ENCLOSURE

U.S. NUCLEAR REGULATORY COMMISSION
REGION IV

NRC Inspection Report: 50-483/95-13

Operating License: NPF-30

Licensee: Union Electric Company
Post Office Box 149 - Mail Code 400
St. Louis, MO 63166

Facility Name: Callaway Plant, Unit 1

Inspection At: Callaway Site, Steedman, MO

Inspection Conducted: October 1 through November 11, 1995

Inspectors: D. Passehl, Senior Resident Inspector
F. Brush, Resident Inspector
L. Ellershaw, Reactor Inspector

Approved: WDJohnson 12/1/95
William D. Johnson, Chief, Project Branch B Date

Inspection Summary

Areas Inspected: Routine unannounced inspection of plant operations, maintenance, surveillance, engineering, and plant support.

Results:

Plant Operations

- In early October 1995, operators shut down the plant due to unidentified reactor coolant system leakage greater than that allowed by Technical Specifications. The ensuing forced outage lasted until mid-October (Section 1).

- Management's response to the unidentified leak was excellent. The decision to shut down the plant was timely. Management was heavily involved in the forced outage planning and execution (Section 1).

- Command and control of the shutdown and subsequent restart activities overall was satisfactory. Some weaknesses were observed in crew communications (Section 1).

- Plant operators experienced some problems with establishing proper system lineups due to inadequate procedures, inadequate communications, or lack of attention to detail. The consequences and safety significance of the misalignment problems were minor (Section 2.3).

Maintenance

- One system alignment error occurred involving the draining of several hundred gallons of primary coolant to the containment sump. The cause was a faulty maintenance procedure (Section 3.2).
- The plant's response to repair a damaged injection check valve from the residual heat removal system to the reactor coolant system was satisfactory. Support from the Operations and Engineering Departments was satisfactory. The Quality Assurance group performed a thorough and self-critical review of this event (Section 3.3).
- The licensee replaced a gasket between the seal housing and casing on Reactor Coolant Pumps B and C due to a buildup of boric acid. Control of foreign material exclusion boundaries was satisfactory. There was a lack of control of foreign material identified in previous NRC inspections (Section 3.4).
- The licensee conducted a maintenance outage on Diesel Generator A, Emergency Service Water System Train A, and the control room air handling system. The outages were well planned and executed. Management involvement was evident. Outage time was appropriately minimized. Probabilistic risk assessment techniques were used (Section 3.5).
- Licensee personnel repaired a leak in the excess letdown line. The plant staff developed a well planned and comprehensive work package; established contingency plans; established excellent interface between line management and craft; implemented the plan in accordance with the specified procedures and instructions; and took a conservative approach in attempting to identify possibly similar conditions, regardless of the preliminarily identified cause (Section 3.6).

Engineering

- Engineering personnel dealt with several technical issues successfully. These included the discovery of defective bolts on Reactor Coolant Pump B and C seal housings, the weld overlay on an excess letdown pipe, the repair of the residual heat removal check valve, and others (Section 5).

Plant Support

- Job-site contamination control during the maintenance activities observed was very good. Health physics technicians were proactive in assuring that personnel remained clear of high dose areas (Section 6.1).

- Security personnel performed their duties in a professional manner. Vehicles were properly controlled or escorted within the protected area. Desigrated vehicles parked and unattended within the protected area were apprcpriately locked with the keys removed. The protected area perimeter was maintained at an excellent level. Proper compensatory measures were observed when a security barrier was inoperable (Section 6.2).

- The inspectors observed various aspects of the licensee's land vehicle bomb threat barrier installation effort. No problems were noted (Section 6.2).

Summary of Inspection Findings:

- A noncited violation was identified (Section 2.3)

Attachment:

- Persons Contacted and Exit Meeting

DETAILS

1 PLANT STATUS

At the beginning of this inspection period, the plant was at 99 percent power. On October 2, 1995, the licensee shut down the plant due to unidentified reactor coolant system leakage of greater than 1 gallon per minute. On October 11, 1995, the generator was placed on line to end the forced outage. At the end of this inspection period, the plant was at 99 percent power.

