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

March 7, 2000

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

SUBJECT: NRC INSPECTION REPORT NO. 50-483/00-01

Dear Mr. Randolph:

This refers to the inspection conducted on January 9 through February 19, 2000, at the Callaway Plant facility. The enclosed report presents the results of this inspection.

Based on the results of this inspection, the NRC has determined that two Severity Level IV violations of NRC requirements occurred. These violations are being treated as noncited violations, consistent with Section VII.B.1.a of the Enforcement Policy. These noncited violations are described in the subject inspection report. If you contest the violations or severity level of these noncited violations, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, 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 Part 2.790 of the NRC's "Rules of Practice," a copy of this letter, its enclosure, and your response, if requested, will be placed in the NRC Public Document 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 No.: 50-483
License No.: NPF-30

Enclosure:
NRC Inspection Report No.
50-483/00-01

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

H. D. Bono, Supervising Engineer
Quality Assurance Regulatory Support
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 Court House
10 East Fifth Street
Fulton, Missouri 65151

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

bcc to DCD (IE01)

bcc electronic distribution from ADAMS by RIV:
Regional Administrator **(EWM)**
DRP Director **(KEB)**
DRS Director **(ATH)**
Senior Resident Inspector **(VGG)**
Branch Chief, DRP/B **(WDJ)**
Senior Project Engineer, DRP/B **(RAK1)**
Branch Chief, DRP/TSS **(LAY)**
RITS Coordinator **(NBH)**

<u>Only inspection reports</u> to the following:
D. Lange **(DJL)**
NRR Event Tracking System **(IPAS)**
Document Control Desk **(DOCDESK)**
CWY Site Secretary **(DVY)**
Wayne Scott **(WES)**

bcc hard copy:
RIV File Room

DOCUMENT NAME: R:_CW\CW2000-01RP-VGG.wpd
To receive copy of document, indicate in box: "C" = Copy without enclosures **"E"** = Copy with enclosures **"N"** = No copy

RIV:RI:DRP/B		SRI:DRP/B		C:DRP/B					
JDHanna		VGGaddy		WDJohnson					
3/7/00 (WDJ)		3/7/00 (WDJ)		3/7/00 /RA/					

OFFICIAL RECORD COPY

ENCLOSURE

U.S. NUCLEAR REGULATORY COMMISSION
REGION IV

Docket No.:	50-483
License No.:	NPF-30
Report No.:	50-483/00-01
Licensee:	Union Electric Company
Facility:	Callaway Plant
Location:	Junction Highway CC and Highway O Fulton, Missouri
Dates:	January 9 through February 19, 2000
Inspectors:	V. G. Gaddy, Senior Resident Inspector J. D. Hanna, Resident Inspector
Approved By:	W. D. Johnson, Chief, Project Branch B

ATTACHMENT: Supplemental Information

EXECUTIVE SUMMARY

Callaway Plant
NRC Inspection Report No. 50-483/00-01

This routine announced inspection included aspects of licensee operations, engineering, maintenance, and plant support activities. This report covers a 6-week period of resident inspection.

Operations

- The licensee's guidance for controlling access to the switchyard while at power was lacking. Operations personnel authorized and security personnel granted access to the switchyard. However, after security personnel granted access, additional personnel and equipment could enter the switchyard without further authorization from operations personnel. Licensee contingency planning procedures did recommend that switchyard entries be limited to operations personnel during critical evolutions such as midloop operations and reduced inventory conditions (Section O1.1).

- Control room operators demonstrated good command and control during the reactor startup following a reactor trip on February 13. Reactor engineering personnel and control room operators were attentive during the approach to criticality and subsequent reactor startup (Section O1.2).

- Breaker protection for a faulted 161 kV power line in southeast Missouri did not operate. This caused significant fluctuations on the licensee’s switchyard buses. These fluctuations caused all reactor coolant pumps and all circulating water pumps to trip. The reactor subsequently tripped on low reactor coolant system flow. Without reactor coolant pumps, decay heat was removed by natural circulation. Operators quickly characterized the event and made restarting reactor coolant pumps a priority. The risk assessment for this event showed that the event had low to moderate risk significance (Section O2.2).

- Control room operators failed to identify a decreasing level in a component cooling water surge tank. Surge tank level decreased below the acceptance criteria. Operators documented that the surge tank level was below the acceptance criteria while taking control room logs. Operators did not document the suspected cause of the low surge tank level or take action to restore the level in the surge tank as required by procedure. Failing to take action to restore the surge tank level when it trended below the minimum acceptance criteria is a violation of 10 CFR Part 50 Appendix B, Criterion V. Operators took action to stop the decreasing level when the surge tank low level annunciator alarmed. This Severity Level IV violation is being treated as a noncited violation, consistent with Section VII.B.1.a of the NRC Enforcement Policy (Section O4.1).

