UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION IV
611 RYAN PLAZA DRIVE, SUITE 400
ARLINGTON, TEXAS 76011-8064

April 20, 2001

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

SUBJECT: CALLAWAY--NRC INSPECTION REPORT 50-483/01-02

Dear Mr. Randolph:

On March 31, 2001, the NRC completed an inspection at the Callaway Plant. The enclosed integrated inspection report documents the inspection findings which were discussed with Mr. Ron Affolter and other members of your staff on March 30, 2001.

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

Based on the results of this inspection, the NRC has identified two issues that were evaluated under the risk significance determination process as having very low safety significance (Green). The NRC has also determined that a violation is associated with one of these issues. This violation is being treated as a noncited violation (NCV), consistent with Section VI.A.1 of the Enforcement Policy. The NCV is described in the subject inspection report. If you contest the violation or significance of the NCV, 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.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter and its enclosure will be made available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARS) component of NRC's document system (ADAMS). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/NRC/ADAMS/index.html (the Public Electronic Reading Room).

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

Sincerely,

/RA/

William D. Johnson, Chief
Project Branch B
Division of Reactor Projects

Docket: 50-483
License: NPF-30

Enclosure:
NRC Inspection Report
50-483/01-02

cc w/enclosure:
Professional Nuclear Consulting, Inc.
19041 Raines Drive
Derwood, Maryland 20855

John O'Neill, Esq.
Shaw, Pittman, Potts & Trowbridge
2300 N. Street, N.W.
Washington, D.C. 20037

Mark A. Reidmeyer, Regional
Regulatory Affairs Supervisor
Quality Assurance
Union Electric Company
P.O. Box 620
Fulton, Missouri 65251

Manager - Electric Department
Missouri Public Service Commission
301 W. High
P.O. Box 360
Jefferson City, Missouri 65102

Ronald A. Kucera, Director
of Intergovernmental Cooperation
P.O. Box 176
Jefferson City, Missouri 65102

Otto L. Maynard, President and
Chief Executive Officer
Wolf Creek Nuclear Operating Corporation
P.O. Box 411
Burlington, Kansas 66839

Dan I. Bolef, President
Kay Drey, Representative
Board of Directors Coalition
for the Environment
6267 Delmar Boulevard
University City, Missouri 63130

Lee Fritz, Presiding Commissioner
Callaway County Courthouse
10 East Fifth Street
Fulton, Missouri 65251

Alan C. Passwater, Manager
Licensing and Fuels
AmerenUE
One Ameren Plaza
1901 Chouteau Avenue
P.O. Box 66149
St. Louis, Missouri 63166-6149

J. V. Laux, Manager
Quality Assurance
Union Electric Company
P.O. Box 620
Fulton, Missouri 65251

Jerry Uhlmann, Director
State Emergency Management Agency
P.O. Box 116
Jefferson City, Missouri 65101

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Regional Administrator **(EWM)**
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Branch Chief, DRP/B **(WDJ)**
Senior Project Engineer, DRP/B **(RAK1)**
Section Chief, DRP/TSS **(PHH)**
RITS Coordinator **(NBH)**

Only inspection reports to the following:
Scott Morris **(SAM1)**
NRR Event Tracking System **(IPAS)**
CWY Site Secretary **(DVY)**
Dale Thatcher **(DFT)**

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ENCLOSURE

U.S. NUCLEAR REGULATORY COMMISSION
REGION IV

Docket:	50-483
License:	NPF-30
Report No.:	50-483/01-02
Licensee:	Union Electric Company
Facility:	Callaway Plant
Location:	Junction Highway CC and Highway O Fulton, Missouri
Date:	January 7 through March 31, 2001
Inspectors:	V. G. Gaddy, Senior Resident Inspector J. D. Hanna, Resident Inspector R. W. Deese, Reactor Inspector P. J. Elkmann, Emergency Preparedness Inspector W. M. McNeill, Reactor Inspector R. P. Mullikin, Senior Reactor Inspector
Approved By:	W. D. Johnson, Chief, Project Branch B

ATTACHMENT: Supplemental Information

SUMMARY OF FINDINGS

Callaway Plant
NRC Inspection Report 50-483/01-02

IR 05000483-01-02; on 01/07-03/31/2001; Union Electric Co; Callaway Plant. Integrated Resident & Regional Report; Personnel Performance During Nonroutine Plant Evolutions and Events, Surveillance Testing.

The inspection was conducted by resident inspectors, a region based emergency preparedness inspector, and three regional reactor inspectors. The inspection identified two Green findings, one of which is a noncited violation. 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 are indicated by "No Color" or by the severity level of the applicable violation. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described at its Reactor Oversight Process website at http://www.nrc.gov/NRR/OVERSIGHT/index.html.

