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

July 29, 2010

EA-10-065

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

Subject: CALLAWAY PLANT - NRC INTEGRATED INSPECTION REPORT 05000483/2010003

Dear Mr. Heflin:

On June 23, 2010, 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 June 23, 2010, with Mr. Fadi Diya, Vice President Nuclear Operations, 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 two NRC-identified findings and two self-revealing findings of very low safety significance (Green). Three 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. In addition, if you disagree with the crosscutting aspect assigned to any finding in this report, you should provide a response within 30 days of the date of this inspection report, with the basis for your disagreement, to the Regional Administrator, Region IV, and the NRC Resident Inspector at the Callaway Plant.

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/ Richard W. Deese for

Geoffrey B. Miller, Chief
Project Branch B
Division of Reactor Projects

Docket: 50-483
License: NPF-30

Enclosure: NRC Inspection Report 05000483/2010003
w/Attachments 1 and 2

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	C:DRS/OB	C:DRS/PSB1	C:DRS/PSB2	C:DRS/EB1
	MHaire	MShannon	GWerner	TFarnholtz
	/RA/ BTLfor	***/RA/***	***/RA/***	***/RA/***
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C:DRS/EB2	C:DRS/TSB	ACES	SRA	C:DRP/B
NO'Keefe	MHay	RKellar	GReplogle	GMiller
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U.S. NUCLEAR REGULATORY COMMISSION

REGION IV

Docket:	05000483
License:	NPF-30
Report:	05000483/2010003
Licensee:	Union Electric Company
Facility:	Callaway Plant
Location:	Junction Highway C and Highway O Fulton, MO
Dates:	March 25 through June 23, 2010
Inspectors:	D. Dumbacher, Senior Resident Inspector J. Groom, Resident Inspector J. Adams, Reactor Inspector T. Buchanan, Reactor Inspector A. Fairbanks, Reactor Inspector C. Graves, Health Physicist N. Greene, Health Physicist B. Henderson, Reactor Inspector N. Hernandez, Project Engineer D. Reinert, Reactor Inspection
Accompanying Inspectors:	J. Braisted, Reactor Inspector
Approved By:	Geoffrey B. Miller, Chief, Project Branch B Division of Reactor Projects

SUMMARY OF FINDINGS

IR 05000483/2010003, 03/25/2010 - 06/23/2010, Callaway Plant, Integrated Resident and Regional Report; Equipment Alignments, Refueling and Other Outage Activities and Event Follow-up.

The report covered a 3-month period of inspection by resident inspectors and an announced baseline inspections by regional inspectors. Three Green noncited violations and one finding of significance 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 finding associated with AmerenUE's failure to take prompt corrective actions for leaking boundary valves in the chemical and volume control system. On April 13, 2010, an attempt to place the train A chemical and volume control system mixed bed in service resulted in leakage past a documented leaking drain valve. The lingering equipment problems resulted in an unplanned 25 gallon per minute loss rate of volume control tank inventory and an emergency action level declaration for excessive reactor coolant system leakage. Later, the declaration was retracted. The licensee placed this issue into the corrective action program as Callaway Action Request 201003146.

  This finding is more than minor because it was associated with the reactor safety Initiating Events Cornerstone attribute of configuration control and affected the objective to limit the likelihood of events that upset plant stability. Using Manual Chapter 0609.04, "Phase 1 - Initial Screening and Characterization of Findings," the inspectors determined that this finding is of very low significance because the condition did not result in the reactor coolant system technical specification leakage limit being exceeded, did not contribute to both the likelihood of a reactor trip and the likelihood that mitigating equipment or functions would be unavailable, and did not increase the likelihood of a fire or internal/external flood. This finding, which involved inadequate scheduling of corrective action related jobs, has a crosscutting aspect in the area of human performance associated with the work control component because AmerenUE did not appropriately coordinate work activities to address the impact of the work on different job activities [H.3(b)] (Section 4OA3).

Cornerstone: Mitigating Systems

- Green. The inspectors identified a noncited violation of Technical Specification 3.5.2, "Emergency Core Cooling Systems." Specifically Technical Specifications Surveillance Requirement 3.5.2.3, "Verify the ECCS piping is full of water," was not being met by licensee Procedure OSP-SA-00003, "Emergency Core Cooling System Flow Path Verification and Venting." On April 22, 2010, the inspectors discovered that the train B residual heat removal system discharge line EJ-024-ECB-10" did not have an accessible high point vent. The line was required by Callaway procedures to be either monitored by venting or tested using an ultrasonic method as described in the procedure's acceptance criteria. Callaway had identified the need to install a vent valve in line EJ-024-ECB-10" per modification MP-08-0016 prior to Refueling Outage 17. The licensee originally scheduled the vent valve installation during Refueling Outage 17, but had inappropriately deferred the maintenance to the next outage in fall 2011. As immediate corrective action, the licensee installed the vent valves in Refueling Outage 17 and placed this issue into the corrective action program as Callaway Action Request 201004078.

  This finding is more than minor because it affected the Mitigating Systems Cornerstone procedure quality attribute and affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Using Manual Chapter 0609.04, "Phase 1 – Initial Screening and Characterization of Findings," the inspectors determined that this finding is of very low significance because it was only a design or qualification deficiency confirmed not to result in loss of operability. This finding has a crosscutting aspect in the area of human performance associated with the decision making component because the licensee failed to use conservative assumptions in decision making and did not adopt a requirement to demonstrate that either venting or ultrasonic testing was needed to verify line EJ-024-ECB-10" was full of water [H.1(b)] (Section 1R04).

- Green. The inspectors identified a self-revealing noncited violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," after the licensee failed to adequately select suitable replacement gaskets essential to the operation of emergency diesel generator train A. On March 30, 2010, during performance of Procedure OSP-NE-00024A, "Standby Diesel Generator A 24-Hour Run and Hot Restart Test," the emergency diesel generator train A unexpectedly lost speed and tripped after 16.7 hours of operation. Posttrip indications revealed that the diesel generator tripped from a stripped splined shaft in the governor drive housing. The failure of the splined shaft was caused by an improperly cut gasket which did not have the required oil port hole to allow proper lubrication of the drive assembly. The licensee replaced the damaged shaft and placed this issue in their corrective action program as Callaway Action Request 201002675.

  This finding was greater than minor because it was associated with the Mitigating Systems Cornerstone attribute of design control and affects the associated cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. The resident inspectors performed the initial significance determination for the diesel gasket finding using the NRC Inspection Manual 0609, Attachment 0609.04, "Phase 1 – Initial Screening and Characterization of Findings." The finding screened to a Phase 2 significance determination because it involved the loss of one train of safety related equipment for greater than its technical specification allowed outage time. A Region IV senior reactor analyst performed a Phase 2 significance determination using the pre-solved worksheet from the "Risk Informed Inspection Notebook for Callaway Nuclear Generating Station," Revision 2.01a. The analyst assumed an exposure period of one year. The finding was potentially Yellow, which warranted further review. The senior reactor analyst subsequently performed a bounding Phase 3 significance determination and found the finding to be of very low safety significance (Green). The dominant cutsets included a loss of offsite power initiating event, failure to recover offsite power in 4 hours, failure of the train B emergency diesel generator, and a reactor coolant pump seal failure. Equipment that mitigated the significance included the operable emergency diesel generator and the turbine-driven auxiliary feedwater pump. This finding did not have a crosscutting aspect since it was not a performance deficiency reflective of current licensee performance (Section 4OA3).

Cornerstone: Barrier Integrity

- Green. The inspectors identified a self-revealing noncited violation of Technical Specification 5.4.1.a, "Procedures," when the licensee's inadequate procedure and failure to control work activities during a reload of the reactor vessel fuel assemblies resulted in deenergization of all available source range nuclear instrument channels. On May 6, 2010, while in Mode 6 – Refueling, licensee testing of nuclear instrument power range channel N44 and maintenance on 120 Vac instrument bus NN03 affecting power range channel N43 made up the logic for permissive P-10. The permissive sent a protective logic signal to deenergize both available source range nuclear instruments. The control room immediately directed the fuel handling crew to stop fuel movement until the source range channels could be restored. A fuel assembly was in the upender ready for transfer to the reactor vessel core location at the time. The licensee placed this issue into the corrective action program as Callaway Action Request 201004301.

  This finding is more than minor because it was associated with the configuration control attribute of the Barrier Integrity Cornerstone and affects the 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 G, Attachment 1, "Shutdown Operations Significance Determination Process Phase 1 - Operational Checklists for Both PWRs and BWRs," this finding was 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 when needed, and did not degrade the licensee's ability to recover decay heat removal once lost. This finding had a crosscutting aspect in the area of human performance associated with the work control component because the licensee failed to coordinate work activities by incorporating actions to address the impact of the work on different job activities and communicate, coordinate, and cooperate with each other during activities in which interdepartmental coordination is necessary to assure plant and human performance [H.3(b)] (Section 1R20).

B. Licensee-Identified Violations

Three 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 request numbers) are listed in Section 4OA7.

REPORT DETAILS

Summary of Plant Status

Callaway Plant was operated at near full power until April 17, 2010, when the licensee shut down the plant to perform Refueling Outage 17 planned maintenance. The original planned restart date was delayed mainly due to discovery of a damaged control rod assembly during startup testing on May 23, 2010. This required cooldown of the plant, replacing the affected control rod and associated guide tube, and evaluating the extent of condition. The plant was started up and the main generator was synchronized to the electrical grid on June 12, 2010. The plant reached 100 percent power on June 16, 2010, and remained near 100 percent for the remainder of the inspection period.

1. REACTOR SAFETY

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

1R01 Adverse Weather Protection (71111.01)

Readiness for Impending Adverse Weather Conditions

a. Inspection Scope

Since thunderstorms with potential tornados and high winds were forecast in the vicinity of the facility on April 30, 2010, the inspectors reviewed the plant personnel’s overall preparations/protection for the expected weather conditions. Specifically, the inspectors walked down the spent fuel pool and residual heat removal systems because their safety-related functions could be affected, or required, as a result of high winds or tornado-generated missiles or the loss of offsite power. The inspectors evaluated the plant staff’s preparations against the site’s procedures and determined that the staff’s actions were adequate. During the inspection, the inspectors focused on plant-specific design features and the licensee’s procedures used to respond to specified adverse weather conditions. The inspectors also toured the plant grounds to look for any loose debris that could become missiles during a tornado. The inspectors evaluated operator staffing and accessibility of controls and indications for those systems required to control the plant. Additionally, the inspectors reviewed the Final Safety Analysis Report and performance requirements for the systems selected for inspection, and verified that operator actions were appropriate as specified by plant-specific procedures. The inspectors also reviewed a sample of corrective action program items to verify that the licensee identified adverse weather issues at an appropriate threshold and dispositioned them through the corrective action program in accordance with station corrective action procedures. Specific documents reviewed during this inspection are listed in the attachment.

These activities constitute completion of one readiness for impending adverse weather condition sample as defined in Inspection Procedure 71111.01-05.

b. Findings

No findings were identified.

1R04 Equipment Alignments (71111.04)

.1 Partial Walkdown

a. Inspection Scope

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

- April 19, 2010, Cold overpressure mitigation system
- May 2, 2010, Engineered safety feature actuation system alignment during plant startups and shutdowns including the feedwater isolation logic (P-4) and main steam dump bypass control permissive (P-12)
- May 5, 2010, Reactor coolant system
- May 7, 2010, Essential service water system train A following restoration from maintenance
- May 22, 2010, Train A and B main steam system as the steam plant temperatures were being raised

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 inspected 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 five partial system walkdown samples as defined in Inspection Procedure 71111.04-05.

b. Findings

No findings were identified.

.2 Complete System Walkdown (Associated with Temporary Instruction (TI) 2515/177, Managing Gas Accumulation in Emergency Core Cooling, Decay Heat Removal, and Containment Spray Systems)

a. Inspection Scope

On April 15, 2010, the inspectors conducted a walkdown of the residual heat removal system in sufficient detail to reasonably assure the acceptability of the licensee's walkdowns (TI 2515/177, Section 04.02.d). The inspectors also verified whether the information obtained during the licensee's walkdowns was consistent with the items identified during the inspector's independent walkdown (TI 2515/177, Section 04.02.c.3).

The inspectors verified that the licensee had isometric drawings that describe the residual heat removal system configurations and had confirmed the accuracy of the drawings (TI 2515/177, Section 04.02.a). The inspectors verified the following related to the isometric drawings:

- High point vents were identified
- High points that do not have vents were acceptably recognizable
- Other areas where gas can accumulate and potentially impact subject system operability, such as at orifices in horizontal pipes, isolated branch lines, heat exchangers, improperly sloped piping, and under closed valves were acceptably described in the drawings or in referenced documentation
- Horizontal pipe centerline elevation deviations and pipe slopes in nominally horizontal lines that exceed specified criteria were identified
- All pipes and fittings were clearly shown
- The drawings were up-to-date with respect to recent hardware changes and any discrepancies between as-built configurations and the drawings were documented and entered into the corrective action program for resolution

The inspectors also verified that piping and instrumentation diagrams accurately described the subject systems, were up-to-date with respect to recent hardware changes, and that any discrepancies between as-built configurations, the isometric drawings, and the piping and instrumentation diagrams were documented and entered into the corrective action program for resolution (TI 2515/177, Section 04.02.b).

Documents reviewed are listed in the attachment to this report.

This inspection effort counts towards the completion of TI 2515/177 which will be closed in a later inspection report. These activities constitute completion of one complete system walkdown sample as defined in Inspection Procedure 71111.04-05.

b. Findings

Introduction. The inspectors identified a Green noncited violation of Technical Specification 3.5.2, “Emergency Core Cooling Systems,” after noting that Technical Specifications Surveillance Requirement 3.5.2.3, “Verify the ECCS piping is full of water,” was not being met by licensee Procedure OSP-SA-00003, “Emergency Core Cooling System Flow Path Verification and Venting.” Procedure OSP-SA-00003 was not capable of monitoring and venting a substantial portion of the accessible train B residual heat removal discharge piping to ensure the emergency core cooling system was full of water as required.

Description. On April 22, 2010, the resident inspectors performed a walkdown of the residual heat removal system per Inspection Procedure 71111.04A in accordance with Temporary Instruction 2515/177 related to Generic Letter 2008-01, “Managing Gas Accumulation in Emergency Core Cooling, Decay Heat Removal, and Containment Spray Systems.” The inspectors discovered that the train B residual heat removal system discharge line EJ-024-ECB-10”, from the 2000’ mechanical pipe penetration room 1323 forward to reactor coolant system loops 3 and 4 did not have an accessible high point vent. Also, the line was not being tested using an ultrasonic method as described in the acceptance criteria step 3.1.2 of Procedure OSP-SA-00003. Callaway engineering had determined that monitoring of safety injection accumulator levels was sufficient to identify and remove any gasses introduced and thus ensure the emergency core cooling system was full of water. The licensee’s option to monitor only accumulator levels did not account for possible reactor coolant system back-leakage through installed check valves or safety injection test header connections to the residual heat removal discharge lines. Additionally, the method used to monitor the accumulator levels inappropriately relied on a computer generated email to the system engineer when the level reached a low alarm setpoint. Normal conservative operator actions to refill the accumulator prior to the alarm level would usually have prevented the alarm and the subsequent alerting of the engineer.

The Callaway Technical Specification Bases for Surveillance Requirement 3.5.2.3 addressed the potential for gas intrusion in stating, “The 31 day frequency takes into consideration the gradual nature of gas accumulation in the ECCS piping and the procedural controls governing system operation.” In its enclosure, Generic Letter 2008-01 discussed the technical considerations for reasonably assuring emergency core cooling system operability. This enclosure identified the following sources of gas:

- Leakage from the accumulators
- Leakage from the reactor coolant system
- Leakage through the test header valves

Section 6 of the enclosure stated, "Where high point vents are not vented, the important questions are whether the licensee is aware of the potential problems, whether controls and practices reflect this awareness, and whether modifications should be accomplished." Callaway responded to Generic Letter 2008-01 on October 13, 2008. On page 33 of the response, the licensee stated the vast majority of the leakage types outlined above would occur very slowly and be identified during the monthly emergency core cooling system venting Surveillance OSP-SA-00003. The monthly surveillance did not take credit for monitoring accumulator levels using computer generated emails. Callaway had identified the need to install a vent valve in line EJ-024-ECB-10" per modification MP-08-0016 prior to Refueling Outage 17. This vent valve was scheduled to be installed during Refueling Outage 17, but the installation was deferred to the fall 2011 outage. Following identification of this issue, by the resident inspectors, the licensee installed the vent valves in Refueling Outage 17. This initial corrective action addresses the inability to meet the 31 day Technical Specification Surveillance Requirement 3.5.2.3.

