ENCLOSURE

U.S. NUCLEAR REGULATORY COMMISSION
REGION IV

Docket No.:	50-483
License No.:	NPF-30
Report No.:	50-483/99-14
Licensee:	Union Electric Company
Facility:	Callaway Plant
Location:	Junction Highway CC and Highway O Fulton, Missouri
Dates:	November 28, 1999, through January 8, 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. 50-483/99-14

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

- Unclear terminology in the annunciator response procedure and control room operators not recognizing the appropriate power supply for the steam generator level control circuitry resulted in an equipment operator being dispatched to verify the status of the wrong power supply feeder breaker. The operator was dispatched to the secondary power supply feeder breaker. He should have been dispatched to the primary power supply feeder breaker. With the primary power supply already inoperable, the equipment operator attempted to ensure that the secondary power supply breaker was fully closed by pushing it to its fully closed position. When this occurred, the secondary power supply momentarily lost power which caused the main feedwater regulating valve for Steam Generator A to close and both main feedwater pumps to go to their low speed stop position. This caused a reactor trip on a low level in Steam Generator A. Operator response following the reactor trip was good (Section O1.2).

- Configuration control, material condition, and alignment of the residual heat removal system were good. This was evident by the sound state of the mechanical and electrical portions of the system and of the associated support systems (e.g., components were properly aligned, adequately supported, reasonably free of oil, boron, or other leakage, and minimal corrosion was observed) (Section O2.1).

- The inspectors verified that the licensee properly controlled the use of overtime during refueling Outage 10 (Section O8.1).

Maintenance

- The inspectors concluded that snubbers in Area 5 (main steam and feedwater piping area) were not adversely affected by elevated temperatures (Section M2.1).

- An inadequate procedure was the cause of a turbine setback that reduced reactor power to approximately 88 percent. The procedure was initially written to be performed while shutdown. The review that was performed later to allow the procedure to be performed at power did not identify that enabling the turbine setback protective function while locking out a circulating water pump would cause a turbine setback. Once operators recognized the turbine setback, they responded quickly to disable the turbine setback protective function and stop the power reduction. This is a violation of 10 CFR Part 50 Appendix B, Criterion V. 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 99-3576 (Section M3.1).

- The licensee continued to experience minor work control errors during the second half of the refueling outage. The licensee identified a potential trend adverse to quality in the workman's protection assurance program (Suggestion-Occurrence-Solution Report 99-3114). The licensee attributed these recent failures to personnel errors due to confusion over tagging requirements (Section M4.1).

Engineering

- On December 3, 1999, the licensee reported that due to a computer software discrepancy the potential existed for a 1 gpm reactor coolant system leak to not be detected within 1-hour as stated in the Updated Safety Analysis Report. The computer programming discrepancy was corrected, restoring compliance. During research, the licensee identified that this problem was identified in 1997. A software change was developed but never implemented. Failing to take correction action when the problem was identified in 1997 was a violation of 10 CFR Part 50, Appendix B, Criterion XVI. 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 E2.1).

- The licensee's training requirements for various engineering positions were comprehensive for the functions being performed. Engineering department contact with industry peers and coordination on emerging technical issues was appropriate (Section E5.1).

Plant Support

- The licensee's compliance with its security plan was verified in the areas of exterior illumination levels, control of unattended vehicles, and security fence integrity (Section S2.1).

Report Details

Summary of Plant Status

The plant began the report period at low power following a reactor trip which occurred on November 26, 1999, due to a low level in Steam Generator A. On November 30, reactor power reached approximately 100 percent. On December 8, a turbine setback reduced reactor power to 88 percent. Full power was reached on December 9, and the plant remained at 100 percent for the remainder of the inspection period.

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. Housekeeping was generally good and discrepancies were promptly corrected. Safety systems were found to be properly aligned. Specific events and noteworthy observations are detailed below.