- The inspectors identified several deficiencies that contributed to a containment isolation valve being removed from service without proper time tracking or retests being specified. These deficiencies included the licensee’s inattention to detail regarding the scheduling and isolation for the containment isolation valve, the lack of communication between the shift supervisor and the shift technical advisor, and not recognizing that removing the valve from service placed the plant in a limiting condition for operation (Section O4.2).

- Training provided to operations personnel on degraded switchyard voltage was well organized and gave guidance on how to detect an undervoltage condition and how to restore offsite power to an operable status (Section O5.1).

- In the event of a strike, the licensee did not plan to train replacement operators as crews prior to standing watch. By not training operators as crews, the licensee would miss an opportunity to develop crew team building and an opportunity to evaluate how replacement crews' members interacted with each other and how they performed under normal and emergency situations (Section O8.1).

Maintenance

- The inspectors noted an increased number of secondary plant steam leaks. These leaks had all been identified by the licensee and had been prioritized for repair. The inspectors also determined that the operability of surrounding equipment was not affected (Section M2).

Engineering

- The licensee was experiencing approximately 2 to 3 gallons per day leakage from a safety injection accumulator. Leakage from the accumulator was occurring through an accumulator test line. The licensee has been proactive in their attempts to stop the leakage from the accumulator and the reactor coolant system. The licensee does not have a program to monitor for nitrogen voids in the safety injection system and the residual heat removal system piping. The current guidance for venting the safety injection and residual heat removal systems did not require the licensee to take any action unless 30 seconds of gas was vented from a vent location. The licensee did not have a technical basis for the 30 second criteria. The licensee planned to install a manual isolation valve in the safety injection test line system to prevent leakage from the accumulator and to stop leakage past reactor coolant system check valves (Section E1.1).

- In 1997, the licensee identified that the reactor coolant system leakage detection system was outside its design basis because a 1 gpm leak could not be detected within 1 hour as required. Although outside design basis, the licensee failed to report this condition as required by 10 CFR Part 50.72. Failing to report this condition was a violation. This Severity Level IV violation is being treated as a noncited violation consistent with Section VII.B.1.a of the NRC Enforcement Policy. This violation was entered into the licensee’s corrective action program as Suggestion-Occurrence-Solution Report 99-3541 (Section E8.2).

Plant Support

- During a containment building entry, health physics technicians demonstrated good ALARA techniques and compliance with the licensee's radiological procedures (Section R1.2).

Report Details

Summary of Plant Status

The plant began the report period at 100 percent power. On February 13, switchyard voltage fluctuations occurred. These voltage fluctuations caused reactor coolant Pump B to trip on a current phase imbalance. Following the reactor coolant pump trip, at approximately 7:34 a.m., the reactor tripped on low reactor coolant system flow. At 9:10 p.m. the reactor was restarted and at 5:18 a.m., on February 14, the main generator was synchronized to the electric grid. Full power was reached on February 16 and the plant remained at 100 percent for the remainder of the inspection period. Details about the reactor trip are discussed in Section O2.2 of this report.

I. Operations

O1 Conduct of Operations

O1.1 General Comments (71707)

The inspectors conducted frequent reviews of ongoing plant operations. In general, the conduct of operations was professional and safety conscious. Plant status, operating problems, and work plans were appropriately addressed during daily turnover and plan-of-the-day meetings. Plant testing and maintenance requiring control room coordination were properly controlled. The inspectors observed several shift turnovers and noted no problems.

The inspectors observed nonlicensed operators performing their duties throughout the inspection period. On February 9, 2000, the inspectors accompanied the outside operator on his rounds. The operator was very knowledgeable and performed his duties in a satisfactory manner. During the tour, the inspectors noted that several bottles of compressed gas were stored in the switchyard. The inspectors asked why the bottles were in the switchyard. The inspectors learned that the bottles were used on February 4 to add gas to switchyard breakers and they were waiting to be removed. The bottles were subsequently removed from the switchyard.

The inspectors asked if there was any procedural guidance that governed switchyard access and equipment that could be taken into the switchyard. Access was authorized by operations personnel and granted by security personnel. However, after security personnel granted access, additional personnel and equipment could enter the switchyard without further authorization from operations personnel.