Analysis. Failure to have the capability to adequately verify that emergency core cooling system piping was full of water, as required by Technical Specification 3.5.2, is a performance deficiency. This finding is more than minor because it affected the Mitigating Systems Cornerstone procedure quality attribute and affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences (i.e., core damage). Using Manual Chapter 0609.04, "Phase 1 – Initial Screening and Characterization of Findings," the inspectors determined that this finding is of very low significance because it was only a design or qualification deficiency confirmed not to result in loss of operability. This finding has a crosscutting aspect in the area of human performance associated with the decision making component because the licensee failed to use conservative assumptions in decision making and did not adopt a requirement to demonstrate that either venting or ultrasonic testing was needed to verify line EJ-024-ECB-10" was full of water. Instead, the licensee assumed that gas intrusion was not likely to occur in the accessible line [H.1(b)].

Enforcement. Technical Specification 3.5.2, "Emergency Core Cooling Systems," Surveillance Requirement 3.5.2.3, required that the licensee verify that emergency core cooling system piping is full of water every 31 days. Contrary to the above, prior to April 22, 2010, AmerenUE Implementing Procedure OSP-SA-00003, "Emergency Core Cooling Flow Path Verification and Venting," was inadequate to meet Technical Specification Surveillance Requirement 3.5.2.3. Because this finding is of very low safety significance and was entered into the licensee's corrective action program as Callaway Action Request 201004078, this violation is being treated as a noncited violation, consistent with Section VI.A.1 of the NRC Enforcement Policy: NCV 05000483/2010003-01, "Inadequate Surveillance Procedure to Verify and Maintain Emergency Core Cooling System Operable."

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:

- April 26, 2010, Essential service water pipe chase room 3101
- April 27, 2010, Emergency diesel generator room B during planned hot work
- April 29, 2010, Reactor building
- May 7, 2010, Auxiliary building 2026' elevation during planned hot work
- May 18, 2010, Turbine building 2000' and 2016' elevations

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 five quarterly fire-protection inspection samples as defined in Inspection Procedure 71111.05-05.

b. Findings

No findings were identified.

1R07 Heat Sink Performance (71111.07)

.1 Annual Review of Heat Sink Performance

a. Inspection Scope

On June 4, 2010, the inspectors completed review of licensee programs, verified performance against industry standards, and reviewed critical operating parameters and maintenance records for the containment cooler heat exchangers. The inspectors verified that performance tests were satisfactorily conducted for heat exchangers/heat sinks and reviewed for problems or errors; the licensee utilized the periodic maintenance method outlined in EPRI Report NP 7552, "Heat Exchanger Performance Monitoring Guidelines," the licensee properly utilized biofouling controls; the licensee's heat exchanger inspections adequately assessed the state of cleanliness of their tubes; and the heat exchanger was correctly categorized under 10 CFR 50.65, "Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants." Specific documents reviewed during this inspection are listed in the attachment.

These activities constitute completion of one annual heat sink inspection sample as defined in Inspection Procedure 71111.07-05.

b. Findings

No findings were identified.

.2 Triennial Review of Heat Sink Performance

a. Inspection Scope

The inspectors reviewed licensee programs, verified performance against industry standards, and reviewed critical operating parameters and maintenance records for the component cooling water train A heat exchanger, residual heat removal train B heat exchanger, ultimate heat sink retention pond and ultimate heat sink cooling tower. The inspectors verified that performance tests were satisfactorily conducted for heat exchangers/heat sinks and reviewed for problems or errors; the licensee utilized the periodic maintenance method outlined in EPRI Report NP 7552, "Heat Exchanger Performance Monitoring Guidelines," the licensee properly utilized biofouling controls; the licensee's heat exchanger inspections adequately assessed the state of cleanliness of their tubes; and the heat exchanger was correctly categorized under 10 CFR 50.65, "Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants." Specific documents reviewed during this inspection are listed in the attachment.

These activities constitute completion of two heat sink inspection samples as defined in Inspection Procedure 71111.07-05.

b. Findings

No findings were identified.

1R08 Inservice Inspection Activities (71111.08)

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

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

a. Inspection Scope

The inspectors reviewed two types of nondestructive examination activities and two welds on the reactor coolant system pressure boundary. None of the above observed or reviewed nondestructive examinations identified any relevant indications and cognizant license personnel stated that no relevant indications were accepted by the licensee for continued service.

The inspectors directly observed the following nondestructive examinations:

SYSTEM	WELD IDENTIFICATION	EXAMINATION TYPE
Safety Injection (EP)	EP01H011231 (Strut)	Visual (VT-3)
Reactor Coolant (BB)	2-BG-21-F012 & 2-BG-21-FO13 Pipe to valve welds, 3.5 inch	Ultrasonic

The inspectors reviewed records for the following nondestructive examination:

SYSTEM	WELD IDENTIFICATION	EXAMINATION TYPE
Main Steam	2-AB-01-F026	Ultrasonic

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. The inspectors also verified the qualifications of all nondestructive examination technicians performing the inspections were current.

The inspectors directly observed a portion of the following welding activities:

SYSTEM	WELD IDENTIFICATION	WELD TYPE
Reactor Coolant (BB)	150-FW02	Manual Gas Tungsten Arc

The inspectors reviewed records for the following welding activities:

SYSTEM	WELD IDENTIFICATION	WELD TYPE
Chemical and Volume Control	08004887.025	Manual Gas Tungsten Arc

The inspectors verified, by review, that the welding procedure specifications and the welders had been properly qualified in accordance with ASME Code, Section IX, requirements. The inspectors also verified through record review that essential variables for the welding process were identified, recorded in the procedure qualification record, and formed the bases for qualification of the welding procedure specifications. Specific documents reviewed during this inspection are listed in the attachment.

These actions constitute completion of the requirements for Section 02.01.

b. Findings

No findings were identified.

.2 Vessel Upper Head Penetration Inspection Activities (71111.08-02.02)

The licensee did not perform any activities in this area. Therefore, the inspectors did not perform any inspections in this area.

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

a. Inspection Scope

The inspectors evaluated the implementation of the licensee's boric acid corrosion control program for monitoring degradation of those systems that could be adversely affected by boric acid corrosion. The inspectors reviewed the documentation associated with the licensee's boric acid corrosion control walkdown as specified in Procedure EDP-ZZ-01004, "Boric Acid Corrosion Control Program." The inspectors also reviewed the visual records of the components and equipment. The inspectors verified that the visual inspections emphasized locations where boric acid leaks could cause degradation of safety-significant components. The inspectors also verified that the engineering evaluations for those components where boric acid was identified gave assurance that the ASME Code wall thickness limits were properly maintained. The inspectors confirmed that the corrective actions performed for evidence of boric acid leaks were consistent with requirements of the ASME Code. Specific documents reviewed during this inspection are listed in the attachment.

These actions constitute completion of the requirements for Section 02.03.

b. Findings

No findings were identified.

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

The licensee did not perform any activities in this area. Therefore, the inspectors did not perform any inspections in this area.

.5 Identification and Resolution of Problems (71111.08-02.05)

a. Inspection scope

The inspection procedure requires review of a sample of problems associated with inservice inspections documented by the licensee in the corrective action program for appropriateness of the corrective actions. The inspectors reviewed several condition reports which dealt with inservice inspection activities and found the corrective actions were appropriate. The specific condition reports 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. Specific documents reviewed during this inspection are listed in the attachment.

These actions constitute completion of the requirements for Section 02.05.

b. Findings

No findings were identified.

1R11 Licensed Operator Requalification Program (71111.11)

a. Inspection Scope

On March 24, 2010, the inspectors observed a crew of licensed operators in the plant's simulator 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 preestablished operator action expectations and successful critical task completion requirements.

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

b. Findings

No findings were identified.

1R12 Maintenance Effectiveness (71111.12)

a. Inspection Scope

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

- Valve EFHV0037, Essential service water system leakage, Callaway Action Requests 20104390 and 201005146
- Non-maintenance rule valve BGHV0017 leakage affecting a maintenance rule scoped valve BGHV182, Callaway Action Requests 201004428 and 201004426. Callaway Action Request 201004426 documented a work process upgrade that had incorrectly removed Valve BGHV0017 from the database of scoped components

The inspectors reviewed events such as where ineffective equipment maintenance has resulted in valid or invalid automatic actuations of engineered safeguards systems and 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 10 CFR 50.65(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 two quarterly maintenance effectiveness samples as defined in Inspection Procedure 71111.12-05.

b. Findings

No findings 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:

- March 31, 2010, Yellow risk during emergent work on emergency diesel generator train A
- April 21, 2010, Yellow risk during planned drain down of reactor coolant system
- April 22, 2010, Elevated risk during planned removal of the reactor vessel head
- April 26, 2010, Yellow risk following reactor core offload when only one train of essential service water was available as a heat sink for the spent fuel pool
- May 11, 2010, Yellow risk during planned drain down of reactor coolant system and installation of reactor vessel head

- May 13, 2010, Yellow risk during planned reduction of the reactor coolant system level to mid-loop for the vacuum fill evolution

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

No findings were identified.

1R15 Operability Evaluations (71111.15)

a. Inspection Scope

The inspectors reviewed the following issues:

- April 28, 2010, Callaway Action Request 201003872, Turbine-driven auxiliary feedwater pump governor valve inboard guide bushing seating area too deep

- May 2, 2010, Callaway Action Request 201003971, Essential service water below minimum wall thickness flaw on line GN-45-HBC-14

- May 13, 2010, Callaway Action Request 201004541, Valve EJHV619 failure to open

- May 25, 2010, Callaway Action Request 20105149, Operation of the residual heat removal system while aligned to the reactor coolant system with temperatures greater than 240 degrees Fahrenheit

- May 26, 2010, Callaway Action Request 201005083, Turbine-driven auxiliary feedwater pump inboard bearing flinger ring found backed out of position

- June 8, 2010, Callaway Action Request 201005587, Void checks performed in the component cooling water system in response to operating experience

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 personnel'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 six operability evaluations inspection samples as defined in Inspection Procedure 71111.15-04

b. Findings

No findings were identified.

1R17 Evaluations of Changes, Tests, or Experiments and Permanent Plant Modifications (71111.17)

a. Inspection Scope

The inspectors reviewed the effectiveness of the licensee's implementation of evaluations performed in accordance with 10 CFR 50.59, "Changes, Tests, and Experiments," and changes, tests, experiments, or methodology changes that the licensee determined did not require 10 CFR 50.59 evaluations. The inspection procedure requires the review of 6 to 12 licensee evaluations required by 10 CFR 50.59, 12 to 25 changes, tests, or experiments that were screened out by the licensee and 5 to 15 permanent plant modifications.

The inspectors reviewed four evaluations required by 10 CFR 50.59; 15 changes, tests, and experiments that were screened out by licensee personnel; and six permanent plant modifications. Document numbers of the evaluations, changes, and modifications reviewed are listed in the attachment.

The inspectors verified that when changes, tests, or experiments were made, that evaluations were performed in accordance with 10 CFR 50.59 and that licensee personnel had appropriately concluded that the change, test or experiment can be accomplished without obtaining a license amendment. The inspectors also verified that safety issues related to the changes, tests, or experiments were resolved. The inspectors reviewed changes, tests, and experiments that licensee personnel determined did not require evaluations and verified that the licensee personnel's conclusions were correct and consistent with 10 CFR 50.59. The inspectors also verified that procedures, design, and licensing basis documentation used to support the changes were accurate after the changes had been made.

In the inspection of modifications the inspectors verified that supporting design and license basis documentation had been updated accordingly and was still consistent with the new design. The inspectors verified that procedures, training plans and other design basis features had been adequately accounted for and updated.

These activities constitute completion of one sample as defined in Inspection Procedure 71111.17-04.

b. Findings

No findings were identified.

1R18 Plant Modifications (71111.18)

.1 Temporary Modifications

a. Inspection Scope

To verify that the safety functions of important safety systems were not degraded, the inspectors reviewed the temporary modification identified as MP-08-0055 FCN 02, Remove SGN01AC2 from service during Refueling Outage 17.

On June 4, 2010, the inspectors completed review of the temporary modification 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 modification did not adversely affect the system operability/availability. The inspectors also verified that the installation and restoration were consistent with the modification documents and that configuration control was adequate. Additionally, the inspectors verified that the temporary modification was 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.

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

b. Findings

No findings were identified.

.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 permanent modifications listed below.

- May 3, 2010, MP-09-0076, Replace emergency diesel generator jacket water and lube oil heat exchangers with new design

- June 3, 2010, MP-10-0023, Install additional diesel generator with tie-in to transformer XNB02

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 modification design assumptions were appropriate; 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.

b. Findings

No findings were identified.

.3 Permanent Plant Modifications Associated with Temporary Instruction 2515/177, Managing Gas Accumulation in Emergency Core Cooling, Decay Heat Removal, and Containment Spray Systems

a. Inspection Scope

The following engineering design package associated with the scope of Generic Letter 2008-01 was reviewed and selected aspects were discussed with engineering personnel:

- May 10, 2010, MP-08-0016 FCN 3, Installation of vent valves on train B residual heat removal cold leg injection lines

The inspectors verified that the licensing basis verification documents have either been updated or are in the process of being updated to reflect the modifications associated with the licensee's resolution of Generic Letter 2008-01 (TI 2515/177, Section 04.01).

The verified documents included technical specifications, technical specification bases, Final Safety Analysis Report, and licensee controlled documents and bases.

In addition, the inspectors verified that the drawings were up-to-date with respect to recent hardware changes and that any discrepancies between as-built configurations and the drawings were documented and entered into the corrective action program for resolution (TI 2515/177, Section 04.02.a.6).

Similarly, the inspectors verified that piping and instrumentation diagrams accurately described the subject systems, that they were up-to-date with respect to recent hardware changes, and any discrepancies between as-built configurations, the isometric drawings, and the piping and instrumentation diagrams were documented and entered into the corrective action program for resolution (TI 2515/177, Section 04.02.b).

Documents reviewed are listed in the attachment to this report.

This inspection effort counts toward completion of TI 2515/177 which will be closed in a later inspection report (See Section 4OA5). These activities constitute completion of three samples for permanent plant modifications as defined in Inspection Procedure 71111.18-05.

b. Findings

No findings 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:

- April 4, 2010, Postmaintenance test of emergency diesel generator A following governor hydraulic actuator and repair of governor hydraulic pump drive shaft, Job 10002124
- April 6, 2010, Postmaintenance test of emergency diesel generator B for extent of condition inspection of its governor hydraulic pump drive shaft, Job 10002189
- April 29, 2010, Postmaintenance test of weld for new residual heat removal system vent valves EJ213 and EJ214, Job 08009444
- May 8, 2010, Postmaintenance test of emergency diesel generator B following heat exchanger replacements
- May 11, 2010, Postmaintenance test of containment isolation valves EFHV0048 and EFHV0050, Job W242069

- May 14, 2010, Postmaintenance test of motor-driven auxiliary feedwater system train A following automatic recirculation check valve maintenance
- May 17, 2010, Postmaintence testing of safety injection system following draining and system refill, Job 10506394
- May 19, 2010, Postmaintenance test of turbine-driven auxiliary feedwater pump casing repairs, Job 10003906
- June 9, 2010, Postmaintenance test of turbine-driven auxiliary feedwater pump governor valve steam leak repairs, Job 10003980
- June 18, 2010, Postmaintenance test of all control rods following likely debris induced damage to control rod R584, Job 08512631

The inspectors selected these activities based upon the structure, system, or component's ability to affect risk. The inspectors evaluated these activities for the following:

- 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 ten postmaintenance testing inspection samples as defined in Inspection Procedure 71111.19-05.

b. Findings

No findings were identified.

1R20 Refueling and Other Outage Activities (71111.20)

a. Inspection Scope

The inspectors reviewed the outage safety plan and contingency plans for the Callaway Plant Refueling Outage 17, conducted from April 17 to June 12, 2010, 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

- 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

- Controls for movement and tracking of fuel assemblies in the spent fuel pool and reactor cavity areas

- 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 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 and other outage inspection sample as defined in Inspection Procedure 71111.20-05.

b. Findings

Introduction. A Green self-revealing noncited violation of Technical Specification 5.4.1.a, "Procedures," was identified when the licensee's failure to control work activities during a reload of the reactor vessel fuel assemblies resulted in deenergization of all available source range nuclear instrument channels.

Description. On May 6, 2010, while in Mode 6 – Refueling, the licensee initiated Procedure ISL-SE-0N44B, "PR N44 Axial Flux Difference," to test nuclear instrument power range channel N44. Maintenance was in progress on 120 Vac instrument bus NN03 which caused the output of power range channel N43 to be sensed as greater than 10 percent power. Procedure ISL-SE-0N44B directed the instrument maintenance technicians to input a test signal causing the output of channel N44 to exceed an equivalent 10 percent power. The combination of deenergized instrument bus NN03 and power range channel N44 output being greater than 10 percent made up the logic for permissive P-10. One of the functions of permissive P-10 is to send a protective logic signal to ensure the source range nuclear instruments are deenergized. This prevents damaging the source range detector by operating in too high a neutron flux. Operator training lesson plans stated that no more than one power range channel may be tested/deenergized when shutdown or P-10 will deenergize both source range detectors. After recognizing the inadvertent loss of the source range monitoring capability, the control room directed the fuel handling crew to stop fuel movement per Technical Specification 3.9.3, "Nuclear Instrumentation," required action B.1. A fuel assembly was in the upender ready for transfer to the reactor vessel core location at the time.