O1.2 Reactor Trip Due to a Low Level in Steam Generator A

a. Inspection Scope (71707 and 93702)

The inspectors assessed the circumstances surrounding the reactor trip which was due to a low level in Steam Generator A.

b. Observations and Findings

On November 26, 1999, at 12:04 p.m., Annunciator 93A, "Power Control System (PCS) Power Failure," alarmed. The licensee determined that the primary power supply (26 VDC) in control Cabinet RP043 had failed, causing the alarm. The secondary power supply (24 VDC) immediately picked up the loads supplied by the primary power supply. The primary power supply provided power to the steam generator level control circuitry.

Control room operators reviewed annunciator response Procedure OTA-RL-RK093, "Windows 93A Through 93F," Revision 5, and directed the equipment operator to verify the position of Breaker PG19GCR218. This was the feeder breaker to the secondary power supply. Control room operators should have directed the equipment operator to verify the position of the feeder breaker to the primary power supply. The annunciator response procedure did not use the terminology of primary/secondary power supplies, but, instead referred to the power supplies by their output voltage ratings. The control room operator mistakenly thought that the 24 VDC power supply was the primary power supply so they directed the equipment operator to verify the status of the 24 VDC power supply and its feeder breaker.

When the equipment operator arrived at the feeder breaker (120/240 Volt Gould molded case breaker) for the secondary power supply, it did not appear to be in its fully closed position. The equipment operator then attempted to ensure that the breaker was closed by pushing it to its fully closed position. When this occurred, main feedwater regulating Valve A closed and both main feedwater pumps shifted to manual control and went to their low speed stops (3600 rpm). Normal speed was approximately 4700 rpm. Operators attempted to take manual control of the main feedwater regulating valve and the main feedwater pumps. They were unsuccessful, and the reactor tripped on a low level in Steam Generator A.

All safety-related equipment operated as designed during the reactor trip.

Following the reactor trip, maintenance personnel tested secondary feeder Breaker PG19GCR218. Testing revealed that, if the breaker was in its closed position but not closed against its stop, attempts to push the breaker against its stop resulted in contact bounce. The bouncing caused momentary loss of power to equipment being supplied by the power supply. Design engineering personnel also determined that a momentary loss of power to the digital main feedwater regulating valve control circuitry would result in a loss of control function for at least five seconds while controls reset. Additionally, the main feedwater pump speed controller lost power. Upon reenergizing, the controller driver came up in the manual mode with zero percent output. Therefore, operators were not able to take manual control of the equipment in time to prevent a reactor trip. Troubleshooting revealed that the primary power supply failed due to a defective fuse holder which was internal to the power supply.

Operator performance following the reactor trip was good. There were no axial offset anomalies during the trip.

Following replacement of the primary power supply and replacement of the feeder breaker to the secondary power supply, the licensee began a reactor startup on November 27. Full power operation was reached on November 30.

Additional corrective actions included the following:

- Revising the annunciator response procedure to clearly identify the feeder breakers for the primary and secondary power supplies,
- Issuing a night order directing operations personnel to not attempt to verify the position of 120/240 Volt Gould molded case breakers,
- Training operators on the response of main feedwater controls following a loss of control power, and

- Coordinating industry notification of the circuit breaker's operational characteristics through the Institute of Nuclear Power Operations.

c. Conclusions

Unclear terminology in the annunciator response procedure and control room operators not recognizing the appropriate power supply for the steam generator level control circuitry resulted in an equipment operator being dispatched to verify the status of the wrong power supply feeder breaker. The operator was dispatched to the secondary power supply feeder breaker. He should have been dispatched to the primary power supply feeder breaker. With the primary power supply already inoperable, the equipment operator attempted to ensure that the secondary power supply feeder breaker was fully closed by pushing it to its fully closed position. When this occurred, the secondary power supply momentarily lost power, which caused the main feedwater regulating valve to Steam Generator A to close and both main feedwater pumps to go to their low speed stop position. This caused a reactor trip on a low level in Steam Generator A. Operator response following the reactor trip was good.