1.1 Forced Outage Due to an Unidentified Reactor Coolant System Leak of Greater Than 1 Gallon Per Minute

Management's response to the unidentified leak was excellent. The decision to shut down the plant was timely. Management was heavily involved in the forced outage planning and execution. During discussions on the best methods to repair the leak, plant safety was emphasized by all personnel involved.

Command and control of the shutdown and subsequent restart activities overall were satisfactory. Some isolated weaknesses were observed in crew communications. Site management was present in the control room at various times to observe the plant shutdown and restart. During both startup and shutdown, shift supervision provided periodic updates of plant status to plant operators. However, management expectations for operator communications to outside groups were not always met. The inspector noted one example when an operator told an instrument and control technician to "do it" when the technician requested permission to troubleshoot a problem recorder. Operations management acknowledged some weaknesses in communication and has implemented plans to improve performance in this area.

At 4:30 a.m. on October 2, 1995, due to a decreasing level in the volume control tank, the licensee identified that a small leak in the reactor coolant system was in progress. The licensee also noted that containment radiation levels and sump flows were increasing. At 6:30 a.m., the licensee determined that the leak rate was 2.1 gallons per minute. At approximately 7:30 a.m., during a containment entry, the licensee determined that the leak was inside the bioshield. At 8 a.m., since the exact source of the leak could not be determined, the licensee commenced a plant shutdown. At 11:26 a.m., when the generator was taken off line, the unit entered a forced outage.

At 11:38 a.m., a high-high water level in Steam Generator D caused a feedwater isolation signal. This occurred during transfer from the normal to the startup feedwater pump. Due to a failed relay, Feedwater Regulation Valve D remained partially open. The licensee replaced the relay and the valve functioned properly.

At 2:25 p.m. on October 11, 1995, the generator was placed on line to end the forced outage.

2 OPERATIONAL SAFETY VERIFICATION (71707)

2.1 Routine Control Room Observations

The inspectors observed operational activities throughout this inspection period to verify that adequate control room staffing and control room professionalism were maintained. Shift turnover meetings were conducted in a manner that provided for proper communication of plant status from one shift to the other. Discussions with operators indicated that they were aware of plant status, equipment status, and reasons for lit annunciators. Control room indications of various valve and breaker lineups were verified for current plant status.

2.2 Plant Tours

The inspectors routinely toured various areas of the plant to assess the safety conditions and adequacy of plant equipment. The inspectors verified that various valve and switch positions were correct for the current plant conditions. Piping and instrumentation drawings and operating instructions posted in vital areas were inspected and found to be current. Personnel were observed obeying rules for escorts, visitors, entry, and exits into and out of vital areas.

2.3 System Alignment Errors

The inspector noted a negative trend in the number and frequency of system misalignment errors that occurred during the last few months. Examples that occurred during this inspection period were:

- On October 11, 1995, a primary equipment operator discovered Reactor Makeup Water to Chemical Volume And Control System Chemical Mixing Tank Isolation Valve BG V-0178 was locked closed. The valve was signed off as restored to the open position during plant heat up from cold shutdown to hot standby following the recent forced outage.

- On October 9, 1995, a leak developed in the reactor coolant system when Reactor Coolant Pump B was taken off its backseat (see Section 3.2). This allowed approximately 700 gallons of reactor coolant to drain to the containment sump.

- On October 15, 1995, the Control Room Supervisor found Reactor Vessel Flange Leakoff Isolation Valve BB-HV-8032 to be closed. This is a normally open valve which is used to detect a failure of the inner O-ring on the reactor vessel flange. The valve was closed on October 3, 1995, during draining of the reactor coolant system during the forced outage. The valve should have been restored to the open position at the completion of venting of the reactor coolant system per Checkoff List 8 of the reactor coolant system vent procedure prior to plant startup.

- On October 13, 1995, operators found the upstream and downstream isolation valves for the nitrogen supply to the pressurizer relief tank, BB HV-8027 and BB HV-8026, respectively, to be open. These two valves are normally closed. The valves were opened on October 3, 1995, during draining of the reactor coolant system during the forced outage. The valves should have been restored to the closed position at the completion of venting of the Reactor Coolant System per Checkoff List 8 of the Reactor Coolant System vent procedure prior to plant startup.

