensee did perform pre-weld overlay examinations on the cold and hot leg dissimilar metal butt welds. The inspectors reviewed the records associated with the volumetric examination of a cold leg safe-end to pipe weld, Weld Number 2-RV-302-121-A. The inspectors verified that the technician was qualified and certified to perform the examination to the requirements of MRP-139, that the technician used a procedure that was qualified to meet the standards of the Performance Demonstration Initiative for this weld, and that the examination was performed to the requirements of the procedure. The examination was successful and no reportable indications were detected.

   The licensee does plan to perform stress improvements on the hot and cold leg dissimilar metal butt welds during their Fall 2011 refueling outage, but the exact method of stress improvement has not yet been selected.

2. Observe or review at least one weld overlay volumetric examination. Verify that the inspection performed is consistent with the NRC staff relief request authorization for the weld overlay. If the inspection coverage warrants further evaluation, review the licensee's documentation of the basis for achieving the required inspection coverage.

The inspectors reviewed the ultrasonic examination records performed on one pressurizer relief valve line (associated with the power-operated relief valve). The examination was performed in accordance with their approved procedure, which meets EPRI Performance Demonstration Initiative requirements and was consistent with the approved relief request for the weld overly.

3. Verify that the examinations were performed by qualified personnel.

   The inspectors verified that the personnel performing the examination were qualified.

4. Verify that any deficiencies identified were appropriately dispositioned and resolved.

   The licensee did not identify any deficiencies (either examination coverage or detected defects).

c. Weld Overlays (TI2515-172-03.03)

MRP-139 addresses inspection of dissimilar metal welds mitigated by weld overlays as part of the strategy to address the dissimilar metal butt weld issue. Inspectors should verify that the proper weld overlay techniques were used. If no examinations are being performed during the current outage, perform a records review of an examination during the previous outage.

1. For at least one weld overlay verify that the welding activities were performed consistent with ASME Code requirements as modified by NRC staff relief request authorizations. The inspectors reviewed the welding records associated with the full structural weld overlay of the pressurizer power operated relief valve, including welding procedure, weld procedure specification, weld bead logs, etc. After review of these records, including a review of interpass temperatures and any remarks from the bead log, the inspectors concluded that the welding was performed in accordance with approved welding requirements, including NRC staff relief request authorizations.

2. Verify that the licensee has submitted a relief request and obtained NRR staff authorization to install the weld overlays, whether full structural or optimized weld overlays.

   The inspectors requested and reviewed the relief request submitted by the licensee and the NRC approval and determined that the welding was performed in accordance with all of the requirements specified in these documents.

3. Verify that welding was performed by qualified personnel.

   The inspectors performed a review of the qualification certificates for several of the welders that performed this weld overlay. Based on this review, the inspectors concluded that the welders that produced the pressurizer power operated relief valve dissimilar metal butt weld overlay weld were qualified and certified in the process.

4. Verify that any deficiencies identified were appropriately dispositioned, and resolved.

   The licensee did not identify any deficiencies in this weld.

d. Mechanical Stress Improvement (TI2515-172-03-04)

MRP-139 addresses inspection of dissimilar metal welds mitigated by stress improvement as part of the strategy to address the dissimilar metal butt weld issue. For each application of stress improvement used, inspectors should review the stress improvement qualification report that describes the essential parameters of the stress improvement process (e.g., the location radial loading is applied and the applied load, as well as the inspection requirements). Inspectors should verify the following for each location where stress improvement was applied.

The licensee has not performed any mechanical stress improvements. While the licensee does plan to perform some form of stress improvement on the hot leg and cold leg dissimilar metal butt welds in the future, the specific stress improvement type has not yet been determined. Therefore, the inspectors did not perform any inspections in this area.

e. In-service Inspection Program (TI2515-172-05)

MRP-139 contains industry mandatory requirements for baseline and in-service inspection. In accordance with MRP-139, in-service inspections are performed based on the categorization of the weld configuration, which are classified as Categories A–I for volumetric examinations and Categories J and K for visual examinations.

1. The inspectors will perform an inspection to verify that the licensee has prepared an MRP-139 in-service inspection program and applicable welds are included in a category consistent with MRP-139 guidelines. The inspectors will verify that the licensee's inspection program and procedures specify inspection frequencies consistent with Tables 6-1 and 6-2 of MRP-139.

   The inspectors reviewed the Callaway plan for implementing MRP-0139 requirements, as included in their overall Alloy 600 inspection plan documented in EDP-ZZ-04070. The inspectors concluded that the dissimilar metal butt welds in the hot and cold legs were correctly categorized, in accordance with MRP-139 and the planned inspection frequencies for these welds meet the requirements of MRP-139.

