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

August 5, 2004

Garry L. Randolph, Senior Vice
President and Chief Nuclear Officer
Union Electric Company
P.O. Box 620
Fulton, MO 65251

SUBJECT: CALLAWAY PLANT - NRC INTEGRATED INSPECTION REPORT 05000483/2004003

Dear Mr. Randolph:

On June 23, 2004, the NRC completed an inspection at your Callaway Plant. The enclosed report documents the inspection findings which were discussed on June 28, 2004, with you and other members of your staff.

This inspection examined activities conducted under your license as they relate to safety and compliance with the Commission's rules and regulations and with the conditions of your license. Within these areas, the inspection consisted of selected examination of procedures and representive records, observations of activities, and interviews with personnel.

This report documents two self-revealing findings of very low safety significance. These findings were determined to involve violations of NRC requirements. Because of the very low safety significance and because they were entered into your corrective action program, these violations are being treated as noncited violations (NCV), consistent with Section VI.A of the Enforcement Policy. Additionally, two licensee-identified violations which were determined to be of very low safety significance are listed in Section 4OA7 of this report. If you contest the violations or significance of these NCVs, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555-0001, with copies to the Regional Administrator, U.S. Nuclear Regulatory Commission, Region IV, 611 Ryan Plaza Drive, Suite 400, Arlington, Texas 76011; the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001; and the NRC Resident Inspector at the Callaway Plant facility.

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter, its enclosure, and your response will be made available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records component of NRC's document system (ADAMS). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Should you have any questions concerning this inspection, we will be pleased to discuss them with you.

Sincerely,

/RA/

David N. Graves, Chief
Project Branch B
Division of Reactor Projects

Docket: 50-483
License: NPF-30

Enclosure:
NRC Inspection Report
05000483/2004003
w/attachment: Supplemental Information

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ENCLOSURE

U.S. NUCLEAR REGULATORY COMMISSION
REGION IV

Docket:	50-483
License:	NPF-30
Report:	05000483/2004003
Licensee:	Union Electric Company
Facility:	Callaway Plant
Location:	Junction Highway CC and Highway O Fulton, Missouri
Dates:	March 25 through June 23, 2004
Inspectors:	M. S. Peck, Senior Resident Inspector J. D. Hanna, Resident Inspector B. D. Baca, Health Physicist R. E. Lantz, Senior Emergency Preparedness Inspector T. O. McKernon, Senior Operations Engineer R. P. Mullikin, Senior Reactor Inspector W. Sifre, Reactor Inspector
Approved By:	D. N. Graves, Chief, Project Branch B

CONTENTS

SUMMARY OF FINDINGS

IR 05000483/2004003; 03/25 - 06/23/2004; Callaway Plant. Temporary Plant Modifications, Postmaintenance Testing.

This report covered a 3-month inspection by resident inspectors and announced inspections by regional health physics, emergency preparedness, and reactor inspectors. Two Green noncited violations and two licensee-identified violations were identified. The significance of most findings is indicated by their color (Green, White, Yellow, Red) using Inspection Manual Chapter 0609, "Significance Determination Process." Findings for which the significance determination process does not apply may be Green or be assigned a severity level after NRC management review. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG 1649, "Reactor Oversight Process," Revision 3, dated July 2000.

A. Inspector-Identified and Self-Revealing Findings

Cornerstone: Barrier Integrity

Green. A self-revealing finding and a noncited violation of Technical Specification 5.4.1, “Procedures,” was identified after maintenance resulted in the failure of a residual heat removal pump seal during shutdown cooling operations. The licensee’s maintenance work instructions were not adequate to ensure the mechanical seal matting ring surface was fully seated when replaced on March 31, 2004. The seal failed on April 11 after about 36 hours of operation.

This finding was greater than minor because it affected the barrier integrity cornerstone attribute of procedure quality, as related to maintenance procedures affecting the functionality of containment. The failed seal provided a containment leakage path for 7 gallons per minute reactor coolant. The inspectors evaluated the finding using the significance determination process for at-power situations because the issue involved the potential degradation of containment barrier integrity during power operations prior to the reactor shutdown on April 10. The finding was only of very low safety significance because the condition did not represent an actual open pathway in the physical integrity of reactor containment during power operation, was not an actual reduction of the atmospheric pressure control function of the reactor containment, and did not represent a degradation of a the control room auxiliary building or spent fuel pool barrier function. The licensee placed the issue into the corrective action program as Callaway Action Request 200402749 (Section 1R19).

Cornerstone: Occupational Radiation Safety

- Green. A self-revealing finding and noncited violation of Technical Specification 5.4.1 was identified after three plant workers were exposed to an unplanned high radiation area. The event was the result of inadequate operational control of the in-core system. The exposure occurred when a reactor engineer removed two in-core detectors from the core after control room personnel authorized a reactor building entry. The procedure used by the reactor engineer to operate the in-core system was not appropriate to the circumstances.

The inspectors used the occupational radiation safety determination processes to analyze the significance of the finding. The finding was greater than minor because it affected the programs and process attribute of the occupational radiation safety cornerstone. The use of the inappropriate procedure could have resulted in unplanned or unintended dose, which could have been significantly greater as a result of a single, minor, alteration of the circumstances. The inspectors concluded that the issue was of very low safety significance because the inspection finding was not related to as low as is reasonably achievable, did not involve an overexposure, and there was no substantial potential for overexposure. The licensee entered this issue into the corrective action program as Callaway Action Request 200402640. This issue was determined to have crosscutting aspects regarding human performance (Section 1R23).

B. <u>Licensee-Identified Violations</u>

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

REPORT DETAILS

Summary of Plant Status: The Callaway Plant was operating at full power at the beginning of the inspection period. The licensee shutdown the reactor on April 10, 2004, and began Refueling Outage 13. The licensee completed refueling activities and restarted the unit on June 12. The licensee returned the reactor to full power on June 23.

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

1R01 Adverse Weather Protection (71111.01)

a. Inspection Scope

The inspectors completed one sample of the licensee's site preparation for an anticipated weather condition. The inspectors performed a detailed review of the station's adverse weather procedure, Emergency Plan Implementation Procedure EIP-ZZ-00231, "Response to Severe Thunderstorm/High Winds/Tornado Watches and Warnings," Revision 16. The inspectors also performed a walkdown of the primary containment during fuel handling operations to verify that the licensee properly implemented required administrative controls. The inspectors performed the walkdown on April 19 while thunderstorm conditions were forecasted for the area. The inspectors selected the primary containment due to the high importance to safety when fuel handling operations are performed. The inspectors also discussed adverse weather precautions with the refueling containment coordinator and control room supervisor. The inspectors reviewed the corrective actions associated with Callaway Action Request (CAR) 200403129, containment equipment hatch shield not closed during severe weather.

b. Findings

No findings of significance were identified.

1R04 Equipment Alignment (71111.04)

a. Inspection Scope

The inspectors performed three partial system walkdowns during the inspection period. On April 6, the inspectors walked down components of essential service water (ESW) Train B located in the auxiliary building, control building, pump house, and diesel generator building. The inspectors performed the walkdown while the redundant ESW train was out of service for testing. On April 13, the inspectors walked down components of residual heat removal (RHR) system Train B located in the auxiliary building and control room. The inspectors performed the walkdown while the redundant RHR train was out of service for corrective maintenance. On May 3, the inspectors walked down components of the 4 kV essential electrical distribution system Train B located in the control building. The inspectors performed the walkdown while the redundant 4 kV essential electrical distribution system was removed from service for preventive maintenance. In each case, the inspectors checked for correct component alignment and evaluated subsystem operability by comparing the selected equipment to the applicable Final Safety Analysis Report (FSAR) and Technical Specification (TS) sections and the procedures and drawings listed in the attachment.

b. Findings

No findings of significance were identified.

1R05 Fire Protection (71111.05)

1. Routine Fire Protection Walkdowns

a. Inspection Scope

The inspectors performed nine walkdowns of the accessible portions of the fire areas described below. These walkdowns were performed to assess the licensee's control of transient combustible materials, ignition sources, fire detection and suppression capabilities, fire barriers, and related compensatory measures. The inspectors used the commitments described in the FSAR, Section 9.5.1, "Fire Protection System," and Appendix 9.5B, "Fire Hazard Analysis," to determine requirements for fire protection design features, fire area boundaries, and combustible loading requirements for each fire area. The inspectors also reviewed CAR 200402450, pickup truck parked overnight within Fire Area T-2 without a fire barrier impediment permit. The inspectors walked down:

- Fire Area T-2, Turbine building 50 feet north of the auxiliary building, on March 31
- Fire Area A-23, Main steam and feedwater valve compartment, on April 5
- Fire Area D-2, West emergency diesel generator (EDG) room, on April 8
- Fire Area A-2A, Containment spray pump Room A, on April 20
- Fire Area A-2B, Centrifugal charging pump Room A, on April 20
- Fire Area A-4B, Containment spray pump Room B, on April 20
- Fire Area RB, Reactor building, on June 7
- Fire Areas F-2 and F-3, Fuel pool cooling heat exchanger rooms, on June 16
- Fire Areas F-4 and F-5, Air handling equipment room, on June 17

b. Findings

No findings of significance were identified.