A. Inspector Identified Findings

Cornerstone: Initiating Events

- Green. During a trip actuating device operational test surveillance, maintenance personnel failed to complete a step in the procedure, resulting in the inadvertent tripping of a reactor trip breaker. This was a violation of Technical Specification 5.4.1.

  This noncited violation was characterized as having very low safety significance through the use of the significance determination process. Equipment designed to mitigate the consequences of a reactor trip was available and the reactor trip bypass breaker had been closed prior to the inadvertent opening of the reactor trip breaker (Section 1R22.2).

Cornerstone: Barrier Integrity

- Green. Although operations personnel had prior indication of a valve alignment problem in the boron thermal regeneration system, they were slow to correctly identify the source of the valve alignment problem. As a result, several valves in the boron thermal regeneration system were overpressurized, resulting in reactor coolant system leakage of approximately 2 gpm.

  This finding was of very low safety significance because once operations personnel identified the valve that was out of alignment they quickly isolated the leak and limited reactor coolant system leakage to approximately 50 gallons (Section R14.2).

B. Licensee Identified Findings

None

Report Details

Summary of Plant Status: The plant began the inspection period at 100 percent power. On February 4, power was reduced to 93 percent for turbine control valve testing. The plant was returned to full power later that day. On March 3, power was reduced to 45 percent to replace Transmitter BBFT0425, the reactor coolant system Loop 2 flow transmitter. The plant returned to full power on March 6. On March 9, a reactor trip occurred when the output breaker for rod drive motor generator Set A tripped open while removing rod drive motor generator Set B from service. The plant returned to full power on March 16.

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

1R02 Changes to License Conditions and Safety Analysis Report (71111.02)

a. Inspection Scope

The inspector reviewed eight safety evaluations to verify that the licensee had appropriately considered the conditions under which the licensee may make changes to the facility or procedures or conduct tests or experiments without prior NRC approval.

The inspector reviewed 11 safety evaluation screenings, in which the licensee determined that safety evaluations were not required, to ensure that the licensee's exclusion of a full evaluation was consistent with the requirements of 10 CFR 50.59.

The inspector reviewed a sample of Suggestion-Occurrence-Solution reports initiated by the licensee that addressed problems or deficiencies associated with 10 CFR 50.59 to ensure that appropriate corrective actions were being taken.

b. Findings

No findings of significance were identified.

1R04 Equipment Alignment (71111.04)

a. Inspection Scope

The purpose of the partial walkdowns was to verify equipment alignment and identify any discrepancies that could impact the function of the systems and therefore increase risk. During the complete walkdown, the inspectors verified that equipment alignments that could impact this mitigating system had been identified and were prioritized for resolution. The inspectors performed the following walkdowns:

- Partial walkdown of containment spray Train B while Train A was out of service for maintenance
- Partial walkdown of emergency exhaust system Train A while Train B was out of service for maintenance

- Partial walkdown of auxiliary feedwater system Trains A and B while the turbine-driven auxiliary feedwater train was out of service for maintenance
- Partial walkdown of centrifugal charging Train A while Train B was out of service for maintenance
- Complete walkdown of the component cooling water system

b. Findings

No findings of significance were identified.

1R05 Fire Protection (71111.05)

a. Inspection Scope

The inspectors reviewed the following areas to determine if the licensee had implemented a fire protection program that adequately controlled combustibles and ignition sources within the plant, effectively maintained fire detection and suppression capabilities, and maintained passive fire protection features in good material condition. The areas reviewed were:

- Fire protection pumphouse and associated equipment
- Service water/essential service water piping area
- Emergency exhaust fan and filter rooms
- Auxiliary building piping penetration rooms
- Control room fire protection Panel KC08 and associated impairments

b. Findings

No findings of significance were identified.

1R06 Flood Protection Measures (71111.06)

a. Inspection Scope

The inspectors performed a periodic review of equipment and flooding mitigation plans in two risk significant areas. The inspectors reviewed these features in the basement of the control building and in the engineered safety features switchgear rooms. The inspectors verified that flood barriers were intact and that floor drains were not obstructed. Seals designed to protect against flood water were adequate. Piping that could cause a flooding event in these areas was also intact.

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalifications (71111.11)

a. Inspection Scope

On February 22, 2001, the inspectors observed a simulator exercise for operations personnel. The inspectors reviewed the scenario, which involved a loss of all ac power during refueling operations. The inspectors evaluated crew communications and performance, command and control, use of procedures, emergency plan usage, and fidelity of the simulator to the actual control room.

b. Findings

No findings of significance were identified.