   The licensee assigned the pressurizer welds as Category B, after performing full structural weld overlays on these welds and subsequent ultrasonic examinations of the new weld material. The inspectors concluded that this was not in accordance with the guidelines of MRP-139 and that these welds should have been assigned as Category F welds. However, since the licensee has performed successful post-weld ultrasonic examinations of these welds and no defects were noted during the examinations, the inspection frequencies, specified in MRP-139 for Category B and Category F welds are identical. Consequently, the inspectors concluded this discrepancy was an observation rather than a finding.

2. The inspectors will determine if any deviations are planned from the inspection guidelines in MRP-139, i.e., frequencies, examination volumes, methods.

   With the possible exception of the pressurizer weld classification (see Paragraph 1, above), the inspectors concluded that there were no deviations from MRP-139.

3. The inspectors will determine if any welds are categorized as H or I and review the licensee's basis for the categorization and the licensee's plans for addressing potential primary water stress corrosion cracking.

   The inspectors concluded that there were no welds categorized as either H or I.

4OA6 Meetings

Exit Meeting Summary

On October 24, 2008, the inspectors presented the radiation protection inspection results to Mr. T. Herrmann, Vice President, Engineering, and other members of the licensee staff. The licensee acknowledged the issues presented. The inspectors asked the licensee whether any materials examined during the inspection should be considered proprietary. No proprietary information was identified.

On October 31, 2008, the inspectors presented the results of the in-service inspection to Mr. T. Herrmann, Vice President, Engineering, and other members of licensee management. Licensee management acknowledged the inspection findings. The inspectors returned proprietary material examined during the inspection.

On December 30, 2008, the inspectors presented the residents integrated inspection report results to Mr. F. Diya, Vice President, Nuclear and other members of the licensee staff. The licensee acknowledged the issues presented. The inspectors confirmed that no proprietary information was retained.

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 noncited violations.

.1 Title 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures and Drawings," requires, in part, that activities affecting quality shall be prescribed by documented instructions, procedures, or drawings of a type appropriate to the circumstances and shall be accomplished in accordance with these instructions, procedures, or drawings. Contrary to the above, on October 18, 2008, the licensee identified that a heavy load lift was performed in violation of Procedure APA-ZZ-00365, Addendum L, "Callaway Plant Lifting Operations." The lift consisted of movement of the reactor vessel stud racks and was performed in the specified heavy load exclusion zone for Mode 6 with the reactor vessel head and upper internals removed and fuel in the reactor vessel. At the time of the lift, only components of residual heat removal Train B were available for shutdown cooling and the exclusion zone was designed to protect Train B of the residual heat removal system. This finding was entered in the licensee's corrective action program as

Callaway Action Request 200810761. This finding is greater than minor because it was associated with the human performance attribute of the Mitigating Systems cornerstone and adversely affected the 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 it did not increase the likelihood of a loss of reactor coolant system inventory, did not degrade the licensee's ability to terminate a leak path or add reactor coolant system inventory and did not degrade the licensee's ability to recover decay heat removal once lost.

2. Technical Specification 5.4.1.a, "Procedures," requires that written procedures be maintained covering the activities specified in Appendix A, "Typical Procedures for Pressurized Water Reactors," of Regulatory Guide 1.33, "Quality Assurance Program Requirements," February 1978. Regulatory Guide 1.33, Appendix A, Item 8.b (j), required procedures for emergency core cooling system surveillance testing. Contrary to the above, on October 17, 2008, the licensee identified that Procedure OSP-EJ-PV04A/B, "Trains A/B RHR and RCS Check Valve In-service Test - IPTE," was inadequate to prevent the residual heat removal pumps, Trains A and B, from achieving a pump run-out condition during surveillance testing. The procedure inappropriately directed residual heat removal heat exchanger bypass Valves EJFCV0618 and EJFCV0619 to be in the open position during testing resulting in decreased system resistance and residual heat removal pump flows above the vendor specified pump run-out conditions. This finding was entered in the licensee's corrective action program as Callaway Action Request 2008010603. This finding is greater than minor because it was associated with the procedural quality attribute of the Mitigating Systems cornerstone and adversely affected the 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 it is a design or qualification deficiency confirmed not to result in a loss of operability.

3. Technical Specification 3.9.4, "Containment Penetrations," requires, in part, that during core alterations the containment equipment hatch be closed and held in place by four bolts or open under appropriate administrative controls. Contrary to the above, during core alterations on October 19, 2008, the licensee moved irradiated fuel assemblies with the containment equipment hatch open and without the required administrative controls. Technical Specification Bases, Section 3.9.4, as well as licensee Procedure OSP-SF-00003, "Pre-core Alteration Verification," specified that the containment purge and exhaust system be in service as administrative controls. This finding was entered in the licensee's corrective action program as CAR 200810729. This finding is greater than minor because it was associated with the configuration control attribute of the Barrier Integrity cornerstone and adversely affected the cornerstone objective to provide reasonable assurance that physical design barriers protect the public from radionuclide releases caused by accidents or events. This finding is of very low safety significance because it did not increase the likelihood of a loss of reactor coolant system inventory, did not degrade the licensee's ability to terminate a leak path or add reactor coolant system inventory and did not degrade the licensee's ability to recover decay heat removal once lost.

SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee Personnel

A. Alley, Engineer
K. Bruckerhoff, Assistant Manager PRS, Emergency Preparedness
F. Diya, Vice President, Nuclear
J. Doughty, Engineer
T. Elwood, Supervising Engineer, Regulatory Affairs/Licensing
B. Farnam, Manager, Radiation Protection
G. Forster, Engineer, In-service Inspection
K. Gilliam, Supervisor, Radiation Protection Operations
L. Graessle, Director, Operations Support
A. Heflin, Senior Vice President and Chief Nuclear Officer
J. Heithold, Associate Engineer. Quality Assurance
T. Herrmann, Vice President, Engineering
M. Hoehn II, Engineer, MSRP-139
G. Hurla, Supervisor, Radiation Protection Operations
L. Kanuckel, Manager, Quality Assurance
C. Kiefer, Supervisor, Technical Programs
S. Maglio, Assistant Manager, Regulatory Affairs
M. McLachlan, Manager, Engineering Services
D. Neterer, Manager, Plant Director
S. Petzel, Engineer, Regulatory Affairs
R. Reed, Engineer
C. Stundebeck, Engineer
S. Thomure, Engineer, Welding/Section XI
D. Trokey, Regulatory Affairs, Specialist
R. Wilson, Engineer

LIST OF ITEMS OPENED AND CLOSED

Opened and Closed

05000483/2008005-01	NCV	Inadequate Shutdown Risk Assessment for Maintenance Activities in the Reactor Building (Section 1R13)
05000483/2008005-02	NCV	Failure to Ensure the Suitability of the Design of the Residual Heat Removal Train A Pump Room Cooler (Section 1R19)
05000483/2008005-03	NCV	Inadequate Maintenance Procedure Results in Residual Heat Removal Mechanical Seal Failure (Section 1R19)
05000483/2008005-04	NCV	Failure to Maintain an Adequate Plant Shutdown Procedure (Section 1R20)

05000483/2008005-05	NCV	Failure to Adequately Implement Plant Equipment Control Tagout Procedure (Section 1R20)
05000483/2008005-06	NCV	Failure to Comply with High Radiation Area Entry Requirements (Section 2OS1)
05000483/2008005-07	NCV	Failure to Terminate Refueling Water Storage Tank Recirculation Results in Inadvertent Loss of Spent Fuel Pool Inventory (Section 4OA3)
05000483/2008005-08	FIN	Failure to Evaluate Material Equivalencies Leads to a Manual Reactor Trip (Section 4OA3)

<u>Closed</u>

05000483/2008002-00	LER	Void Found in Line EM-023-HCB – Residual Heat Removal Pump A to Safety Injection Pumps (Section 4OA3)
05000483/2008003-05	VIO	Failure to Prevent Recurrence of Voids in ECCS Cold Leg Recirculation Piping (Section 4OA3)
05000483/2008003-00	LER	Inadvertent P-14 Feedwater Isolation Signal Actuation Followed by a Reactor Trip Actuation Due to Steam Generator Low-Low Water Narrow Range Trip (Section 4OA3)
05000483/2008004-00	LER	Failure to Maintain Containment Purge and Exhaust System In-service During Core Alterations with the Equipment Hatch Open (Section 4OA3)
05000483/2008005-00	LER	Reactor Manually Tripped Due to Main Feed Pump B Tripping on Low Lube Oil Pressure (Section 4OA3)

LIST OF DOCUMENTS REVIEWED

<u>Section 1RO4: Equipment Alignment</u>

<u>DRAWINGS</u>

NUMBER	TITLE	REVISION/DATE
M-22EC01	Piping and Instrumentation Diagram Fuel Pool Cooling and Clean-Up System	24
M-22EC02	Piping and Instrumentation Diagram Fuel Pool Cooling and Clean-Up System	31
M-25BG24	Hanger Location Drawing CVCS Auxiliary Spray Reactor Building	11

CALLAWAY ACTION REQUESTS

200809886	200811257

MISCELLANEOUS

Letter L-4189-00-1, Dominion Engineering, Inc. to Nicole Weber, AmerenUE, Subject: Structural Integrity Evaluation of Leaking Flaw on Callaway Pressurizer Auxiliary Spray Line, Revision 0

Section 1RO5: Fire Protection

DOCUMENTS

NUMBER	TITLE	REVISION/DATE
	Transient combustible permit for AB 2000 Rooms 1329 and 1331	10/27/08
MP 07-0066 Section 13. Fire Protection	Engineering Screen of Hazards Review	0