The licensee had previously identified instances of system misalignment and had, in the Spring of 1995, formed a multi-discipline task team consisting of members from various departments to review, evaluate, and recommend corrective actions to address the system misalignment concerns. The root causes were varied and related to inadequate procedures, inadequate communciations, and/or lack of attention to detail. The licensee developed several programs and policies, some of which were recently implemented, to address the issues. Although there have been some successes, the corrective actions had not been implemented long enough for the inspector to gauge their effectiveness, and will continue to be evaluated.

The instances of system misalignment identified in this inspection period constitute a violation of procedural requirements dictated by Technical Specifications. Based on the facts that these violations were licensee identified or self-revealing, corrective actions have been implemented to address the root causes, and there was little safety significance or consequences associated with the events; these are being treated as examples of a noncited violation, consistent with Section VII of the NRC Enforcement Policy.

2.4 Excellent Operator Observation of Changing System Conditions

During the October forced outage, an operator noted what appeared to be a flow anomaly in the residual heat removal system. Upon investigation, the licensee determined that a check valve had failed. The disk fell out of its holder, which caused a reduction in shutdown cooling return flow to the reactor coolant system. The operator's alertness and knowledge of residual heat removal system operation prevented any significant problems from occurring. This event is discussed further in Section 3.3.

3 MAINTENANCE OBSERVATIONS (62703)

The maintenance activities listed below were observed and documentation reviewed to verify that the activities were conducted in a manner which resulted in reliable safe plant operation.

3.1 Maintenance Observations

The following maintenance activities were observed:

- W575187 Reactor Coolant Pump B Seal Housing Work
- W575186 Excess Letdown Line Weld Overlay
- G526257 Generic Work Request - Valve Cleaning and Packing Adjustment
- P559478 Diesel Generator A Lube Oil Keep Warm Pump, Repack Coupling
- W173417 Replace Gasket on Centrifugal Charging Pump B Drain Line Flange
- W141862 Replace Centrifugal Charging Pump Discharge Downstream Drain Valve

3.2 Inadvertent Loss of 700 Gallons of Reactor Coolant Due to an Inadequate Procedure

At 4:10 a.m. on October 9, 1995, a leak developed in the reactor coolant system when Reactor Coolant Pump B was taken off its backseat. The pump was on its backseat to allow replacement of the seal housing to pump casing gasket. At 4:30 a.m., the pump was returned to its backseat, stopping the leak. Approximately 700 gallons of water drained from the reactor coolant system.

The licensee determined that a temporary drain line had not been returned to its normal configuration prior to removing the pump from its backseat. This was due to a deficiency in Maintenance Procedure MPM-BB-QP001, "Reactor Coolant Pump Seal Removal and Replacement."

During maintenance on Reactor Coolant Pump B, a temporary modification was used to install a drain from the number one seal injection line. The drain line was used to prevent the reactor coolant pump cavity from filling with water. The maintenance procedure directed personnel to install the temporary modification but did not direct them to remove it before removing the pump from its backseat.

Due to an oversight during original plant construction, a drain line was not installed on Reactor Coolant Pump B. However, the other three reactor coolant pumps had a drain line installed and used the component tagout process to control its use. The licensee initiated a request for resolution to evaluate installing a permanent drain line on Reactor Coolant Pump B.

As a corrective action, the licensee formed a task team to review the reactor coolant pump maintenance procedures for additional deficiencies. The team was scheduled to complete its effort by December 16, 1995. Also, earlier in 1995, the licensee implemented a program to review all maintenance procedures to correct any deficiencies. This extensive effort was scheduled to be completed in 1997.

3.3 Failure of Residual Heat Removal to Reactor Coolant System Loop 4 Cold Leg Injection Check Valve

The maintenance department's response to repair a damaged injection check valve from the residual heat removal system to reactor coolant system was satisfactory. Support from the Operations and Engineering Departments was good. The valve was refurbished with new internals and tested satisfactorily. The quality assurance group performed a thorough and self-critical review of this event.