1R12 Maintenance Rule Implementation (71111.12)

a. Inspection Scope

The inspectors verified proper implementation of the maintenance rule to assess the effectiveness of maintenance efforts. Specifically, the inspectors verified structure and component scoping, characterization, safety significance, performance criteria, and the appropriateness of goals and corrective actions. These aspects of the Maintenance Rule were reviewed for the following components:

- 125 Vdc vital battery charger Number 4
- Centrifugal charging Pump A discharge miniflow to seal water heat exchanger isolation Valve BGHV8110
- Chemical volume and control system letdown heat exchanger outlet header pressure relief Valve BGV0020
- Excore neutron monitoring system
- Containment purge exhaust radiation gas Monitor GTRE0033
- Chemical and volume control system moderating heat exchanger to reactor coolant filter control Valve BGUV7022
- Essential service water Pump B

b. Findings

No findings of significance were identified.

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

a. Inspection Scope

Throughout the inspection period, the inspectors reviewed the daily and weekly schedules to determine when risk significant activities were scheduled and to verify how the licensee managed risk. The inspectors discussed selected activities with operations and work control personnel regarding risk evaluations and plant configuration control. The inspectors evaluated the following risk evaluations:

- Removing Breaker NG0216 (tie breaker between electrical Bus NG02 and Bus NG04) from service for maintenance
- Simultaneous inoperability of essential service water Pump B and containment cooler Train A
- Risk assessment for weeks beginning January 22 and February 12
- Risk assessment for extended operation in Mode 2

b. Findings

No findings of significance were identified.

1R14 Personnel Performance During Nonroutine Evolutions and Events (71111.14)

.1 Routine Observations

a. Inspection Scope

The inspectors observed operator performance during a surveillance test that measured the moderator temperature coefficient. The surveillance was performed once every 18 months using Procedure ESP-ZZ-00010, “At-Power Moderator Temperature Coefficient Measurement,” Revision 16. The inspectors reviewed the plans for the surveillance, procedure, briefing, and contingency plans.

b. Findings

No findings of significance were identified.

.2 Boron Thermal Regeneration System

a. Inspection Scope

The inspectors assessed the circumstances surrounding an unidentified reactor coolant system leak in excess of limits allowed by Technical Specifications.

b. Findings

Early on the day shift on January 10, 2001, operations personnel had indication that there was a problem with the boron thermal regeneration system valve alignment. The indication was that the dilute light on Handswitch HIS-27 on the main control board was not lit. A prerequisite for placing a cation demineralizer in the boron thermal regeneration system in service was that the light be lit. The cation demineralizer was to be placed in service later that evening on the night shift. The cation demineralizer was used to reduce lithium concentration and fission products in the reactor coolant system. Operations personnel were aware that a valve alignment problem in this system could potentially overpressurize the boron thermal regeneration system and cause a reactor coolant system leak.

During the day shift and during the shift turnover at approximately 6:30 p.m., operations personnel discussed that a valve alignment verification would have to be performed prior to bringing the cation demineralizer into service.

To verify the valve alignment, operations personnel had to verify the position of 13 valves that provided input to the light on Handswitch HIS-27. Twelve of the valves were verified using the plant computer. The remaining valve (chemical volume and control system moderating heat exchanger to reactor coolant filter control Valve BGUV07022) could not be verified by the plant computer. The position of this valve had to be locally verified. However, operations personnel concluded that this valve was in its correct position (open) because there had been no significant change in letdown flow. Since there was no significant change in letdown flow, operations personnel concluded that there was a position indication problem with Valve BGUV07022 that was causing the valve alignment problem.

At 8:58 p.m., operators entered Off-Normal Procedure OTO-BB-00003, “Reactor Coolant System Excessive Leakage,” Revision 3, when they observed a decreasing level in the volume control tank. The decrease in volume control tank level was equated to a 2 gpm unidentified reactor coolant system leak. Technical Specification 3.4.13 was entered due to the leakage exceeding the allowed 1 gpm. Using Procedure OTO-BB-00003, operators quickly identified the cause of the leak and isolated the boron thermal regeneration system. Procedure OTO-BB-00003 was exited at 9:10 p.m.

During the subsequent evaluation, operations personnel discovered that the air supply valve to Valve BGUV07022 was approximately 80 percent closed. This valve failed closed on loss of air. With the air supply valve 80 percent closed, Valve BGUV07022 drifted closed and valves in the boron thermal regeneration system were overpressurized. The diaphragms of these valves were overpressurized and leaked reactor coolant system inventory. The valve diaphragms were rated at 300 psig.

The licensee determined that, on January 9, quality control personnel entered the high radiation area where the air isolation valve was located to perform a surveillance. The licensee believed that, during this surveillance, the air supply valve was inadvertently bumped out of position. Since the air supply valve was 80 percent closed,

Valve BGUV07022 drifted closed. This drift caused the time delay between the air supply being bumped and Valve BGUV07022 going closed.