Section 1R06: Flood Protection

CALCULATIONS

NUMBER	TITLE	REVISION/DATE
M-FL-10	Flooding of Diesel Building Rooms	0

Section 1RO8: In-service Inspection Activities

DRAWINGS

NUMBER	TITLE	REVISION
C-2L2961	Pressure Boundary Drawing Reactor Building Personnel Access Hatch	0
E-CSCA-156-001	Long Version Canopy Seal Clamp Assembly	7
M-OS-BG24(Q),	Small Pipe Spool Isometric CVCS Auxiliary Spray Reactor Building	1
M-25EG03(Q),	Hanger Location Drawing, Component Cooling Water System, Aux Bldg Train B	0

EXAMINATION RECORDS

NUMBER	TITLE	DATE
Customer PO # 203840 SR Rev 1	Certificate of Compliance – Weld Wire	05/02/05
Job Task # 08006823.475	Weld Control Record	10/21/08
Welder ID: TPL 06498	Interim Update to Welder Qualification Summary	10/01/08
Welder ID: TPL 06498	AmerenUE Welder Qualification Record	04/29/08

EXEMPTION LETTERS

UL-NRC-05271, Docket Number 50-483 Union Electric Company Callaway Plant 10CFR50.55a Requests for Relief from ASME Section XI In-service Inspection Requirements for Third 10-Year Inspection Interval, dated March 28, 2006

UL-NRC-05185, Docket Number 50-483 Union Electric Company Callaway Plant Request for Relief from ASME Section XI Code In-service Examination Requirements, dated August 10, 2005

UL-NRC-05291, Docket Number 50-483 Union Electric Company Callaway Plant 10CFR50.55a Requests for Relief from ASME Section XI In-service Inspection Requirements for Third 10-Year Inspection Interval (Relief Requests 13R-05 and 13R-06, dated May 18, 2006

TAC NO. MD1155., Callaway Plant, Unit 1- Third 10-Year Interval In-service Inspection Program Relief Request 13R-01, dated January 3, 2007

TAC NO. MC8176, Callaway Plant, Unit 1- Authorization of Relief Request NO. 13R-03 for Snubber Visual Examination and Functional Testing Related to the Third 10-Year Interval In-service Inspection Program, dated March 7, 2006

TAC NO. MD2031 and MD2032, Callaway Plant, Unit 1- Relief Request 13R-05 and 13R-06 for the Third 10-Year In-service Inspection Interval, dated January 17, 2006

MISCELLANEOUS

Job Task # 08006823.475, ASME Section XI Repair/Replacement Plan Pressurizer Auxiliary Spray Line Prefab Pipe Weld, dated 10/21/08

Job Task # 08006823.475, List of ASME Section XI Replacement Materials, Parts, and Components, dated 10/21/08

Letter NED -01-105, and Program Doc., Pressure Boundary ASME Section XI, Subsection IWE Inspection Program, Revision 2, dated 11/15/2001

PERSONNEL QUALIFICATION RECORDS

Q-NIC-100 Rev. 21, Certification Record – Terrence J. McConnell, dated 05/27/08

Q-NIC-100 Rev. 21, Certification Record – Robert Nicholas, dated 09/24/08

Q-NIC-100 Rev. 21, Certification Record – Robert K. Gordon, dated 08/06/08

Q-NIC-100 Rev. 21, Certification Record – Larry M. Zahara, dated 03/17/08

Q-NIC-100 Rev. 21, Certification Record – Alfred H. Cote, dated 06/22/08

IQC-560, International Quality Consultants, Inc. Nondestructive/Visual Examination Certification Record – William T. Sims, dated 09/29/08

IQC-560, International Quality Consultants, Inc. Nondestructive/Visual Examination Certification Record – Jonathan Holzworth, dated 09/05/08

IQC-560, International Quality Consultants, Inc. Nondestructive/Visual Examination Certification Record – Sam Calvert, dated 09/20/06

IQC-560, International Quality Consultants, Inc. Nondestructive/Visual Examination Certification Record – Gerald J. Bitner, dated 03/27/06

Quality Control Inspector Recertification Record – Ted E. Stevens, dated 02/27/08

PROCEDURES

NUMBER	TITLE	REVISION
770.511.0113.A4FL	Time of Flight Diffraction Ultrasonic Examination	0
AUE-UT-98-1	Manual Ultrasonic Examination of Ferritic Piping Welds	1
AUE-UT-98-14	Manual Ultrasonic Examination of Nozzle Inside Radius, Excluding Reactor Pressure Vessel	1
AUE-UT-98-15	Ultrasonic Examination of Class 1 and 2 Vessel Welds over 2 Inches Thick	0
AUE-UT-98-2	Manual Ultrasonic Examination of Austenitic Piping Welds	1
AUE-UT-98-3	Ultrasonic Through-Wall Sizing in Piping Welds	0