During the forced outage, the plant experienced a failure of Reactor Coolant System Loop 4 Cold Leg Injection Check Valve EP 8818D. Evidence of the failure was a partial loss of residual heat removal flow in Train B. Plant operators aligned residual heat removal flow to Train A and removed Train B from service to investigate. The licensee found that the check valve disc had dislodged from the inconel arm and settled in the bottom of the valve. Both the disc and disc arm exhibited signs of wear.

Valve EP 8818D is one of four residual heat removal to reactor coolant system cold leg injection check valves. The others are EP 8818A, EP 8818B, and EP 8818C. The valves supply residual heat removal flow to Reactor Coolant System Loops 1, 2, and 3, respectively. All four valves are Westinghouse 6-inch swing check, ASME Code Section III, Class 1 valves.

The licensee determined the cause of the failure of Valve EP 8818D to be accelerated wear of the valve disc. Valve EP 8818D was located 15 inches downstream of a flow orifice and in the turbulent flowpath which the orifice generated.

The licensee had been aware of the accelerated wear phenomena because of previous problems experienced with the same valve. In response to a failure of the valve in 1985, Westinghouse was contacted and recommended that the valve disc be replaced on a periodic basis because of vibrational wear concerns. The licensee established a preventive maintenance activity to disassemble and inspect Valve EP 8818D every other refueling outage.

The licensee discovered an additional problem with the check valve upon disassembly. The retaining ring that held the disc in place was undersized in diameter by 0.007 inch. The licensee believed that the undersized retaining ring in conjunction with the high residual heat removal flow rate allowed for extra movement on the disc arm and the disc rotation, which led to the premature failure of this valve. Plant engineers reviewed procurement records and associated documentation and found no evidence that other check valves had undersized retaining rings.

The inspector reviewed maintenance histories and flow configurations for the other loop injection check valves. The corresponding reactor coolant system Loop 1 cold leg injection check Valve EP 8818A is also located a short distance downstream of a flow orifice and was subject to the same preventive maintenance activity as Valve EP 8818D. Valve EP8818A had been disassembled and inspected in the Spring 1995, refueling outage. No unusual wear was detected and the valve was reassembled and tested satisfactorily. Valves EP 8818B and EP 8818C are not in a turbulent flowpath and, hence, have not exhibited the same degradation problem.

The licensee was in the process of evaluating possible design changes to prevent future residual heat removal injection check valve failures. During the next refueling outage, the chosen solution would be implemented for both Valves EP 8818A and EP 8818D to preclude other failures.

3.4 Reactor Coolant Pump Seal Leakage Due to Defective Bolts

During the forced outage discussed in Section 1, the licensee noted a buildup of boric acid crystals on Reactor Coolant Pump B and C seal housings. A flexitallic gasket provides the pressure boundary seal between the seal housing and reactor coolant pump casing. During the refueling outage in the spring of 1995, also due to boric acid crystal buildup, a vendor replaced the gaskets on these reactor coolant pumps. However, due to a manufacturing error, the bolts that hold down the seal housing were not properly tightened. This allowed a very small amount of reactor coolant to seep by the flexitallic gaskets which caused the buildup of boric acid crystals.

The bolts that held down the reactor coolant pump seal housing had a hole in the center. This hole allowed measuring the stretch of the bolts when they were tightened. Because of a burr left in the bottom of the hole, the bolt stretch measurements were inaccurate. Instead of approximately 10 Mils, the bolts had 2 to 3 Mils of stretch. Therefore, the flexitallic gasket did not provide an adequate seal. The licensee replaced the bolts with studs and measured the stretch using an ultrasonic device. During this work, control of foreign material exclusion boundaries was satisfactory. The lack of control of foreign material was a weakness discussed in previous NRC inspection reports.

The licensee also inspected Reactor Coolant Pumps A and D. There was no indication of boric acid crystal buildup, although the pumps had bolts installed. During the fall 1996 refueling outage, the licensee planned to replace the bolts with studs on Reactor Coolant Pumps A and D. The inspectors will continue to monitor the licensee's plans for work on these pumps.