2. Fire Drill Observation

a. Inspection Scope

The inspectors observed an announced fire drill conducted on June 21. The purpose of the drill was to evaluate the fire brigade's response to a fire using foam as a firefighting agent. The simulated fire occurred outside the power block buildings at the solvent storage building. The inspectors observed the drill to evaluate fire brigade members as they donned protective clothing, entered the fire area, and utilized fire preplan strategies. The inspectors evaluated the fire brigade and control room communications and whether sufficient firefighting equipment was available. The fire drill was conducted using fire drill Scenario 04A02, which the inspectors reviewed for objectives and acceptance criteria. The inspectors also reviewed the fire drill critique and CAR 200405230.

b. Findings

No findings of significance were identified.

1R07 Heat Sink Performance (71111.07)

a. Inspection Scope

The inspectors completed one sample of heat sink performance during the inspection period. The inspectors selected the component cooling water heat exchanger Train A performance test due to the risk importance of the component. The inspectors performed the review to verify adequate cooling water flow and heat exchanger capacity. The inspectors reviewed "Heat Exchanger Inspection Report," A675979B, performed on May 1 and "Eddy Current Testing of Heat Exchanger EEG01A," A215478J, completed on May 3. The inspectors completed an in-office review of the test data on May 13. The inspectors compared the test results against TS requirements and guidance provided in Generic Letter 89-13, "Service Water System Problems Affecting Safety-Related Equipment."

b. Findings

No findings of significance were identified.

1R08 Inservice Inspection Activities (71111.08)

a. Inspection Scope

.1 Performance of Nondestructive Examination Activities Other than Steam Generator Tube Inspections

The inspectors observed the ultrasonic system calibration and observed ultrasonic and magnetic particle examinations. The inspectors observed 20 examinations, which are listed in the attachment.

During the review of these examinations, the inspectors verified that the correct nondestructive examination procedure was used, examinations and conditions were as specified in the procedures, and test instrumentation or equipment was properly calibrated and within the allowable calibration period. The inspectors also reviewed the documentation to determine if revealed indications were compared against the American Society of Mechanical Engineers (ASME) Code specified acceptance standards and whether the indications were appropriately dispositioned. The nondestructive examination certifications of those personnel observed performing examinations or identified during review of completed examination packages were reviewed by the inspectors.

.2 Steam Generator Tube Inspection Activities

The inspection procedure specified, with respect to in-situ pressure testing, performance of an assessment of in-situ screening criteria to assure consistency between assumed nondestructive examination flaw sizing accuracy and data from the Electric Power Research Institute (EPRI) examination technique specification sheets. It further specified assessment of appropriateness of tubes selected for in-situ pressure testing, observation of in-situ pressure testing, and review of in-situ pressure test results.

The inspectors selected and reviewed the acquisition technique sheets and their qualifying EPRI examination technique specification sheets to verify that the essential variables regarding flaw sizing accuracy had been identified and qualified through demonstration.

The inspection procedure specified confirmation be made that the steam generator tube eddy-current testing scope and expansion criteria meet TS requirements, EPRI guidelines, and commitments made to the NRC. The inspectors' review determined that the steam generator tube eddy-current testing scope and expansion criteria were being met.

The inspection procedure also specified that, if the licensee identified new degradation mechanisms, then verify that the licensee had fully enveloped the problem in an analysis and had taken appropriate corrective actions before plant startup. At the time of this inspection, no new degradation mechanisms had been identified.

The inspection procedure also required confirmation that all areas of potential degradation were being inspected, especially areas that were known to represent potential eddy-current challenges (e.g., top-of-tubesheet, tube support plates, and U-bends). The inspectors confirmed that all known areas of potential degradation, including eddy-current-challenged areas, were included in the scope of inspection and were being inspected.

The inspection procedure further required that repair processes being used were approved in the TSs for use at the site. At the time of this inspection, the licensee had not performed or used the designated TS-approved repair processes, thus, there was no opportunity to observe implementation of any potential repairs (e.g., plugging operations) or in-situ pressure testing.

The inspection procedure also required confirmation that the TS plugging limit was being adhered to and determination of whether depth-sizing repair criteria were being applied for indications other than wear or axial primary water stress corrosion cracking in dented tube support plate intersections. The inspectors confirmed that the licensee was adhering to these specifications.

The inspection procedure stated, if steam generator leakage greater than 3 gallons per day was identified during operations or during postshutdown visual inspections of the tubesheet face, to assess whether the licensee had identified a reasonable cause and corrective actions for the leakage based on inspection results. The inspectors did not conduct any assessment because this condition did not exist.

The inspection procedure required confirmation that the eddy-current testing probes and equipment were qualified for the expected types of tube degradation and assessment of the site-specific qualification of one or more techniques. The inspectors observed portions of all eddy-current testing performed. During these examinations, the inspectors verified that: (1) the probes appropriate for identifying the expected types of indications were being used, (2) probe position location verification was performed, (3) calibration requirements were adhered to, and (4) probe travel speed was in accordance with procedural requirements.

Finally, the inspection procedure specified the review of one to five samples of eddy-current testing data if questions arose regarding the adequacy of eddy-current testing data analyses. The inspectors did not identify any results where eddy-current testing data analyses adequacy was questionable.

.3 Temporary Instruction (TI) 2515/152, "Reactor Pressure Vessel Lower Head Penetrations" (NRC Bulletin 2003-02)

The inspectors reviewed the procedures and analysis guidelines for the inspection of the bottom mounted instrument nozzles in the reactor pressure vessel bottom. The inspectors determined that the procedures had been issued and qualified in accordance with the licensee's quality assurance program. The licensee had equipment, procedures, and personnel in place to perform ultrasonic and eddy-current examinations of the nozzles. The inspectors reviewed the qualifications for the personnel performing the examinations. The nozzle examinations had not taken place at the time of this inspection. The inspectors will review the inspection results when they become available.

.4 Identification and Resolution of Problems

The inspectors reviewed Inservice inspection-related condition reports issued during the current and past refueling outage, and verified that the licensee identified, evaluated, corrected, and trended problems. In this effort, the inspectors evaluated the effectiveness of the licensee's corrective action process, including the adequacy of the technical resolutions.

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification Activities Review by Resident Staff (71111.11Q)

a. Inspection Scope

The inspectors observed one licensed operator simulator training exercise and critique. The inspectors observed the exercise to assess operator performance during high-risk operator actions associated with the emergency plan, lessons learned items, and plant operational experiences. The inspectors observed Licensed Operator Continued Training Simulator Scenario 04-02, "Failure of ACPTO506," on June 18.

b. Findings

No findings of significance were identified.

1R12 Maintenance Effectiveness (71111.12Q)

a. Inspection Scope

The inspectors reviewed two samples of equipment maintenance problems. The inspectors performed the review to verify that the licensee's maintenance efforts met 10 CFR 50.65, "Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants." The inspectors focused on maintenance rule characterization of failed components, risk significance, determination of the (a)(1) classification, corrective actions, and the appropriateness of performance goals and monitoring criteria. The inspectors also evaluated emergent equipment issues to determine if problems were identified at the appropriate level and entered into the corrective action program. The inspectors used Administrative Procedure EDP-ZZ-01128, "Maintenance Rule Program," Revision 6, during the review. The inspectors performed an in-office review of the following Maintenance Rule (a)(1) evaluations:

- CAR 200308178, Loss of Bus PB122 while restoring circulating and service water power to normal alignment

- CAR 200303919, Containment spray Pump B acted abnormally during surveillance

b. Findings

No findings of significance were identified.

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

a. Inspection Scope

The inspectors reviewed five maintenance risk assessments. The inspectors compared the licensee's risk assessment and risk management activities against the requirements of 10 CFR 50.65(a)(4); the recommendations of Nuclear Management and Resource Council 93-01, "Industry Guidelines for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," Revision 3; and Engineering Department Procedure EDP-ZZ-01129, "Callaway Plant Risk Assessment," Revision 2. The inspectors also reviewed the effectiveness of the licensee's contingency actions to mitigate increased risk resulting from degraded equipment. The inspectors evaluated the following risk assessments by in-office review and control room walkdowns:

- CAR 200402529, Unplanned failure of EDG B on April 5. The inspectors observed the licensee's elevated risk compensatory actions from the control room and the switchyard.

- Preventive Maintenance P721255, ESW Train B out of service due to special testing on April 6. The inspectors observed the licensee's elevated risk compensatory actions from the control room and the control building.

- CAR 200402789, Both trains of the RHR system were inoperable April 12 while in Mode 5. The inspectors observed the licensee's elevated risk compensatory actions from the control room, auxiliary building, diesel generator building, and switchyard.

- Reduced reactor vessel inventory operations on April 18 and May 9. The inspectors observed the licensee's elevated risk compensatory actions from the control room, reactor building, and control building.

- Essential 4 kV electrical distribution Train A was removed from service on May 3, 4, and 5, for planned maintenance. The inspectors verified that the licensee implemented compensatory actions by performing walkdowns in the control building, auxiliary building, control room, and switchyard.

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations (71111.15)

a. Inspection Scope

The inspectors reviewed four operability determinations to verify that the licensee properly evaluated the operability of plant components and systems. The inspectors compared the technical adequacy of the evaluations to requirements stated in the TSs, the FSAR, and associated design-bases documents.