AUE-UT-98-5	Ultrasonic Examination of Studs/Bolts Greater than Two Inches in Diameter	0
AUE-UT-98-6	Manual Ultrasonic Examination of Reactor Pressure Vessel Welds	0
AUE-UT-98-7	Manual Ultrasonic Through Wall and Length Sizing of Ultrasonic Indications in Reactor Pressure Vessel Welds	0
AUE-UT-98-PA-1	Manual Phased Array Ultrasonic Examination of Weld Overlaid similar and Dissimilar Metal Welds	0
EDP-ZZ-01004	Boric Acid Corrosion Control Program	6
ESP-ZZ-01016	Callaway Plant Nuclear Engineering ASME Section XI, Subsection IWE, Containment Pressure Boundary Inspection	5
MDP-ZZ-LM001	Attachment 2 Leakage Categorization and Acceptance Criteria	2
MTW-ZZ-WP514	Welding of P-8 Materials	14
NSD-ENG-EP-366	Procedure for Installing and Removing Spare Capped CRDM Canopy Seal Clamp Assembly (CSCA) and Dummy Can Assemblies	0
QCP-ZZ-05000	Liquid Penetrant Examination	19
QCP-ZZ-05010	Magnetic Particle Examination	14
QCP-ZZ-05042	Visual Examination to ASME VT-3	17

QA SURVEILLANCE REPORTS

NUMBER	TITLE	DATE
AP08-010	Quality Assurance Audit of In-Service Inspection	10/14/08
SP07-029	Quality Assurance Surveillance Report	05/24/07

CALLAWAY ACTION REQUESTS

200700468	200700495	200703186	200703551
200703707	200703890	200704015	200704370

200704512	200704522	200704631	200704783
200704908	200706049	200707147	200707221
200708195	200709633	200710008	200711763
200800378	200800705	200800712	200800732
200800768	200801209	200802643	200803107
200803577	200803992	200804909	200805918
200806305	200807812	200808033	200808262
200808945	200809176	200809446	200809449
200809456	200809481	200809493	200809573
200810295	200810451	200810457	200810463
200810578	200810613	200810794	200811213

Section 1R11: Licensed Operator Requalification Program

CALLAWAY ACTION REQUESTS

200812294

MISCELLANEOUS

American Nuclear Society ANSI/ANS-3.5-1998, Nuclear power plant simulators for use in operator training and examination.

Section 1R12: Maintenance Effectiveness

PROCEDURES

NUMBER	TITLE	REVISION/DATE
EDP-ZZ-01128	Maintenance Rule Program	10

Section 1R13: Maintenance Risk Assessment and Emergent Work Controls

PROCEDURES

NUMBER	TITLE	REVISION/DATE
EDP-ZZ-01129	Callaway Plant Risk Assessment	12

CALLAWAY ACTION REQUESTS

200811037

Section 1R15: Operability Evaluations

DRAWINGS

NUMBER	TITLE	REVISION/DATE
C-2L2933	Reactor Building Liner Plate Misc Details	6

PROCEDURES

NUMBER	TITLE	REVISION/DATE
ESP-ZZ-00356		10/09/2008

CALLAWAY ACTION REQUESTS

200703313	200809886	200810222	200810241	200810719
200811097	200811463	200811479	200811563	

MISCELLANEOUS:

ANSI/ANS-56.8-2002, Containment System Leakage Testing Requirements

Calculation GP-2, Containment Leakage Rates, Revision 0

MP 00-1008, Install New stainless steel liner plate in Containment Normal Sumps, Revision A

Section 1R18: Plant Modifications

DRAWINGS

NUMBER	TITLE	REVISION/DATE
118E03	Pressurizer Relief Tank Vol: 1800ft3	7
118E03	Pressurizer Relief Tank Vol: 1800ft3	8

BB-202	RHR Suction Relief Discharge Line	0
M-22BB03	Piping Isometric Reactor Coolant System Reactor Building	3
SK-6D31020	Pressurizer Relief Tank 4" Sparger Pipe Layout and Details	0

PROCEDURES

NUMBER	TITLE	REVISION/DATE
APA-ZZ-00150, Appendix E	Readiness Review Performance Checklist RHR/PRT Modifications	08/15/08
OTG-ZZ-00004, Addendum 2	End of Life Coastdown Operation	0

CALLAWAY ACTION REQUESTS

200809979	200811463	200811576	20081158

JOBS

NUMBER	TITLE	REVISION/DATE
07000966	RHR Suction Relief Valve Discharge Piping Modification	12/13/07
08008349.510	Train B containment cooler temporary modification installing blank flange on one tubing bundle pass	

MODIFICATION PACKAGES

NUMBER	TITLE	REVISION/DATE
MP-08-0029	CTMT Sump Base Plate Replacement	0
MP-07-0007	Modify the RHR Suction Relief Discharge Piping	0
MP-06-0006	Cycle 16 Reload Core Design – End of Cycle Tavg Coastdown	0
MP-07-0066	Train A ESW Outage supply/return replacement	0

MISCELLANEOUS

Calculation EJ-29, Residual Heat Removal Pump NPSH Margin During Recirculation, Revision 2

SFS-WC/CW-PA-7300, Wolf Creek/Callaway Sure Flow Strainer “A-Sump” Bottom Platform Assembly – Plan and Views, Revision 8