O2 Operational Status of Facilities and Equipment

O2.1 Residual Heat Removal System Walkdown

a. Inspection Scope (71707)

The inspectors performed a detailed walkdown of the accessible portions of the residual heat removal system to independently verify its operability. The review included portions of the auxiliary building, the control building, and the control room. The inspectors also conducted a walkdown of ac and dc electrical systems and equipment that supports the residual heat removal system.

b. Observations and Findings

The inspectors checked the interiors of electrical circuit breaker cabinets and verified them to be free of debris, loose material, and unauthorized jumpers. Power supplies and breakers were correctly aligned, functional, and available for components that must activate upon receipt of an actuation signal. The mechanical piping penetration rooms, the pump rooms, and the heat exchanger rooms were found to be free of ignition sources and flammable materials. Cleanliness was acceptable in all areas inspected.

Components were properly labeled and correctly positioned in accordance with system drawings and the Updated Safety Analysis Report. Valves were found to be free of excessive packing or boron leakage. Pipe hangers and supports were adequate. The inspectors also performed a review of the residual heat removal system against the Technical Specifications, design basis documents, and system lineup Procedure OTN-EJ-00001, "Residual Heat Removal System," Revision 14, to verify consistency with the as-built configuration. No discrepancies were noted.

c. Conclusions

Configuration control, material condition, and alignment of the residual heat removal system were good. This was evident by the sound state of the mechanical and electrical portions of the system and of the associated support systems (e.g., components were properly aligned, adequately supported, reasonably free of oil, boron, or other leakage, and minimal corrosion was observed).

O8 Miscellaneous Operations Issues (92700)

O8.1 Review of Licensee's Use of Overtime

a. Inspection Scope (71707)

The inspectors reviewed the licensee's use of overtime for licensed reactor and senior reactor operators and others engaged in safety-related activities during refueling Outage 10.

b. Observations and Findings

The inspectors randomly reviewed the use of overtime for two operating crews. The inspectors selected October because approximately 85 percent of refueling Outage 10 was performed during that month. The inspectors reviewed the working hours of 20 individuals, including shift supervisors, operating supervisors, reactor operators, and equipment operators. The inspectors verified the licensee's conformance with the requirements of Technical Specification 6.2.2.f and Generic Letter 82-12, Nuclear Power Plant Staff Working Hours.

The inspectors found that the working hour limitations of Technical Specification 6.2.2.f and Generic Letter 82-12 were met for all of the selected individuals. The personnel did not exceed the limitations of the above guidance (e.g., no more than 16 hours continuously, no more than 16 hours in any 24-hour period, etc.).

c. Conclusions

The inspectors verified that the licensee properly controlled the use of overtime during refueling Outage 10.

O8.2 (Closed) Licensee Event Report (LER) 50-483/9908-00: reactor trip due to a low steam generator level resulting from the loss of power to a feedwater control cabinet. Details surrounding this reactor trip are discussed in Section O1.2 of this report.

II. Maintenance

M1 Conduct of Maintenance

M1.1 General Comments

a. Inspection Scope (62707)

The inspectors observed all or portions of the following maintenance activities:

- A506702A, Remove Core Load Jumper for Valve EMHV8807B
- P506702, Service Limitorque Operator for Valve EMHV8807B
- C644794, Rewiring of Control Circuitry for Valve EMHV8923
- C644791, Hot Shot Rewiring for Valve EMHV8821B

b. Observations and Findings

The inspectors identified no concerns.

M1.2 General Comments on Surveillance Activities

a. Inspection Scope (61726)

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

- Test Procedure OSP-NE-0001B, "Standby Diesel Generator 'B' Periodic Tests," Revision 5,

- Test Procedure OSP-SB-0001A, "Reactor Trip Breaker 'A' - Trip Actuating Device Operational Test," Revision 7,

- Test Procedure ISF-SB-00A29, "Fctnal - Anal : SSPS Train 'A' Fctnl Test," Revision 20,

- Test Procedure OSP-AL-P0001A, "Motor Driven Aux Feedwater Pump 'A' Inservice Test," Revision 22,

- Test Procedure OSP-AL-V0001B, "Train 'B' Auxiliary Feedwater Valve Operability," Revision 17,

- Test Procedure OSP-EM-P001B, "Safety Injection Train 'B' Inservice Test," Revision 20, and

- Test Procedure OSP-EM-V001B, "Safety Injection System Train 'B' Valve Inservice Test," Revision 14.