The licensee's contingency planning procedures recommended that switchyard entries be limited to operations personnel during critical evolutions such as midloop and reduced inventory.

O1.2 Observation of Reactor Startup

a. Inspection Scope (71707)

On February 13, operators commenced reactor startup and achieved criticality following a reactor trip due to a fault on the electrical grid in southeast Missouri. The inspectors observed the plant startup and related activities.

b. Observations and Findings

Prior to commencing the startup, operators conducted a briefing and limited personnel access to the control room to minimize distractions. The reactor startup was conducted in accordance with Procedure OTG-ZZ-00002, "Reactor Startup," Revision 26. Control room operators adhered to procedural requirements and performed the reactor startup cautiously and methodically. Operators utilized three-way communications during the evolution. A reactor engineer was stationed in the control room to perform independent verifications of subcritical multiplication and reactivity calculations as required by the procedure. Criticality was achieved within the range allowed by the estimated critical rod position calculation. The inspectors found that the licensee maintained proper control room decorum and performed the reactor startup with few problems. A detailed assessment of the reactor trip and associated operator response appears in Section O2.2 of this report.

c. Conclusions

Control room operators demonstrated good command and control during the reactor startup following a reactor trip on February 13. Reactor engineering personnel and control room operators were attentive during the approach to criticality and subsequent reactor startup.

O2 Operational Status of Facilities and Equipment

O2.1 Engineered Safety Feature System Walkdowns (71707)

The inspectors walked down accessible portions of the following engineered safety features and vital systems:

- auxiliary feedwater system
- safety injection system
- containment spray system

Equipment operability, material condition, and housekeeping were acceptable.

02.2 Automatic Reactor Trip Due to Low Reactor Coolant System Flow

a. Inspection Scope (71707)

The inspectors followed up to determine the circumstances surrounding a reactor trip.

b. Observations and Findings

On February 13, 2000, at 7:33 a.m., voltage fluctuations occurred on the licensee's offsite power buses. Normal switchyard bus voltage was approximately 360 kV. The fluctuations varied voltage from 337 kV to 373 kV. At 7:34 a.m., reactor coolant Pump B tripped on a current phase imbalance. Immediately after the reactor coolant pump tripped, the reactor tripped on low reactor coolant system flow. The remaining three reactor coolant pumps and all three circulating water pumps tripped on a current phase imbalance. Without forced reactor coolant flow, decay heat was removed by natural circulation. The voltage fluctuations lasted for approximately 12 minutes.

At approximately 8 a.m., the licensee started reactor coolant Pumps A and D, restoring forced circulation. At 9:40 a.m., the remaining two reactor coolant pumps were started.

Following a reactor trip, pressure increases were normally controlled by the pressurizer spray valves. However, with all reactor coolant pumps tripped, the pressurizer spray driving force was unavailable. As a result, pressurizer pressure increased, causing a pressurizer power-operated relief valve to lift and then reseat. The power-operated relief valve's setpoint was 2335 psig. At the same time that the power-operated relief valve lifted, the safety valve open annunciator alarmed in the control room. Although the safety valve open annunciator alarmed, there were no other indications that a safety valve lifted. Safety valve tail pipe temperature remained normal and pressure remained constant. While troubleshooting the annunciator, the licensee determined that a reed switch that input to the annunciator was out of adjustment. The reed switch was repaired and the annunciator cleared.

When the circulating water pump tripped, condenser vacuum was lost, and the steam dump valves were unavailable to regulate pressure. Steam dumps normally operated at 1092 psig at no load. Temperature and pressure were controlled using steam generator atmospheric dump Valve D. The relief setpoint for this valve was 1125 psig. At approximately 9 a.m., the licensee restarted two circulating water pumps, reestablished condenser vacuum, and closed steam generator atmospheric dump Valve D.

All safety systems operated as designed and all control rods inserted. The offsite power sources remained operable.

The cause of the fluctuating switchyard voltage and subsequent reactor trip was a downed 161 kV power line in southeast Missouri. Breakers designed to isolate the downed line did not operate. This caused repeated high current until a 345 kV/161 kV transformer, that provided the interface with the licensee's electrical grid, failed. The affected power line, breakers, and transformers were owned and serviced by another (non-nuclear) utility. The licensee planned to meet with this utility to determine why their line protection devices failed. The licensee was also evaluating methods to reduce their susceptibility to similar events.

Operators' performance, following the reactor trip, was good. They properly characterized the event and established proper priorities to mitigate the event. Restarting reactor coolant pumps to reestablish forced circulation was a priority. Although operators had indication that a safety valve had lifted, operators quickly determined that this indication was incorrect. This incorrect indication had minimal effect on how operators responded to the event.