SFS-WC/CW-PA-7306, Wolf Creek/Callaway Sure Flow Strainer “B-Sump” Bottom Platform Assembly – Plan and Views, Revision 9

RFR 20083801, Revision 0

Letter LTR-NEW-07-227, Westinghouse Electric Company to Nickolas Sutherland, AmerenUE, Subject: Transmittal of PRT Sparger Design Change Documents, December 26, 2007

Technical Specification Amendment 186, OL1282 commitment summary

Equipment Out of Service Log, EOSL Record #'s 16132 and16133, dated 12/4/2008

ULNRC-05541, Request for Extension of Enforcement Discretion (ESW)

Night Order, Commitments associated with the 14 day LCO for the "A" ESW Train, and Job tasks for these Commitments.

Section 1R19: Postmaintenance Testing

DRAWINGS

NUMBER	TITLE	REVISION/DATE
E-23GL05	Schematic Diagram Pump Room Coolers	2

PROCEDURES

NUMBER	TITLE	REVISION/DATE
MPM-EJ-QP001	Residual Heat Removal Pump Overhaul	3
MPM-EJ-QP001	Residual Heat Removal Pump Overhaul	8

CALLAWAY ACTION REQUESTS

200810223	200810241	200810241	200810242	200810933
200811023	200811517			

JOBS

NUMBER	TITLE	REVISION/DATE
07005322	Perform OSP-NE-0001A	10/23/2008
07006649	Functional PMT per MP 01-1003	10/08/2008

07502389	Perform Offline Motor Test DSGL10A	10/08/2008
08003279	RHR Pump A PMT of mechanical seal	10/08/2008

Section 1R20: Refueling and Other Outage Activities

PROCEDURES

NUMBER	TITLE	REVISION/DATE
APA-ZZ-00365	Callaway Plant Lifting and Rigging Program	16
APA-ZZ-00365, Addendum L	Callaway Plant Lifting Operations	4
ETP-BB-03138	Disassembly of the Core Exit Thermocouple Nozzle Assembly (CETNA)	16
ETP-BB-03147	Reactor Vessel Head Removal – IPTE	15
OTG-ZZ-00001	Plant Heatup Cold Shutdown to Hot Standby	61
OTG-ZZ-00001, Addendum 1	Auxiliary Spray Operation	1
OTG-ZZ-00001, Addendum 2	Safety Injection Accumulator Preparation	0
OTG-ZZ-00001, Addendum 3	Enabling Pressurizer and Steamline Pressure Safety Injection	1
OTG-ZZ-0001A	Shutdown bank Withdrawal	16
OTG-ZZ-00002	Reactor Startup –IPTE	42
OTG-ZZ-00003	Plant Startup Hot Zero Power to 30% Power – IPTE	43
OTG-ZZ-00004	Power Operation	72
OTG-ZZ-00004, Addendum 1	Reactor Control During Power Operation	1
OTG-ZZ-00005	Plant Shutdown 20% Power to Hot Standby	32
OTG-ZZ-00005, Addendum 2	Control Bank Insertion	0

OTG-ZZ-00005, Addendum 3	Maintaining Mode 1 with the Turbine Tripped	1
OTG-ZZ-00006	Plant Cooldown Hot Standby to Cold Shutdown	50
OTG-ZZ-00006, Addendum 2	Shutdown Bank Insertion	1
OTG-ZZ-00006, Addendum 3	Opening Reactor Trip Breakers	4
OTG-ZZ-00006, Addendum 4	Initial RCS Depressurization and SI Block	5
OTG-ZZ-00006, Addendum 6	Securing Safety Injection Accumulators	3
OTG-ZZ-00006, Addendum 8	Pressurizer Auxiliary Spray Operation	3
OTG-ZZ-00007	Refueling Preparation, Performance and Recovery	27
OTG-ZZ-00004, Addendum 2	End of Life Coastdown Operation	0
OTN-BB-00002	Reactor Coolant System Draining	37
OTN-BB-00002, Addendum 3	Placing Nitrogen Cover Gas on the Reactor Vessel Head	1
OTN-BB-00002, Addendum 4	Venting the Reactor Vessel Head to Atmosphere	1
OTN-BB-00002, Addendum 6	Draining the RCS to Limited Inventory or Reduced Inventory – IPTE	9
OTN-BB-00002, Addendum 6	Draining the RCS to Limited or Reduced Inventory – IPTE	6

CALLAWAY ACTION REQUESTS

199300862	200708627	200810480	200810484	200810495
200810503	200810664	200811336	200811360	200811550