- CAR 200402353, Operability determination for nonconservative containment cooler TS and Night Order O120.0015, containment cooler TS limiting condition for operation change. The inspectors completed a control room walkdown and in-office review on March 31.

- CAR 20040529, Operability determination for degradation of the EDG Train B lockout relay circuit. The inspectors completed an in-office review and an EDG building equipment walkdown on April 8.

- CAR 200402749, RHR Pump A seal leakage. The inspectors completed an in-office and an RHR room walkdown on May 21.

- CAR 20040498, Operability determination of nuclear instrument Channel N42 following greater than 10 percent axial flux deviation. The inspectors completed a control room walkdown and in-office review on June 17.

b. Findings

No findings of significance were identified.

1R16 Operator Workarounds (71111.16)

a. Inspection Scope

The inspectors completed an evaluation of one operator workaround during the inspection. The inspectors reviewed the affect that the failure of steam generator blowdown sample Valve BMHV0035 had on the licensee’s ability to implement the Emergency Operating Procedures. The inspectors review included the May 17

Operator Workaround List to assess the cumulative effect of the workarounds and to verify that they did not challenge the operators' capability to respond to plant transients and events. The inspectors also performed a review of the control room Computer Point Status Log; Procedure ODP-ZZ-00013, "Control and Tracking of Plant Computer Points," Revision 8; the Temporary Modification Log; and Procedure APA-ZZ-00605, "Temporary System Modifications," Revision 14, to verify that no other conditions existed that should have been treated as workarounds. The inspectors completed the review on May 18.

b. Findings

No findings of significance were identified.

1R19 Postmaintenance Testing (71111.19)

a. Inspection Scope

The inspectors reviewed four postmaintenance retests that could potentially affect risk significant systems or components. The inspectors completed an in-office review to verify that each test adequately demonstrated system operability and capability. The inspectors used plant TSs, the FSAR, and ASME Section XI to determine system and component requirements. The inspectors' review included the following postmaintenance retests:

- Retest R229160B, RHR Pump A mechanical seal replacement. The inspectors observed portions of the test from the control room on April 1 and subsequently performed an in-office review of the completed test package.

- Retest EDG Train A (Surveillances S724284 and S700489) after the EDG failed hot start criteria following the 24-hour run on May 9. The inspectors observed portions of the test from the EDG and control rooms on May 9 and reviewed CAR 200403870, EDG A failed hot start criteria.

- Retest R611297A, CAR 200403872, auxiliary feedwater (AFW) Valve ALHV0031, performed on May 9. The inspectors performed an in-office review of the completed test package.

- Request for Resolution 23528, Revision B, "Allowable Vibration for AFW," and Surveillance 721725, "AFW pump Inservice Test," postmodification testing of the AFW system following piping and recirculation valve modifications. The inspectors observed testing from the auxiliary building and control room which was performed between May 8 and June 6. The inspectors also completed an in-office review of the test packages.

b. Findings

Introduction. A self-revealing Green finding and noncited violation (NCV) of Technical Specification 5.4.1, Procedures," was identified after maintenance resulted in the failure of a residual heat removal pump seal during shutdown cooling operations. The licensee's maintenance work instructions were not adequate to ensure the mechanical seal was fully seated when previously replaced.

Description. The RHR Train A pump seal failed after 36 hours of shutdown cooling operations on April 12. The seal failed because the pump seal mating ring was not fully seated when installed on March 31. The seal replacement procedure, Mechanical Maintenance Procedure MPM-EJ-QP001, "Residual Heat Removal Pump Overhaul," Revision 8, did not adequately specify that the seal needed to be fully seated.

The pump shaft was equipped with a disaster bushing. The disaster bushing limited the leakage rate 7.5 gallons per minute after the seal failed. The pump shutdown cooling and emergency core cooling functions remained intact with leakage limited to 7.5 gallons per minute. However, the leakage exceeded the value assumed by the licensee in postaccident dose calculations. FSAR TABLE 15.6-6, "Parameters Used in Evaluating the Radiological Consequences of a Loss-of-coolant-accident," stated that 2.0 gallons per minute total emergency core cooling system leakage was assumed for the postaccident dose analysis. Also, Technical Specification 5.5.2, "Primary Coolant Sources Outside Containment," required a program be established to insure the integrity of the 2.0 gallon per minute assumption.

Analysis. The inspectors used the at-power situations significance determination processes to analyze the finding. The finding affected the barrier integrity cornerstone because of potential release of emergency core cooling system (ECCS) recirculation leakage in amounts greater than assumed in the postaccident does analysis. The finding was greater than minor because the issue affected the quality of maintenance procedure attributes related to the functionality of containment. The finding was only of very low safety significance because the condition did not represent an actual open pathway of reactor containment or an actual reduction of the containment atmospheric pressure control function and it did not represent a degradation of a the control room, auxiliary building, or spent fuel pool barrier functions.

Enforcement. TS 5.4.1(a) required written procedures to be established, implemented, and maintained as recommended by Regulatory Guide 1.33, Revision 2, Appendix A, February 1978. Appendix A recommends procedures for maintenance that can affect the performance of safety-related systems. Contrary to this requirement, Procedure MPM-EJ-QP001 did not contain enough detail to successfully replace the RHR pump seal on March 31. Because of the very low safety significance and the licensee's action to place this issue in their corrective action program as CAR 200402749, this violation is being treated as an NCV in accordance with Section VI.A.1 of the Enforcement Policy (NCV 50-483/2004003-01).

1R20 Refueling and Outage Activities (71111.20)

a. Inspection Scope

The inspectors evaluated and observed selected refueling outage activities to ensure that the licensee appropriately considered plant risk when developing outage schedules and controlling plant configuration. The inspectors also reviewed refueling activities to verity that the licensee developed appropriate mitigation strategies for losses of key safety functions and complied with operating license and TS requirements.

Outage Plan Review

Prior to the outage, the inspectors performed an in-office review of the refueling risk analysis and schedule to verify that the licensee appropriately considered risk, industry experience, and previous site-specific problems. The inspectors compared the licensee's outage plan with Administrative Procedure APA-ZZ-00150, "Outage Preparation and Execution," Revision 15, and Nuclear Utility Management and Resource Council 91-06, "Guidelines for Industry Actions to Assess Shutdown Management," December 1991, as a basis for acceptability. The inspectors also reviewed refueling activities against the requirements of Procedure APA-ZZ-00322, "Integrated Work Management Process Description," Revision 1.

Monitoring of Shutdown Activities

The inspectors reviewed reactor coolant system (RCS) cooldown on April 10, 11, and 12 (Surveillance S720430), to verify that the licensee did not exceed TS cooldown limits. The inspectors compared the plant cooldown data against Procedure OSP-BB-0007, "Reactor Coolant System Heatup and Cooldown Limitations," Revision 6.

Licensee Control of Outage Activities

The inspectors attended daily outage status meetings and observed the licensee's control of outage activities to verify that activity defense-in-depth risk was commensurate with the outage risk control plan. The inspectors compared the licensee's evaluation of emergent work risk with Engineering Department Procedure EDP-ZZ-1129, "Callaway Plant Risk Assessment," Revision 0; and Nuclear Utility Management and Resource Council 91-06, "Guidelines for Industry Actions to Assess Shutdown Management."

Inventory Control

The inspectors observed the reactor cavity flood-up from the control room and the containment building on April 15 to verify that the flow paths, configurations, and alternative means for inventory addition were consistent with the outage risk plan. The inspectors also observed the licensee's actions following an April 12 unplanned loss of

RHR, as described in CAR 20040402749, to verify that there were adequate controls in place to prevent inventory loss or other conditions with the potential to cause a loss of inventory.

Reactivity Control

The inspectors reviewed the licensee’s outage reactivity controls to verify that the TS reactivity control requirements were met. The inspectors performed auxiliary building and control room walkdowns on April 20 to verify that the licensee maintained the required boron injection flow paths. The inspectors also reviewed outage activities that could cause unexpected reactivity changes.

Containment Closure

The inspectors conducted a walkdown and interviewed the containment coordinator on April 20 to verify that the licensee was capable of containment closure as required by Off-Normal Operating Procedure OTO-KE-00001, "Fuel Handling Accident," Revision 7, and TSs. The inspectors compared plant observations with Administrative Procedure APA-ZZ-00150, “Outage Preparation and Execution,” Revision 12; Operations Surveillance Procedure OSP-ZZ-00001, “Control Room Shift and Daily Log Readings and Channel Checks,” Revision 39; and Operations Surveillance Procedure OSP-SF-00003, “Pre-Core Alteration Verifications,” Revision 12. The inspectors also reviewed CAR 200403129, failure to close containment hatch during approaching thunderstorm.

Reduced Inventory and Midloop Conditions

The inspectors reviewed the licensee’s commitments from GL 88-17, “Loss of Decay Heat Removal,” and confirmed, by sampling, that these commitments were effectively implemented. The inspectors observed reduced inventory operations on May 9 and 20 from the control room, the reactor building, and the technical support center. The inspectors assessed the effect of distractions, due to unexpected plant conditions or emergent activities, on the operators’ abilities to maintain the required reactor vessel level during midloop operations. The inspectors used Normal Operating Procedure OTN-BB-00001, "Reactor Coolant System," Revision 17, as the bases for acceptability.