Analysis. The performance deficiency associated with this finding was an inadequate procedure that allowed work on power range nuclear instruments when two operable source range channels were required. This finding is more than minor because it was associated with the configuration control 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, Appendix G, Attachment 1, "Shutdown Operations Significance Determination Process Phase 1 - Operational Checklists for Both PWRs and BWRs," this finding was determined to be 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 when needed, and did not degrade the licensee's ability to recover decay heat removal once lost. This finding had a crosscutting aspect in the area of human performance associated with the work control component because the licensee failed to coordinate work activities by incorporating actions to address the impact of the work on different job activities and communicate, coordinate, and cooperate with each other during activities in which interdepartmental coordination is necessary to assure plant and human performance [H.3(b)].

Enforcement. Technical Specification 5.4.1.a, "Procedures," required that written procedures be established, implemented and maintained covering the activities specified in Appendix A, "Typical Procedures for Pressurized Water Reactors," of Regulatory

Guide 1.33, “Quality Assurance Program Requirements,” February 1978. Appendix A, Item 3.t, required procedures for power range and source range nuclear instrument system operation. Contrary to the above, prior to May 6, 2010, Procedure ISL-SE-0N44B, “PR N44 Axial Flux Difference,” was not adequate for power range and source range nuclear instrument system operation because it did not contain any applicable prerequisites or initial conditions to prohibit its performance during refueling operations. As a result, on May 6, 2010, operations and instrument maintenance department personnel failed to ensure that no more than one power range channel was tested/deenergized when shutdown to prevent the P-10 interlock from deenergizing both source range detectors. Because this finding is of very low safety significance and was entered into the licensee's corrective action program as Callaway Action Request 201004301, this violation is being treated as a noncited violation, consistent with Section VI.A.1 of the NRC Enforcement Policy: NCV 05000483/2010003-02, “Failure to Maintain Two Operable Source Range Channels During Core Alterations.”

1R22 Surveillance Testing (71111.22)

a. Inspection Scope

1. The inspectors reviewed the Final Safety Analysis Report, procedure requirements, and technical specifications to ensure that the 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
- Annunciators and alarms setpoints

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

- April 17, 2010, Routine surveillance shutdown margin while subcritical, Job 10506157
- April 17, 2010, Routine surveillance reactor coolant system cooldown surveillance, PM 0913014
- April 17, 2010, Routine surveillance centrifugal charging pump B incapable of injection discharge path alignment surveillance, Job 09502047
- April 17, 2010, Routine surveillance safety injection pumps incapable of injection discharge path alignment surveillance, Job 09502048
- April 17, 2010, Routine Surveillance Procedure OSP-AB-V002A – stroke of atmospheric steam dumps, Job 08500878
- April 18, 2010, Routine Surveillance Procedure OSP-SA-2413B, Diesel generator train B and sequencer testing, Job 08512051
- May 1, 2010, Routine Surveillance Procedure OSP-NE-0024B, Standby diesel generator B 24-hour run and hot restart, Job 08510540
- May 9, 2010, Routine Surveillance Procedure OSP-NE-0024A, Standby diesel generator A 24-hour run and hot restart, Job 08510902
- May 11, 2010, Containment isolation valve overall cumulative local leak rate Surveillance Procedure ESP-SM-01001, Job 10506893
- May 12, 2010, Containment isolation valves BGHV8105 and BGHV8160 (penetrations 80 and 23) local leak rate tests, Jobs 07505181 and 06522087
- April 30, 2010, Inservice Surveillance Procedure OSP-EJ-V002A, Residual heat removal pump containment sump suction and refueling water storage tank suction inservice test - IPTE, Jobs 08511808/500 and 08004477/905

- May 19, 2010, Inservice test of the pressurizer power operated relief valves, Job 08512521

- May 19, 2010, Inservice test of the main steam isolation valves, Job 08512511

- May 20, 2010, Inservice test of the auxiliary feedwater pump discharge check valve, Job 08512160

- June 6, 2010, Reactor coolant system inventory balance to verify reactor coolant system leakage

2. Surveillance Testing Associated with Temporary Instruction 2515/177, Managing Gas Accumulation in Emergency Core Cooling, Decay Heat Removal, and Containment Spray Systems

When reviewing and observing performance of Procedure OSP-SA-00003, "Emergency Core Cooling System Flow Path Verification and Venting," on April 30, 2010, the inspectors verified that the fill and vent procedure was acceptable for testing train B of the residual heat removal system following refueling outage maintenance, and subject system modifications.

The inspectors reviewed the procedure used for conducting surveillances and determination of void volumes to ensure that the void criteria was satisfied and will be reasonably ensured to be satisfied until the next scheduled void surveillance (TI 2515/177, Section 04.03.a). Also, the inspectors reviewed the procedure used for filling and venting following conditions which may have introduced voids into the subject systems to verify that the procedures acceptably addressed testing for such voids and provided acceptable processes for their reduction or elimination (TI 2515/177, Section 04.03.b). Specifically, the inspectors verified that:

- Gas intrusion prevention, refill, venting, monitoring, trending, evaluation, and void correction activities were acceptably controlled by approved operating procedures (TI 2515/177, Section 04.03.c.1)

- Procedure ensured the system did not contain voids that may jeopardize operability (TI 2515/177, Section 04.03.c.2)

- Procedure established that void criteria were satisfied and will be reasonably ensured to be satisfied until the next scheduled void surveillance (TI 2515/177, Section 04.03.c.3)

- The licensee entered changes into the corrective action program as needed to ensure acceptable response to issues. In addition, the inspectors confirmed that a clear schedule for completion is included for corrective action program entries that have not been completed (TI 2515/177, Section 04.03.c.5)

- Procedure included independent verification that critical steps were completed (TI 2515/177, Section 04.03.c.6)

The inspectors verified the following with respect to surveillance and void detection:

- Specified surveillance frequencies were consistent with technical specification surveillance requirements (TI 2515/177, Section 04.03.d.1)

- Surveillance frequencies were stated or, when conducted more often than required by technical specifications, the process for their determination was described (TI 2515/177, Section 04.03.d.2)

- Surveillance method was acceptably established to achieve the needed accuracy (TI 2515/177, Section 04.03.d.3)

- Surveillance procedure included up-to-date acceptance criteria (TI 2515/177, Section 04.03.d.4)

- Procedure included effective follow-up actions when acceptance criteria are exceeded or when trending indicates that criteria may be approached before the next scheduled surveillance (TI 2515/177, Section 04.03.d.5)

- Measured void volume uncertainty was considered when comparing test data to acceptance criteria (TI 2515/177, Section 04.03.d.6)

- Venting procedure and practice utilized criteria such as adequate venting durations and observing a steady stream of water (TI 2515/177, Section 04.03.d.7)

- An effective sequencing of void removal steps was followed to ensure that gas does not move into previously filled system volumes (TI 2515/177, Section 04.03.d.8)

- Qualitative void assessment methods included expectations that the void will be significantly less than allowed by acceptance criteria (TI 2515/177, Section 04.03.d.9)

- Venting results were trended periodically to confirm that the systems are sufficiently full of water and that the venting frequencies are adequate. The inspectors also verified that records on the quantity of gas at each location are maintained and trended as a means of preemptively identifying degrading gas accumulations (TI 2515/177, Section 04.03.d.10)

- Whether surveillances were conducted at all location where a void may form, including high points, dead legs, and locations under closed valves in vertical pipes (TI 2515/177, Section 04.03.d.11)

- The licensee ensures that systems were not preconditioned by other procedures that may cause a system to be filled, such as by testing, prior to the void surveillance (TI 2515/177, Section 04.03.d.12)

- Procedure included gas sampling for unexpected void increases if the source of the void is unknown and sampling is needed to assist in determining the source (TI 2515/177, Section 04.03.d.13)

The inspectors verified the following with respect to filling and venting:

- Revisions to fill and vent procedure to address new vents or different venting sequences were acceptably accomplished (TI 2515/177, Section 04.03.e.1)

- Fill and vent procedure provided instructions to modify restoration guidance to address changes in maintenance work scope or to reflect different boundaries from those assumed in the procedure (TI 2515/177, Section 04.03.e.2)

The inspectors verified the following with respect to void control:

- Void removal methods were acceptably addressed by approved procedures (TI 2515/177, Section 04.03.f.1)(See Section 1R04)

This inspection effort counts towards the completion of TI 2515/177 which will be closed in a later inspection report. This surveillance for the emergency core cooling system venting is considered an additional routine surveillance.

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

These activities constitute completion of 9 routine surveillances, 2 containment isolation valve surveillances, 4 inservice tests and one reactor coolant system leak rate surveillance for a total of 16 surveillance testing inspection samples as defined in Inspection Procedure 71111.22-05.

b. Findings

A finding associated with monitoring of gas accumulation in the emergency core cooling system is described in Section 1R04 of this report.

2. RADIATION SAFETY

Cornerstone: Occupational and Public Radiation Safety

2RS01 Radiological Hazard Assessment and Exposure Controls (71124.01)

a. Inspection Scope

This area was inspected to: (1) review and assess licensee's performance in assessing the radiological hazards in the workplace associated with licensed activities and the implementation of appropriate radiation monitoring and exposure control measures for both individual and collective exposures, (2) verify the licensee is properly identifying and reporting Occupational Radiation Safety Cornerstone performance indicators, and (3) identify those performance deficiencies that were reportable as a performance indicator and which may have represented a substantial potential for overexposure of the worker. 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 walkdowns of various portions of the plant, performed independent radiation dose rate measurements and reviewed the following items:

- Performance indicator events and associated documentation reported by the licensee in the Occupational Radiation Safety Cornerstone

- The hazard assessment program, including a review of the license's evaluations of changes in plant operations and radiological surveys to detect dose rates, airborne radioactivity, and surface contamination levels

- Instructions and notices to workers, including labeling or marking containers of radioactive material, radiation work permits, actions for electronic dosimeter alarms, and changes to radiological conditions

- Programs and processes for control of sealed sources and release of potentially contaminated material from the radiologically controlled area, including survey performance, instrument sensitivity, release criteria, procedural guidance, and sealed source accountability

- Radiological hazards control and work coverage, including the adequacy of surveys, radiation protection job coverage, and contamination controls; the use of electronic dosimeters in high noise areas; dosimetry placement; airborne radioactivity monitoring; controls for highly activated or contaminated materials (non-fuel) stored within spent fuel and other storage pools; and posting and physical controls for high radiation areas and very high radiation areas

- Radiation worker and radiation protection technician performance with respect to radiation protection work requirements

- Audits, self-assessments, and corrective action documents related to radiological hazard assessment and exposure controls since the last inspection

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

These activities constitute completion of the one required sample as defined in Inspection Procedure 71124.01-05.

b. Findings

No findings were identified.

2RS02 Occupational ALARA Planning and Controls (71124.02)

a. <u>Inspection Scope</u>

This area was inspected to assess performance with respect to maintaining occupational individual and collective radiation exposures ALARA. 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 licensee personnel and reviewed the following items:

- Site-specific ALARA procedures and collective exposure history, including the current 3-year rolling average, site-specific trends in collective exposures, and source-term measurements

- ALARA work activity evaluations/postjob reviews, exposure estimates, and exposure mitigation requirements

- The methodology for estimating work activity exposures, the intended dose outcome, the accuracy of dose rate and man-hour estimates, and intended versus actual work activity doses and the reasons for any inconsistencies

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

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

- Audits, self-assessments, and corrective action documents related to ALARA planning and controls since the last inspection

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

These activities constitute completion of the one required sample as defined in Inspection Procedure 71124.02-05.

b. <u>Findings</u>

No findings were identified.

2RS03 In-plant Airborne Radioactivity Control and Mitigation (71124.03)

a. <u>Inspection Scope</u>

This area was inspected to verify that in-plant airborne concentrations are being controlled consistent with ALARA to the extent necessary to validate plant operations as reported by the PI and to verify that the practices and use of respiratory protection devices on-site do not pose an undue risk to the wearer. The inspectors interviewed licensee personnel and reviewed the following:

- The licensee's use, when applicable, of ventilation systems as part of its engineering controls

- The licensee's respiratory protection program for use, storage, maintenance, and quality assurance of NIOSH certified equipment, qualification and training of personnel, and user performance

- The licensee's capability for refilling and transporting SCBA air bottles to and from the control room and operations support center during emergency conditions, status of SCBA staged and ready for use in the plant and associated surveillance records, and personnel qualification and training

- Self-assessments, audits, corrective actions, and reports related to the respiratory protection program and devices

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

These activities constitute completion of the one sample as defined in Inspection Procedure 71124.03-05.

b. Findings

No findings were identified.

4. OTHER ACTIVITIES

4OA1 Performance Indicator Verification (71151)

.1 Data Submission Issue

a. Inspection Scope

The inspectors performed a review of the performance indicator data submitted by the licensee for the first quarter 2010 performance indicators for any obvious inconsistencies prior to its public release in accordance with Inspection Manual Chapter 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 were identified.

.2 Safety System Functional Failures (MS05)

a. Inspection Scope

The inspectors sampled licensee submittals for the safety system functional failures performance indicator for the period from the second quarter 2009 through the first quarter 2010. To determine the accuracy of the performance indicator data reported during those periods, the inspectors used definitions and guidance contained in NEI Document 99-02, "Regulatory Assessment Performance Indicator Guideline," Revision 6, and NUREG 1022, "Event Reporting Guidelines 10 CFR 50.72 and 50.73." The inspectors reviewed the licensee's operator narrative logs, operability assessments, maintenance rule records, maintenance work orders, issue reports, event reports, and NRC integrated inspection reports for the period of April 2009 through March 2010 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. Specific documents reviewed are described in the attachment to this report.

These activities constitute completion of one safety system functional failures sample as defined in Inspection Procedure 71151-05.

b. Findings

No findings were identified.

.3 Mitigating Systems Performance Index - Heat Removal System (MS08)

a. Inspection Scope

The inspectors sampled licensee submittals for the mitigating systems performance index - heat removal system performance indicator for the period from the second quarter 2009 through the first quarter 2010. To determine the accuracy of the performance indicator data reported during those periods, the inspectors used definitions and guidance contained in NEI Document 99-02, "Regulatory Assessment Performance Indicator Guideline," Revision 6. 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 April 2009 through March 2010 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. 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 were identified.

.4 Reactor Coolant System Leakage (BI02)

a. Inspection Scope

The inspectors sampled licensee submittals for the reactor coolant system leakage performance indicator for the period from the second quarter 2009 through the first quarter 2010. To determine the accuracy of the performance indicator data reported during those periods, the inspectors used definitions and guidance contained in NEI Document 99-02, "Regulatory Assessment Performance Indicator Guideline," Revision 6. 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 April 2009 through March 2010 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. 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 were identified.

.5 Occupational Exposure Control Effectiveness

a. Inspection Scope

Cornerstone: Occupational Radiation Safety

The inspectors reviewed performance indicator data for the second quarter 2009 through the first quarter 2010. The objective of the inspection was to determine the accuracy and completeness of the performance indicator data reported during these periods. The inspectors used the definitions and clarifying notes contained in NEI Document 99-02, "Regulatory Assessment Performance Indicator Guideline," Revision 6, as criteria for determining whether the licensee was in compliance.

The inspectors reviewed corrective action program records associated with high radiation area (greater than 1 R/hr) and very high radiation area non-conformances. The inspectors reviewed radiological, controlled area exit transactions greater than 100 millirems. The inspectors also conducted walkdowns of high radiation areas (greater than 1 R/hr) and very high radiation area entrances to determine the adequacy of the controls of these areas.

These activities constitute completion of the occupational exposure control effectiveness sample as defined in Inspection Procedure 71151-05.

b. Findings

No findings were identified.

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

a. Inspection Scope

Cornerstone: Public Radiation Safety

The inspectors reviewed performance indicator data for the second quarter 2009 through the first quarter 2010. The objective of the inspection was to determine the accuracy and completeness of the performance indicator data reported during these periods. The inspectors used the definitions and clarifying notes contained in NEI Document 99-02, "Regulatory Assessment Performance Indicator Guideline," Revision 6, as criteria for determining whether the licensee was in compliance.

The inspectors reviewed the licensee's corrective action program records and selected individual annual or special reports to identify potential occurrences such as unmonitored, uncontrolled, or improperly calculated effluent releases that may have impacted offsite dose.

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 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 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 were identified.