MISCELLANEOUS

Simple Self Assessment SA07-PE-S05, Control of Heavy Lifts, October 25, 2007

Calculation BB-18, Reactor Vessel Head Drop Analysis, Revision 0

ITR Report 08-025, Review of Refuel 16 Outage Risk Assessment, October 20, 2008

JOBS FOR STARTUP/MODE CHANGE SURVEILLANCES

04503937	05516932	07500932	08511116	08511116
08511230				

Section 1R22: Surveillance Testing

DRAWINGS

NUMBER	TITLE	REVISION/DATE
E-23EJ04A	Schematic Diagram RHR Pump 1 to Charging Pump Valve	9

PROCEDURES

NUMBER	TITLE	REVISION/DATE
EDP-ZZ-01114	Motor Operated Valve Program Guide	15
OSP-EJ-PV04A	RHR Train A and RCS Check Valve In-service Test – IPTE	1
OSP-EJ-PV04A	RHR Train A and RCS Check Valve In-service Test – IPTE	2
OSP-EJ-V003A	RHR Train A Mode 5 Valve In-service Test	14
OSP-SA-2413A	Diesel Generator Train A and Sequencer Testing	8
OSP-NE-0024B	Standby Diesel Generator B 24-Hour Run and Hot Restart Test	25
OSP-SF-00005	Estimated Critical Position Calculation	17
OSP-BB-00009	RCS Inventory Balance	22

CALLAWAY ACTION REQUESTS

200810598	200810603	200811557	200811559

JOBS

NUMBER	TITLE	REVISION/DATE
07504731/500	Train A RHR Comprehensive Pump Test	10/17/08
0754910/500	Diesel Generator Train A and Sequencer Testing	10/11/08
07509412/500	Standby Diesel Generator B 24-Hour Run and Hot Restart Test	10/03/08
08007549/910	EJHV8804A/RHR Train A Charging Pumps Supply Isolation	10/17/08
08510068	Motor-Driven Auxiliary Feedwater Train A Pump In-service Test	11/4/08

Section 2OS1: Access Controls to Radiologically Significant Areas

CALLAWAY ACTION REQUESTS

200809854	200810107	200810130	200810271	200810320
200810495	200810495	200810567	200810571	200810576
200810640	200810644	200810763	200810771	200810781
200810835	200810920			

PROCEDURES

NUMBER	TITLE	REVISION/DATE
APA-ZZ-01000	Callaway Plant Radiation Protection Program	29
APA-ZZ-01004	Radiological Work Standards	13
APA-ZZ-01106	Lock and Key Control	18
HDP-ZZ-01500	Radiological Postings	31
HDP-ZZ-03000	Radiological Survey Program	31
HTP-ZZ-06001	High Radiation/Very High Radiation Area Access	34

RADIATION WORK PERMITS

850121RPJOBCOV	Radiation Protection Job Coverage in the Refuel Cavity
890401HRA	Engineering Activities in High Radiation Areas
890401NONHRA	Engineering Activities in The RCA

SAMPLE RESULTS AND SURVEYS

Particulate air sample outside hatch (from 01:20 October 21, 2008 to 06:15 October 22, 2008)

MISCELLANEOUS

Callaway Plant Long Range Dose and Source Term Reduction Plan; Revision 2
Form CA0417, Radiation Dose Evaluation, for J. Picard
Form CA0417, Radiation Dose Evaluation, for C. Whiteley

Section 2OS2: ALARA Planning and Controls

AUDITS, SELF-ASSESSMENTS, AND SURVEILLANCES

NUMBER	TITLE	REVISION/DATE
APA-ZZ-01000	Callaway Plant Radiation Protection Program	29
APA-ZZ-01004	Radiological Work Standards	13
HDP-ZZ-01100	ALARA Planning and Review	8
HDP-ZZ-01200	Radiation Work Permits	10

CALLAWAY ACTION REQUESTS

200802837	200803673	200803829	200804502	200805135
200810316	200810337	200810340	200810575	200810602
200810664	200810771	200810775	200810776	200810777
200810813	200810824	200810833	200810835	

RADIATION WORK PERMITS

840903AREA5	Maintenance Activities in the Auxiliary Building
850820RBBIO	Install/Remove Temporary Shielding in the Reactor Building Bioshield
850901LHRA	Work Activities in the Reactor Building
07008244	Refuel ESW Pipe Replacement Inside the Reactor Building

Section 4OA1: Performance Indicator Verification

PROCEDURES

NUMBER	TITLE	REVISION/DATE
RRA-ZZ-00001	NRC Performance Indicator Program	5

Section 4OA2: Identification and Resolution of Problems

DRAWINGS

NUMBER	TITLE	REVISION/DATE
M22EP01	Piping and Instrumentation Diagram Accumulator Safety Injection	16

CALLAWAY ACTION REQUESTS

200604909	200705863	200707793	200800831	200802926
200806902	200807613	200807812	200807881	200808262
200808723	200808868	200809210	200809385	200809445
200809449	200809468	200809472	200809493	200809577
200809757	200809802	200809886	200810222	200810223
200810241	200811418	200811576	200811692	200811711