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

M2.1 Snubber Performance

a. Inspection Scope (61726)

The inspectors evaluated whether elevated temperature in Area 5 (main steam and feedwater piping area) had a detrimental effect on snubbers in the area.

b. Observations and Findings

During a tour of Area 5, the inspectors noted that the temperature was elevated due to the main steam and feedwater piping that passed through the area. The inspectors asked engineering personnel if the elevated temperature had an adverse effect on snubbers in the area. The licensee determined that there were 95 snubbers in Area 5. All were manufactured by Pacific Scientific. Since plant startup, 53 surveillances had been performed on the snubbers. None had failed.

The inspectors asked how the elevated temperature affected the grease inside the snubbers. The inspectors learned that all 95 snubbers used Chevron NRR Grease 95. This grease had an operating range of 10°F to 325°F. The licensee randomly measured the surface temperature of a few snubbers. Surface temperatures ranged from 87°F to 135°F. When the surface temperatures were recorded, the average room temperature was 101°F.

During fuel Cycle 10, the Area 5 temperature ranged from approximately 83°F to 105°F. Based on this information, it did not appear that the snubbers in Area 5 were being adversely affected by temperature.

c. Conclusions

The inspectors concluded that snubbers in Area 5 (main steam and feedwater area) were not adversely affected by elevated temperatures.

M3 Maintenance Procedure and Documentation

M3.1 Main Turbine Setback and Accompanying Decrease in Reactor Power

a. Inspection Scope (72707 and 71707)

The inspectors reviewed the circumstances of a main turbine setback, which occurred during surveillance testing of the main circulating water pumps, on December 8, 1999.

b. Observations and Findings

On December 8, the reactor was at 100 percent power and relay services group personnel were performing surveillances on the circulating water pump breakers. This activity was being coordinated and controlled by control room operators. At approximately 12:41 p.m., main condenser vacuum decreased to 4 inches of mercury due to the relatively warm atmospheric temperatures. In accordance with Procedure ODP-ZZ-00016, "Reactor Operator Watchstation Practices & Logs," Revision 40, the operators placed the turbine setback switch, on the main control board, to "enable." The turbine setback protective function serves to automatically reduce main turbine load to 75 percent when the circuitry perceives a loss of a main circulating water pump. This helps ensure that the reactor does not trip due to the loss of condenser vacuum.

At approximately 3:51 p.m., relay services personnel initiated a lockout signal on circulating water Pump C during the performance of tripping sequence checks. This action was directed by Procedure MPE-ZZ-NY142, "Operational Test Sequence of 4.16 KV Circulating Water Pump DPDA 2101C Air Circuit Breaker 152PB12302," Revision 6. With the turbine setback switch on the control board in "enable," in conjunction with the circulating water pump lockout signal, the circulating pump lockout breaker relay initiated a turbine setback.

The operators properly responded to the event in accordance with Procedure OTO-MA-00007, "Turbine Setback," Revision 3. The turbine setback was stopped at approximately 88 percent power when the shift supervisor placed the control board switch to "disable." When the plant was stabilized, operators commenced raising reactor power to 100 percent.

The inspectors reviewed the circumstances surrounding this transient. The inspectors found that the control room operators and relay services personnel properly followed the test procedure. The test procedure did not address the position of the control board setback arming switch. Prior to 1996 this test had been performed while in a shutdown condition, thereby making this protective function not applicable. This lockout test was successfully performed once at power, in 1997. However, in 1997, there was adequate condenser vacuum and it was not necessary to place the turbine setback switch in the "enable" position. Since the procedure was originally written to be performed while shutdown, the review that allowed the procedure to be performed at power did not identify that a turbine setback would occur if the turbine setback protective function was enabled and a circulating water pump was locked out. Therefore, the procedure was inadequate and was a violation of 10 CFR Part 50 Appendix B, Criterion V. 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 99-3576 (50-483/99014-01).