Failing to aggressively pursue and identify the source of a valve alignment problem that overpressurized the boron thermal regeneration system had an actual impact on safety, affected the integrity of the reactor coolant system, and resulted in an unidentified reactor coolant system leak in excess of Technical Specification limits. However, once operators identified the leak, they took quick action to isolate the leak and limited the leakage to approximately 50 gallons. As a result, this finding is considered to have very low safety significance (Green) using the significance determination process qualitatively with the assistance of a senior reactor analyst. This issue was entered into the licensee's corrective action program as Suggestion-Occurrence-Solution Report 01-0095.

.3 Operator Response to Reactor Trip

a. Inspection Scope

The inspectors evaluated operator response following a reactor trip that occurred when the output breaker of rod drive motor generator Set A tripped when rod drive motor generator Set B was being removed from service for maintenance. Operators properly characterized the reactor trip and responded appropriately to the plant. All safety-related equipment operated as designed. The licensee planned to report the circumstances surrounding this reactor trip in Licensee Event Report 50-483/2001-003-00.

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations (71111.15)

a. Inspection Scope

The inspectors reviewed the following evaluations to ensure that operability was properly justified and the component or system remained available:

- Suggestion-Occurrence-Solution Report 00-3185, load swings during periodic one-hour run of Diesel Generator A
- Suggestion-Occurrence-Solution Report 00-2401, centrifugal charging Pump A discharge miniflow to seal water heat exchanger valve
- Safety injection Pump A room Cooler SGL09A
- Motor-driven auxiliary feedwater to steam Generator B hand Valve ALHV0009

- Suggestion-Occurrence-Solution Report 01-0515, essential service water Pump B
- Callaway Action Request 01-01168, essential service water to turbine-driven auxiliary feedwater pump check valve ALV0012
- Asiatic clams found in the essential service water supply line to motor-driven auxiliary feedwater Pump B

b. Findings

No findings of significance were identified.

1R16 Operator Workarounds (71111.16)

a. Inspection Scope

The inspectors reviewed the cumulative effects of existing operator workarounds on the reliability, availability, and potential for misoperation of a system, the potential for increasing the frequency of an initiating event that could affect multiple mitigating systems, and the effects on the ability of operations personnel to respond in a correct and timely manner to plant transients and accidents.

b. Findings

No findings of significance were identified.

1R17 Permanent Plant Modifications (71111.17B)

a. Inspection Scope

The inspectors reviewed procedures governing plant modifications to evaluate the effectiveness of the programs for implementing modifications to risk-significant systems, structures, and components to ensure these changes did not adversely affect the design and licensing basis of the facility. The inspectors also reviewed 10 permanent plant modification packages to verify that they were performed in accordance with plant procedures.

The inspectors conducted field walkdowns of three permanent plant modifications. The inspectors interviewed the cognizant design and system engineers for the identified modifications as to their understanding of the modification packages.

The inspectors evaluated the effectiveness of the licensee's corrective action process to identify and correct problems concerning the performance of permanent plant modifications. In this effort, the inspectors reviewed Callaway action requests and the subsequent corrective actions pertaining to licensee-identified problems and errors in the performance of permanent plant modifications.

b. Findings

No findings of significance were identified.

1R19 Postmaintenance Testing (71111.19)

a. Inspection Scope

The inspectors observed or evaluated the following postmaintenance tests to determine whether they were adequate to verify system operability and functional capabilities:

- Component cooling water Pump B
- Component cooling water heat Exchanger B, Valve EGTV0030
- Essential service water Pump B
- Emergency Diesel Generator A
- Turbine-driven auxiliary feedwater pump
- Turbine-driven auxiliary feedwater pump to Steam Generator B, Valve ALHV0012

b. Findings

No findings of significance were identified.

1R20 Refueling and Outage Activities (71111.20)

a. Inspection Scope

The inspectors reviewed the outage risk control plan to verify that the licensee considered appropriate risk, industry experience, and previous site problems. The inspectors verified that the licensee had mitigation strategies in place to respond to losses of key safety functions during the refueling outage scheduled to begin April 7, 2001.

b. Findings

No findings of significance were identified.

1R22 Surveillance Testing (71111.22)

.1 Routine Surveillance Testing

a. Inspection Scope

The inspectors observed or reviewed the following surveillance tests to ensure the systems tested were capable of performing their safety function and to assess their operational readiness. Specifically, the inspectors verified that the following surveillance tests met Technical Specifications, ASME Section XI test requirements, the Final Safety Analysis Report, and licensee procedural requirements:

- Procedure ESP-ZZ-00010, "At-Power Moderator Temperature Coefficient Measurement," Revision 16
- Procedure OPS-AL-P001A, "MDAFP 'A' Pump Inservice Test," Revision 24
- Procedure OSP-EN-P001A, "Containment Spray Pump 'A' Inservice Test," Revision 20
- Procedure OSP-AC-00003, "Turbine Control Valve Stroke Test," Revision 7
- Procedure OSP-AL-P0002, "Turbine Driven Aux Feedwater Pump Operability Inservice Test," Revision 29
- Procedure OSP-AL-V001C, "TD Auxiliary Feedwater Valve Operability," Revision 21

b. Findings

No findings of significance were identified.