3.5 Review of Online Maintenance Activities

During the week of October 30, 1995, the licensee conducted a maintenance outage on Diesel Generator A, Emergency Service Water System Train A, and the control room air handling system. The inspectors reviewed the outage schedule, observed various maintenance activities, and discussed the use of the Individual Plant Examination for maintenance planning with the licensee.

The outage was well planned and executed. Management was present at many of the activities. Since various safety-related components were removed from service, the maintenance tasks were worked on an around-the-clock schedule. The inspectors did not have any concerns as a result of this effort.

The licensee did not have a formal method for factoring Individual Plant Examination results into maintenance planning. However, as a result of the maintenance rule implementation effort, the licensee was developing a matrix that will be included in the maintenance planning procedure. The matrix indicates which component out-of-service combinations are not recommended. The basic premise in determining whether various combinations of components can be taken out of service, other than those prohibited by Technical Specifications, was if the basic core damage frequency would be increased by a factor of 10. The inspectors reviewed the matrix and did not note any obvious problems.

3.6 Repair of Excess Letdown Piping Through-Wall Crack

The licensee identified a through-wall crack in the 2-inch diameter excess letdown branch line (BB-074-BCA-2) from the Reactor Coolant Loop 4 (D-Loop) cross-over leg. This defect resulted in a reactor coolant system leak rate greater than 2 gpm and required the reactor to be shut down and taken to Mode 5, with the reactor coolant system depressurized. Work Request (WR) W575186 (the licensee's ASME Section XI Repair/Replacement Plan), was initiated on October 3, 1995, to effect the necessary repairs, including nondestructive examination to quantify the defect, mechanical peening, welding, and final nondestructive examination. The inspector noted that the licensee invoked the use of alternative rules established by ASME Code Case N-504, which had been approved in NRC Regulatory Guide 1.147.

The licensee conducted a formal safety evaluation and concluded that current accident analyses bound a postulated break of the excess letdown branch line, thus, there was no increase in consequences or probabilities and no Unreviewed Safety Question existed. It was further concluded that the proposed repair activity would not reduce the margin of safety as defined in the basis for any Technical Specification and that sufficient safety systems required to cope with a postulated break would remain available to maintain adequate core cooling and containment of leakage. The inspector, after reviewing the types of reactor coolant pressure boundary accidents described in Chapter 15 of the Final Safety Analysis Report, considered the licensee's conclusion to be appropriate.

The licensee informed the inspector that the repair work would be performed by PCI Energy Services, a wholly owned subsidiary of Westinghouse Electric Corporation. The inspector verified that PCI Energy Services was on the licensee's qualified suppliers list and that they were approved for the scope of repair activities.

The inspector reviewed and compared Work Request W575186 to the alternative rules for repair of austenitic stainless steel piping specified in ASME Code Case N-504 and verified that the weld reinforcement design had been performed and that all of the specified requirements were incorporated into the work request, including provisions for documenting the completion of those requirements.

Location of the defect was determined to be in the heat affected zone of the 90-degree elbow immediately adjacent to the elbow-to-downstream pipe weld. By using the ultrasonic examination method, licensee nondestructive examination personnel quantified the extent of the defect as being circumferential and approximately 180 degrees from about the 3 o'clock position to the 9 o'clock position, with an approximate 1/2 inch through-wall condition at about 5 o'clock (looking at the pipe from downstream). The inspector learned that licensee nondestructive examination personnel had reviewed the original radiographs of the weld to determine if any indications existed. The inspector did not review the radiographs; however, licensee personnel stated that there were no observable indications.

During repair preparations, personnel established contingency plans to contain reactor coolant in the event the defect opened up during peening or welding. This was accomplished by attaching a nonmetallic sleeve and clamps downstream of the area to be repaired. If necessary, the sleeve and clamps could have been quickly positioned and secured, with minimum loss of coolant. As previously stated, the reactor coolant system was depressurized from the normal operating pressure of 2196 psig to approximately 47 psia at the repair location (equal to the static head pressure).