MISCELLANEOUS

Simple Surveillance Report SP08-009, Leak Management Program, Dated March 3, 2008
Job 07010080
Job 08008207
Job 08008511

Section 4OA3: Event Follow-Up

PROCEDURES

NUMBER	TITLE	REVISION/DATE
EDP-ZZ-04015	Evaluating and Processing Requests for Resolution	32
OTN-EC-00001, Addendum 3	RWST Cleanup Operations	6
OTN-EC-00001, Addendum 6	Filling the Spent Fuel Pool	3

CALLAWAY ACTION REQUESTS

200811692	200811781	200811821

Section 4OA5: Other Activities

CALLAWAY ACTION REQUESTS

200809586

DRAWINGS

NUMBER	TITLE	REVISION
10058C06	Callaway Unit 1 Pressurizer Relief Nozzle SWOL Design 2-TBB03-4-W	0
10017D92	Callaway Unit 1 Pressurizer Relief Nozzle SWOL Field Implementation 2-TBB03-4-W	0

MISCELLANEOUS

PDI-ISI-254-SE, “Remote In-service Examination of Reactor Vessel Nozzle to Safe End, Nozzle to pipe, and Safe End to Pipe Welds, ” Revision 2

SP07-020, “Quality Assurance Surveillance Report,” April 13, 2007

SP07-020, “Quality Assurance Surveillance Report,” April 13, 2007

WDI-PJF-1303575-FSR-001. “Calloway Pressurizer PORV Nozzle SWOL Examination Coverage Summary”, April 2007

900708-05, “Report of Non-Destructive Examination Liquid Penetrant Examination,” April 7, 2007

8MC-GTAW, “ASME IX Weld Procedure Specification,” Revision 10

Letter from Thomas G. Hiltz (NRC) to Charles D. Naslund (UEC), “Callaway Plant, Unit 1 – Alternatives for Application of Structural Weld Overlays to Pressurizer Dissimilar Metal Nozzle Welds (TAC NO. MD2815),” July 10, 2007

Letter from David T. Fitzgerald (AmerenUE) to U.S. Nuclear Regulatory Commission, “Docket Number 50-483 Callaway Plant Unit 1 Union Electric Co. Inspection/Mitigation Plans for Alloy 82/182 Pressurizer Butt Welds,” January 31, 2007

ULNRC -05385 Letter from David T. Fitzgerald (AmerenUE) to U.S. Nuclear Regulatory Commission, “Docket Number 50-483 Union Electric Company Callaway Plant Response to Request for Additional Information Regarding 10 CFR50.55a Request for Relief from ASME Section XI Repair and Replacement Requirements Proposed Alternatives for Application of Structural Weld Overlays to Pressurizer Nozzle Welds,” March 26, 2007

ULNRC-05395, Letter from David T. Fitzgerald (AmerenUE) to U.S. Nuclear Regulatory Commission, “Docket Number 50-483 Union Electric Company Callaway Plant Clarification of Response to Request for Additional Information Regarding 10 CFR50.55a Request for Relief from ASME Section XI Repair and Replacement Requirements Proposed Alternatives for Application of Structural Weld Overlays to Pressurizer Nozzle Welds,” April 5, 2007

PROCEDURES

NUMBER	TITLE	REVISION
AUE-UT-98-12	Ultrasonic Examination of Class 1 & 2 Vessel Welds Over 2 Inches Thick	1
AUE-UT-CP-2	Procedure for Inspection System Performance	1

	Checks & Beam Spread Measurements	
AUE-UT-98-8	Manuel Ultrasonic Examination of Weld Overlay Similar & Dissimilar Metal Welds	9
PDI-UT-8	PDI Generic Procedure for Ultrasonic Examination of Weld Overlay Similar & Dissimilar Metal Welds	F
EPRI 1015134	Nondestructive Evaluation: Procedure for Manual Phased Array UT of Weld Overlays, Technical Update	October 2007
EDP-ZZ-04070	Reactor Coolant System Materials Degradation Management Plan	3
MRS-SSP-2063	Appendix C Structural Weld Overlay (SWOL) Layout and UT Thickness Location Guidelines	0
MRS-SSP-2063	Appendix D Structural Weld Overlay Template Preparation	0
MRS-SSP-2063	Appendix E Profile Grinding	0
UT.ASME.PA.	Ultrasonic Examination Using the Phased Array Technique	0

QUALIFICATION CERTIFICATIONS

UT NDE Technician-1
Welders - 16

Section 4OA7: Licensee-Identified Violations

PROCEDURES

NUMBER	TITLE	REVISION/DATE
APA-ZZ-00365	Callaway Plant Lifting and Rigging Program	16
APA-ZZ-00365, Addendum L	Callaway Plant Lifting Operations	4
OSP-EJ-PV04A	RHR Train A and RCS Check Valve In-service Test – IPTE	1
OSP-EJ-PV04A	RHR Train A and RCS Check Valve In-service Test – IPTE	2

CALLAWAY ACTION REQUESTS

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