.3 Semi-Annual Trend Review

a. Inspection Scope

The inspectors performed a review of the licensee's 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 January through June 2010 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.

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

b. Observations and Findings

The inspectors found that the licensee did identify the following trends of significance:

- Callaway Action Request 201001217, Emerging trend in untimely reporting of changes in work hours and Callaway Action Request 201004453, Adverse trend in working hour limitations

- Callaway Action Request 201002916, Adverse trend in use of non-OEM gaskets in safety related structures, systems and components

- Callaway Action Request 201004669, Adverse trend in foreign material exclusion controls

- Callaway Action Request 201005158, Adverse trend in maintenance worker practices in turbine-driven auxiliary feedwater pump maintenance

The resident inspectors concurred with these items as being noteworthy trends needing additional corrective actions. Additionally the inspectors noted adverse trends in:

- NRC performance indicator reporting difficulties. Specifically, the inspectors observed that the licensee had submitted incorrect component demand estimates which results in an artificially low index value. In addition to these errors, the licensee discovered incorrect system availability accounting. Once corrected, the net results of these errors was a higher index value but still below the green-white threshold. In the case of the turbine-driven auxiliary feedwater pump, the results were impacted two 2009 failures of the pump. The licensee plans to provide an update to the performance indicator for this component at the conclusion of the second quarter 2010 showing the mitigating systems performance index for the heat removal system at the green-white threshold of 1.0 E-6.

- Work control scheduling of corrective actions and coordination of interdepartmental maintenance activities

No findings 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:

- Callaway Action Request 201003580, Corrective actions to address extent of condition concerns for outside diameter stress corrosion cracking in alternate charging, excess letdown, and auxiliary pressurizer spray systems

- Callaway Action Request 201005424, Corrective actions associated with operating experience at a similar plant that experienced voiding in the component cooling water system following a refueling outage

- Callaway Action Request 201005096, Control bank C control rod R584 damaged and stuck during rod drop testing

These activities constitute completion of three selected follow-up inspection samples as defined in Inspection Procedure 71152-05.

b. Findings

No findings were identified.

.5 In-depth Review of Operator Workarounds

a. Inspection Scope

During a review of items entered in the licensee's corrective action program, the inspectors reviewed an operator workaround associated with:

- Callaway Action Request 201001813, Nonconservative essential service water calculation not addressing the limiting single active failure. This included inspector review of the operator workaround compensatory measure.

This activity constitutes completion of one operator workaround inspection sample as defined in Inspection Procedure 71152-05.

b. Findings

No findings were identified.

4OA3 Event Follow-up (71153)

.1 Inadequate Equipment Control Procedure Resulted in Loss of Volume Control Tank Inventory

a. Inspection Scope

The inspectors reviewed the circumstances and the operator response to the volume control tank level transient that occurred on April 13, 2010.

b. Findings

Introduction. The inspectors identified a Green finding associated with the licensee's failure to take prompt corrective actions for leaking boundary valves in the chemical and volume control system. The lingering equipment problems resulted in an unplanned 25 gallon per minute loss of volume control tank inventory and an emergency action level declaration for excessive reactor coolant system leakage that was later retracted.

Description. On April 13, 2010, an attempt to place the train A chemical and volume control system mixed bed in service resulted in leakage past documented leaking train A chemical and volume control system mixed bed demineralizer drain valve BGV0237. The event resulted in a 25 gallon per minute loss of water inventory from the volume control tank. Previous operating experience documented in Callaway Action Requests 200700528 and 200702968 identified that the plant continued to experience problems with seat leakage past manually operated diaphragm valves associated with the chemical and volume control demineralizers. Several instances of seat leakage had resulted in loss of volume control tank level. Prior to 2007, the demineralizer tank boundary valves had been caution carded as "leaking valves." Adverse trend Callaway Action Request 200700525 designated as significance level 3, required as corrective action CA# 4, that each valve identified as leaking have a job initiated to replace the valve diaphragm and O-rings. Both the adverse trend and the effectiveness Callaway action requests were closed out by July 2009. The completion date for valve BGV0237 was deferred several times and was finally scheduled to be complete on January 2, 2010. This did not occur. On March 19, 2010, new resin was placed in the train A chemical and volume control mixed bed and the tank was left filled and vented. There was no knowledge or discussion of the leaking valve history in the April 13, 2010, prejob brief. There was no walkdown of the system or check that the bed was ready for alignment to the chemical and volume control system. Callaway Procedure APA-ZZ-00500, Appendix 14, "Adverse Condition - Significance Level 3," required that for Callaway action request actions being transferred to a job, the equipment related deficiencies and issues documented in the Callaway action request must be completely covered in the job/task description. This did not prevent the repeated rescheduling and use of the degraded valve prior to placing the demineralizer bed in service. The immediate corrective action was to isolate the leaking drain valve. Subsequently, the licensee repaired the drain valve.

Analysis. Failure to take actions prescribed in corrective action documents to prevent loss of volume control tank inventory is a performance deficiency associated with licensee Procedure APA-ZZ-00500, “Corrective Action Program.” This finding is more than minor because it was associated with the reactor safety Initiating Events Cornerstone attribute of configuration control and affects the objective to limit the likelihood of events that upset plant stability. Using Manual Chapter 0609.04, “Phase 1 - Initial Screening and Characterization of Findings,” the inspectors determined that this finding is only of very low significance because the condition did not result in the reactor coolant system technical specification leakage limit being exceeded, did not contribute to both the likelihood of a reactor trip and the likelihood that mitigating equipment or functions would be unavailable, and did not increase the likelihood of a fire or internal/external flood. This finding, which involved inadequate scheduling of corrective action related jobs, has a crosscutting aspect in the area of human performance associated with the work control component because AmerenUE did not appropriately coordinate work activities to address the impact of the work on different job activities [H.3(b)].

Enforcement. Enforcement action does not apply because the performance deficiency did not involve a violation of regulatory requirements. This finding is of very low safety significance and the issue was entered into the licensee's corrective action program as Callaway Action Request 201003146: FIN 05000483/2010003-03, “Failure to Ensure Completion of Corrective Actions for Degraded Chemical and Volume Control System Valves.”

.2 Callaway Plant Notice of Enforcement Discretion (NOED No. 10-4-001, EA 2010-065) for Technical Specification 3.8.1 - AC Sources – Operating, Required Actions B.4

a. Inspection Scope

The inspectors reviewed the circumstances leading to the licensee’s April 2, 2010, request that the NRC exercise discretion to not enforce compliance with the actions required in Callaway Plant, Technical Specification 3.8.1, “AC Sources – Operating,” due to the failure of the train A emergency diesel generator. The licensee’s request for a notice of enforcement discretion was granted on April 2, 2010, which allowed the licensee an additional 48 hours to repair the diesel engine’s governor drive assembly.

b. Findings

Introduction. The inspectors identified a self-revealing Green noncited violation of 10 CFR Part 50, Appendix B, Criterion III, “Design Control,” after the licensee failed to select suitable replacement gaskets essential to the operation of emergency diesel generator train A.

Description. On March 30, 2010, during performance of Procedure OSP-NE-00024A, “Standby Diesel Generator A 24-Hour Run and Hot Restart Test,” the emergency diesel generator train A unexpectedly lost speed and tripped after about 16.7 hours of operation. Posttrip indications revealed that the diesel generator tripped on reverse power. Troubleshooting by the licensee revealed that a stripped splined shaft caused the diesel engine’s governor drive to fail. Disassembly of the failed drive revealed the governor overspeed base to drive assembly gasket did not have the required oil port hole to allow proper lubrication of the drive assembly. The gasket found during disassembly was not an original equipment manufacturer part and had been field cut and installed on October 11, 1999, under Work Request W646151. Because of the time required to repair and retest the failed governor drive assembly, on April 2, 2010, the licensee requested that the NRC exercise discretion to not enforce compliance with the specified completion time for Technical Specification 3.8.1 “AC Sources – Operating,” Required Actions B.4. Following the licensee’s request, the NRC granted a notice of enforcement discretion which allowed the licensee an additional 48 hours to repair the governor drive assembly. Repairs and retests of the governor drive assembly were complete on April 4, 2010.

Hardware failure analysis performed by the licensee with input from the diesel vendor confirmed that the lack of lubricating oil caused by the improperly cut gasket resulted in the failure of the splined shaft that couples the governor actuator to the diesel engine. An extent of condition review for the emergency diesel generator train B revealed that engine had an original equipment manufacturer gasket installed. Proper oil flow to the train B emergency diesel generator governor drive was verified by inspection on April 5, 2010. Corrective actions to prevent recurrence by the licensee included revising work instructions to ensure the oil port for the governor overspeed base to drive assembly gasket is properly aligned. Additional corrective actions included adding a quality assurance inspection point to the work instruction for installation of the gasket.

Analysis. The performance deficiency associated with this finding involved the licensee’s failure to adequately select and review the suitability of replacement parts essential to the operation of emergency diesel generator train A. This finding was more than minor because it was associated with the Mitigating Systems Cornerstone attribute of design control and affects the associated cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. The resident inspectors performed the initial significance determination for the diesel gasket finding using the NRC Inspection Manual 0609, Attachment 0609.04, “Phase 1 – Initial Screening and Characterization of Findings.” The finding screened to a Phase 2 significance determination because it involved the loss of one train of safety related equipment for greater than its technical specification allowed outage time. A Region IV senior reactor analyst performed a Phase 2 significance determination using the pre-solved worksheet from the “Risk Informed Inspection Notebook for Callaway Nuclear Generating Station,” Revision 2.01a. The analyst assumed an exposure period of one year. The finding was potentially Yellow, which warranted further review. The senior reactor analyst subsequently performed a bounding Phase 3 significance determination and found the finding to be of very low safety significance (Green). The dominant cutsets included a loss of offsite power initiating event, failure to recover offsite power in 4 hours, failure of the train B emergency diesel generator, and a reactor coolant pump seal failure. Equipment that mitigated the significance included the operable emergency diesel generator and the turbine-driven auxiliary feedwater pump. The results of the senior reactor analyst’s review are included as Attachment 2.

This finding did not have a crosscutting aspect since it was not a performance deficiency reflective of current licensee performance.

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 and parts that are essential to the safety related functions of structures, systems, and components. Contrary to the above, from October 11, 1999, through March 30, 2010, the licensee failed to ensure the suitability of repair parts essential to the safety-related function of emergency diesel generator train A. Because this violation is of very low safety significance and has been entered into the licensee's corrective action program as Callaway Action Request 201002675, this violation is being treated as a noncited violation, consistent with Section VI.A.1 of the NRC Enforcement Policy: NCV 05000483/2010003-04. “Failure to Correctly Fabricate Replacement Gasket for Emergency Diesel Generator Train A” (EA-10-065).

.3 (Closed) Licensee Event Report 05000483/2010-001-00 and 05000483/2010-001-01, Emergency Core Cooling System Mode 4 Operating Practices Prohibited by Current Technical Specification 3.5.3

On January 22, 2010, Callaway Plant identified that Procedure OTO-BB-00010, “Shutdown LOCA,” did not instruct operators to align the residual heat removal cross-tie valves to supply emergency core cooling flow to each of the four reactor coolant system cold leg injection nozzles. This capability is required in the event of a loss of coolant accident during Mode 4 per Plant Technical Specification 3.5.3, “Emergency Core Cooling System (ECCS) Shutdown.” This procedure was the approved procedure for Mode 4 operations during Refueling Outage 16 (October 2008). Prior to that time, the licensee had no specific Mode 4 loss of coolant accident procedure. As corrective action, the licensee’s Mode 4 loss of coolant accident procedure was revised to ensure that the required residual heat removal low head emergency core cooling system subsystem is capable of being aligned to inject into the reactor coolant system as an injection source via the four cold leg injection nozzles when plant conditions necessitate. This event is reportable as a condition prohibited by technical specifications and as a condition that could have prevented fulfillment of a safety function due to a procedural deficiency. The inspectors reviewed the licensee’s most recent submittal and determined that the report adequately documented the summary of the event including the potential safety consequences and corrective actions required to address the performance deficiency. The inspectors reviewed a licensee-identified violation of Technical Specification 3.5.3, “Emergency Core Cooling System (ECCS) Shutdown.” The enforcement aspects of the violation are discussed in Section 4OA7 of this report. This licensee event report is closed.

4OA5 Other Activities

.1 NRC TI 2515/177, Managing Gas Accumulation in Emergency Core Cooling, Decay Heat Removal, and Containment Spray Systems NRC Generic Letter 2008-01

As documented in Section 1R04 and 1R22, the inspectors confirmed the acceptability of the described licensee’s actions. This inspection effort counts towards the completion of TI 2515/177 which will be closed in a later inspection report.

.2 (Closed) Temporary Instruction 2515/179, Verification of Licensee Responses to NRC Requirement for Inventories of Materials Tracked in the National Source Tracking System Pursuant to Title 10, Code of Federal Regulations, Part 20.2207 (10 CFR 20.2207)

a. Inspection Scope

An NRC inspection was performed to confirm that the licensee has reported their initial inventories of sealed sources pursuant to 10 CFR 20.2207 and to verify that the National Source Tracking System database correctly reflects the Category 1 and 2 sealed sources in custody of the licensee. Inspectors interviewed personnel and performed the following:

- Reviewed the licensee's source inventory
- Verified the presence of any Category 1 or 2 sources
- Reviewed procedures for and evaluated the effectiveness of storage and handling of sources
- Reviewed documents involving transactions of sources
- Reviewed adequacy of licensee maintenance, posting, and labeling of nationally tracked sources

b. Findings

No findings were identified.

.3 (Closed) Temporary Instruction 2515/172, Reactor Coolant System Dissimilar Metal Butt Welds

Temporary Instruction 2515/172 was previously performed at the Callaway Plant during Refueling Outage 16. The results of this inspection are documented in Inspection Report 05000483/2008005. Activities that were completed prior to the current refueling outage (Refueling Outage 17) are specifically indicated in the discussion below.

a. Inspection Scope

Portions of Temporary Instruction 2515/172 were performed at the Callaway Plant, during Refueling Outage 17, which took place in the spring of 2010. In the discussion below, NRC inspection activities not otherwise specified were completed during Refueling Outage 17 and are documented in this inspection report. Specific documents reviewed during this inspection are listed in the attachment. This unit has the following dissimilar metal butt welds:

- One 14-inch pressurizer surge line nozzle was mitigated during Refueling Outage 15 using a weld overlay process, and was categorized as Category F following the weld overlay process

- Three 6-inch pressurizer safety nozzles were mitigated during Refueling Outage 15 using a weld overlay process, and all were categorized as Category F after the weld overlay

- One 4-inch pressurizer spray nozzle was mitigated during Refueling Outage 15 using a weld overlay process, and was categorized as Category F after the weld overlay

- One 6-inch pressurizer pressure operated relief valve nozzle was mitigated during Refueling Outage 15 using a weld overlay process, and was categorized as Category F after the weld overlay

- Four 32-inch hot leg nozzles. These welds have not been remediated and are categorized as Category D

- Four 34-inch cold leg nozzles. These welds have not been remediated and are categorized as Category E

Licensee's Implementation of the Materials Reliability Program (MRP-139) Baseline Inspections (03.01)

MRP-139 baseline inspections: The inspectors reviewed records of structural weld overlays and nondestructive examination activities associated with the licensee's pressurizer structural weld overlay mitigation effort. These baseline inspections of the pressurizer dissimilar metal butt welds, including an assessment of deviations from MRP-139, if any, were completed during Refueling Outage 16 and are documented in Inspection Report 05000483/2008005.

Volumetric Examinations (03.02)

The licensee performed volumetric examinations of the pressurizer dissimilar metal butt welds, after remediation by full structural weld overlay, during Refueling Outage 15 and Refueling Outage 16. NRC inspection activities are documented in Inspection Report 05000483/2008005.

Weld Overlays (03.03)

The licensee performed full structural overlay welds on all six pressurizer dissimilar metal butt welds during Refueling Outage 15. NRC inspection activities are documented in Inspection Report 05000483/2008005.

Mechanical Stress Improvement (03.04)

The licensee has not employed a mechanical stress improvement process.

Inservice Inspection Program (03.05)

The licensee has prepared an MRP-139 inservice inspection program. All the welds in the MRP-139 inservice inspection program are appropriately categorized in accordance with MRP-139. The inservice inspection frequencies are consistent with the inservice inspection frequencies called for by MRP-139.

b. Findings

No findings were identified.

.4 (Closed) Callaway Plant Notice of Enforcement Discretion for Technical Specification 3.8.1 – AC Sources – Operating, Required Action B.4 (NOED Number 10-4-001, EA-10-065)

The inspectors reviewed the circumstances leading to NOED Number 10-4-001. This review resulted in documentation of a self-revealing noncited violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control" (See Section 4OA3). No other enforcement issues were noted.