The licensee performed a risk assessment of the event. The analysis showed that this event was of low to moderate risk significance. Region IV senior reactor analysts reviewed the results of the licensee's risk assessment and concluded that the licensee had used a reasonable approach in assessing the overall risk of the event.

c. Conclusions

Breaker protection for a faulted 161 kV power line in southeast Missouri did not operate. This caused significant fluctuations on the licensee's switchyard buses. These fluctuations caused all reactor coolant pumps and all circulating water pumps to trip. The reactor subsequently tripped on low reactor coolant system flow. Without reactor coolant pumps, decay heat was removed by natural circulation. Operators quickly characterized the event and made restarting reactor coolant pumps a priority. The risk assessment for this event showed that the event had low to moderate risk significance.

O4 Operator Knowledge and Performance

O4.1 Decreasing Level in the Component Cooling Water Surge Tank

a. Inspection Findings

The inspectors followed up to determine why control room operators did not notice a decreasing level in a component cooling water surge tank.

b. Observations and Findings

At midnight on January 26, 2000, reactor operators' logs indicated that the component cooling water surge Tank A level was 72 percent. At 10:31 p.m. on January 27, the component cooling water surge Tank A low level annunciator alarmed. The low level alarm setpoint was 45 percent. Operators entered off-normal Procedure OTO-EG-00001, "Component Cooling Water System Malfunction." Operators started component cooling water Pump B as directed by procedure, transferred all safety loads to the opposite train, and began troubleshooting to isolate the leak from the component cooling water system.

During troubleshooting, the licensee narrowed the source of leakage from the surge tank to the radiation waste building. As directed by procedure, operators closed Valves EGHV69 and EGHV70 (radiation waste building supply and return valves). Closing these valves stopped the leakage. The leakage was estimated to be 0.6 gallons per minute. From midnight January 2, until the low level alarm, component cooling water level in Surge Tank A decreased approximately 27 percent.

Operations personnel reviewed the reactor operators' logs and noted that the level in component cooling water surge Tank A had steadily decreased over the previous 20 hours. On January 26, the midnight, 8 a.m., and 4 p.m. surge tank levels were 72 percent, 72 percent, and 68 percent, respectively. On January 27, the midnight, 8 a.m., and 4 p.m. surge tank readings were recorded as 62 percent, 55 percent, and 49 percent, respectively. The January 27, 4 p.m. reading was below the 50 percent acceptance criteria. After each set of readings was taken, they were reviewed by the control room supervisor. The control room supervisor did not investigate to determine the cause of the decrease in surge tank level.

The inspectors learned that, during the time that the surge tank level was decreasing, the licensee was performing work on component cooling water Pump B and on valves in the radiation waste building. The licensee indicated that operators may have assumed that the steady decrease in surge tank level was due to ongoing work.

Although a decrease in surge tank level had developed, operators did not investigate to determine the reason for the negative trend. The 4 p.m. surge tank reading was below the acceptance criteria and operators did not take action to correct the condition as required by Procedure ODP-ZZ-00016, "Reactor Operator Watchstation Practices and Logs." Steps 3.3.2 and 3.3.3 of this procedure required reactor operators to report conditions which were out of specification to the shift supervisor or control room supervisor and document the suspected cause and action taken to correct the condition. Step 3.2.3 required the shift supervisor or control room supervisor to provide guidance to reactor operators to correct the out of specification condition. Failing to recommend and take action to correct the out of specification component cooling water surge tank level was a violation of 10 CFR Part 50 Appendix B, Criterion V. This Severity Level IV violation is being treated as noncited violation, consistent with Section VII.B.1.a of the NRC Enforcement Policy. This violation was entered into the licensee's corrective action program as Suggestion-Occurrence-Solution Report 00-0185 (50-483/00001-01).

The licensee identified other operator performance issues. To address this adverse trend, the licensee initiated Suggestion-Occurrence-Solution Report 00-00375.

c. <u>Conclusions</u>

Control room operators failed to identify a decreasing level in a component cooling water surge tank. Surge tank level decreased below the acceptance criteria. Operators documented that the surge tank level was below the acceptance criteria while taking control room logs. Operators did not document the suspected cause of the low surge tank level or take action to restore the level in the surge tank as required by procedure. Failing to take action to restore the surge tank level when it trended below the minimum acceptance criteria is a violation of 10 CFR Part 50, Appendix B, Criterion V. Operators stopped the decreasing level when the surge tank low level annunciator alarmed. This Severity Level IV violation is being treated as a noncited violation, consistent with Section VII.B.1.a of the NRC Enforcement Policy.