Refueling Activities

The inspectors observed fuel handing activities from containment and the control room on April 20 to verify that operations were performed in accordance with the FSAR, plant TSs, and approved procedures. The inspectors observed fuel handing coordination from the control room to verify that the licensee tracked the location of fuel assemblies from core offload through core reload. The inspectors observed the lower reactor internals removal from the reactor vessel on April 26.

Containment Closure Walkdown

The inspectors performed independent walkdowns of containment on June 7 and 9 to verity that the licensee had removed debris and foreign material that could adversely affect emergency core cooling equipment by blockage of the containment sump following an accident.

Monitoring of Heatup and Startup Activities

The inspectors observed reactor startup activities from the control room and the technical support center on June 8, 9, and 13 to verify that plant TSs, license conditions, and other requirements, commitments, and administrative procedure prerequisites for mode change and power ascension were met.

b. Findings

Potential for Containment Sump Blockage by Unqualified Lead Blankets

The inspectors identified 132 lead radiation shield blankets in containment during the closeout inspection at the end of the outage. The inspectors were concerned that blanket failure during a loss of coolant accident (LOCA) could adversely affect the function of safety-related components. The licensee had not performed a safety evaluation demonstrating that a failure of the blanket coverings/coatings would not result in a blockage of the ECCS containment recirculation sump. Failure of the containment recirculation sump during a LOCA could result in fuel failure. In response to the inspectors' concerns, the licensee removed all of the lead blankets from the reactor building prior to restarting the reactor.

The licensee had blankets within the bioshield and in close proximity to high energy RCS piping. These blankets were covered/coated with a double layer of 0.027-inch thick Herculite fabric with a vinyl laminate. The blankets contained about 3,876 square feet of the vinyl laminate fabric. Table 1 describes when the licensee installed the blankets. Herculite Products, the coating manufacturer, stated that the vinyl coverings/coatings may separate from the polyester and release the fabric reinforcement when exposed to temperatures above 275EF. The licensee determined that the predicted post-LOCA containment bulk atmospheric temperatures range from 274EF for the small LOCA case to 308EF for the large break LOCA. The temperatures in the vicinity of the blankets would be expected to be higher due to the close proximity of the RCS piping and the physical barrier provided by the bioshield wall.

During an accident, some of the blanket coverings/coatings would deteriorate into foreign material. Some of this foreign material would be expected to fall to the containment floor and be transported to the containment sump by the flow from the RCS pipe break. Once at the sump, this foreign material may challenge ECCS recirculation function by reducing the available net positive suction head (NPSH) to the RHR and containment spray pumps.

Table 1 - Installation and Location of Lead Blankets

Number of Blankets	Scope	Installed Refuel (RF)	Year	Added Vinyl Laminate Fabric (ft^3)	Total Vinyl Laminate Fabric (ft^3)
20	Regen Heat Exchanger	RF 2	1987	618	618
40	Regen Heat Exchanger	RF 6	1993	700	1518
74	Pressurizer Spray Line	RF 11	2000	1,693	3,011
18	Southside Walkway	RF 13	2004	864	3,875

The Callaway plant design provided very little ECCS recirculation design margin due to low RHR pump NPSH margin and the small wetted sump screen crosssectional surface area. The facility was equipped with two containment sumps. Each sump was designed to provide the ECCS water inventory source during the recirculation phase of accident mitigation. This inventory source must be available to successfully meet long-term postaccident core cooling requirements. FSAR Table 6.2.2-7, “Input and Results of NPSH Analysis,” indicated that the RHR pumps had only 1.9 feet of NPSH margin at the time of ECCS suction swapover to the containment sump. FASR Table 6.2.2-6, "Water Sources and Water Losses Which Contribute to the Water Level Within the Reactor Building Following a Large LOCA,” indicated the minimum containment water level would be 2,001'-11" at the initiation of ECCS switchover. Based on the sump dimension provided in Drawing C-2S2935, "Reactor Building Misc Steel Recirculation Sump Enclosure," Revision 4, the wetted area on the inner sump screen would only be about 68 square feet at the time of the initiation of ECCS switchover. The low NPSH margin and small sump area placed the ECCS function at risk due to RHR pump cavitation with the additional foreign material from the degraded blankets.

The plant licensing bases included a commitment to Regulatory Guide 1.82., "Water Sources for Long-Term Recirculation Cooling Following a LOCA,” Revision 0, June 1974. The Regulatory Guide allowed the licensee to assume that 50 percent of the sump area would be unavailable due to debris. However, the 50 percent sump blockage assumption did not provide for the addition of nonqualified coverings/coatings and other foreign material to be placed in containment without an appropriate safety analysis. The licensee did not adequately address post-LOCA recirculation sump function in any of the modification design packages that have been identified as of the end of this inspection period that were used to install the shield blankets. A summation of the licensee’s safety evaluations are shown in Table 2.

Table 2 - Summary of Lead Blanket Safety Evaluations

Modification	Installed	Safety Evaluation
86-0037A	RF 2 (1987)	Stated lead shielding would remain intact during a seismic event. Did not address the blanket coverings/coatings, containment sump or the LOCA environmental conditions.
92-1025	RF 6 (1993)	Stated a formal safety evaluation was not required. Did not address the blanket coverings/coatings, containment sump or the LOCA environmental conditions.
99-1003	RF 11 (2000)	Stated the melting point of lead (630EF) is greater than the containment postaccident temperature of 400EF. Therefore, the lead wool inside the blankets would remain intact. Should the blanket material be destroyed by a water or steam jet, the loose pieces would be similar to insulation and soft material debris in the containment. This type of debris was enveloped in the previous sump blockage analysis. The inspectors determined there was no previous sump blockage analysis, other than the Regulatory Guide 1.82 assumption that total blockage would be limited to 50 percent. The safety evaluation did not provide any bases that this assumption would remain valid with the additional lead blankets.
01-1026-2	RF 13 (2004)	Only addressed the weight of the lead blankets - Did not address the blanket coverings/coatings, containment sump or the LOCA environmental conditions.

In June 2003, the NRC issued Bulletin 2003-01, "Potential Impact of Debris Blockage on Emergency Sump Recirculation at Pressurized-Water Reactors." The bulletin informed the licensee of the results of NRC-sponsored research identifying the potential susceptibility of pressurized-water reactor recirculation sump screens to debris blockage. The bulletin stated that the 50 percent sump blockage assumption used in plant licensing may not be conservative. Bulletin 2003-01 stated that pressurized-water reactor sump screens were typically designed assuming that relatively small structural loadings would result from the differential pressure associated with debris blockage. Consequently, reactor sump screens may not be capable of accommodating the substantial structural loadings that would occur due to debris beds covering the sump screen surface.

The licensee planned to test a representative sample of the lead blankets in a chamber by simulating the postaccident environment. The licensee anticipates this testing will yield the blanket failure mechanism and the type of debris formed. This data would provide a bases for estimating how much of the blanket covering/coating debris may be transported to the containment sumps during a LOCA. This issue was unresolved pending the conclusion of the licensee’s environmental testing and further evaluation by the NRC to determine if the sumps would have remained functional under LOCA conditions with the blankets installed (URI 50-483/2004003-02).

1R22 Surveillance Testing (71111.22)

a. Inspection Scope

The inspectors observed and/or reviewed four surveillance tests to assess system operational readiness and to verify that the tests demonstrated system safety function. The inspectors compared the following surveillance tests against requirements in plant TSs, ASME Code Section XI, the FSAR, and licensee procedural requirements:

- Surveillance S721208, Emergency ventilation system 10-hour run, performed on April 5. The inspectors observed portions of the test from the auxiliary building and control room and completed an in-office review of the completed surveillance package.

- Special Test P721255, ESW Train B underground piping leakage test, performed on April 6. The inspectors observed portions of the test from the control building and the ESW pumphouse and completed an in-office review the completed surveillance package.

- Surveillance S720430, Reactor coolant system heatup and cooldown plot performed April 10, 11, and 12. The inspectors completed an in-office review of the completed surveillance package. The inspectors also reviewed CAR 200403130, the RCS loop wide hot leg temperature appeared to exceed the plant heatup limit, April 20.

- Surveillance S700265, Diesel Generator Train B, 24-hour run and hot restart, performed on May 11. The inspectors observed portions of the test in the diesel generator and control rooms, and completed an in-office review of the data.

b. Findings

No findings of significance were identified.

1R23 Temporary Plant Modifications (71111.23)

a. Inspection Scope

The inspectors sampled two temporary plant modifications based on the safety significance of the affected systems. The inspectors performed an in-office review and walked downed affected plant equipment to verify that the installation was consistent with the modification documents. The inspectors reviewed the configuration control of the modification to verify that the plant documents, such as drawings and procedures, were appropriately updated. The inspectors reviewed postinstallation test results to verify that the actual impact of the temporary modifications on permanent plant systems were satisfactory. The inspectors compared temporary modification documentation against the requirements established in Administrative Procedure APA-ZZ-00605, "Temporary System Modifications," Revision14, and system requirements contained in FSAR Section 9.1.3.