As corrective action, the licensee changed the procedure to require the turbine setback switch to be placed in the "disable" position as an initial test condition.

c. Conclusions

An inadequate procedure was the cause of a turbine setback that reduced reactor power to approximately 88 percent. The procedure was initially written to be performed while shutdown. The review that was performed later to allow the procedure to be performed at power did not identify that enabling the turbine setback protective function while locking out a circulating water pump would cause a turbine setback. Once operators recognized the turbine setback, they responded quickly to disable the turbine setback protective function and stop the power reduction. This is a violation of 10 CFR Part 50 Appendix B, Criterion V. 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 99-3576.

M4 Maintenance Staff Knowledge and Performance

M4.1 Work Control Errors During Refueling Outage 10

a. Inspection Scope (62707)

In NRC Inspection Report 50-483/99-09, the inspectors reviewed the licensee's work control process and tagging program in response to several errors that occurred during refueling Outage 10. That assessment covered the time period from the beginning of the refueling outage (October 2) until the close of the inspection period (October 16). Specifically, the inspectors reviewed the following aspects of the work control process:

- Adequacy of work instructions,
- Operator/worker compliance with work instructions, and
- Control of the maintenance process and the work being performed.

The inspectors reviewed these same aspects of the licensee's work control process, from October 17 to the completion of the refueling outage on November 5.

b. Observations and Findings

The inspectors reviewed seven work control errors that occurred during the second half of the outage. These errors were entered in the licensee's corrective action program under the following Suggestion-Occurrence-Solution report numbers: 99-2717, 99-2738, 99-2751, 99-2896, 99-2985, 99-3103, and 99-3217.

The inspectors observed that these work control errors were similar to the previous errors in three aspects. First, lack of attention to detail was a common contributing cause. Second, personnel involved in the incidents were not from a single functional area, but from various departments (e.g., maintenance, operations, etc.). Third, the inspectors observed that a comparable number of work control errors occurred over a similar time period. (In the first half of the outage, six mispositioning events occurred in 12 days, while in the second half seven events occurred in 17 days.)

The inspectors reviewed the licensee's corrective actions for these events. The remedial actions included counseling of individuals and entering the occurrences in the licensee's corrective action program. Similar to NRC Inspection Report 50-483/99-09, the inspectors found these corrective actions to be adequate in scope, but largely unsuccessful at preventing recurrence. This observation is based on the licensee's continuing challenges in the work control area.

On October 28, the licensee initiated Suggestion-Occurrence-Solution Report 99-3114 which identified that there may be a trend, adverse to quality, in the workman's protection assurance program. In this Suggestion-Occurrence-Solution the licensee attributed recent errors to personnel errors due to confusion over tagging requirements. A formal root cause analysis was recommended.

The inspectors noted that the actual/potential safety impact of work control errors in the second half of the refueling outage was less significant than that of the first. This was, in part, due to the use of multiple barriers to prevent damage to equipment or risk to personnel. For example, during the October 16 event, residual heat removal heat Exchanger A to safety injection pump suction upstream isolation Valve EMHV8924 was found electrically disconnected without any workman's protection tagging to prevent reenergizing the valve wiring. Personnel verified that the wiring was deenergized and a tie-wrap was installed to prevent inadvertent operation of the breaker. Although this personnel error resulted in a lack of workman's protection tagging, there was little possibility of injury to workers. Other errors that occurred during this time period also had minimal actual/potential safety impact.

c. Conclusions

The licensee continued to experience minor work control errors during the second half of the refueling outage. The licensee identified a potential trend adverse to quality in the workman's protection assurance program (Suggestion-Occurrence-Solution Report 99-3114). The licensee attributed these recent failures to personnel errors due to confusion over tagging requirements.

III. Engineering

E2 Engineering Support of Facilities and Equipment

E2.1 Reactor Coolant System Leakage Detection

a. Inspection Scope (37551 and 92903)

The inspectors reviewed information to determine why the licensee may not have been able to detect a 1 gpm leak in the reactor coolant system within 1 hour.

b. Observations and Findings

On December 2, 1999, the licensee was contacted by another nuclear utility to inform them that the utility's containment normal sump level measurement system and containment air cooler condensate flow rate systems were not capable of performing their design function in all cases. The licensee performed an evaluation and determined that their systems were also susceptible. On December 3, the Callaway Plant reported this condition to the NRC in accordance with 10 CFR Part 50.72 (b)(1)(ii)B and entered the applicable limiting condition for operation.