O4.2 Errors Associated with Work on Containment Isolation Valve

a. Inspection Scope (71707)

On January 12, 2000, control room operators recognized that a retest had not been performed for containment isolation Valve EJHCV8825. Maintenance work on this valve had been completed the previous day.

b. Observations and Findings

On January 6, 2000, the licensee placed workman's protection assurance tags during a containment entry. This was done in preparation for spring adjustment work to be performed on containment isolation Valve EJHCV8825 the following week. (The placement of this workman's protection assurance isolation did not render the valve inoperable. Therefore, entry into Technical Specification Action Statement 3.6.3 for containment isolation valves was not required.) The shift supervisor then placed a hold off tag on the isolation valve in order to prevent the work from being performed without the control room staff being informed. On January 7, the work document was reviewed and approved by the licensee's planning department.

On January 10, the shift technical advisor performed a review of workman's protection assurance isolation for scheduled work. The containment isolation valve work did not appear on this report because it had not been scheduled. Because these work documents and associated isolation did not appear on this report, an equipment out-of-service list was not generated at the time. (An equipment out-of-service list is the licensee's mechanism to track Technical Specification allowed outage times and other time restricted items in order to verify completion. Typically, this mechanism is controlled by the shift technical advisor.) Later that morning, the work document was added to the schedule for the following day. No retest was scheduled for the containment isolation valve. The retest was added after the daily planning meeting; therefore, the retest document did not appear on the schedule until the following day.

On January 11, at 7:07 a.m., the hold-off tag was removed and the work was performed. The licensee considered the valve to be inoperable at the time the hold-off tag was removed. The shift technical advisor was not informed that the hold-off tag had been removed. Consequently, the licensee did not generate an equipment out-of-service list for the valve.

On January 12, the shift supervisor noted that the retest document for Valve EJHCV8825 had not been completed and informed the control room immediately. Operators then realized that Valve EJHCV8825 was a containment isolation valve. The licensee entered Technical Specification Action Statement 3.6.3 due to the inoperability of the valve. Further investigation revealed that the Technical Specification action statement requirements were met, during the period in question, by having the upstream valve, Valve EJHV8840, closed with its power removed. The equipment out-of-service list was retroactively entered by the licensee and the retests were satisfactorily performed.

The inspectors did not identify any noncompliance with the licensee's Technical Specifications or procedures. The inspectors did identify several deficiencies that contributed to this event. These included:

- the licensee's inattention to detail regarding the retest scheduling and the associated workman's protection isolation for the valve,

- lack of communication between the shift supervisor and the shift technical advisor with respect to the lifting of the hold-off tag on the work, and

- not recognizing that removing the valve from service placed the plant in a limiting condition for operations.

The inspectors reviewed the licensee's corrective actions for this event and found them to be adequate. The licensee entered this occurrence into its corrective action program as Suggestion-Occurrence-Solution Report 00-0061.

c. Conclusions

The inspectors identified several deficiencies that contributed to a containment isolation valve being removed from service without proper time tracking or retests being specified. These deficiencies included the licensee's inattention to detail regarding the scheduling and isolation for the containment isolation valve, the lack of communication between the shift supervisor and the shift technical advisor, and not recognizing that removing the valve from service placed the plant in a limiting condition for operations.

O5 Operator Training and Qualification

O5.1 Degraded Switchyard Voltage Training (71707)

On January 28, 2000, the inspectors attended training given to operations personnel regarding degraded switchyard voltage. This issue was the subject of an NRC special inspection. Results of the inspections are documented in NRC Inspection Report 50-483/99-15.

The training provided an overview of the issue, short- and long-term corrective actions, and a review of procedures for ensuring that offsite voltage remained above the minimum requirement. The training was well organized and gave guidance on how to detect a degraded switchyard condition as well as guidance on how to restore offsite power to an operable status.

O8 Miscellaneous Operations Issues (92901)

O8.1 Strike Contingency Planning

a. Inspection Scope (71707 and 92709)

The inspectors assessed the licensee’s strike contingency plan.

b. Observation and Findings

In preparation for a potential work stoppage, the licensee developed a contingency plan. The plan was dated October 7, 1999. The plan identified replacement personnel and outlined their responsibilities if a work stoppage occurred. In addition to senior reactor operators that currently stand watch, other personnel from operations and training were identified as replacement operators. The inspectors reviewed the contingency plan and noted that, although replacement operators were qualified, not all routinely stood watch. The inspectors asked if replacement operators were going to be given additional training prior to standing watch to evaluate how they perform as a crew and to give operators an opportunity to work together. The licensee stated that no additional training was planned for the crews prior to assuming watch. The licensee did reconfigure the crews to ensure that only the most qualified individuals would stand watch. No other issues with the contingency plan were identified.

c. Conclusions

In the event of a strike, the licensee did not plan to train replacement operators as crews prior to standing watch. By not training operators as crews, the licensee would miss an opportunity to develop crew team building and an opportunity to evaluate how replacement crews' members interacted with each other and how they performed under normal and emergency situations.