- Work Package A69376G, special test for installation and troubleshooting of the in-core digital controls. The inspectors walked down the affected plant equipment in the control room and attended the event review team meeting conducted on April 8.
- Callaway Plant Modification, CMP 02-1018, installation of motor-driven AFW pump automatic recirculation control discharge check valves and separate recirculation lines. The inspectors observed modification planning meetings, installation, troubleshooting, and testing conducted during May and June.

b. Findings

Inadequate Operational Control of Equipment Resulted in an Unexpected High Radiation Field

Introduction. A self-revealing Green finding and NCV of TS 5.4.1 was identified after three plant workers were exposed to an unplanned high radiation area. The exposure occurred when a reactor engineer removed two in-core detectors from the core after control room personnel authorized a reactor building entry.

Description. On April 7, two electricians and one health physics technician unexpectedly walked into an unplanned high radiation area while inside the reactor building. Peak radiation doses were measured at 3,000 millirem/hr. The exposure occurred because a reactor engineer had remotely withdrawn two in-core detectors from the core. The reactor engineer was transferring the detectors from the core to the storage location following postmodification testing (PMT) of the in-core system.

The PMT scope involved troubleshooting activities on the in-core system. The troubleshooting required two detectors to be inserted into the core. The reactor engineer's role was to transfer the detectors between the core and storage location. The supporting instrument maintenance personnel and test engineer held a prejob brief to discuss the PMT sequence prior to beginning work. However, neither the reactor engineer nor the health physics representative attended the prejob brief. The test engineers received written permission to operate the in-core system per Nuclear Engineering Test Procedure ETP-SR-00020, "Flux and Thermocouple Mapping," Revision 20. The control room staff also notified the Health Physics Department that movement of the detectors was authorized. Health Physics restricted access to the reactor building. The test engineer notified the reactor engineer that he had permission to operate the in-core system. The reactor engineer withdrew two detectors from storage and placed them into the core and left the control room. The test engineers and instrument maintenance personnel performed the PMT. Later, the operators verbally requested the test engineer to leave the in-core detectors in the core. The operators then notified Health Physics that reactor building entries could resume.

The test engineers completed testing for the day and left the area. The reactor engineer returned to the control room and transferred the detectors from the reactor core back to the storage location per Procedure ETP-SR-00020. The reactor engineer was unaware of the verbal direction the Control Room Supervisor had provided the test engineer. Procedure ETP-SR-0020 did not require the reactor engineer to notify operations again before the detector was returned to the storage location. The high radiation fields were created as the two detectors were transferred from the core to the storage location.

. The event occurred due to a lack of operational formality. The Control Room Supervisor had transferred control of the in-core system to the test and reactor engineers. The control room staff then notified Health Physics that reactor building entries could resume without formal control over the in-core system. Also, the PMT package stated that the in-core system was to be operated under Procedure ETP-SR-00020. Procedure ETP-SR-00020 was not appropriate for this application. Procedure ETP-SR-00020 was used for core flux mapping. Procedure ETP-SR-00020 prerequisites were not met prior to operating the in-core system.

Analysis. The inspectors used the occupational radiation safety determination processes to analyze the significance of the finding. The inspectors concluded that the finding was greater than minor because the issue was associated with the occupational radiation safety cornerstone attribute limiting individual workers' potential unplanned or unintended dose resulting from actions or conditions contrary to licensee procedures. This attribute addresses dose which could have been significantly greater as a result of a single, minor, reasonable alteration of the circumstances. The inspectors concluded the issue was of very low safety significance because the inspection finding was not related to as low as is reasonably achievable, did not involve an overexposure, and there was no substantial potential for overexposure. This issue was determined to have crosscutting aspects regarding human performance.

Enforcement. TS 5.4.1(a) required written procedures to be established, implemented, and maintained as recommended by Regulatory Guide 1.33, Revision 2, Appendix A, February 1978. Appendix A recommends procedures for operation of the in-core system. Contrary to this requirement, plant engineering personnel operated the in-core system without using an appropriate procedure on April 7. Because of the very low safety significance and the licensee’s action to place this issue in their corrective action program (CAR 200402640), this violation is being treated as an NCV in accordance with Section VI.A.1 of the Enforcement Policy (NCV 50-483/2004003-03).

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

a. Inspection Scope

The inspectors performed an in-office review of Revision 26 to the Callaway Radiological Emergency Response Plan, submitted May 19. The revision incorporated Change Notice 03-001 that was preapproved by the NRC. The revision also added an automated calling system, as a backup public notification system, and made other administrative and editorial corrections.

The revision was compared to the previous revisions, to the criteria of NUREG-0654, "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants," Revision 1, and to the requirements of 10 CFR 50.47(b) and 50.54(q) to determine if the revisions decreased the effectiveness of the plan. The inspectors completed one sample during the inspection.

b. Findings

No findings of significance were identified.

2. RADIATION SAFETY
Cornerstone: Occupational Radiation Safety

2OS1 Access Control to Radiologically Significant Areas (71121.01)

a. Inspection Scope

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

- Performance indicator (PI) events and associated documentation packages reported by the licensee in the occupational radiation safety cornerstone

- Controls (surveys, posting, and barricades) of radiation, high radiation, or airborne radioactivity areas

- Radiation work permit, procedure, and engineering controls and air sampler locations

- Conformity of electronic personal dosimeter alarm setpoints with survey indications and plant policy; workers' knowledge of required actions when their electronic personal dosimeter noticeably malfunctions or alarms.

- Barrier integrity and performance of engineering controls of airborne radioactivity areas

- Physical and programmatic controls for highly activated or contaminated materials (non-fuel) stored within spent fuel and other storage pools

- Self-assessments and audits related to the access control program since the last inspection

- Corrective action documents related to access controls

- Radiation Work Permit briefing and worker instructions

- Adequacy of radiological controls such as required surveys, radiation protection job coverage, and contamination controls during job performances

- Dosimetry placement in high radiation work areas with significant dose rate gradients

- Changes in licensee procedural controls of high dose rate - high radiation areas and very high radiation areas

- Controls for special areas that have the potential to become very high radiation areas during certain plant operations

- Posting and locking of entrances to all accessible high dose rate - high radiation areas and very high radiation areas

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

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

- Adequacy of the licensee's internal dose assessment for any actual internal exposure greater than 50 millirem committed effective dose equivalent

- Licensee actions in cases of repetitive deficiencies or significant individual deficiencies

- Licensee event reports and special reports related to the access control program since the last inspection

The inspectors completed 21 of the required 21 samples.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA1 PI Verification (71151)

a. Inspection Scope

The inspectors sampled licensee submittals for the five PIs listed below for the period from March 2003 through March 2004. The inspectors used the definitions and guidance contained in Nuclear Energy Institute (NEI) 99-02, "Regulatory Assessment Indicator Guideline" to verify the accuracy of the PI data reported by the licensee.

Reactor Safety Cornerstone

- High pressure injection safety system unavailability
- Safety system functional failures

The inspectors reviewed a selection of LERs, portions of operator log entries, daily morning reports, the monthly operating reports, and PI data sheets to determine whether the licensee adequately identified the number of unavailable hours for the AFW and RHR systems. This number was compared to the number reported for the PI during the current quarter. In addition, the inspectors also interviewed licensee personnel associated with PI data collection, evaluation, and distribution.

Occupational Radiation Safety Cornerstone

- Occupational Exposure Control Effectiveness PI

Licensee records reviewed included corrective action documentation and electronic dosimetry alarms that identified occurrences of locked high radiation areas (as defined in the licensee's TSs), very high radiation areas (as defined in 10 CFR 20.1003), and unplanned personnel exposures (as defined in NEI 99-02). Additional records and items reviewed included as low as is reasonably achievable and radiation work permit records, corrective action requests, whole body counts of selected individual exposures, radiological controlled area entries, and electronic dosimetry set points. The inspectors interviewed licensee personnel that were accountable for collecting and evaluating the PI data. In addition, the inspectors toured plant areas to verify that high radiation, locked high radiation, and very high radiation areas were properly controlled.

Public Radiation Safety Cornerstone

- Radiological Effluent TS/Offsite Dose Calculation Manual
- Radiological Effluent Occurrences

Licensee records reviewed included corrective action documentation that identified occurrences for liquid or gaseous effluent releases that exceeded PI thresholds and those reported to the NRC. The inspectors interviewed licensee personnel that were accountable for collecting and evaluating the PI data.

b. Findings

No findings of significance were identified.

4OA2 Identification and Resolution of Problems (71152)

1. Routine Review of Identification and Resolution of Problems

a. Inspection Scope

The inspectors performed detailed in-office reviews and walkdowns of plant equipment related to two significant conditions adverse to quality. The inspectors reviewed licencee CAR reports to verify that the full extent of the issues was identified, that the licensee performed appropriate evaluations, and that corrective actions were specified and prioritized. The inspectors evaluated the reports against the requirements of Administrative Procedure APA-ZZ-00500, "Corrective Action Program," Revision 35 and 10 CFR Part 50, Appendix B. The inspectors also attended the CAR Screening Meeting on May 9. The inspectors reviewed the following two samples:

- CAR 200402318, Nuclear Safety Review Board Special Assessment Report following adverse trend in human performance, March 31

- CAR 200403367, Unplanned trip of V51, poor prejob brief on April 15

b. Findings

No findings of significance were identified.