The containment normal sump level measurement and containment air cooler condensate flow rate systems were required by Technical Specification 3.4.6.1(b) and (c) for the reactor coolant system leakage detection system. The Updated Safety Analysis Report stated that this system met the requirements of Regulatory Guide 1.45 which required the leakage detection system to detect a 1 gpm reactor coolant system leak within 1 hour. The software program used to calculate the containment normal sump level measurement and containment air cooler condensate flow rates did not always provide adequate leak detection to ensure that a 1 gpm reactor coolant system leak was detected within 1 hour.

The containment normal sump level measurement and containment air cooler condensate flow rate systems measured changes in sump or standpipe level, converted the change in level to gallons, and then divided by the amount of time. Time was reset when the sump pump turned off or the standpipe dump valves closed following pumping down the sump or draining the standpipe. If there was a long period of little or no leakage, the amount of time that the change in sump level was divided by became a larger number. Therefore, a 1 gpm leak may not be detected within 1 hour since the amount of time the level change was divided by was large.

As corrective action, the licensee changed the leakage detection software. These changes caused the leak rate program to reinitialize after a maximum of 30 minutes to prevent a large time divisor. The change also increased the sump calculation frequency from once per 15 minutes to once per 5 minutes. The standpipe calculation remained once per minute. Operations procedures were also changed to reflect the new calculation methodology.

During followup of this problem, the licensee discovered that this problem was identified in May 1997 and documented in Suggestion-Occurrence-Solution Report 97-0592. Software Change Request 5354 was written to modify the program so it would respond with an accurate flow in less than 1 hour. The change was not implemented. Also, no reportability or operability review was conducted in 1997. The inspectors asked why the software change was not implemented in 1997. The licensee stated that, due to an oversight, the software changes were not made and no reportability evaluation was performed.

The licensee knew in 1997 that the leakage detection method could mask the detection of a 1 gpm reactor coolant system leak within 1 hour, yet, no corrective action was taken. Failing to take corrective action is a violation of 10 CFR Part 50, Appendix B, Criterion XVI. 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 (50-483/99014-02).

c. Conclusions

On December 3, 1999, the licensee reported that due to a computer software discrepancy the potential existed for a 1 gpm reactor coolant system leak to not be detected within 1 hour as stated in the Updated Safety Analysis Report. The computer programming discrepancy was corrected, restoring compliance. During research, the licensee discovered that this problem had been identified in 1997. A software change was developed but never implemented. Failing to take correction action when the problem was identified in 1997 was a violation of 10 CFR Part 50, Appendix B, Criterion XVI. 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.

E5 Engineering Staff Training and Qualification

E5.1 Staffing and Training of Engineering Department Personnel

a. Inspection Scope (37551)

The inspectors examined the staffing and training of engineering department personnel as well as the coordination with industry peers. These aspects were evaluated in an effort to assess the effectiveness of the engineering department.

b. Observations and Findings

The inspectors first reviewed the formal training required by the licensee prior to certification of an individual for an engineering position. Documents reviewed included the following:

- Procedure TDP-ZZ-00065, "Training and Qualification of Engineering Personnel," Revision 5,
- Qualification module checkoff lists for various engineering functions (e.g., safety analysis, performing design changes, etc.), and
- American National Standards Institute Guideline 3.1-78, "Selection, Qualification, and Training of Personnel for Nuclear Power Plants" (invoked by reference in Procedure TDP-ZZ-00065).

The inspectors found that the training requirements for engineering personnel were comprehensive for the functions being performed. As an example, in order to perform the safety analysis function, an individual must be "checked-out" on three different computer analysis codes, 20 accident analysis basis documents, 19 administrative and engineering procedures, etc. The inspectors also noted that the functional areas an individual must be certified in are consistent with the responsibilities for specific engineering jobs.