II. Maintenance

M1 Conduct of Maintenance

M1.1 General Comments - Maintenance

a. Inspection Scope (62707)

The inspectors observed or reviewed portions of the following work activities:

P601455	Turbine-driven auxiliary feedwater pump outboard bearing oil inlet switch
P633799	Turbine-driven auxiliary feedwater pump speed controller calibration

b. Observations and Findings

All work observed was performed with the work packages present and in active use. The inspectors inspected the inside of the speed controller cabinet. No dust or debris accumulation was noted. Internal wiring was intact and properly terminated. The inspectors frequently observed supervisors and system engineers monitoring job progress and quality control personnel were present when required.

M1.2 General Comments - Surveillance

a. Inspection Scope (61726)

The inspectors observed or reviewed all or portions of the following test activities:

- Surveillance Procedure OSP-NE-00001B, "Standby Diesel Generator B Periodic Tests," Revision 5,

- Test Procedure OSP-GK-0001A, "A Train Control Room Filtration and Pressurization System Monthly Operability Verification," Revision 0, and

- Surveillance Procedure OSP-NE-00001A, "Standby Diesel Generator A Periodic Tests," Revision 5.

b. Observations and Findings

The surveillance testing was conducted satisfactorily and in accordance with the licensee's approved programs and the Technical Specifications.

M2 Maintenance and Material Condition of Facilities and Equipment

M2 Review of Material Condition During Plant Tours

During this inspection period, the inspectors noted an increasing number of secondary plant steam leaks. The inspectors noted five steam leaks. The valves were:

- AFV00087, first stage reheat drain Tank B emergency drain upstream isolation valve,

- AFV00615, first stage reheat drain tank to high pressure Heater 6B drain valve,

- AFV00538, first stage reheat drain Tank D downstream isolation valve,

- AEV00978, feedwater high pressure Heater 7B pressure relief valve, and

- BMV00077, steam generator nonregenerative heat exchanger tube side drain.

The valves had body-to-bonnet leakage or would not completely seat when closed. The steam leaks did not appear to affect the operation of other equipment in the area. The leaking valves had been identified by the licensee and were prioritized for repair.

III. Engineering

E1 Conduct of Engineering

E1.1 Reactor Coolant System Check Valve and Safety Injection Accumulation Leakage

a. Inspection Scope (37551)

The inspectors followed the licensee's actions in addressing reactor coolant system check valve leakage.

b. Observations and Findings

After starting up from refueling Outage 10, the licensee noted that safety injection Accumulator D level was increasing. Level increased approximately 10 gallons per day. There was no leakage into the remaining three accumulators. There was no increase in safety injection or residual heat removal discharge pressures.

To eliminate the leakage, the licensee systematically isolated manual valves in the safety injection system to identify the source of the leakage. They increased the spring tension closing force on selected air-operated valves in the safety injection test line system.

The licensee narrowed the source of the leakage into the accumulator to two leaking check valves on reactor coolant system Loop 4, Valve EPHV8877D (safety injection Accumulator D isolation valve), Valve EMHV8889A (safety injection Pump B loop hot leg isolation hand valve), and Valve EJHCV8825 (safety injection test line isolation valve). Other valves could also be leaking.

The licensee placed the safety injection test regulator in service. The regulator was controlled by Procedure OTN-EM-00001, "Safety Injection System." Placing the regulator in service caused leakage into the accumulator to divert to the recycle holdup tank. This eliminated the operator work-around of draining the accumulator to maintain the level required by Technical Specifications.

Following isolation of manual valves, increasing spring tension closing force, and installation of the test regulator, leakage to the recycle holdup tank was reduced to approximately 0.25 gpm.

Leakage into the accumulator stopped when the test regulator was placed in service. With Valve EPHV8877D (safety injection Accumulator D isolation valve) leaking by, approximately 2 to 3 gallons of water per day are now being lost from the accumulator.