2. Semi-Annual Review of Identification and Resolution of Problems

a. Inspection Scope

The inspectors performed a semi-annual review of significant conditions adverse to quality to identify adverse trends that might indicate the existence of a more significant safety issue. The inspectors included within the scope of this review repetitive or closely related issues that may have been documented by the licensee outside the normal corrective action program, such as in trend reports or performance indicators, 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, maintenance rule assessments, or corrective action backlog lists.

b. Findings

No findings of significance were identified. However, the inspectors observed the following issues which were discussed with licensee management:

- The licensee identified an adverse trend in human performance in April. This trend consisted of five events in the first quarter followed by six additional events in April. The NRC also documented three human performance related findings in Callaway Inspection Report 05000483/2004002. The licensee discussed this adverse trend in CAR 200403258.

- Licensee personnel initiated 2,598 adverse condition CARs between January 1 and June 23. Of these, the licensee attributed the cause of about half to personnel error. However, most of these conditions were very minor in scope such as the failure to perform monthly cleanliness inspections. A significant number of these issues involved minor material control issues in areas such as welding and qualification of items. Twelve personnel errors involving mis-positioned valves or components also occurred during the first two quarters.

4OA3 Event Followup (71153)

1. (Closed) LER 50-483/2004-002-00: Reactor trip due to faulty electrical relay.

   On January 27, the reactor tripped from 100 percent power. The inspectors reviewed the licensee's corrective actions and the operating experience assessment associated with the failed main generator distance relay contact. The inspectors reviewed the LER and no findings of significance were identified. Additionally the inspectors did not identify any commonality between the causes of the February 3 reactor trip and this reactor trip, other than that both events were caused by the poor material condition of main generator protection devices. The licensee documented the failed equipment in CAR 200400629. This LER is closed.

2. (Closed) LER 50-483/2004-003-00: Reactor trip due to faulty relay.

   On February 3, the reactor tripped from 100 percent power. Subsequent to the reactor trip, the turbine driven AFW pump tripped due to an overspeed condition. The inspectors reviewed the licensee's corrective actions and the operating experience assessment associated with the failed main generator dead machine protection circuits and the turbine driven AFW control systems. The inspectors reviewed the LER and no findings of significance were identified. The licensee documented the failed equipment in CAR 200400791. This LER is closed.

3. (Closed) LER 50-483/2004-004-00: Safety injection while conducting plant heatup to normal operating pressure and temperature.

   On February 11, an operator error resulted in an unplanned safety injection and main steamline isolation during a reactor heatup. The safety injection was generated from the combination of reactor pressure above the Permissive P-11 setpoint (1,970 psig) and main steamline pressure less than 615 psig. This issue was dispositioned as a finding of very low safety significance (FIN 50-483/2004002-03) in Callaway Plant Integrated Inspection Report 50-483/2004003 and was entered into the licensee's corrective action program as CAR 200401076. The inspectors reviewed the LER and no additional findings of significance were identified. This LER is closed.

4. <u>(Closed) LER 50-483/2004-005-00:</u> Inadequate feedwater heating during plant start-up causes turbine trip and subsequent reactor trip.

On February 15, an operator error resulted in an unplanned reactor trip during power ascension. The reactor tripped from low steam generator water levels 28 minutes after operations personnel synchronized the main turbine-generator to the grid. Steam generator level oscillations began immediately after the generator output breakers were closed. The level oscillations were caused by a combination of steam generator shrinkage induced by decreasing feedwater temperatures and the high rate of load increase. This issue was dispositioned as a finding of very low safety significance (FIN 50-483/2004002-01) in Callaway Plant Integrated Inspection Report 50-483/2004002 and was entered into the licensee's corrective action program as CAR 20040116. The inspectors reviewed the LER and no additional findings of significance were identified. This LER is closed.

4OA4 <u>Crosscutting Aspects of Findings</u>

Section 1R23 documents a finding with human performance crosscutting aspects. Inadequate operational control of the incore monitoring system resulted in three individuals inadvertently entering a high radiation area.

4OA5 <u>Other Activities</u>

1. <u>TI 2515/152, "Reactor Pressure Vessel Lower Head Penetrations" (NRC Bulletin 2003-02)</u>

a. <u>Inspection Scope</u>

The inspectors completed a review of the licensee's reactor pressure vessel lower head bare metal visual examination on April 16. Licensee personnel accessed the lower head area and did not need remote-controlled camera equipment to perform the examination. A certified Level II nondestructive examination licensee employee performed the examination. The inspectors reviewed the qualifications and certification of the inspection personnel to assess the licensee's ability to detect small boric acid deposits on the reactor vessel head.

The examination was conducted in accordance with Procedure QCP-ZZ-05048, "Boric Acid Walkdown for RCS Pressure Boundary," Revision 1, and Surveillance S714762, "Perform Leakage Examination of RCS". Licensee personnel were able to identify, disposition, and resolve any deficiencies. They were also able to identify whether there was any pressure boundary leakage or reactor pressure vessel lower head corrosion as described in the bulletin. The examiner and other support personnel made numerous photographs of the bottom head and bottom head area.

There were no obstructions to the visual inspection of the head. The head did have boric acid stains which the licensee attributed to cavity seal ring leakage during past refueling outages. The licensee also cleaned portions of the bottom head to remove small amounts of corrosion products that had the potential to obstruct sighting of boric acid leakage.

The licensee's examination consisted of a 360-degree coverage of all of the nozzles. The licensee had the capability to identify small boric acid leaks as described in Bulletin 2003-02. Nondestructive examination personnel noted a small amount of red corrosion products and boric acid material around a few penetrations. The examiners determined that this boric acid was not due to a penetration leak but was from a cavity seal ring leak. There was no boric acid in the interface between the vessel and penetrations.

The licensee removed samples of the deposits, but did not perform chemical analyses of the material because volumetric examinations of the penetrations were conducted with satisfactory results. The licensee also examined the reactor vessel for pressure boundary leaks and none were identified.

The licensee concluded that the deposits were from cavity seal ring leakage due to the seals not seating properly. There were no material deficiencies that required repair. There were no impediments to an effective examination.

On April 21 NRC and licensee personnel participated in a conference call to discuss the initial results of the examination. All personnel were provided photographs of the bottom head examination. There were no issues identified concerning reactor pressure vessel bottom head integrity during the call.

b. Findings

No findings of significance were identified.

2. TI 2515/153, "Reactor Containment Sump Blockage" - Review Licensee's Implementation of Interim Compensatory Measures

a. Inspection Scope

In June 2003, the NRC issued Bulletin 2003-01, "Potential Impact of Debris Blockage on Emergency Sump Recirculation at Pressurized-Water Reactors." The bulletin informed the licensee of the results of NRC-sponsored research identifying the potential susceptibility of pressurized-water reactor recirculation sump screens to debris blockage in the event of a high-energy line break requiring recirculation operation of the ECCS or containment spray system. The bulletin informed the licensee of the potential for additional adverse effects due to debris blockage of flow paths necessary for ECCS and containment spray system recirculation and containment drainage and need for interim compensatory measures, including more aggressive containment cleaning and increased foreign material controls. In response to the bulletin, the licensee committed to more aggressive containment cleaning and increased foreign material controls to ensure that containment drainage paths would remain unblocked (letter ULNRC-04884, August 1, 2003).

The inspectors reviewed the licensees' interim compensatory actions in response to Bulletin 2003-01. The inspectors performed this review to verify that the licensee implemented the commitments contained in the bulletin response and that the licensee's interim compensatory measures were effective in reducing risk associated with potential degraded or nonconforming containment sump recirculation functions. The inspectors also performed walkdown inspections of both containment sumps and observed the containment closeout inspection to verify the absence of material that could be transported to and block the containment sump screens. The results of the containment walkdown inspections were discussed in Section 1R20 of this report.

The inspectors also reviewed the licensee's containment walkdown used to quantify potential debris which was conducted during Refueling Outage 12. The inspectors compared the scope and breadth of the licensee's walkdown against the criteria in NEI 02-01, "Condition Assessment Guidelines: Debris Sources Inside Pressurized Water Reactor Containment," September 2002. The inspectors also walked down portions of the RHR hot leg injection line located inside containment and compared the results with piping insulation/lagging with plant design documents.