4OA6 Meetings

Exit Meeting Summary

On April 23, 2010, the inspectors debriefed the inservice inspection results to Mr. A. Heflin, Senior Vice President and Chief Nuclear Officer, and other members of the licensee staff. The licensee acknowledged the issues presented. The inspectors acknowledged review of proprietary material during the inspection which had been or will be returned to the licensee.

On April 30, 2010, the inspectors presented the results of the radiation safety inspection to Mr. A. Heflin, Senior Vice President and Chief Nuclear Officer, 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.

The inspectors briefed Mr. F. Diya, Vice President, Nuclear Operations, Mr. C. Reasoner, Vice President, Engineering, and other members of the licensee's staff, on the results of the inspection on June 17, 2010. The licensee acknowledged the findings presented. The inspectors asked the licensee whether any materials examined during the inspection should be considered proprietary. Proprietary information had been provided and will be treated in accordance with NRC governing procedures.

On June 22, 2010, the inspectors presented the resident integrated inspection results to Mr. F. Diya, Vice President Nuclear Operations, 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 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.

- Technical Specification 3.5.3, "Emergency Core Cooling System (ECCS) Shutdown," requires, in part, that in Mode 4 one emergency core cooling system train shall be operable. The operable emergency core cooling system train must provide a flow path from the refueling water storage tank to the reactor coolant system via each of the four cold leg injection nozzles. Contrary to the above, on January 22, 2010, Callaway Plant identified that Procedure OTO-BB-00010, "Shutdown LOCA," did not instruct operators to align the residual heat removal cross-tie valves to supply emergency core cooling system flow to each of the four reactor coolant system cold leg injection nozzles. This finding was entered in the licensee's corrective action program as Callaway Action Request 201000601. 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. The inspectors performed the initial significance determination for the failure to translate technical specification requirements into operating procedures using NRC Inspection Manual 0609, Attachment 0609.04, "Phase 1 – Initial Screening and Characterization of Findings." The finding screened to a Phase 2 significance determination, using NRC Inspection Manual 0609, Appendix G, "Shutdown Operations Significance Determination," because the finding affected the licensee's ability to add reactor coolant system inventory when needed. A Region IV senior reactor analyst performed a Phase 2 significance determination using Inspection Manual 0609, Appendix G. In this particular case, the finding only involved the failure to align all four injection pathways to the discharge of the residual heat removal pump that was aligned for coolant injection. Two pathways were still available. This alignment, while not consistent with technical specification requirements, was adequate to meet the probabilistic safety function in the significance determination process. Consequently, there was no quantifiable change in risk using the significance determination process worksheets. The applicable worksheets included those for loss of inventory and loss of level control. Therefore, the finding was of very low safety significance (Green).

- Technical Specification 5.4.1.a, "Procedures," requires, in part, that written procedures be established, implemented, and maintained as recommended by Regulatory Guide 1.33, Revision 2, Appendix A, February 1978. Appendix A, Section 3.d, recommends procedures for operation of the emergency core cooling system. Contrary to this requirement, on May 25, 2010, Procedure OTG-ZZ-00006, "Plant Cooldown Hot Standby to Cold Shutdown," did not contain sufficient detail for operators to determine if a residual heat removal system aligned to the reactor coolant system with temperatures greater than 240 degrees Fahrenheit could satisfy the requirements of Technical Specification 3.5.3, "ECCS – Shutdown." This finding was entered in the licensee's corrective action program as Callaway Action Request 201005149. 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. Using Manual Chapter 0609.04, "Phase 1 – Initial Screening and Characterization of Findings," the issue screened as very low safety significance (Green) 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 the technical specification allowed outage time and did not affect seismic, flooding, or severe weather initiating events.

- Technical Specification 3.3.2, Function 6.g., "Engineered Safety Feature Actuation System (ESFAS) Instrumentation," requires, in part, that two channels of auxiliary feedwater actuation upon trip of all main feedwater be operable in Modes 1 and 2. Contrary to the above, on February 19, 2010, Callaway Plant reviewed industry operating experience and identified that the actuation logic for the start of the motor-driven auxiliary feedwater pumps on a trip of all main feedwater pumps could not be satisfied when one main feedwater pump is operating and the second main feedwater pump is secured and reset. A review by the licensee discovered eleven times in the past three years that Callaway was in the condition where one main feedwater pump is operating and the second main feedwater pump is secured and reset. In each case, the occurrence constituted a violation of Technical Specifications 3.3.2. This finding was entered in the licensee's corrective action program as Callaway Action Request 201000574. This finding is greater than minor because it was associated with the configuration control 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. Using Manual Chapter 0609.04, "Phase 1 – Initial Screening and Characterization of Findings," the issue screened to a Phase 2 significance determination because it involved an actual loss of safety function. A Region IV senior reactor analyst performed a Phase 2 significance determination using the pre-solved worksheet from the "Risk Informed Inspection Notebook for Callaway Nuclear Generating Station," Revision 2.01a. The analyst noted that this particular finding only affected the auxiliary feedwater pump automatic start feature in response to a trip of both main feedwater pumps (when one pump was in reset). Specifically, if one main feedwater pump tripped and the other was in a reset mode (not operating) the function would not work. However, this feature was not credited in the plant's safety analysis and the other auto-start circuits remained fully functional (lo-lo steam generator water level, anticipated transient without scram, and the loss of coolant accident sequencer or shutdown sequencer, and undervoltage on either 4160 Vac safety bus (turbine-driven pumps only)). The manual start feature was also still available. The Phase 2 worksheets only consider the ability to perform the safety function (not specifically which automatic feature is used to accomplish the function). Using the Phase 2 worksheets, no change in the quantifiable risk was obtained. Therefore, the finding was of very low safety significance (Green).

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee Personnel

K. Austgen, Engineer
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F. Diya, Vice President Nuclear Operations
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C. Graham, Health Physicist, Radiation Protection
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B. Huhmann, Supervisory Engineer
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S. Petzel, Engineer, Regulatory Affairs
C. Reasoner, Vice President Engineering
A. Schnitz, Regulatory Affairs
D. Thompson, Health Physicist, Radiation Protection
S. Thomure, Welding, R&R Engineer
D. Trokey, Regulatory Affairs Specialist
R. Wishau, Health Physicist, Radiation Protection

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened and Closed

05000483/2010003-01	NCV	Inadequate Surveillance Procedure to Verify and Maintain Emergency Core Cooling System Operable (Section 1R04)
05000483/2010003-02	NCV	Failure to Maintain Two Operable Source Range Channels During Core Alterations (Section 1R20)
05000483/2010003-03	FIN	Failure to Ensure Completion of Corrective Actions for Degraded Chemical and Volume Control System Valves (Section 4OA3)
05000483/2010003-04	NCV	Failure to Correctly Fabricate Replacement Gasket for Emergency Diesel Generator Train A (Section 4OA3)

Closed

05000483/2010-001-00 05000483/2010-001-01	LER	Emergency Core Cooling System Mode 4 Operating Practices Prohibited by Current Technical Specification 3.5.3 (Section 4OA3)
2515/179	TI	Verification of Licensee Responses to NRC Requirement for Inventories of Materials Tracked in the National Source Tracking System Pursuant to Title 10, Code of Federal Regulations, Part 20.2207 (10 CFR 20.2207) (Section 4OA5)
2515/172	TI	Reactor Coolant System Dissimilar Metal Butt Welds (Section 4OA5)

Discussed

2515/177	TI	Managing Gas Accumulation in Emergency Core Cooling Decay Heat Removal, and Containment Spray Systems NRC Generic Letter 2008-01 (Section 4OA5)

LIST OF DOCUMENTS REVIEWED

Section 1RO1: Adverse Weather Protection

PROCEDURES

NUMBER	TITLE	REVISION / DATE
OTO-ZZ-00012	Severe Weather	20

Section 1RO4: Equipment Alignment

PROCEDURES

NUMBER	TITLE	REVISION
ESP-EF-0002B	Essential Service Water Train B Flow Verification	14
ODP-ZZ-00004	Locked Component Control	37
OSP-BB-00003	PORV/RHR Coms Alignment Verification	13
OTG-ZZ-00002	Reactor Startup – IPTE	40-44
OTG-ZZ-00005	Plant Shutdown 20% Power to Hot Standby	32-36

OTG-ZZ-00005, Addendum 1	Opening Reactor Trip Breakers in Mode 2 - IPTE	1-3
OTO-MA-0008	Rapid Load Reduction	11
OTO-MA-0008	Rapid Load Reduction	20
OTS-FC-00006	TD AFW Pump Post-Maintenance Test Run on Aux Steam	0

DRAWINGS

NUMBER	TITLE	REVISION
M-22BB01(Q)	Piping and Instrumentation Diagram Reactor Coolant System	30
M-22EF01(Q)	Piping and Instrumentation Diagram Essential Service Water System	72
M-22EF02(Q)	Piping and Instrumentation Diagram Essential Service Water System	71
M-22EG02(Q)	Piping and Instrumentation Diagram Component Cooling Water System	19
J-24BB01(Q)	Instrument Isometric Drawing Reactor Coolant Loop 1 Crossover Leg	7
J-24BB04(Q)	Instrument Isometric Drawing Reactor Coolant Loop 2 Crossover Leg	3
J-24BB07(Q)	Instrument Isometric Drawing Reactor Coolant Loop 3 Crossover Leg	1
J-24BB10(Q)	Instrument Isometric Drawing Reactor Coolant Loop Crossover Leg	4
M-22AB02(Q)	Piping and Instrumentation Diagram Main Steam System	15

CALLAWAY ACTION REQUESTS

201003364	201002240	201003492	201004885	20104078

JOBS

09502047	09502048

MISCELLANEOUS DOCUMENTS

NUMBER	TITLE	REVISION
Calculation BB-74	Flow Tubing Analysis	0
Calculation BB-74, Addenda 1	BB Instrument Tubing Re-routing – Stress Analysis	0
Calculation BB-74, Addenda 2	RCS Loop Seismic Anchor Movement Changes	0

Section 1RO5: Fire Protection

CALLAWAY ACTION REQUESTS

201005426

Section 1R07: Heat Sink Performance

PROCEDURES

NUMBER	TITLE	REVISION
APA-ZZ-00500, Appendix 4	Transient Evaluation (TRAN)	2
APA-ZZ-00542	Event Review and Post Transient Evaluation	15
APA-ZZ-01025	Raw Water Systems Control Program	0
APA-ZZ-01025, Appendix A	Raw Water Chemistry Strategic Optimization Plan	0
CDP-ZZ-00200, Appendix D	Closed Cooling Systems Tables	9
E-1	Loss of Reactor or Secondary Coolant	13
EDP-EF-UHS01	UHS Cooling Tower Operational Guidance Following a LOCA	3
EDP-ZZ-01007	Mechanical Snubber Program	22
EDP-ZZ-01110	Predictive Maintenance Program	21
EDP-ZZ-01112	Heat Exchanger Predictive Performance Manual	16

ETP-ZZ-03001	GL89-13 Heat Exchanger Inspection	7
OSP-EF-0003A	Train A UHS Cooling Tower Fans Test	9
OSP-EF-0003B	Train B UHS Cooling Tower Fans Test	8
OSP-EF-P001A	ESW Train A Inservice Test	62
OSP-EJ-P001A	RHR Train A Inservice Test – Group A	48
OSP-EJ-PV04A	Train A RHR and RCS Check Valve Inservice Test – IPTE	3
OTA-RK-00016, Addendum 30E	NG07 Bus Undervoltage / Overvoltage	1
OTA-RK-00016, Addendum 31E	NG08 Bus Undervoltage / Overvoltage	1
EOP, Addendum 17	Securing ESW Train Due to UHS Cooling Tower Trouble	0
ESP-EF-0001A	Ultimate Heat Sink Train A Cooling Tower Fill Inspection	3
ESP-EF-0001B	Ultimate Heat Sink Train B Cooling Tower Fill Inspection	3
ESP-EF-03002	Ultimate Heat Sink Retention Pond In-Service Inspection	6
ESP-ZZ-01013	Maintenance Rule Walkdown Report Structure UHS Cooling Tower	1
ETP-EG-00001	Component Cooling Water Heat Exchanger Test	9
ETP-EG-00002	Component Cooling Water System Flow Verification	13
ETP-EG-00003	Thermal Performance Test of the ‘A’ CCW Heat Exchanger	0
OTA-RK-00020, Addendum 54A	ESW A Pressure Low / Flow Low	2
OTA-RK-00020, Addendum 54E	UHS Cooling Tower Trouble	1

OTA-RK-00020, Addendum 55A	ESW B Pressure Low / Flow Low	1
OTA-RK-00020, Addendum 55B	ESW Pump Trouble	0
OTA-RK-00020, Addendum 55C	ESW Strainer Differential Pressure High	1
OTA-RK-00020, Addendum 55D	UHS Pond Level High or Low	1
OTN-EF-00001, Addendum 1	ESW Train A – Fill and Vent	3
OTN-EF-00001, Addendum 2	ESW Train B – Fill and Vent	3
OTN-EG-00001	Component Cooling Water System	43
OTN-EG-00001, Addendum 2	Supplying CCW to Idle Train Safety Loads	2
OTN-EJ-00001	Residual Heat Removal System	25
OTN-EJ-00001, Addendum 3	Placing A RHR Train in Service for RCS Cooldown	14
OTN-EJ-00002	Fill and Vent of the RHR System	6

<u>CALCULATIONS</u>

<u>NUMBER</u>	<u>TITLE</u>	<u>REVISION / DATE</u>
08-176	Thermal Performance Test Data Evaluation and Uncertainty Analysis for CCW Heat Exchangers EEG01A and EEG01B	A
EF-54	Ultimate Heat Sink Thermal Performance Analysis	3
EG-20	Max. CCW Temp. During Post-LOCA	July 14, 1983
EJ-12	Comparison of RHR System Heat Removal Capability vs. Core Decay Heat After Shutdown	0

EG-42	Calculate the number of Tubes that can be Plugged	0

DRAWINGS

NUMBER	TITLE	REVISION / DATE
5736	Vertical Residual H.E. Outline Drawing	6
C-U101	Ultimate Heat Sink Retention Pond Plan & Sections	6
C-U102 (Q)	Ultimate Heat Sink discharge & Outlet Strictures Plans & Sections	6
M-015-U003	UHS Cooling Tower Cross Section and Key Plan for 2 at M94-3740	7
M-015-U0049	Instruction Manual for Installation, Operation, Maintenance, and Storage of Ultimate Heat Sink Cooling Towers	12
M-22EJ01(Q)	Piping and Instrumentation Diagram – Residual Heat Removal System	59
M-072-0001	Setting Plan for Component Cooling Water Heat Exchangers 76” I.D. X 37’ -0” Tube Lgth.	17
M-072-022	Component Cooling Water Heat Exchangers Appendix A	1
M-1055-00203	Component Cooling Water Heat Exchanger A EEG01A	0
M-1055-00206	Residual Heat Removal Heat Exchanger B EEJ01B	0
M-072-022-01	Heat Exchanger Data Sheet SNUPPS Component Cooling Water Heat Exchangers Appendix A	5
M-22-EG01 (Q)	Piping and Instrumentation Diagram Component Cooling Water System	9
M-22-EG02 (Q)	Piping and Instrumentation Diagram Component Cooling Water System	19
51-812-613-404	Outline: T.E.F.C. Ball Brg Motor C.I. Cond 80x	3

DRAWINGS

NUMBER	TITLE	REVISION / DATE
	w/10.0 Drop Aud Aul.Cond. Box (attached) – SP.Htrs-Grd Bolt Std.Shaft Ext – Rear Ext. 2.3756475	

MISCELLANEOUS

NUMBER	TITLE	REVISION / DATE
	A and B CCW Chemistry Sampling Results from 9/9/2009-5/2/2010	
	A ESW Strainer (FEF02A) Work History	June 6, 2010
	B ESW Strainer (FEF02B) Work History	June 6, 2010
	Cooling Tower Performance and Associated Curves – Letter from SPX Cooling Technologies	February 18, 2007
	DCEF01A UHS Fan A HS Motor Test Results	March 27, 2009
	FEF02A-B CARS	June 6, 2010
	SEF01 CM Job List	
	System Monitoring Plan: EF – Essential Service Water (ESW) System	
	System Monitoring Plan: EG Component Cooling Water System	
10466-M-072	Design Specification for Component Cooling Water Heat Exchangers for the standardized Nuclear Unit Power Plant System (SNUPPS)	11
BOP FR-1	Functional Requirements and Design Criteria Standard Single and Twin Units 212, 312, & 412 Plants Component Cooling System	3
CDP-ZZ-00200 App. D	Closed Cooling Systems Tables	9