.2 Trip Actuating Device Operational Test of the Reactor Trip Breaker

a. Inspection Scope

On January 12, 2001, the inspectors observed portions of the trip actuating device operational test of reactor trip breaker Train A to ensure the systems tested were capable of performing their safety function and to assess their operational readiness.

b. Findings

Operations and maintenance personnel were performing Train A trip actuating device operational testing in accordance with the following procedures:

- Procedure OSP-SB-0001A, "Reactor Trip Breaker 'A'-Trip Actuating Device Operational Test," Revision 9

- Procedure ISF-SB-00A29, “Fctnal-Anal; SSPS Train A Fctnal Test,” Revision 22

The rod drive electrical alignment was such that both the bypass and the reactor trip breakers were shut, in accordance with the procedure. Shortly after 9 a.m., maintenance personnel failed to place the input error inhibit switch in the inhibit position as directed by step 6.2.6 of Procedure ISF-SB-00A29. At 9:25 a.m. the maintenance personnel did not receive proper indication when logic Switch A at the test cabinet was rotated to Position 14 as directed by step 6.3.2 of Procedure ISF-SB-00A29. While cycling the test switch to clean the contacts, the technicians rotated the switch one full turn back to Position 14 and reactor trip Breaker A opened. (Power was maintained to hold the control rods through reactor trip bypass Breaker A and reactor trip Breaker B.) The control room supervisor then directed all personnel to stop testing. A review by the control room staff and personnel involved revealed the following:

- Source range nuclear Instrument N31 flux trip caused the reactor trip breaker to open. This occurred when the test switch was rotated through Position 21. The plant computer recorded that source range Channel 1 alarmed in the “Hi-Hi” condition and then indicated “BAD.”

- The input error inhibit switch was not in the inhibit position as directed by step 6.2.6 of Procedure ISF-SB-00A29, yet the procedural step had been initialed as being performed. (The input error inhibit switch serves to block a high voltage signal from being provided to the source range detector and consequently damaging it.)

Both procedures were then resumed and testing was completed without further incident. The inspectors verified that the licensee met all applicable Technical Specification action statements. The licensee made the correct notifications in the time required in accordance with 10 CFR 50.72(b)(2)(ii) for an event or condition that results in a manual or automatic actuation of any engineered safety feature, including the reactor protection system. The licensee considered the source range power detector, Instrument N31, inoperable because it was potentially damaged from exposure to a power range neutron flux for approximately 4 minutes. Since the source range detectors are normally de-energized at power, the licensee could not determine if damage had been done until the plant was in a shutdown condition. Subsequent to the reactor trip on March 9, 2001, source range nuclear Instrument N31 was determined to be operable.

Technical Specification 5.4.1 requires, in part, that written procedures shall be established, implemented, and maintained covering the applicable procedures recommended in Appendix A of Regulatory Guide 1.33, Revision 2, February 1978. Regulatory Guide 1.33, Section 8.l, addresses “Reactor Protection System Tests and Calibrations.” Failure to follow the trip actuating device operational test procedure resulted in an unplanned reactor protection system actuation and was a violation of Technical Specification 5.4.1. This violation is being treated as a noncited violation, consistent with Section VI.A.1 of the NRC Enforcement Policy (50-483/0102-01). This violation was entered into the licensee’s corrective action program as Callaway Action Request 01-00116.

This finding had a credible impact on safety due to its potential to cause or increase the frequency of an initiating event (i.e., reactor trip). This noncited violation was characterized as having very low safety significance (Green) through the use of the significance determination process. Although the potential existed for a slight increase in the likelihood of an uncomplicated reactor trip, other equipment designed to mitigate the effect of a reactor trip was available and the reactor trip bypass breaker had been closed prior to the inadvertent opening of the reactor trip breaker.