This valve is scheduled to be repaired during refueling Outage 11. The inspectors asked if there was a concern that leakage out of the accumulator could cause nitrogen voids in the safety injection and residual heat removal systems' piping. The inspectors also asked if there was a program to monitor nitrogen accumulation. The licensee stated that they did not have a nitrogen monitoring program and any nitrogen accumulation concerns were mitigated because each month the safety injection and residual heat removal systems were vented. In November 1999, December 1999, and February 2000, no gas was noted while venting. However, in January 2000, gas was noted in two locations in the residual heat removal system and one location in the safety injection system. The licensee stated that the gas could have been from the accumulator but, since it was not analyzed, its exact source could not be determined. The licensee indicated that if gas continues to be observed, the venting frequency would be increased.

The licensee's venting procedure required an analysis to be performed if gas was vented for approximately 30 seconds from a vent location. If 30 seconds of gas was vented, the procedure only required the chemistry department to sample for hydrogen, not nitrogen. The procedure then required that the surveillance completion form be forwarded to engineering for review and trending. In January 2000, 9 seconds was the longest that gas was vented from any location. The inspectors asked what the basis was for the 30 second requirement. The licensee did not have a basis for the 30 second requirement.

To prevent leakage to the recycle holdup tank and to stop leakage from safety injection Accumulator D, the licensee planned to install a manual isolation valve downstream of Valve EMHV8889A (safety injection Pump B Loop 1 hot leg injection valve) during this cycle. Once the manual isolation valve was installed, the licensee planned to remove the safety injection test regulator from service. This issue was being tracked under Suggestion-Occurrence-Solution Report 00-00026.

c. Conclusions

The licensee was experiencing approximately 2 to 3 gallons leakage from a safety injection accumulator per day. Leakage from the accumulator was occurring through an accumulator test line. The licensee has been proactive in their attempts to stop the leakage from the accumulator and the reactor coolant system. The licensee does not have a program to monitor for nitrogen voids in the safety injection system and the residual heat removal system piping. The current guidance for venting the safety injection and residual heat removal systems did not require the licensee to take any action unless 30 seconds of gas was vented from a vent location. The licensee did not have a technical basis for the 30 second criteria. The licensee planned to install a manual isolation valve in the safety injection test line system to prevent leakage from the accumulator and to stop leakage past reactor coolant system check valves.

E8 Miscellaneous Engineering Issues (92903)

E8.1 (Closed) Licensee Event Report 50-483/98003-01: inadvertent actuation of the engineered safety features actuation system due to high water level in Steam Generator A during refueling Outage 9.

This licensee event report was initially discussed and closed in NRC Inspection Report 50-483/98-12, issued August 12, 1998. On May 12, 1999, the licensee revised their corrective action to prevent recurrence and issued Revision 1 to the licensee event report. Originally the licensee stated that Procedure OSP-AE-V0003B, "Feedwater Supply Check Valve Closure Test," would be revised to jumper out the feedwater isolation signal during Mode 5 surveillance testing to prevent this signal from occurring. This signal was not required to be functional during Mode 5. The May 1999 submittal indicated that Procedure OSP-AE-V0003B would be revised to incorporate a lower initial steam generator level condition for nitrogen addition to reduce the potential for a feedwater isolation signal actuation. The inspectors had no further concerns.

E8.2 (Closed) Licensee Event Reports 50-483/99009-00 and 50-483/99009-01: reactor coolant system leakage detection system is outside design basis because a 1 gpm leak may not be detected within 1 hour.

This issue was discussed in NRC Inspection Report 50-483/99-14. This report discussed the failure to take corrective action when it was identified that the containment normal sump measurement and containment air cooler condensate flow rate system may not be able to detect a 1 gpm leak within 1 hour as required by the Updated Final Safety Analysis Report. This discrepancy was initially identified in 1997.

Although the discrepancy was identified in 1997, the licensee failed to report that the plant was outside its design basis as required by 10 CFR Part 50.72(b)(1)(ii)B. Failing to report that the plant was outside its design basis was a violation. This Severity Level IV violation is being treated as a noncited violation consistent with Section VII.B.1.a of the NRC Enforcement Policy. This violation was entered into the licensee's corrective action program as Suggestion-Occurrence-Solution Report 99-3541 (50-483/00001-02).

IV. Plant Support

R1 Radiological Protection and Chemistry Controls

R1.1 General Comments (71750)

The inspectors observed health physics personnel, including supervisors, routinely touring the radiologically controlled areas. Licensee personnel working in radiologically controlled areas exhibited good radiation worker practices.