The inspectors reviewed the following documents during the inspection:

- Procedure OSP-SA-00004, "Visual Inspection of Containment for Loose Debris," Revision 12

- Operations Information Report which provided operations with information associated with NRC Bulletin 2003-01, January 16, 2004

- Licensed Operator Continuing Training Simulator Scenario 2003-05 S04, "ECCS Degraded Recirculation," November 15, 2003

- Procedure OSP-EJ-00003, "Containment Recirculation Sump Inspection," Revision 4

- Emergency Operating Procedure ECA-1.1, "Loss of Emergency Coolant Recirculation," Revision 1B1

- CAR 200304409, Required responses to NRC Bulletin 2003-01, potential impact of debris blockage on emergency sump recirculation at pressurized water reactors, June 12, 2003

- Work Request W227955, Containment cleaning, completed May 26

- Work Request W229953, Inspect and repair containment sump enclosures and screens, completed April 27, and CAR 200403498, Containment sump screen did not meet acceptance criteria

- Preventive Maintenance P702010, Lower cavity preparation for power operations, completed May 22

- Preventive Maintenance P699593, Lower cavity refueling preparations, completed April 11

- Request for Resolution 23255, "Containment sump screen size evaluation," Revision A, competed January 29

b. Findings

.1 Risk Insights Were Not Effectively Used When Determining Compensatory Measures

The inspectors identified 132 lead shield blankets in containment during a walkdown at the end of the refueling outage. As described in Section 1R23 of this report, the licensee did not evaluate the potential affect of the blanket coverings/coatings on the ECCS water inventory source during the recirculation phase of accident mitigation. The licensee installed 18 of these blankets during the refueling outage following the NRC issuance of the bulletin. The licensee's interim compensatory actions were not effective to prevent the installation of these additional lead blankets or to identify the potential risk to the containment sump of the blankets previously installed.

.2 Inspection of the Containment Sump and Condition Assessments

The inspectors performed an in-office review of the containment condition assessments. The inspectors compared the scope, breadth and depth of the assessment with the guidance provided in NEI 02-01. Table 3 provides the results of this comparison.

Table 3 Comparison of the Licensee's Walkdown with NEI 02-01

Piping Insulation Walkdown	
NEI 02-01 Recommendations	**Licensee's Action**
• Walkdown all piping, equipment structures, penetrations and fire barriers and survey the installed insulation. • Documentation (as a minimum) of each type of insulation in detail. • Plant configuration control programs that control and document the locations and conditions of various types of insulation may be used as either an alterative or supplemental source of information.	• The licensee spot-checked 12 examples of containment pipe insulation against the design documents. • The licensee concluded the type and quantity of piping insulation may be recovered from the design documents when needed. No attempt was made to quantify the amount of insulation that was present in containment. • The licensee did not assess the condition of piping insulation (other than the 12 spot-check examples).

Coatings Walkdown	
NEI 02-01 Recommendations	**Licensee's Action**
• Identify the location of qualified and unqualified containment coatings.	• The licensee spot-checked 10 examples of containment coatings.
• Document the locations on plant drawings.	• The licensee credited a preventive maintenance activity (P674099) which performs an 18 month inspection of nonqualified coatings (tracked on FSAR Table 6.1-3 Containment Components - Coating Schedule.)
• Determine if multiple coating systems have been used in a particular location.	• The licensee did not address multiple coating systems.
Foreign Material	
NEI 02-01 Recommendations	**Licensee's Action**
• Perform a walkdown to identify the amount and location of foreign materials: tape, equipment labels, construction and maintenance debris, temporary equipment (ladders, lead shielding, Herculite), dirt, dust and lint.	• The licensee did not perform the recommended walkdown. The licensee relied on Operational Surveillance Procedure OSP-SA-00004, "Visual Inspection of Containment for Loose Debris," prior to restart following an outage.
• Record buildup of dirt, dust and lint	• The licensee did not perform the recommend walkdown or produce the inventory
• Document location and type of foreign material characterized with respect to specific gravity and particle size.	

3. TI 2515/156, "Operational Readiness of Offsite Power Systems"

a. Inspection Scope

The inspectors collected data from licensee maintenance records, event reports, corrective action documents and procedures and through interviews of station engineering, maintenance, and operations staff, as required by TI 2515/156. The data was gathered to assess the operational readiness of the offsite power systems in accordance with NRC requirements such as Appendix A to 10 CFR Part 50, General Design Criterion 17; Criterion XVI of Appendix B to10 CFR Part 50, plant TSs for offsite power systems; 10 CFR 50.63; 10 CFR 50.65 (a)(4), and licensee procedures. Documents reviewed for this TI are listed in the attachment.

b. Findings

No findings of significance were identified. Based on the inspection, no immediate operability issues were identified. In accordance with TI 2515/156 reporting requirements, the inspectors provided the required data to headquarters staff for further analysis.

4OA6 Management Meetings

Exit Meeting Summary

On June 28, the resident inspectors presented their inspection results to Mr. G. Randolph, Senior Vice President-Generation and Chief Nuclear Officer, and other members of his staff who acknowledged the findings.

The inspectors conducted an exit interview by telephone on June 4, and presented inspection results to Mr. K. Brukerhoff, Supervisor, Emergency Preparedness.

On April 19, the inspectors presented results of the inservice inspection effort to Mr. R. Affolter, Vice President Nuclear, and other members of licensee management who acknowledged the inspection findings.

On April 30, the inspectors presented inspection results to Mr. W. Witt, Plant Manager, Nuclear Operations, and other members of his staff who acknowledged the findings.

On May 13, a telephonic re-exit of the access control to radiologically significant area inspection was conducted with Mr. K. Young, Manager, Regulatory Affairs and Mr. M. Hale, Superintendent, Health Physics.

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

4OA7 Licensee-Identified Violations

The following findings of very low safety significance (Green) were identified by the licensee and are violations of NRC requirements which meet the criteria of Section VI of the NRC Enforcement Policy, NUREG-1600, for being dispositioned as NCVs.

- TS 5.4.1(a) requires written procedures to be established, implemented, and maintained as recommended by Regulatory Guide 1.33, Revision 2, Appendix A, February 1978. Appendix A recommends procedures for equipment control. Regulatory Guide 1.33 recommends procedures for combating emergencies and other significant events. Contrary to this requirement, emergency operating Procedure E-3, could not be implemented by the operators consistent with the time stated in the FSAR for a main steamline break accident. Other related issues involved station blackout and isolation of the chemical and volume control system during varied event scenarios. The licensee’s analysis indicated that the single event that could not be met was the operator response times for a main steamline break accident. This finding affected the mitigating systems cornerstone and was of very low safety significance since it did not represent the loss of any safety function. The licensee has taken interim compensatory actions until the specific emergency operating procedure is upgraded and any necessary changes to the FSAR are made.

This finding also documents closure of Unresolved Item 50-483/2003006-02, "Critical operator emergency operating procedure response times exceeded." The licensee entered the issues into their corrective action program as CARs 200308666 and 200308667.

- TS License Condition 2.C(5)(c), requires the licensee to implement and maintain all provisions of the approved fire protection program. The Callaway fire protection program commits to 10 CFR Part 50, Appendix R, Section III.G.2, which requires a 3-hour fire barrier between the redundant safe shutdown equipment. Contrary to this requirement, the licensee identified 11 fire seals inadequately installed to provide the required 3-hour fire rating, as documented in CAR 200400717. The licensee identified these fire seals following operating experience information received concerning similar deficiencies at the Wolf Creek Generating Station. Using Manual Chapter 0609, Appendix F, Attachment 2, "Additional Guidance for the Assessment of Findings Using Significance Determination Process Entry," the inspectors determined the following.

  There was no credible fire scenario that would affect more than one safe shutdown area. This was based on either a lack of combustible material, lack of ignition sources, location of the seal in regard to safe shutdown equipment, the presence of detection and suppression, the amount of seal degradation, or the distance from the seal to any combustible material. In addition, the inspectors determined that the fire barriers did not have a significant gap between fire areas and, in some cases, the fire seal material was butted up against each other at a right angle.

  These as-found conditions did not provide the required 3-hour fire rating, but would hamper the propagation of smoke and hot gases between the fire areas for an untested amount of time. The inspectors concluded that the degraded fire barriers would provide a minimum of 20 minutes fire endurance protection, and the fixed and in-situ fire ignition sources and combustible flammable materials were positioned such that, even considering fire spreading to secondary combustibles, the degraded fire barriers would not be subject to direct flame impingement.

  This finding affected the mitigating systems cornerstone and was of very low safety significance because, overall, the fire barriers would have provided protection needed for credible fire scenarios.