.3 Inoperability of Essential Service Water Pump B

a. Inspection Scope

The inspectors reviewed the circumstances surrounding the inoperability of essential service water Pump B. The pump became inoperable when a 20 foot section of reinforced tygon hose fell into the suction bay of the pump and blocked a portion of the pump suction.

b. Findings

At 5:07 a.m., on February 14, 2001, the licensee removed containment cooler Train A from service for a routine breaker changeout. Later that morning, Surveillance OSP-SA-00017A, “Train A SIS/CSAS Slave Relay Test,” was scheduled. At 8:51 a.m., essential service water Pump B was started as directed by the surveillance procedure. Essential service water was the safety-related supply to the containment coolers and also the containment spray pump's room coolers. Following the start, the pump failed to achieve rated flow and discharge pressure. The pump was secured and inspected by maintenance and engineering personnel. The pump was restarted and failed to achieve rated flow and discharge pressure. Normal discharge pressure and discharge flow were approximately 140 psi and 7 Mlbm/hr, respectively. Actual discharge pressure and discharge flow were approximately 80 psi and 4 Mlbm/hr, respectively. The discharge flow and pressure were insufficient for the pump to perform its safety function. Additionally, pump vibration was excessive. Normal vibration measured at the motor was approximately 0.1 inches/second. Actual vibration at the motor was approximately 0.5 inches/second. The inservice testing required action level for motor vibration was 0.5 inches/second. As a result, essential service water Pump B was declared inoperable as of 8:51 a.m.

With essential service water Pump B inoperable and containment cooler Train A removed from service for breaker maintenance, three of the four trains used for containment cooling were now inoperable. Technical Specification 3.6.6.F.1 required that, with three of the four trains used to cool containment inoperable, Technical Specification 3.0.3 be entered. At 11:15 a.m., Technical Specification 3.0.3 was retroactively entered as of 8:51 a.m. The reason for the delay in entering Technical Specification 3.0.3 was that the operations staff had difficulty using Procedure ODP-ZZ-00027, “Safety Function Determination Program.” This procedure was used by operators to assist in determining whether or not a loss of a safety function had occurred. Once operators determined that a loss of safety function for the containment spray pump room cooler and the containment cooler had occurred,

Technical Specification 3.0.3 was entered. At 11:23 a.m., the licensee exited Technical Specification 3.0.3 when containment cooler Train A was returned to service.

The condition at the time of discovery could have prevented the fulfillment of a safety function for components needed to control the release of radioactive material or mitigate the consequences of an accident. As a result, this condition was reported to the NRC at 4:05 p.m. in accordance with 10 CFR 50.72(b)(3)(v)(D).

A diver was dispatched to inspect the suction bay of the pump. The diver discovered a 20-foot section of reinforced tygon hose wrapped around the rotor assembly in the first stage impeller of the pump casing, blocking a portion of the pump's suction path. The hose was approximately 1.5 inches in diameter. There was also a hose clamp and camlock fitting attached to the hose. The tygon hose had been used to direct leakage from the pump's prelube tank to the suction bay of the pump. The tygon hose, camlock fitting, and clamp were removed and the discharge strainer was inspected. No material from the tygon hose was found during the strainer inspection. The pump passed subsequent retests and was declared operable at 2:31 a.m., on February 15.

The licensee performed an investigation and determined that on February 9 operations personnel were notified that the prelube tank in essential service water Pump B room was leaking and spilling onto the floor. The prelube tank was not required for pump operability. Operations personnel directed plant helpers to install a funnel under the tank and direct the leakage to the pump suction bay using a tygon hose. Plant helpers installed the funnel, attached a 20-foot section of tygon hose to the funnel using duct tape, and routed the hose through a 6-inch floor penetration into the suction bay. The funnel and hose were installed at approximately 2 p.m. Later that afternoon, different plant helpers were directed by other operations personnel to install a hose from the funnel that was hanging from the prelube tank. Due to inadequate communication between plant helpers and operations personnel, no one questioned the location of the first hose or why a second hose needed to be hung. Plant helpers did wonder why the funnel was installed with no hose attached, but their concern was not brought to the attention of their supervisor. A second tygon hose was hung from the funnel at approximately 7 p.m. The first tygon hose had fallen off the funnel and through the 6-inch floor penetration into the suction bay of the pump. The tygon hose remained undetected inside the suction bay of the pump from February 9-14, when the pump was started as directed by Surveillance OSP-SA-0017A.

During the root cause evaluation completed March 12, 2001, the licensee determined that there was not a procedure for installing temporary drain hoses. Since there was no procedure, there also was no method for controlling the installation and removal of drain hoses. Hoses installed to route system leakage were also exempted from temporary modification requirements.

During the event review team meeting, plant helpers stated that on February 12 they recognized that a second hose had been hung, but they concluded that the first hose could not have come loose and fallen into the suction bay. As a result, the location of the first hose was not questioned.

The Technical Specification allowed outage time for an inoperable essential service water pump was 72 hours. The tygon hose in the suction bay of the pump caused the pump to be inoperable from February 9-14 (approximately 132 hours). Since the allowed outage time was exceeded, Technical Specification 3.7.8.B required that the plant be placed in Mode 3 within 6 hours and in Mode 5 within 36 hours. The licensee failed to comply with this Technical Specification requirement. This issue will remain unresolved pending a review of the safety significance by an NRC Senior Reactor Analyst (Unresolved Item 50-483/0102-02). This issue is in the licensee's corrective action program as Solution-Occurrence-Solution Report 01-0515.