Prior to initiation of the repair work on October 4, 1995, the inspector noted that wireless communications were established with the control room. However, the noise levels were quite high in the area, thus it was difficult for the communicators to understand messages from the control room or elsewhere. The inspector observed a large part of the mechanical peening (blunt nosed punch and hammer, without the addition of wire material) and shielded metal arc welding of the repair. This work constituted the seal welding portion which was used to seal off the leak. Code Case N-504 requires that low carbon austenitic stainless steel weld material be applied 360 degrees around the circumference of the pipe in order to seal off the defect prior to deposition of the structural portion of the weld overlay. Since the elbow-to-downstream pipe weld was approximately 6 inches off the floor, peening and welding on the bottom side of the pipe was very difficult and time consuming. There were several occurrences where the peening and welding appeared to be successful; however, during a subsequent weld pass, the defect would open up at a different location (i.e., 4 o'clock, 6 o'clock, etc.) and dripping would recur. Upon completion of the seal welding portion, a liquid penetrant examination was successfully performed on October 5, 1995.

To assure sufficient material existed prior to the deposition of the weld overlay, additional weld buildup was performed. This was accomplished by using the gas tungsten arc welding process. Upon completion of the buildup, a liquid penetrant examination and an ultrasonic examination (to establish thickness) was successfully performed.

While these activities were in progress, the inspector observed and verified that the welding variables were being complied with, nondestructive examinations were properly performed, and delta ferrite measurements were taken after each layer of weld material had been deposited (as required by the code case).

The inspector also noted that line management and craft frequently communicated and that work did not proceed until there was a clear understanding of what was expected. Significant attention was paid to radiation and ALARA (as low as reasonbly achievable) concerns, as evidenced by health physics personnel continuous monitoring, placement of personnel in low-dose areas, and limiting the number of personnel in the area. Plant areas which were observed exhibited the use of good housekeeping practices and equipment appeared to be in good material condition.

The inspector did not observe deposition of the structural weld layers, which occurred during October 6-8, 1995, or the final liquid penetrant and ultrasonic examinations performed on October 8, 1995. However, the completed WR package, which contained all weld control and nondestructive examination records (procedures used, welder and examiner identifications, weld material used, dates and sign-offs), was subsequently provided to the inspector for in-office review. This review did not identify any instances where work was not performed in accordance with the work package.

The inspector also reviewed welder qualifications, welding procedure specifications (WPS-0808S01 - shielded metal arc, and WPS-8 - gas tungsten arc), their respective procedure qualification records, and weld material certified test reports and found these to be in accordance with ASME Code requirements.

Code Case N-504 required the completed repair to undergo a system hydrostatic test. However, the inspector noted that the final step in the WR (Step 15.8) stated that "QC shall perform a VT-2 (visual examination) on the completed repair. This VT-2 satisfies all hydrostatic testing requirements, as provided by ASME Code Case N-416-1." Since this code case had not been approved in NRC Regulatory Guide 1.147, the inspector requested licensee representatives to provide the basis for the use of Code Case N-416-1. The inspector was provided a copy of NRC Safety Evaluation on ASME Code Case N-416-1, dated October 28, 1994, which was in response to a relief request submitted by the licensee on April 25, 1994. The Safety Evaluation authorized the use of Code Case N-416-1 until such time as the code case is published in a future revision of Regulatory Guide 1.147.

The licensee had initiated Design Input Report SOS 95-1891, on October 4, 1995, to address the condition, its repair, and determine a root cause. The preliminary assessment indicated that the flaw was caused by thermally induced stress. A plant walkdown had identified one area in which an interference was occurring. A downstream flange associated with a drain valve had been contacting a surface mounted plate used for two pipe supports. When the plant was in a cold condition, there was no interference; however, when the plant was heated up and operating, it could be seen where the flange had actually contacted and deformed the edge of the surface mounted plate. It was estimated that in excess of 50 ksi loads were transmitted back to the elbow where the flaw occurred. Additional stress analyses were being performed by Westinghouse to see what other stresses were created by this interference. In any event, the Design Input Report stated that confirmation of the root cause will be obtained by metallurgical analysis. This will be performed following the plant's next refueling outage scheduled for the fall of 1996, when the repaired section will be removed and replaced.

Licensee personnel performed a walkdown of the other three loops in order to identify potential interference points. One possible interference point was located on the Loop B drain line and was eliminated. All of the branch piping welds upstream of the identified interference on both Loops B and D were liquid penetrant examined and ultrasonically examined. No indications were identified.