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee

R. Affolter, Vice President - Nuclear
B. Corder, Steam Generator Engineer
M. Evans, Manager, Nuclear Engineering
K. Gilliam, Rad/Chem Supervisor, Operations Health Physics
L. Graessle, Superintendent, Protective Services
C. Graham, Senior Health Physicist, Technical Support Health Physics
M. Hale, Superintendent, Health Physics
J. Hiller, Engineer, Regulatory Affairs
G. Hurla, Rad/Chem Supervisor, Operations Health Physics
L. Kanuckel, Superintendent, Quality Assurance
S. McCracken, Welding Engineer
B. Montgomery, ISI Engineer
D. Neterer, Superintendent, Operations
H. Osborn, Senior Health Physicist, Technical Support Health Physics
S. Petzel, Regional Regulatory Affairs Engineer
D. Purvis, Quality Control Supervisor
M. Reidmeyer, Supervisor, Regional Regulatory Affairs
E. Thornton, Quality Assurance Engineering Evaluator
W. Witt, Plant Manager
K. Young, Manager, Regulatory Affairs

LIST OF ITEMS OPENED AND CLOSED

Opened

50-483/2004003-01	NCV	Inadequate Work Instructions Resulted in the Failure of Residual Heat Removal Pump Seal (Section 1R19)
50-483/2004003-02	URI	Potential for Containment Sump Blockage by Unqualified Lead Blankets (Section 1R20)
50-483/2004003-03	NCV	Personnel Operated In-Core System Without Appropriate Procedure (Section 1R23)

Closed

50-483/2004003-01	NCV	Inadequate Work Instructions Resulted in the Failure of Residual Heat Removal Pump Seal (Section 1R19)
50-483/2004003-03	NCV	Personnel Operated In-Core System Without Appropriate Procedure (Section 1R23)
50-483/2004-002-00	LER	Reactor Trip Due to Faulty Electrical Relay (Section 4OA3)

50-483/2004-003-00	LER	Reactor Trip Due to Faulty Relay (Section 4OA3)
50-483/2004-004-00	LER	Safety Injection While Conducting Plant Heatup to Normal Operating Pressure and Temperature (Section 4OA3)
50-483/2004-005-00	LER	Inadequate Feedwater Heating During Plant Start-up Causes Turbine Trip and Subsequent Reactor Trip (Section 4OA3)
50-483/2003006-02	URI	Critical Operator Emergency Operating Procedure Response Times Exceeded (Section 4OA7)

DOCUMENTS REVIEWED

Procedures

APA-ZZ-00150, Outage Preparation and Execution, Revision 15

APA-ZZ-00322, Integrated Work Management Process Description, Revision 1

APA-ZZ-00352, Callaway Quality Control Inspection Program, Revision 8

APA-ZZ-00500, Corrective Action Program, Revision 35

ETP-SA-ST001, MSFIS Mod Test, Revision 0

HDP-ZZ-01300, Internal Dosimetry Program, Revision 21

HDP-ZZ-01500, Radiological Posting, Revision 16

HDP-ZZ-03000, Radiological Survey Program, Revision 21

HTP-ZZ-01302, Response to Positive Whole Body Counts, Revision 1

HTP-ZZ-01490, Determination of Beta Skin Dose, Revision 24

HTP-ZZ-03100, Performing Radiological Surveys, Revision 3

HTP-ZZ-03200, Performing Contamination Surveys, Revision 5

HTP-ZZ-03300, Airborne Radioactivity Surveys, Revision 6

HTP-ZZ-06009, Personnel Contamination Incidents, Revision 27

HTP-ZZ-06028, Radiological Controls for Pools that Contain or Store Spent Fuel, Revision 4

MPM-BB-QP001, Reactor Coolant Pump Seal Removal and Replacement, Revision 27

OSP-EJ-P001A, RHR Train A In-Service Test, Revision 33

OSP-NB-00001, Class 1E Electrical Source Verification, Revision 20

OTN-EF-0001, ESW System, Revision 26

PDI-ISI-254, Remote Inservice Examination of Reactor Vessel Shell Welds, Revision 5.

PDI-ISI-254-SE, Remote Inservice Examination of Reactor Vessel Nozzle to Safe End, Nozzle to Pipe, and Safe End to Pipe Welds, Revision 1

PDI-ISI-254-NZ, Remote Inservice Examination of Reactor Vessel Nozzle to Shell Welds, Revision 0

RRA-ZZ-00001, NRC Performance Indicator Program, Revision 1

UEND-Radiological-01, Expectations for Work in the Radiological Controlled Area, Revision 4

WDI-STD-011, Liquid Penetrant Examination, Revision 0

WDI-STD-088, Underwater Remote Visual Examination of Reactor Vessel internals, Revision 2

WDI-STD-133, Paragon ET Imaging for the Inspection of Reactor Vessel BMI Tube Penetrations, Revision 0

WDI-STD-134, Paragon UT Procedure for Inspection of RPV Bottom Mounted Instrument Tube Penetrations, Revision 0

WDI-STD-141, Bottom Mounted Instrumentation UT Analysis Guidelines for Use With Paragon, Revision 0

WDI-STD-142, Paragon ET Analysis Guidelines for Inspection of Reactor Vessel BMI Tube Penetrations, Revision 0

WDI-STD-146, ET Examination of Reactor Vessel Pipe Welds Inside Surface, Revision 1

WDP-9.2, Qualification and Certification of Personnel in Nondestructive Examination, Revision 3

WDP-9.2.2, ASME Code Section XI; Appendix VIII Qualification Program, Revision 1

WCAL-002, Pulser/Receiver Linearity Procedure, Revision 3

Green Book (Callaway Switchyard Voltage Requirements), June 18, 2001

Callaway Action Requests

20038092	20041174	20041858	20042604
20042770	20043000	20043080	20043134
20043454	20043462	20043465	20043488
20043513			

Drawings

Piping and Instrumentation Diagram M-22EF01, ESW System, Revision 20
Piping and Instrumentation Diagram M-22EF01, ESW System, Revision 42
Piping and Instrumentation Diagram M-22EF02, ESW System, Revision 48

Event Review Team Meeting Summaries

Problems during engineered safety feature actuation system Train B testing, May 28

Autostart of ESW Train A during restoration of AFW low suction pressure bistable, June 6

Inadvertent radiation espouser due to in-core operation, April 8

Radiation Work Permits

RWP 453323EC	453322INSTALL	C707691	P699659
S685396	W212681		

Self-Assessment and Quality Verification

ESW System

Quality Assurance Surveillance Reports

AP04-003

AP04-05, dated April 7, 2004, Fire Protection Program Audit

SA03-HP-S05

SP03-013

SP03-018

SP04-004, dated April 7, 2004, Surveillance of Corrective Actions Following Operational Events

SP04-005, dated April 7, 2004, Surveillance of Plant Activities During Recent Shutdown

SP04-008, dated April 7, 2004, Surveillance of Access Authorization Program

SP04-010, dated April 20, 2004, Surveillance of Employee Concerns Program Audit

SP04-012, dated May 4, 2004, Surveillance of Control Room Control of Operations

SP04-017, dated May 13, 2004, Surveillance of Assessment of Human Performance

SP04-030, dated May 21, 2004

SP04-026, dated June 10, 2004, Surveillance of Control Room Shift Turnovers and Briefs During Midloop Operations

SP04-029, dated June 15, 2004, Followup Surveillance on Prejob Briefs

SP04-032, dated June 8, 2004, Surveillance of Safety-Related Train Work Activities

SP04-033, dated June 11, 2004, Surveillance of Bottom Mounted Instrumentation Inspections

SP04-037, dated June 17, 2004, Surveillance of Implementation Commitments to NRC Bulletin 2003-01 for Containment Recirculation Sump Screen Inspections and Repairs and Containment Cleaning

NRC Performance Indicator Transmittal Reports

2003 - Third Quarter
2003 - Fourth Quarter
2004 - First Quarter

Quarterly Dose Assessments

Liquid Effluents (ST-11004): 2003 - Third and Fourth Quarter, and 2004 - First Quarter
Noble Gases (ST-11009): 2003 - Third and Fourth Quarter, and 2004 - First Quarter

Repair & Replacement Documents

PO109719B
A708415A
P670494
W227580

Examinations Observed

System	Component/Weld Identification	Examination Method
Main Feedwater	Elbow to Pipe Weld	Ultrasonic Examination, Magnetic Particle Exam
Main Feedwater	Elbow to Pipe Weld	Ultrasonic Examination, Magnetic Particle Exam
Main Feedwater	Elbow to Pipe Weld	Ultrasonic Examination, Magnetic Particle Exam
Reactor Stud 8CH	Full Length	Ultrasonic Examination

System	Component/Weld Identification	Examination Method
Reactor Stud 11CH	Full Length	Ultrasonic Examination
Reactor Stud 12CH	Full Length	Ultrasonic Examination
Reactor Stud 14CH	Full Length	Ultrasonic Examination
Reactor Stud 17CH	Full Length	Ultrasonic Examination
Reactor Stud 20CH	Full Length	Ultrasonic Examination
Reactor Stud 21CH	Full Length	Ultrasonic Examination
Reactor Stud 23CH	Full Length	Ultrasonic Examination
Reactor Stud 26CH	Full Length	Ultrasonic Examination
Reactor Stud 29CH	Full Length	ltrasonic Examination
Reactor Stud 30CH	Full Length	Ultrasonic Examination
Reactor Stud 32CH	Full Length	Ultrasonic Examination
Reactor Stud 33CH	Full Length	Ultrasonic Examination
Reactor Stud 36CH	Full Length	Ultrasonic Examination
Reactor Stud 38CH	Full Length	Ultrasonic Examination
Reactor Stud 39CH	Full Length	Ultrasonic Examination
Reactor Stud 41CH	Full Length	Ultrasonic Examination

LIST OF ACRONYMS

AFW	auxiliary feedwater
ASME	American Society of Mechanical Engineers
CAR	Callaway Action Request
ECCS	emergency core cooling system
EDG	emergency diesel generator
EPRI	Electric Power Research Institute
ESW	essential service water
FSAR	Final Safety Analysis Report
LER	licensee event report
LOCA	loss of coolant accident
NCV	noncited violation
NEI	Nuclear Energy Institute
NPSH	net positive suction head
PI	Performance Indicator
PMT	postmodification testing
RCS	reactor coolant system
RHR	residual heat removal
TI	temporary instruction
TS	Technical Specification