1R23 Temporary Plant Modifications (71111.23)

a. Inspection Scope

Since April 1, 2000, the licensee has not installed any risk significant temporary modifications. As a result, none were available for review.

b. Findings

No findings of significance were identified.

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

a. Inspection Scope

The inspectors reviewed Change Notice 00-008 to Revision 23 of the Callaway Plant Radiological Emergency Response Plan, transmitted by the licensee on February 5, 2001, and Procedure EIP-ZZ-00101, "Classification of Emergencies," Revision 27, transmitted by the licensee on February 6, 2001, to determine if the revisions decreased the effectiveness of the emergency plan.

b. Findings

No findings of significance were identified.

1EP6 Drill Evaluation (71114.06)

a. Inspection Scope

The inspectors observed simulator exercises conducted on January 12 and February 22, 2001. The purpose of these observations was to evaluate operator performance, licensee event classification, notification of state and local authorities, and adequacy of protective action recommendations. The inspectors also reviewed the licensee's critiques of the exercises to determine if they were self-critical in the identification of strengths and performance issues.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA1 Performance Indicator Verification

a. Inspection Scope

The inspectors reviewed the 2000 PI data submittal to verify the accuracy and completeness of data used to calculate and report safety system unavailability for the residual heat removal system and emergency diesel generators.

b. Findings

No findings of significance were identified.

4OA3 Event Followup (71153)

(Closed) Licensee Event Report 50-483/1998-010-00 and Licensee Event Report 50-483/1998010-01: failure to verify the position of several valves in the system flow path or that serviced other safety-related equipment. The licensee initiated corrective action program document Callaway Action Request 199803836. The licensee event report was reviewed and no findings of significance were identified. This licensee event report is closed.

(Closed) Licensee Event Report 50-483/1999-004-00: missed Technical Specification surveillance due to erroneous procedure guidance. The licensee initiated corrective action program document Suggestion-Occurrence-Solution Report 99-1558. The licensee event report was reviewed and no findings of significance were identified. This licensee event report is closed.

(Closed) Licensee Event Report 50-483/2000-007-00: unplanned emergency diesel generator actuation resulting from inadvertent actuation of the local emergency start device. This issue was previously discussed in NRC Inspection Report 00-14. This licensee event report is closed.

(Closed) Licensee Event Report 50-483/2000-001-00: hydrogen Recombiner A inoperable for over 30 days due to swapped leads. No new information was provided. This licensee event report is closed.

(Closed) Licensee Event Report 50-483/2001-001-00: missed step in test procedure results in an invalid reactor trip breaker actuation due to the system being improperly removed from service. This event was discussed in Section 1R22.2. This licensee event report is closed.

4OA6 Management Meetings

Exit Meeting Summary

The emergency preparedness inspector presented inspection results to Mr. M. Evans, Manager, Operations Support, and other members of licensee management by telephone on March 2, 2001. The region based reactor inspectors presented inspection results to Mr. M. Taylor, Manager, Nuclear Engineering, and other members of licensee management at the conclusion of their inspections on March 9 and 23, 2001. The resident inspectors presented inspection results to Mr. R. Affolter, Vice President Nuclear, and other members of licensee management at the conclusion of their inspection on March 30, 2001. The licensee acknowledged the inspection results.

The inspectors asked the licensee whether any materials examined during the inspection should be considered proprietary. No proprietary information was identified.

ATTACHMENT

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee

R. Affolter, Vice President, Nuclear
D. Bettenhausen, Supervising Engineer
J. Blosser, Manager, Corporate Nuclear Services
S. Bond, Supervising Engineer
M. Evans, Manager, Operations Support
T. Herrmann, Superintendent, Steam Generator Replacement/Accident Analysis
M. Hillstrom, Engineer
D. Hollabaugh, Superintendent, Design Engineering
R. Lamb, Superintendent, Work Control
J. Laux, Manager, Quality Assurance
D. Maxwell, Supervisor, Mechanical Engineering Design
J. McGraw, Superintendent, Technical Support Engineering
T. Moser, Superintendent, Systems Engineering
W. Muskopf, 50.59 Program Engineer
A. Passwater, Plant Manager
M. Reidmeyer, Supervisor, Regional Regulatory Affairs
K. Schoolcraft, Senior Engineer, Quality Assurance Regulatory Support
C. Slizewski, Supervising Engineer
M. Taylor, Manager, Nuclear Engineering
D. Waller, Supervisor, Electrical Engineering Design
W. Witt, Plant Manager