Since the licensee had not been able to clearly identify the cause of the flaw in the Loop D drain piping, and recognizing that other mechanisms may have been involved, it was decided to take a conservative approach and perform liquid penetrant examinations and ultrasonic examinations on all branch drain lines off the Loops A and C. The inspector was informed that no indications were identified.

The inspector concluded that licensee personnel: developed a well planned and comprehensive work package; established contingency plans; established excellent interface between line management and craft; implemented the plan in accordance with the specified procedures and instructions; and took a conservative approach in attempting to identify possibly similar conditions, regardless of the preliminarily identified cause.

4 SURVEILLANCE OBSERVATIONS (61726)

The inspectors observed the surveillance testing listed below to verify that the activities were performed in accordance with the licensee's approved programs and the Technical Specifications.

4.1 Surveillance Observations

The following surveillance activities were observed:

- OSP-EN-P001B Containment Spray Test
- OSP-EF-P001A Diesel Generator A Postmaintenance Test
- OSP-NE-0002 Emergency Service Water Flow Balance

The inspectors did not note any problems in the surveillance area.

4.2 Duplicate Noncited Violations Documented

NRC Inspection Report 50-483/94-10, paragraph 3.b, documented a noncited violation when several surveillances of the contents of the gas decay tank that was in service had not been properly performed. NRC Inspection Report 50-483/95-11 erroneously closed out the associated licensee event report with an additional noncited violation. The noncited violation issued in NRC Inspection Report 50-483/95-11 is withdrawn.

5 ENGINEERING

During the forced outage, engineering personnel dealt with several technical issues successfully. This included determining the root cause of the reactor coolant pump seal leakage (see Section 3.4) and the residual heat removal check valve failure (see Section 3.3), developing the work package for the weld overlay on the letdown pipe (see Section 3.6), and other emergent issues. No significant issues were identified.

6 PLANT SUPPORT ACTIVITIES (71750)

The inspectors sampled selected activities in the different areas of plant support and verified that they were implemented in conformance with licensee procedures and regulatory requirements.

6.1 Radiological Protection Program Observations

During this inspection period, the inspectors verified that selected activities of the licensee's radiological protection program were properly implemented. Health physics personnel were observed routinely touring the radiologically controlled areas. Contaminated areas and high radiation areas were properly posted, and restricted high radiation areas were found to be locked, as required.

The inspectors accompanied licensee personnel on an at-power containment entry. The health physics technician was very proactive in assuring that personnel were aware of potential high dose areas and remained clear of them. The inspectors also reviewed contamination controls at various job sites. Licensee personnel followed proper procedures to prevent the spread of contamination.

6.2 Security Program Observations

The inspectors observed various aspects of the licensee's security program. Security personnel were found to perform their duties in a professional manner. Vehicles were properly controlled or escorted within the protected area. Designated vehicles parked and unattended within the protected area were found to be locked and the keys removed. The inspectors routinely toured the protected area perimeter and found it maintained at an excellent level. Proper compensatory measures were observed when a security barrier was inoperable.

The inspectors observed various aspects of the licensee's land vehicle bomb threat barrier installation effort. No problems were noted.

ATTACHMENT

1 PERSONS CONTACTED

Licensee Personnel

R. D. Affolter, Manager, Callaway Plant
H. D. Bono, Supervising Engineer, Site Licensing
M. S. Evans, Superintendent, Health Physics
G. A. Hughes, Supervising Engineer, ISEG
L. H. Kanuckel, Supervising Engineer, QA
R. T. Lamb, Superintendent, Operations
C. D. Naslund, Manager, Nuclear Engineering
G. L. Randolph, Vice President, Nuclear Operations
M. A. Reidmeyer, Senior Engineer, Quality Assurance

The above licensee personnel attended the exit meeting. In addition to these personnel, the inspectors contacted other personnel during this inspection period.

2 EXIT MEETING

An exit meeting was conducted on November 14, 1995. During this meeting, the inspectors reviewed the scope and findings of the report. The licensee did not identify as proprietary any information provided to, or reviewed by, the inspectors.