EF-ESW	ESW System Health Report	May 28, 2010
EG – CCW	CCW System Health Report	May 28, 2010
EJ – RHR	RHR System Health Report	May 28, 2010
EPRI NP-7552	Heat Exchanger Performance Monitoring Guidelines	December 1991
EPRI TR-106438	Water Hammer Handbook for Nuclear Plant Engineers and Operators	May 1996
EPRI TR-107397	Service Water Heat Exchanger Testing Guidelines	March 1998
Heat Exchanger	Heat Exchanger Health Report	May 28, 2010
Lesson Plan 5	Essential Service Water – EF	
Lesson Plan 7	Residual Heat Removal – EJ	
Lesson Plan 10	Component Cooling Water – EG	
MOV Predictive Performance Report	EFHV0051 – Essential Service Water Train A to Component Cooling Water Heat Exchanger A Isolation Valve	May 13, 2004
NMR 90-I00164	Drift Eliminator Blades broken or missing	June 25, 1990
OSP-EF-P001A Attachment 3	ESW Pump A Vibration Test Data Sheet	January 4, 2010
Surveillance 05510851	Ultimate Heat Sink Retention Pond Walkdown	July 14, 2008
ULNRC-2324	Callaway Plant Response to Generic Letter 89-13 Service Water System Problems affecting Safety-Related Equipment	November 14, 1990
ULNRC-05425	Callaway Plant Unit 1 Union Electric Co. Facility Operating License NPF-30 Cycle 15 Commitment Change Summary Report	July 16, 2007
08007639/51 0	Job Report IEEG01A / Comp Cool Wtr HX A	November 30, 2009
04503552.560	Heat Exchanger Inspection Report A CCW Hx	April 19, 2007
08007639.510	Heat Exchanger Inspection Report A CCW Hx	May 3, 2010

07505143.500	Heat Exchanger Inspection Report A CCW Hx	October 18, 2008
ULNRC 2146	Letter from D. F. Schnell (UE) to the US NRC Docket NO. 50-483 Callaway Plant Response to Generic Letter 89-13 Service Water System Problems Affecting Safety-related Equipment	January 29, 1990
I0466-M-072 (Q)	Design Specification for Component Cooling Water Heat Exchangers for the Standardized Nuclear Unit Power Plant System (SNUPPS)	11
FSAR 9.2.2	Cooling System for Reactor Auxiliaries	OL-17
APAZZ01025	Raw Water Systems Control Program	0
EDP-ZZ-01121	Raw Water Systems Predictive Performance Program	14
WEST-0435	Westinghouse Instruction Manual for auxiliary Heat Exchangers Prepared for SNUPPS	1
NRC Information Notice 81-21	Potential Loss of Direct Access to Ultimate Heat Sink	July 21, 1981

JOBS

04503164	05514601	05515812	05516379	05516708
06519720	07505357	07506014	07506067	07506068
07507658	08502518	08504012	08509993	08512900
08512901	10507422	10507989		

CALLAWAY ACTION REQUESTS

199500293	199500850	200100208	200102273	200202475
200204566	200304463	200802638	200803108	200807193
200810348	200810719	200811563	200812639	200900576
200908095	200909091	200909684	201000433	201000757
201000956	201001152	201001813	201001900	201002250
201002505	201003021	201004411	201005657	201005694

201005865	201005900	201005901	201005905	200704197
200811500				

COMMITMENT TRACKING NUMBERS

42088	42089	42090	42091	42092	42097
42098	42101	42157	50076	50077	50078
50079	50080	50081	50082	50083	50084
50122					

Section 1R08: Inservice Inspection Activities

PROCEDURES

NUMBER	TITLE	REVISION
MRW-ZZ-WP514	Welding of P-8 Materials	14
MRS-SSP-2063	Callaway Pressurizer Structural Weld Overlay	0
EDP-ZZ-01004	Boric Acid Corrosion Control Program	9
MDP-ZZ-LM001	Fluid Leak Management Program	6
QCP-ZZ-05049	RPV Head Bare Metal Examination	2

WELD PROCEDURE SPECIFICIATIONS

NUMBER	TITLE	REVISION
WPS-0808T01	GTAW of P8 Materials [Max Deposit Thickness ≤ ¾”]	2
WPS=0808TS01	GTAW/SMAW of P8 Materials [1/16” Min. to 8.0” Max]	2

PROCEDURE QUALIFICATION RECORDS

PQR 8 PQR 42

CALLAWAY ACTION REQUESTS

200811641	200812205	200900375	200902177	200903210
200903365	200903612	200904321	200905536	200908018
200909507	201003171	201003361	201003420	201003504

Section 1R12: Maintenance Effectiveness

CALLAWAY ACTION REQUESTS

201004350	201004344

Section 1R13: Maintenance Risk Assessment and Emergent Work Controls

PROCEDURES

NUMBER	TITLE	REVISION
APA-ZZ-00150, Appendix M	Containment Closure	4
EDP-ZZ-01129	Callaway Plant Risk Assessment	23
ETP-BB-03147	Reactor Vessel Head Removal – IPTE	16
ETP-BB-03147, Addendum 2	Preparation for Reactor Vessel Head Lift	4
OTN-BG-00006	Boration Flow Paths in Modes 4, 5, and 6	1

CALLAWAY ACTION REQUESTS

200704250	201003411

MISCELLANEOUS DOCUMENTS

NUMBER	TITLE	REVISION / DATE
Job 10002900	Risk management contingency of 75 foot of cable for power B train spent fuel pool cooling pump	

Section 1R15: Operability Evaluations

PROCEDURES

NUMBER	TITLE	REVISION
EDP-ZZ-03000	Containment Building Coatings	14
OTG-ZZ-00006	Plant Cooldown Hot Standby to Cold Shutdown	58
OTO-BB-00010	Shutdown LOCA	1
OTN-EJ-00001,	Placing A RHR In Service for RCS Cooldown	14

Addendum 3

DRAWINGS

NUMBER	TITLE	REVISION
J-24EJ11	Instrument Isometric Drawing RHR Heat Exchanger Vent	1
M-21-00066	Governor Valve and Linkage Section	4
M-21-00069	Terry Turbine Parts List	6
M-22-EG01	Piping and Instrumentation Drawing Component Cooling Water System	9
M-22-EJ01	Piping and Instrumentation Drawing Residual Heat Removal System	59
M23EG01	Piping Isometric Component Cooling Water Sys Aux Bldg A Train	6

CALLAWAY ACTION REQUESTS

201003872	201003971	201005083	201005149	201005587
201005401	201004541			

JOBS

10003131	10004398	10004404	10003528

Section 1R17: Evaluations of Changes, Tests, or Experiments and Permanent Plant Modifications

50.59 SCREENS

NUMBER	TITLE	REVISION
FSARCN 07-017	Update FSAR Table 5.3-10 Capsule Number Z Withdrawal Time	0
FSARCN 07-046	Remove Reference to ASHRAE 52 Testing in FSAR SP 9.4.2.4 and 9.4.6.4	0
FSARCN 07-051	Revision of Stroke Time for Main Steam Isolation Valve Bypass Valves	
RFR 23186	Evaluate Using MOV-Long-Life Grease on All MOVs	A

MP 07-0007	Modify the RHR Suction Relief Discharge Piping	0
MP 02-1018	Installation of MDAFP's Discharge Automatic Recirculation Control Check Valve	A
MP 00-1009B	MSIV Actuator Replacement	0
AL-18	Verification of Adequate Water Inventory for TDAFP Startup after SSE and LOOP without CST Available	0
CSP-ZZ-07042	Accumulator Boron Concentration	26
CTP-ZZ-00480	Steam Generator Blowdown Outage	11
ESP-EF-0002A	Essential Service Water Train A Flow Verification	8
BO-04	Minimum CST Inventory for 4 hour SBO	3
KA-37	Backup Nitrogen Supply System Design Pressure Losses	0

50.59 EVALUATIONS

NUMBER	TITLE	REVISION / DATE
10-01	Prompt OD 201001813 (Rev. 3); EOP E-1, "Loss of Reactor or Secondary Coolant," (Rev. 13); EOP Addendum 17, Securing ESW Train Due to UHS Cooling Tower Trouble, (Rev. 0); OTA-RK-00016, Annunciator Response Procedure – MCB Panel RK016	3
07-04	Replacement of Essential Service Water Buried Piping with High Density Polyethylene	0
08-01	Reduction in Subcritical Decay Time for Core Offloads from 100 hours to 72 hours per FSARCN 08-028	September 29, 2008
08-02	Implement the Westinghouse Margin Recovery Program	

PERMANENT PLANT MODIFICATIONS

NUMBER	TITLE	REVISION
MP 00-1009B	Replace Electro-Hydraulic Actuators for the Main Steam Isolation Valves with System Medium Actuators	0

PERMANENT PLANT MODIFICATIONS

NUMBER	TITLE	REVISION
MP 02-1018	Replace Motor-Driven Auxiliary Feedwater Pump Discharge Check Valves With Automatic Recirculation Control Check Valves	A
MP 06-0191	Evaluate nonconformances for repaired ESW pump	0
MP 07-0007	Modify the RHR Suction Relief Discharge Piping	0
MP 07-0159	ESW UHS Cooling Tower Equipment Room Sump Pumps and Motors	0
MP 07-0175	Settings for Containment Spray Pump Time Delay Relays	0

MISCELLANEOUS

NUMBER	TITLE	REVISION / DATE
AABD 20.A.0	Steamline Break Mass & Energy Releases Inside Containment	2
Limitorque Technical Update – 02-01	Exxon Nebula EP Grease Replacement	November 13, 2002
RFR 23186A	Evaluate Using MOV-Long-Life Grease on All MOVs	February 13, 2007
RFR 23186A	Licensing Impact Review for RFR 23186	
RFR 23186	Applicability Determination for RFR 23186	A
EPRI Technical Document 1003483	Comparative Analysis of Nebula and MOV Long Life Greases for Limitorque Main Gearbox Applications	December 2002
EVAL-07-26	Westinghouse Evaluation of Main Steam Isolation Valve (MSIV) Stroke Time Increase	0
06116104/965	ABHV0014, ABHV0017, ABHV0020, and ABHV011 Low Mode 3 (250-350 psig) Stroke Test Data	May 5, 2007

06116104/966	ABHV0014, ABHV0017, ABHV0020, and ABHV011 Mid Mode 3 (500-600 psig) Stroke Test Data	May 6, 2007
06116104/967	ABHV0014, ABHV0017, ABHV0020, and ABHV011 High Mode 3 (800-900 psig) Stroke Test Data	May 6, 2007
06116104/968	ABHV0014, ABHV0017, ABHV0020, and ABHV011 NOP/NOT (1050-1125 psig) Stroke Test Data	May 6, 2007
MP 08-01	FSAR CN 08-028: Change in Subcritical Time Requirement from 100 hours to 72 hours Prior to Fuel Off-load	October 1, 2008
FSAR SP, Section 3B.2.1	Earthquake Analysis Assumptions	OL-17i
FSAR SP, Section 3B.4.1	Auxiliary Feedwater Pump Rooms	OL-17i
FSAR 10.3.5.1	Chemistry Control Basis	OL-17
Crane Technical Paper No. 410	Flow of Fluids Through Valves, Fittings, and Pipe	03/06
TS 3.5-1	Accumulators	133
TS 3.7-5	Auxiliary Feedwater (AFW) System	176
TS 3.7-6	Condensate Storage Tank (CST)	176
TS 3.7-8	Essential Service Water System (ESW)	186
OSP-AL-P0002	TDAFW Startup Traces from Job 07514181	February 13, 2008
FSAR 16.9.5	Decay Time	OL-17
FSAR 15.7.4	Fuel Handling Accidents	OL-17
TSB 3.7.15	Fuel Storage Pool Water Level	8
TSB 3.9.4	Containment Penetrations	8
FSAR Table 15.7-7	Parameters Used in Evaluating the Radiological Consequences of a Fuel-Handling Accident	OL-17
FSAR Table 15.7-8	Radiological Consequences of a Fuel Handling Accident	OL-17

OTS-EF-P001A	Performance Testing of Essential Service Water Pump A	5
ESP-EF-0002A	Essential Service Water Train A Flow Verification (task 0551509	11
TS 3.7.8	Essential Service Water System (ESW)	186
TS 3.8	Electrical Power Systems	
FSAR 9.2.1.2.2.2	Component Description	OL-17
RFR 200603478	Evaluate Nonconformances for Repaired ESW Pump	
07004499	Work Order – PEF01B – Essential Service Water Pump B	0
FSAR 5.4.12	Valves	OL-17j
FSAR 5.4.7	Residual Heat Removal System	OL-17j
TS 3.4.12	Cold Overpressure Mitigation System (COMS)	133
TSB 3.4.12	Cold Overpressure Mitigation System (COMS)	8
CN-CPS-08-13	Callaway Overtemperature and Overpower Delta-T Reactor Trip Setpoint Uncertainty Calculation	0
LTR-TA-08-116	Non-LOCA Input to Callaway (SCP) Margin Recovery Program Report	1
M-762-00390	Instruction Manual for Power Range Uncompensated Ionization Chamber Assembly	3
M-762-00412	Instruction Manual for Source and Intermediate Range Housing Assembly	4
RFR 8029A	New Source Range Detector Approval	June 7, 1990
RFR 8668A	Use of Power Range Detectors with Long Cables	October 10, 1991

CALCULATIONS

NUMBER	TITLE	REVISION / DATE
WCAP-16140-P	Callaway Replacement Steam Generator Program NSSS	1

	Engineering Report	
ARC-197	Review Auxiliary Feedwater for an Operating Temperature of 120F	0
AL-18	Verification of Adequate Water Inventory for TDAFP Startup after SSE and LOOP without CST Available	1
AL-57	Air Transport Time to ALV0001 Following a CST Postulated Line Break	0
ARC-727	PAL02 Suction Pressure Evaluation Prior to ESW Pump Start	0
EF-45	Acceptance Criteria Used in Essential Service Water Flow Balance Procedures	8
GN-17	Containment Cooler Water Hammer Analysis	0
RFR 14020	Operability of TDAFP After SSE and LOOP	B
EC-32	Fuel Cycle 16 Decay Heat Load Calculation	0
ZZ-406	Reactor Building Fuel Handling Accident with less than 100 hours Subcritical Decay Time	3
ZZ-83	Fuel Building Fuel Handling Accident with less than 100 hours Subcritical Decay Time	1
HI-971795	Holtec International Calculation – Radiological Evaluation of the Callaway and Wolf Creek Spent Fuel Pools	3
NCR 4668	Flowserve Calculation – Callaway 37 KXH suction bell flange stress calculation	December 12, 2005
NCR 4671	Flowserve Calculation – First Stage Impeller Keyway not Machined to Print	December 21, 2005
BB-54	RHR Suction Relief Valve to PRT Delta P	0
BB-54, Add. 1	Modification of RHR Suction Relief Valve Discharge Piping	0
KA-37	Backup Nitrogen Supply System Design Pressure Losses	0
ZZ-525	Containment Pressure/Temperature Calculation for MSLB Including LOCA Sequencer, Addendum 1	1

PROCEDURES

NUMBER	TITLE	REVISION / DATE
OSP-AB-V0005	Main Steam Isolation Bypass Valve Inservice Retest	8
OSP-SA-0005A	Train A SLIS Slave Relay Test	16
EOP E-1	Loss of Reactor or Secondary Coolant	13
EDP-ZZ-01126	Lubrication Predictive Maintenance Program	8
OSP-AB-V2CSD	Main Steam Isolation Valve Functional Stroke	6
OSP-AB-V02HS	Main Steam Isolation Valve Inservice Test	4
ZZ-006	Engineering Changes	23
APA-ZZ-00600	Design Change Control	35
CSP-ZZ-07042	Accumulator Boron Concentration	28
CTM-RM-00001	RM Panel Alarm Annunciator	25
CTP-ZZ-00480	Steam Generator Blowdown Outage	14
ESP-EF-0002A	Essential Service Water Train A Flow Verification	8
OSP-SJ-V001A	Solenoid Valve Position Indication Test	11
MTM-EA-NP001	Service Water Pump Overhaul	8
MTM-DA-NP001	Circulating Water Pump Overhaul	12
CA1339	Licensing Impact Review	February 21, 2008
CA2510	Applicability Determination	November 5, 2007
CA2511	50.59 Screen	June 2, 2008
CA2512	50.59 Evaluation	November 5, 2007

STARS–ENG–5000–8.1	Change Assessment Matrix	0
STARS–ENG–5001–8.1	Engineering Disposition	0
STARS–ENG–5001–8.3	Engineering Screen: Hazards Review	0
STARS–ENG–5001–8.4	Engineering Screen: Programs Review	0
STARS–ENG–5001–8.6	Field Change Notice	0
STARS–ENG–5001–8.7	Information Request Sheet	0
ESP-EF-002B	Essential Service Water Train B Flow Verification	15

WORK ORDERS

13412

CALLAWAY ACTION REQUESTS

199300224	200101383	200101383	200401458	200507755
200605986	200607835	200609805	200702063	200800461
200801118	200808674	200810024	200810180	200903216
200906154	200909091			