Contaminated areas and high radiation areas were properly posted. The inspectors checked a sample of doors, required to be locked for the purpose of radiation protection, and found no problems.

R1.2 Observation of Health Physics' Coverage During Containment Building Walkdown

a. Inspection Scope (71750)

On February 17, 2000, the licensee entered the containment building while at power to perform several maintenance activities. These activities included a general walkdown of the containment building. The inspectors accompanied the health physics technicians and system engineers during this tour in order to assess the radiological controls implemented.

b. Observations and Findings

In preparation for the containment entry, health physics technicians held a prejob ALARA (as low as is reasonably achievable) briefing for all participating individuals. The inspectors assessed this briefing for its thoroughness and compliance with licensee Procedure HTP-ZZ-01102, “Pre-job ALARA Planning and Briefing,” Revision 14. The individual presenting the brief identified areas of concern in the containment building from the perspectives of personnel safety, ALARA, and reactor safety. These concerns included heat stress effects on individuals, low dose wait areas, and components that posed a potential reactor trip hazard. The individual specified various elevated radiation areas in containment (e.g., bioshield access doors) and techniques to avoid these areas. Based on these observations, the inspectors found the briefing to be comprehensive and well performed.

The inspectors then accompanied two licensee engineers and two health physics technicians during their tour of the containment building. (This tour was performed for the purposes of a general walkdown in order to look for leaking components and other deficient conditions, obtain temperature data on certain components, and evaluate the steam leakage from a feedwater check valve bypass valve.) The inspectors observed the technicians' compliance with Procedure HTP-ZZ-03100, “Performing Radiation Surveys,” Revision 3. Health physics technicians displayed good practices by maintaining visual contact with all individuals in the group and performing radiation surveys ahead of the group as it moved through different areas in containment. This ensured that individuals did not accidentally enter an abnormally high radiation area. The health physics technicians demonstrated good ALARA practices by frequently performing radiation surveys and having members of the group move to lower dose areas.

c. Conclusions

During a containment building entry, health physics technicians demonstrated good ALARA techniques and compliance with the licensee's radiological procedures.

V. Management Meetings

X1 Exit Meeting Summary

The exit meeting was conducted on February 18, 2000. The licensee did not express a position on any of the findings in the report.

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

ATTACHMENT

SUPPLEMENTAL INFORMATION

PARTIAL LIST OF PERSONS CONTACTED

Licensee

R. D. Affolter, Manager, Callaway Plant
G. N. Belchik, Supervising Engineer, Operations
J. D. Blosser, Manager, Operations Support
D. G. Cornwell, Supervisor, Electrical Work Control
G. J. Czeschin, Superintendent, Training
J. W. Dowling, Supervisor, Electrical Work Control
M. S. Evans, Superintendent, Protective Services
R. F. Farnam, Supervisor, Health Physics Operations
J. W. Hiller, Engineer, Quality Assurance
R. T. Lamb, Superintendent, Work Control
J. V. Laux, Manager, Quality Assurance
G. L. Randolph, Vice President and Chief Nuclear Officer
M. A. Reidmeyer, Supervisor, Regional Regulatory Affairs
R. R. Roselius, Superintendent, Radiation Protection and Chemistry
J. D. Schnack, Supervising Engineer, Quality Assurance Corrective Action
K. C. Schoolcraft, Senior Engineer, Quality Assurance
M. E. Taylor, Manager, Nuclear Engineering
R. C. Wink, Engineer, System Engineering

INSPECTION PROCEDURES USED

37551	Onsite Engineering
61726	Surveillance Observations
62707	Maintenance Observations
71707	Plant Operations
71750	Plant Support Activities
92700	Onsite Follow Up of Written Reports of Nonroutine Events at Power Reactor Facilities
92709	Licensee Strike Contingency Plan
92903	Follow Up - Engineering
93702	Prompt Onsite Response to Events At Operating Power Plants

ITEMS OPENED AND CLOSED

Opened

00000-01	NCV	Failure to correct the out of specification component cooling water surge tank level (Section O4.1).
00000-02	NCV	Reactor coolant system leakage detection system is outside design basis (Section E8.2).

Closed

00000-01	NCV	Failure to correct the out of specification component cooling water surge tank level (Section O4.1).
98003-01	LER	Inadvertent actuation of the engineered safety features system is (Section E8.1).
99009-00	LER	Reactor coolant system leakage detection system is outside design basis (Section E8.2).
99009-01	LER	Reactor coolant system leakage detection system is outside design basis (Section E8.2).
00000-02	NCV	Reactor coolant system leakage detection system is outside design basis (Section E8.2).