The inspectors reviewed the contact with industry peers and coordination on emerging technical issues. The inspectors reviewed a list of the off site meetings attended by members of the engineering department during 1999. The inspectors found that approximately the same number of trips were taken by personnel of the three subdivisions in the engineering department (design control, systems, and technical support). Approximately 127 individuals traveled on 245 trips during this time period. The inspectors found that engineering personnel stayed abreast of emerging technical issues through participation in these meetings. Examples of conferences of particular relevance to the Callaway Plant included: axial offset meeting with Westinghouse, ultrasonic fuel cleaning discussion with Dominion Engineering, and Electrosleeving™ meeting with Framatome.

c. Conclusions

The licensee's training requirements for various engineering positions were comprehensive for the functions being performed. Engineering department contact with industry peers and coordination on emerging technical issues was appropriate.

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. The inspectors also observed portions of sampling the reactor coolant that was performed on December 20. The evolution was performed in accordance with Procedure COA-ZZ-07600, "Obtaining Pressurized Samples," Revision 4, and Procedure CTP-ZZ-2550, "Pressurized Reactor Coolant Sample," Revision 13. The inspectors did not identify any concerns.

Contaminated areas and high radiation areas were properly posted. Area surveys posted outside rooms in the auxiliary building were current. The inspectors checked a sample of doors, required to be locked for the purpose of radiation protection, and found no problems.

S2 Status of Security Facilities and Equipment

S2.1 Walkdown of Protected Area Barrier

a. Inspection Scope (71750)

The inspectors performed a walkdown of the protected area to determine compliance with the security plan. The areas inspected included the general area illumination levels and access to licensee controlled vehicles.

b. Observations and Findings

On January 3, 2000, the inspectors performed a walkdown of the protected area. The inspectors examined unattended licensee controlled vehicles. The inspectors verified that these vehicles, approximately eight light and heavy trucks, were secured with the ignition keys removed, in accordance with the licensee's security plan.

The inspectors also performed a walkdown of the protected area border to verify the integrity of the security fence. The inspectors observed security guards performing routine tours of the security fence. The inspectors also verified that illumination levels were adequate in exterior areas in accordance with the security plan. No discrepancies were noted.

c. Conclusions

The licensee's compliance with its security plan was verified in the areas of exterior illumination levels, control of unattended vehicles, and security fence integrity.

V. Management Meetings

X1 Exit Meeting Summary

The exit meeting was conducted on January 10, 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

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J. D. Blosser, Manager, Operations Support
G. W. Hamilton, Supervising Engineer, Quality Assurance
R. T. Lamb, Superintendent, Work Control
D. S. Hollabaugh, Superintendent, Design Engineering
J. A. McGraw, Superintendent, Technical Support Engineering
J. T. Patterson, Superintendent, Mechanical Maintenance
J. R. Peevy, Manager, Emergency Preparedness
G. L. Randolph, Vice President and Chief Nuclear Officer
R. R. Roselius, Superintendent, Radiation Protection and Chemistry
L. S. Sandbothe, Superintendent, Operations
J. D. Schnack, Supervising Engineer, Quality Assurance Corrective Action
C. E. Slizewski, Supervising Engineer, Quality Assurance
T. P. Sharkey, Supervising Engineer, Safety Related Mechanical Systems

INSPECTION PROCEDURES USED

37551	Onsite Engineering
61726	Surveillance Observations
62707	Maintenance Observations
71707	Plant Operations
71750	Plant Support Activities
92700	Onsite Followup of Written Reports of Nonroutine Events at Power Reactor Facilities
92902	Followup - Maintenance
92903	Followup - Engineering

ITEMS OPENED AND CLOSED

Opened

99014-01	NCV	Inadequate procedure resulted in turbine setback (Section M3.1)
99014-02	NCV	Failure to take corrective action for reactor coolant system leakage detection discrepancy (Section E2.1)

Closed

99008-00	LER	Reactor trip due to a low steam generator level (Section O8.2).
99014-01	NCV	Inadequate procedure resulted in turbine setback (Section M3.1)
99014-02	NCV	Failure to take corrective action for reactor coolant system leakage detection discrepancy (Section E2.1).