Section 1R18: Plant Modifications

DRAWINGS

NUMBER	TITLE	REVISION / DATE
C-2C1211(Q)	Auxiliary Building – Area 1 Concrete Neat Lines Plan – Floor El 2026’-0”	5
C-2S1903(Q)	Auxiliary Building Structural Steel Sections and Details	0

E-23KJ06(Q)	Diesel Generator KKJ01A Governor Control Schematic Diagram	
M-22KJ01(Q)	Piping and Instrumentation Diagram Standby Diesel Generator A Cooling Water System	22
M-22KJ03(Q)	Piping and Instrumentation Diagram Standby Diesel Generator A Lube Oil System	19
M-22KJ04(Q)	Piping and Instrumentation Diagram Standby Diesel Generator B Cooling Water System	20
M-22KJ06(Q)	Piping and Instrumentation Diagram Standby Diesel Generator B Lube Oil System	18
M-23EF089(Q)	Piping Isometric Essential Service Water – Diesel Generator Building	6

CALLAWAY ACTION REQUESTS

200903892	200909091	200909152

JOBS

08007977	09000318	09000319

MISCELLANEOUS DOCUMENTS

NUMBER	TITLE	REVISION / DATE
Report 0901166.402	Component Cooling Water Heat Exchanger Design Report Addendum	December 2009
Modification Package 09-0076	Replace EDG Jacket Water and Lube Oil Heat Exchangers with New Design	0

Section 1R19: Postmaintenance Testing

PROCEDURES

NUMBER	TITLE	REVISION
ETP-NE-ST01A	A Emergency Diesel Generator Governor Post Maintenance Test Procedure – ITPE	0
OSP-AL-P001A	Motor Driven Aux. Feedwater Pump A Inservice Test – Group A	54

OSP-SA-00003	Emergency Core Cooling System Flow Path	36
ESP-SF-00001	Rod Drop Testing using the Plant Computer - ITPE	18

CALLAWAY ACTION REQUESTS

201004071	201004108	201004900	201005718

JOBS

08512631	10002124	10003980
09002778	10002189	10506180

MISCELLANEOUS DOCUMENTS

NUMBER	TITLE	REVISION / DATE
NOED 10-4-001	Request for Enforcement Discretion for Compliance with Technical Specification 3.8.1, “AC Sources – Operating”	April 2, 2010

Section 1R20: Refueling and Other Outage Activities

PROCEDURES

NUMBER	TITLE	REVISION
ESP-SF-00001	Rod Drop Testing Using the Plant Computer – IPTE	17
ESP-ZZ-00024	Low Power Physics Testing Data Acquisition	9
ETP-BB-03148	Reactor Vessel Upper Internals Removal – IPTE	16
ETP-SF-00005	Control Rod H02 Movement Test	0
ETP-ZZ-00012	Inverse Count Rate Ratio (ICRR) Monitoring for Approach to Criticality	13
ETP-ZZ-00035 Attachment 14	Refueling Performance Refueling Sequence Transfer Checklist	32
ETP-ZZ-00035 Attachment 8	Refueling Performance Spent Fuel Pool Shuffle to Maintain Coolable Geometry Checklist	32
ETP-ZZ-00035 Attachment 4	Refueling Performance Spent Fuel Pool after Offload Checklist	32
ISL-SE-0N44B	PR N44 Axial Flux Difference	32

ITL-AP-000L4	Loop Level; Condensate Storage Tank Level	6
ITL-AP-000L4	Loop Level; Condensate Storage Tank Level	10
OSP-AL-V0003	Auxiliary Feedwater Pump Discharge Check Valve Closure Test	13
OSP-BB-00007	Reactor Coolant System Heatup and Cooldown Limitations	12
OSP-GT-00003	Containment Closure	17
OSP-SA-00004	Visual Inspection of Containment for Loose Debris	22
OSP-SA-2413A	Train A Diesel Generator and Sequencer Testing	10
OSP-SF-00003	Pre-Core Alteration Verifications	21
OSP-SF-00003	Pre-Core Alteration Verifications	22
OSP-SF-00005	Estimated Critical Position Calculation	18
OSP-SF-00500	Control Rods Incapable of Withdrawal	5
OTG-ZZ-00001	Plant Heatup Cold Shutdown to Hot Standby	69
OTG-ZZ-00001, Addendum 3	Enabling Pressurizer and Steamline Pressure Safety Injection	0
OTG-ZZ-00002	Reactor Startup – IPTE	44
OTG-ZZ-00003	Plant Startup Hot Zero Power to 30% Power – IPTE	50
OTG-ZZ-00004, Addendum 3	Planned Power Changes from Full Power	6
OTG-ZZ-00006, Addendum 10	Pressurizer Solid Operation - IPTE	9
OTG-ZZ-00006, Addendum 4	Initial RCS Depressurization and SI Block	5
OTG-ZZ-00007	Refueling Preparation, Performance and Recovery	29
OTG-ZZ-00007	Refueling Preparation, Performance and Recovery	30
OTN-BB-00001	Reactor Coolant System – IPTE	30
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	12
OTN-EJ-00001, Addendum 3	Placing a RHR Train in Service for RCS Cooldown	13
OTS-KE-00013	Refueling Machine	28
OTS-KE-00018	Draining Refueling Pool	32

DRAWINGS

NUMBER	TITLE	REVISION
CD-7C	Instrument Loop Diagram Condensate Transfer and Storage System. Condensate Storage Tank Level	0

CALLAWAY ACTION REQUESTS

200704250	200812606	201003361	201004847	201005096
201004153	201004412	201003411	201004301	201005444
201005278				

JOBS

04500285	07510847	08511183 under vessel inspection	0913014	08512631

CALCULATIONS

NUMBER	TITLE	REVISION
Calculation M-AP-02, Addendum 2	Calculate Required CST Capacity to Support Station Blackout Event	0
Calculation AP-05	Calculate the Volume of the Condensate Storage Tank	0
Calculation AL-22	Auxiliary Feedwater Pumps Suction Pressure Setpoints	3
Calculation BO-04	Condensate Storage Tank Inventory for a Four Hour Station Blackout	3

MISCELLANEOUS DOCUMENTS

NUMBER	TITLE	DATE
AUCA 10-045	Event Review Team summary of Loss of Both Source Range channels	May 6, 2010

Section 1R22: Surveillance Testing

PROCEDURES

NUMBER	TITLE	REVISION
ESP-EF-0002A	Essential Service Water Train A Flow Verification	10
OSP-AB-V002A	Steam Generator Atmospheric Steam Dump (ASD) Valve Inservice Test	33
OSP-BB-V002A	PORV Inservice Test	12
OSP-BG-00002	Verify One Centrifugal Charging Pump Incapable of Injection into Reactor Coolant System	16
OSP-EJ-PV04B	Train B Residual Heat Removal and Reactor Coolant System Check Valve Inservice Test	3
OSP-EM-00002	Rendering Safety Injection Pumps Incapable of Injection	19
OSP-EM-V003A	CCP A and B Full Flow Test – IPTE	21
OSP-NE-0024A	Standby Diesel Generator A 24 hour Run and Hot Restart	31
OSP-NE-0024B	Standby Diesel Generator B 24 hour Run and Hot Restart	31
OSP-SA-2413B	Train B Diesel Generator and Sequencer Testing	14
OSP-SF-00001	Shutdown Margin Calculations	38
OSP-AL-V0003	Auxiliary Feedwater Pump Discharge Check Valve closure test	13

CALLAWAY ACTION REQUESTS

201003385	20104078	201004090	201003492

JOBS

08511394	08511717	08512051	10506157	08510878
08512521	10506893			

2RS01 Radiological Hazard Assessment and Exposure Controls

PROCEDURES

NUMBER	TITLE	REVISION
APA-ZZ-01000	Callaway Plant Radiation Protection Program	33
APA-ZZ-01004	Radiological Work Standards	17
HDP-ZZ-01500	Radiological Postings	34
HTP-ZZ-02004	Control of Radioactive Sources	30
HTP-ZZ-02023	Unconditional Release of Material from Radiological Controls	12
HTP-ZZ-06001	High Radiation/Locked High Radiation/Very High Radiation Area Access	38

CALLAWAY ACTION REQUESTS

200908788	200908892	200908893	200910284	201001185
201003315				

RADIOLOGICAL SURVEYS

NUMBER	TITLE	DATE
1309	Residual Heat Removal Heat Exchanger Room B	April 19, 2010
1309	Residual Heat Removal Heat Exchanger Room B	April 24, 2010
1310	Residual Heat Removal Heat Exchanger Room A	February 2, 2010
1310	Residual Heat Removal Heat Exchanger Room A	April 20, 2010
FB2047	Fuel Assembly Handling Tool	April 22, 2010
RB-2047-CAV	Reactor Building 2047 Refuel Cavity Area	April 18, 2010
RB-2026-VC	Reactor Building-B C Loops- Letdown Valve Cubicle 2026’	April 23, 2010
RB-2026-VC	Reactor Building-B C Loops- Letdown Valve Cubicle 2026’	April 24, 2010

RADIATION WORK PACKAGES

RWP NUMBER	RWP DESCRIPTION
170813208	Remove and Replace Reactor Vessel Head O-Ring
08004886500	Replace Valves BGV0001, BGV0002, BGV0003 and Other Related Work in Letdown Valve Cubicles
08511183	Inspect Bottom of Reactor Vessel in In-Core Tunnel

2RS02 Occupational ALARA Planning and Controls

PROCEDURES

NUMBER	TITLE	REVISION
APA-ZZ-01001	Callaway Plant ALARA Program	13
HDP-ZZ-01100	ALARA Planning and Review	10
HDP-ZZ-01200	Radiation Work Permits	12
HTP-ZZ-01101	Administrative Controls for Radiation Shielding	16

CALLAWAY ACTION REQUESTS

200908657	200908785	201001360	201003499	201003902
201003907	201003964			

RADIATION WORK PACKAGES

RWP NUMBER	RWP DESCRIPTION
090701NRC	NRC Tours and Inspection in the RCA during Refueling Outage 17
040903AREA5	Refueling Outage 17 Maintenance Activities in Area 5 of Auxiliary Building
040903AB2026	Refueling Outage 17 Maintenance Activities on the 2026' Level of the Auxiliary Building
050901RBMAINT	Refueling Outage 17 Miscellaneous Maintenance Activities in the Reactor Building Requiring Minimal Radiation Protection Support
07010204	Work on Loose Parts Monitoring System in In-Core Tunnel
170806568	Lower Cavity Penetrations for Power Operations

170822708	Set-up and Remove UT and Reconstitution Equipment, Provide UT and Reconstitution Support, UT Fuel, and Repair Fuel as Necessary

RADIOLOGICAL SURVEYS

NUMBER	TITLE	DATE
RB-2021-NOZ	Reactor Building 2021' – Cavity Posted LHRA; Posted LHRA inside annulus	April 21, 2010
RB-2021-NOZ	Reactor Building 2021' – Annulus Opening De-posted from LHRA	April 22, 2010

MISCELLANEOUS DOCUMENTS

TITLE	DATE
Refuel 16 ALARA Outage Report	November 7, 2008
Callaway Plant Long Range Dose and Source Term Reduction Plan	September 2009
Nuclear Oversight Performance Report – 4th Quarter 2009	April 7, 2010
08512403/500 - Temporary shielding for RHX Job Package	April 17, 2010
07505589/500 – Temporary Shielding for RCS S/G B Crossover Drain Valves Job Package	April 19, 2010
Refueling Outage 17 Outage Active RWP Dose Comparison	April 27, 2010 – April 29, 2010

2RS03 In-Plant Airborne Radioactivity Control and Mitigation

PROCEDURES

NUMBER	TITLE	REVISION
HDP-ZZ-03000 Appendix D	Performing Airborne Radioactivity Surveys	3
APA-ZZ-01004	General Instructions for Donning and Removing Respiratory Equipment	6

MISCELLANEOUS DOCUMENTS

TITLE	DATE
Breathing Air Sample Data Sheet	December 2009
Breathing Air Sample Data Sheet	February 2010

Weekly/Monthly SCBA Inspection Logs	June 2009 – April 2010

AIR ACTIVITY CONCENTRATION WORKSHEETS

TITLE	DATE
2000 Reactor Building	April 26, 2010
Spent Fuel Building West Side	April 27, 2010
Reactor Building 2047 West Side	April 27, 2010
Top Pressurizer Clean Studies	April 27, 2010
B Loop Cube 2000’	April 27, 2010
Reactor Coolant Pump B	April 27, 2010
Containment Equipment Hatch	April 23, 2010
Containment Equipment Hatch	April 24, 2010
Containment Equipment Hatch	April 25, 2010

4OA1: Performance Indicator Verification

PROCEDURES

NUMBER	TITLE	REVISION
APA-ZZ-00500, Appendix 9	Mitigating Systems Performance Index (MSPI)	3

CALLAWAY ACTION REQUESTS

201004229	201004284	200500187

Section 4OA2: Identification and Resolution of Problems

DRAWINGS

NUMBER	TITLE	REVISION
M-25BG21(Q)	Hanger Location DWG. CVCS-Normal and Alternate Charging Reactor Building	15
M-25BG24(Q)	Hanger Location Drawing. CVCS Auxiliary Spray Reactor Building	11

M-25BG23(Q)	Hanger Location DWG. CVCS Charging and Excess Letdown Reactor Building	15
J-24EJ11	Instrument Isometric Drawing RHR Heat Exchanger Vent	1
M23EG01	Piping Isometric Component Cooling Water Sys – Aux Building “A” Train	6
M23EG03	Piping Isometric Component Cooling Water Sys – Aux Building “B” Train	7

CALLAWAY ACTION REQUESTS

200206970	201001900	200902154	200906288	201000527
200207379	201003591	201000773	200705072	200909120
200905004	201003580	200910268	200909866	201005435
201005440	201005424			

JOBS

10000139	10000141	10000142

MISCELLANEOUS DOCUMENTS

NUMBER	TITLE	DATE
NRC Information Notice 81-21	Potential Loss of Direct Access to Ultimate Heat Sink	July 21, 1981

Section 4OA3: Event Follow-up

PROCEDURES

NUMBER	TITLE	REVISION
APA-ZZ-00107	Review of Current Industry Operating Experience	13
APA-ZZ-00500, Appendix 17	Screening Process guidelines	11
APA-ZZ-00500, Appendix 21	Other Issues- Significance Level 6	8

DRAWINGS

NUMBER	TITLE	REVISION
E-23AL01A(Q)	Schematic Diagram Motor Driven Auxiliary Feedwater Pump A	8
E-23KJ03A(Q)	Schematic Diagram Diesel Generator KKJ01B Engine Control (Start/Stop Circuitry)	13
E-23KJ04(Q)	Schematic Diagram Diesel Generator KKJ01B Annunciator and Miscellaneous Circuits	12
E-23NE11(Q)	Schematic Diagram 4.16KV DG NE02 Feeder Breaker 152NB0211	10
E-23NE13(Q)	Schematic Diagram Diesel Generator KKJ01B Exciter/Voltage Control	10
M-22KJ02(Q)	Piping and Instrumentation Diagram Standby Diesel Generator "A" Intake Exhaust, F.O and Start. Air Sys	20
J-22FC19(Q)	Steam Generator Feed Pump Turbines ESFAS Block Control Logic Diagram	0

CALLAWAY ACTION REQUESTS

201000574	201003524	201003146	201002675	201003524
200910153				

Section 4OA5: Other Activities

PROCEDURES

NUMBER	TITLE	REVISION
EDP-ZZ-04070	Appendix A: Alloy 600 Management Plan	3

CALLAWAY ACTION REQUESTS

200908682	200910027	201002428	201002429	201002997
201003655	201003765	201004305		

MISCELLANEOUS DOCUMENTS

NUMBER	TITLE	DATE
NRC Form 748	National Source Tracking Transaction Report	January 2010

	National Source Tracking System Annual Inventory 2010 for Callaway Plant	
WDI-PJF-1303575-FSR-001	AmerenUE Callaway April 2007 Refueling Outage 15 PORV	April 24, 2007
EDP-ZZ-04070	Appendix A: Alloy 600 Management Plan	3
E170.0105	Callaway Nuclear Power Plant Third Interval Inservice Inspection Program Plan	3
MRS-SSP-2063	Callaway Pressurizer Structural Weld Overlay	0

Attachment 2
Failure of EDG A Governor Drive Lack of Lubrication
SDP Phase 3 Analysis

Performance Deficiency:

Inadequate maintenance resulted in train A emergency diesel generator failing to run on March 30, 2010. The event was caused by the use of a licensee manufactured gasket that did not include a lubrication oil port. The governor drive subsequently failed. The diesel was not recoverable.

Assumptions:

1. The analyst assumed that the governor drive degraded only during times that the diesel generator was running. There was no assumption of accelerated degradation associated with diesel starts or any degradation while the unit was in standby. It was further assumed that the failure was a deterministic outcome set to occur after a specific number of operating hours.