ITEMS OPENED AND CLOSED

Opened

50-483/0102-01	NCV	Inadvertent reactor protection system actuation during trip actuating device operational testing (Section 1R22.2)
50-483/0102-02	URI	Inoperable essential service water pump (Section 1R22.3)

Closed

50-483/0102-01	NCV	Inadvertent reactor protection system actuation during trip actuating device operational testing (Section 1R22.2)
50-483/1998-010-00	LER	Failure to verify the position of several valves in the system flow path or that serviced other safety-related equipment (Section 4OA3)

50-483/1998-010-01	LER	Failure to verify the position of several valves in the system flow path or that serviced other safety-related equipment (Section 4OA3)
50-483/1999-004-00	LER	Missed Technical Specification surveillance due to erroneous procedure guidance (Section 4OA3)
50-483/2000-007-00	LER	Unplanned emergency diesel generator actuation resulting from inadvertent actuation of the local emergency start device (Section 4OA3)
50-483/2000-001-00	LER	Hydrogen Recombiner A inoperable for over 30 days due to swapped loads (Section 4OA3)
50-483/2001-001-00	LER	Missed step in test procedure results in an invalid reactor trip breaker actuation due to the system being improperly removed from service (Section 4OA3)

DOCUMENTS REVIEWED

The following documents were selected and reviewed by the inspectors to accomplish the objectives and scope of the inspection and to support any findings:

Equipment Alignment

Final Safety Analysis Report Section 9.2.2, “Cooling System For Reactor Auxiliaries”

Checkoff List 1, “Fuel Building HVAC System Valve and Damper Lineup”

Checkoff List 2, “Fuel Building HVAC System Breaker Lineup”

Checkoff List 1, “Component Cooling Water System Normal Power Supply”

Checkoff List 2, “Component Cooling Water System Normal Valve Lineup”

Checkoff List 5, “Component Cooling Water System Control Board Lineup”

M-22EN01(Q), “Piping and Instrumentation Diagram Containment Spray System,” Revision 6

M-22AL01(Q), “Piping and Instrumentation Diagram Auxiliary Feedwater System,” Revision 17

M-22BG03(Q), “Piping and Instrumentation Diagram Chemical and Volume Control System,” Revision 44

Drawing M-22EN01(Q), “Piping & Instrumentation Diagram Containment Spray System,” Revision 6

Drawing M-22EG01(Q), “Piping and Instrumentation Diagram Component Cooling Water System,” Revision 11

Procedure OTN-GG-0001, “Fuel Building HVAC System,” Revision 10

Procedure OTN-EG-0001, “Component Cooling Water System,” Revision 18

Drawing M-22AL01(Q), “Piping & Instrumentation Diagram Auxiliary Feedwater System,” Revision 17

Drill Evaluation

Procedure EIP-ZZ-00212, “Protective Action Recommendations,” Revision 18
Procedure EIP-ZZ-00201, “Notifications,” Revision 33
Procedure EIP-ZZ-00101, “Classification of Emergencies,” Revision 27

Maintenance Rule Implementation

Suggestion-Occurrence-Solution Reports 00-1926 and 00-2821
Expert Technical Panel Meeting Minutes NET 00-155, dated November 15, 2000
Expert Technical Panel Meeting Minutes NET 00-169, dated December 13, 2000
Expert Technical Panel Meeting Minutes NET 01-011, dated January 17, 2001
Expert Technical Panel Meeting Minutes NET 01-024, dated February 16, 2001

Operability Evaluations

Safety evaluation for Suggestion-Occurrence-Solution Report 01-0515
Safety evaluation for Callaway Action Request 01-00876
Safety evaluation for Callaway Action Request 01-01168

Operator Workarounds

Workaround list dated February 2, 2001

Postmaintenance Testing

Retest Documents R633019A, R204968A, S668799, R203834A, S666457

Surveillance Testing

Surveillance Task Sheets S664321 and S664308
Callaway Action Request 01-00116
Surveillance Task Sheets S664224, S663918,

Performance Indicator Verification

Equipment out of service logs
Operator logs
Various licensee event reports
Maintenance rule reporting data
Various corrective action documents

Safety Evaluations

MP 99-1044	Replace EDG Governor Control System
RFR 20681	Determine Maximum Allowable Stroke Time for Motor Operated Valves
CMP 98-1025	Change Power Source for DG Supply Fans
CMP 98-1004	Replace Feed Water Isolation Valve Actuators
RFR 13311	Change BGV 0638 to a Normally Opened Valve
RFR 20601	Change Stroke Time for EG System Containment Isolation Valves
RFR 06597	Operational Evaluation of EDG's During Cross-Connected Operations
RFR 20019	Operational Evaluation for Twelve Motor Operated Valves