   The diesel was run at the following times (in reverse order):

   03/30/10 – ran for 16 hrs 43 min, could have operated 16 hrs 43 minutes
   03/24/10 – ran for 2 hrs 38 min, could have operated 19 hrs, 21 minutes
   03/18/10 – ran for 1 hr 19 min, could have operated for 20 hrs, 40 minutes
   03/05/10 – ran for 2 hrs 18 min, could have operated for 22 hrs, 58 minutes
   02/05/10 – ran for 13 min, could have operated for 23 hrs, 11 min
   02/03/10 – ran for 1 hr, 45 min, could have operated for 24 hours (max mission time in PRA)

   From the above information, the analyst established the following bins for analysis. Because of the nature of the failure, recovery was not credited once the EDG had failed.

Bin	Run time before failure	Exposure Period
3/30 (after failure) for next 105 hours	0	105 hours
3/24-3/30	17 hrs	6 days
3/18-3/24	19 hrs	6 days
3/05-3/18	21 hrs	13 days
2/05-3/05	23 hrs	28 days
2/03-2/05	24 hrs	None, no quantitative risk impact

1. The analyses used the Callaway SPAR Model, Revision 3.51 to analyze the finding. The SPAR initiating event frequency for a loss of offsite power was 3.59E-2/year. This value was generic to United States reactors based on information from NUREG 6890, "Reevaluation of Station Blackout Risk at Nuclear Power Plants, Analysis of Loss of Offsite Power," dated 2005. However, the NUREG contained more specific loss of offsite power frequencies based on the plant's location within the United States power grids. The applicable generic frequency for Callaway was 2.12 E-2/year. The analyst based the remainder of the internal assessment on this initiating event frequency.

2. The analyst additionally adjusted the loss of offsite power initiating event frequency to account for compensatory measures taken by the licensee in response to the diesel failure. Specifically, the licensee prohibited all switchyard work during the entire repair time. NUREG 6890 delineates that the switchyard centered portion of the loss of offsite power frequency was 1.04E-2/year. Further, the portion of the frequency associated with human error initiated events was 30%. Therefore, the analyst reduced the loss of offsite power initiating event frequency by 1.04E-2 * .30 = 3.12E-3. The resultant loss of offsite power initiating event frequency was 1.81E-2/year. This initiating event frequency was utilized for the repair time portion of the calculation.

5. Common cause vulnerabilities for the train B emergency diesel generator did not exist. The licensee verified that a proper manufacturer supplied gasket was installed on that diesel.

6. A cutset truncation of 1.0E-13 was used. Average test and maintenance was assumed for all other components.

Internal Events Analysis:

A. **<u>Bin 1: Risk Estimate for the 105 hour repair period</u>:**

The train A emergency diesel generator (EDGA) was unavailable and unrecoverable during the repair period. The analyst calculated a new baseline core damage frequency (CDF) assuming a loss of offsite power initiating event frequency of 1.81E-2. The base CDF was 2.14E-5/year. The analyst used the SPAR model to calculate the Risk Achievement Worth (RAW) for the NE01 diesel generator. The licensee had stipulated that, because of an artificiality in the risk model, the model incorrectly assumed non symmetric loading of the safety buses, which resulted in a slightly different RAW value for each diesel. The licensee advised the analyst that the RAW used for the significance determination should be the average of the emergency diesel generator RAWs. This average value was 3.95 in the SPAR model for the test and maintenance basic events. The test and maintenance RAW value was used because the model eliminates test and maintenance on the other diesel when calculating the RAW value (which is appropriate for this case). The delta-CDF for this portion was:

Delta-CDF = 2.14E-5/year * (3.95-1) * 105 hours/8760 hours = 7.5E-7/year

B. Bin 2: Risk Estimate for the 6-day period between 3/24 to 3/30:

During this exposure period, EDGA was capable of running for 17.0 hours. Accordingly, the failure would only affect the loss of offsite power sequences that were beyond the 17.0 hour point. Therefore, the analyst adjusted the loss of offsite power frequency to be consistent with these assumptions.

The base loss of offsite power frequency for Callaway was 2.12 E-2/year. The analyst did not credit the post-failure compensatory measures in the switchyard because they were not yet in place. The licensee established the measures following the diesel failure. The 17.0-hour non-recovery of offsite power was 2.65E-2. Therefore, the frequency for a loss of offsite power event that cannot be recovered in 17.0 hours was 2.12E-2 * 2.65E-2 = 5.6E-4/year. This was the new loss of offsite power initiating frequency.

Resetting event time t=0 to 17.0 hours following the event required that the recovery factors for offsite power be adjusted. Because the new LOOP frequency included the assumption that offsite power was not recovered in the first 17.0 hours, the analyst calculated new nonrecovery values for offsite power given the nonrecovery occurred at t=17.0 hours. The analyst used the following standard statistical equation for the calculation:

P(BIA) = P(A*B)/P(A), where:

P(BIA) = probability that offsite power will not be recovered given that it was not recovered at t=17.0 hours.

P(A) = nonrecovery probability of offsite power at t=17.0 hours. Equal to 2.654E-2.

P(B) = SPAR nonrecovery of offsite power at time = x

Since B is a subset of A and the two probabilities are dependent, P(A*B) = P(B). The equation then reduces to:

P(BIA) = P(B)/(PA)

For example, for the 2-hour SPAR sequences, the offsite power nonrecovery probability at 19.0 hours, given that offsite power was not recovered at 17.0 hours is:

P(BIA) = 2.332E-2/2.654E-2 = .8787

Finally, the analyst adjusted the model to account for lower decay heat generating conditions. Specifically, the decay heat generation rate at 1.0 hour after shutdown is 2.4 times the decay heat generated at 18 hours after shutdown (Reference ANSI/ANS 5.1, 2005). This would allow more time to mitigate an initiating event before water above the core could boil off to cause core damage.

The decay heat adjustment method involves using a nonrecovery probability in the numerator (of the previous equation) that is delayed beyond the values used thus far. For example, for the 19.0 hour sequences (two hours after the diesel failed), instead of using 2.332E-2 (the offsite power nonrecovery at 19.0 hours) in the numerator, the analyst conservatively used the offsite power nonrecovery value at 19.0 hours + approximately 2.0 hours (= 21 hours), which was 2.080E-2. This would generate a nonrecovery probability of .7837 for the 2.0 hour SPAR slot. The analyst noted that the SPAR model only provides data out to 24.0 hours. Therefore, for nonrecovery values associated with 24.0 hours or more, the analyst conservatively used the value for 24.0 hour nonrecovery value in the numerator.

The following table represents adjustments the analyst made to the SPAR model to accommodate these changes:

Offsite Power Recovery Period	Normal Offsite Power Non-Recovery Value	Adjusted Offsite Power Non-Recovery Value (i.e. +17 hours)	Notes:
30 min	7.314E-1	8.536E-1	
1.0 Hour	5.303E-1	8.286E-1	
90 min	4.031E-1	8.062E-1	
2.0 Hours	3.181E-1	7.837E-1	Example cited above
3.0 hours	2.149E-1	7.434E-1	
4.0 hours	1.566E-1	7.072E-1	
5.0 hours	1.205E-1	6.745E-1	
6.0 hours	9.637E-2	6.745E-1	
7.0 hours	7.949E-2	6.745E-1	
Greater than 7.0 hours		0	PRA does not calculate beyond 24.0 hours.

The analyst performed two calculations, a “base case,” and a “current case.” The base case assumed that EDGA did not fail because of the performance deficiency. The current case assumed that EDGA did fail (at 17.0 hours), and was not recoverable. As a surrogate for a non-recovered emergency diesel generator, the analyst set the basic event for test and maintenance to True. Random failures associated with EDGB could be recovered at the nominal recovery probabilities. The analyst then solved only the LOOP sequences.

> Note: The SPAR model uses a single set of basic events to model emergency diesel generator recovery for both trains. Some of the cutsets could include failure of both emergency diesels. Since the train A would not be recoverable and train B would be recoverable, the analyst set the train A test and maintenance basic event to True (which would have a very similar impact in the model as not being recoverable). This adjustment allowed the train B diesel to be recoverable consistent with the model’s assumptions.

For a one year exposure period, the base case conditional core damage probability was 5.3E-7 and the current case probability was 4.9E-6. The incremental conditional core damage probability was 4.4E-6. For a 6 day exposure period, the delta-CDF was:

Delta-CDF = 4.4E-6 * 6/365 = 7.3E-8/year

C. <u>Bin 3: Risk Estimate for the 6-day period between 3/18 to 3/24</u>:

During this exposure period, EDGA was capable of running for 19.0 hours. The analyst followed the same method described in Bin 2, except the event started at the 19.0 hour point instead of the 17.0 hour point.

The 19.0-hour non-recovery of offsite power was 2.33E-2. Therefore, the frequency for loss of offsite power events that are not recovered in 19.0 hours was 2.12E-2 * 2.33E-2 = 4.9E-4/year. This was the new loss of offsite power initiating frequency.

The adjusted offsite power nonrecovery values are noted in the following table:

Offsite Power Recovery Period	Normal Offsite Power Non-Recovery Value	Adjusted Offsite Power Non-Recovery Value	Notes:
30 min	7.314E-1	8.613E-1	
1.0 Hour	5.303E-1	8.306E-1	
90 min	4.031E-1	8.177E-1	
2.0 hours	3.181E-1	8.049E-1	

3.0 hours	2.149E-1	7.676E-1	
4.0 hours	1.566E-1	7.676E-1	
5.0 hours	1.205E-1	7.676E-1	
Greater than 5.0 hours		0	PRA does not calculate beyond 24.0 hours.

For a one year exposure period, the base case conditional core damage probability was 4.8E-7 and the current case probability was 4.6E-6. The incremental conditional core damage probability was 4.1E-6. For a 6 day exposure period, the delta-CDF was:

Delta-CDF = 4.1E-6 * 6/365 = 6.7E-8

D. <u>Bin 4: Risk Estimate for the 13-day period between 3/5 to 3/18</u>:

During this exposure period, EDGA was capable of running for 21.0 hours.

The 21.0-hour non-recovery of offsite power was 2.08E-2. Therefore, the frequency for loss of offsite power events that are not recovered in 21.0 hours was 2.12E-2 * 2.08E-2 = 4.4E-4/year. This was the new loss of offsite power initiating frequency.

The adjusted offsite power nonrecovery values are noted below:

Offsite Power Recovery Period	Normal Offsite Power Non-Recovery Value	Adjusted Offsite Power Non-Recovery Value (i.e. +19 hours)	Notes:
30 min	7.314E-1	8.815E-1	
1.0 hour	5.303E-1	8.686E-1	
90 min	4.031E-1	8.686E-1	
2.0 hours	3.181E-1	8.686E-1	
3.0 hours	2.149E-1	8.686E-1	
Greater than 3.0 hours		0	PRA does not calculate beyond 24.0 hours.

For a one year exposure period, the base case conditional core damage probability was 2.1E-7 and the current case probability was 1.3E-6. The incremental conditional core damage probability was 1.1E-6. For a 13 day exposure period, the delta-CDF was:

Delta-CDF = 1.1E-6 * 13/365 = 3.8E-8

E. <u>Bin 5: Risk Estimate for the 28-day period between 2/5 to 3/5</u>:

During this exposure period, EDGA was capable of running for 23.0 hours.

The 23.0-hour non-recovery of offsite power was 1.877E-2. Therefore, the frequency for loss of offsite power events that are not recovered in 23.0 hours was 2.12E-2 * 1.877E-2 = 4.0E-4/year. This was the new loss of offsite power initiating frequency.

The adjusted offsite power nonrecovery values are noted below:

Offsite Power Recovery Period	Normal Offsite Power Non-Recovery Value	Adjusted Offsite Power Non-Recovery Value (i.e. +19 hours)	Notes:
30 min	7.314E-1	9.536E-1	
1.0 hour	5.303E-1	9.536E-1	
Greater than 1.0 hours		0	PRA does not calculate beyond 24.0 hours.

For a one year exposure period, the base case conditional core damage probability was 9.8E-8 and the current case probability was 5.1E-7. The incremental conditional core damage probability was 4.1E-7. For a 28 day exposure period, the delta-CDF was:

Delta-CDF = 4.1E-7 * 28/365 = 3.6E-8/year

The total delta-CDF from internal events was:

Delta-CDF = 7.5E-7 +7.3E-8 + 6.7E-8 + 3.8E-8 + 3.2E-8 = 9.6E-7/year

External Events Analysis: The analyst reviewed the Callaway, “Individual Plant Examination of External Events (IPEEE),” dated June 30 1995 to determine the contribution of external events to delta-CDF. The analyst noted that high winds (including tornados), floods and transportation accidents were screened from the analysis, as the licensee met the 1975 Standard Review Plan screening criteria. Therefore, the analyst did not consider these areas further.

Seismic: The seismic events of interest included only those that could result in a loss of offsite power. The analyst referenced the "Risk Assessment of Operational Events Handbook," Volume 2, "External Events," dated January 2008, and obtained the seismically induced loss of offsite power frequency for the Callaway station. The frequency was 4.14E-5/year.

Next the analyst used the information calculated in the internal events sections to estimate the seismic contribution to delta-CDF. Specifically, the analyst set up the same bins that were used in the internal events section. Then, the seismic contribution was equal to the internal delta-CDF multiplied by a ratio of the seismic initiating event frequency over the general loss of offsite power initiating event frequency assumed for the particular bin.

Bin 1 delta-CDF = (4.14E-5/1.81E-2)*7.5E-7 = 1.7E-9

Bin 2 delta-CDF = (4.14E-5/2.12E-2)*7.3E-8 = 1.4E-10

Bin 3 delta-CDF = (4.14E-5/2.12E-2)*6.7E-8 = 1.2E-10

Bin 4 delta-CDF = (4.14E-5/2.12E-2)*3.8E-8 = 7.4E-11

Bin 5 delta-CDF = (4.14E-5/2.12E-2)*3.2E-8 = 5.8E-11

The cumulative delta-CDF for seismic contributors was:

Delta-CDF = 2.1E-9/year

Fire: The fire events of interest only included those that could result in a loss of offsite power. The analyst identified three fire zones from the IPEEE that could concurrently cause a loss of offsite power. The frequency for a fire induced loss of offsite power for each area was:

Fire Area	Loss Due to Fire	Fire Frequency	Probability of Nonsupression	Frequency of LOOP
C-5	Offsite power only	3.9E-4	.02	7.8E-6
TB-3	Offsite power only	4.10E-4	.05	2.1E-6
SWYD	Offsite power only	1.1E-4	1.0	1.1E-4

Note: Additional areas were identified in the IPEEE but the analyst verified from the licensee that a fire in those areas would not actually result in a loss of offsite power. The cumulative fire initiated loss of offsite power initiating event frequency was 1.2E-4/year.

Similar to the seismic calculation, the analyst calculated the contribution to delta-CDF from fire initiated loss of offsite power events:

Bin 1 delta-CDF = (1.2E-4/1.81E-2)*7.5E-7 = 5.0E-9

Bin 2 delta-CDF = (1.2E-4/2.12E-2)*7.3E-8 = 4.1E-10

Bin 3 delta-CDF = (1.2E-4/2.12E-2)*6.7E-8 = 3.7E-10

Bin 4 delta-CDF = (1.2E-4/2.12E-2)*3.8E-8 = 2.1E-10

Bin 5 delta-CDF = (1.2E-4/2.12E-2)*3.2E-8 = 1.8E-10

The cumulative delta-CDF for fire initiated loss of offsite power events was:

Delta-CDF = 6.2E-9

Delta-CDF from all causes: The total delta-CDF was:

Delta-CDF = 9.6E-7+2.1E-9+6.2E-9 = 9.7E-7/year

Adjustment for Capacity Factor: The above delta-CDF value is for an entire year of reactor operation. However, on average the plant does not operate for a full 12 months in any year. The licensee indicated that the capacity factor for Callaway was approximately 0.9. Using this factor, the final delta-CDF was:

Delta-CDF = 9.7E-7 * 0.9 = 8.7E-7

Therefore, the finding was of very low safety significance (Green). The analyst noted that the results were conservative and bounding because of the limitations in modeling significantly lower decay heat conditions. The dominant cutsets included a loss of offsite power initiating event, failure to recover offsite power in 4.0 hours, failure of the train B emergency diesel generator, and a reactor coolant pump seal failure. Equipment that mitigated the significance included the operable emergency diesel generator and the turbine driven auxiliary feedwater pump.

Large Early Release Frequency: To evaluate the change to the large early release frequency (LERF), the analyst used Inspection Manual Chapter 0609, Appendix H, “Containment Integrity Significance Determination Process.” Callaway has a large dry containment. The finding screened as having very low safety significance for LERF because it did not affect the intersystem loss of coolant accident or steam generator tube rupture accident categories.