ENCLOSURE

U.S. NUCLEAR REGULATORY COMMISSION
REGION IV

Docket No.:	50-483
License No.:	NPF-30
Report No.:	50-483/97-14
Licensee:	Union Electric Company
Facility:	Callaway Plant
Location:	Junction Highway CC and Highway O Fulton, Missouri
Dates:	June 22 through August 2, 1997
Inspectors:	D. G. Passehl, Senior Resident Inspector D. N. Graves, Senior Project Engineer, Project Branch B W. M. McNeill, Reactor Inspector
Approved By:	W. D. Johnson, Chief, Project Branch B

ATTACHMENT: Supplemental Information

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EXECUTIVE SUMMARY

Callaway Plant
NRC Inspection Report 50-483/97-14

Operations

- Management's direction to reduce load to 95 percent reactor power to gain additional margin to the limits for the heat flux hot channel factor F(q) and axial offset was conservative and timely (Section O1.2).

- Operators responded well to an Unusual Event involving failures in the plant annunciator system. Operators remained focused on plant parameters. Operators carried out compensatory actions until the annunciator system was fully restored. Management and technical support were very good (Section O1.3).

- There were weaknesses in plant procedures regarding the criteria for declaring the Unusual Event, which led to about a 6-hour delay in declaring the Unusual Event. There was confusion on what constituted minimum acceptable field power supply voltage and what constituted a failure of "most or all annunciators" (Section O1.3).

Maintenance

- Material condition and housekeeping of accessible areas of the auxiliary building, the fuel building, the essential service water pumphouse, and most areas of the turbine building were very good (Section M2.1).

- A noncited violation was identified during Licensee Event Report 96-008 closeout review. Plant electricians performed a weekly battery surveillance on the wrong train of batteries. This resulted in Train B of the station batteries being without a current surveillance for approximately 17 hours (Section M8.2).

- A noncited violation was identified during Licensee Event Report 97004 closeout review. This involved missed surveillances on feedwater isolation and turbine trip slave relays (Section M8.4). The licensee had, on several occasions, performed the surveillances at power instead of "during refueling."

Engineering

- An axial offset anomaly has caused shutdown margin to reduce at a faster rate than predicted. The licensee was in compliance with the Technical Specifications and was aggressively monitoring the shutdown margin and other plant parameters to ensure the plant remained within operating limits (Section E1.1).

- A noncited violation was identified during Unresolved Item 97-007-05 closeout review. The inspectors identified a discrepancy between the Final Safety Analysis Report and the actual response time of control room ventilation isolation system radiation monitors. The licensee did not perform a 10 CFR 50.59 evaluation to ensure that an unreviewed safety question did not exist (Section E8.1).

Report Details

Summary of Plant Status

The plant began the inspection report period near full power operation.

On July 10, 1997, operators began a gradual load reduction to 95 percent reactor power. The licensee was operating close to the limits for axial flux difference and the heat flux hot channel factor F(q). The licensee reduced power to gain additional margin to these limits.

On July 19, 1997, an Unusual Event occurred as a result of a failure of the plant annunciator system. The licensee properly responded and restored the annunciator system to operable status the following day.

The plant ended the inspection report period at 95 percent power.

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.

O1.2 Gradual Power Reduction

a. Inspection Scope (71707)

The inspectors reviewed the licensee's actions for the gradual power reduction. The inspectors' review of the licensee's compliance with requirements associated with the heat flux hot channel factor F(q), axial offset, and shutdown margin is addressed in Section E1.1.

b. Observations and Findings

On July 10, 1997, the licensee began a gradual load reduction to 95 percent power. There were no plant parameters, test results, or reactivity measurements that required the load reduction. The licensee was experiencing an axial offset anomaly and had been operating close to operating limits for the heat flux hot channel factor F(q) and axial offset. Power was reduced to gain additional margin for these parameters. The licensee accomplished the load reduction at approximately one half percent per day.

The inspectors had no concerns with operator performance during the power reduction. The inspectors found that management's direction to reduce load to gain additional margin to the heat flux hot channel factor F(q) and axial offset was conservative and timely.

O1.3 Unusual Event for Failure in Plant Annunciator System

a. Inspection Scope (93702)

The inspectors responded to the site to review the licensee's actions in response to a failure in the plant annunciator system.

The inspectors reviewed:

- Suggestion-Occurrence-Solution Report 97-0852;
- Procedure OTO-RK-00001, "Loss of Control Room Alarms," Revision 5; and
- Procedure EIP-ZZ-00101, "Classification of Emergencies," Revision 20.

b. Observations and Findings

On July 19, 1997, at approximately 4 p.m. (CDT), lightning strikes in the area of the water treatment plant resulted in the failure of numerous annunciators in the control room. Plant operators verified plant parameters were within limits, contacted instrument and control technicians, and began to perform Procedure OTO-RK-00001. This procedure described the method for monitoring various vital equipment and plant parameters to allow continued safe operation of the plant.

Approximately 30 minutes after the event began, licensee personnel discovered and disconnected four failed connector cards in the "RK" nonsafety related plant annunciator system. These failed cards had caused the voltage for four annunciator field power supplies to drop to 25 VDC. The normal voltage for these power supplies was approximately 125 to 130 VDC. After disconnecting the cards, the voltage of the field power supplies returned to normal and approximately 90 percent of the annunciators were restored.

Several system engineers assisted in the subsequent troubleshooting. At approximately 10 p.m., the licensee determined that, during the first 30 minutes of the event, the majority of the annunciators may not have been functional. The licensee notified the NRC at 10:34 p.m., that an Unusual Event condition existed between 4:15 p.m. and 4:30 p.m. because of the failed annunciators.

Callaway Plant remained stable during the event, with no damage to any safety-related equipment. All control room instrumentation and plant computer displays were available to operators to monitor plant status throughout the event.

The licensee appropriately performed compensatory measures directed by Procedure OTO-RK-00001. Although most of the annunciators were restored after the first 30 minutes of the event, the licensee continued to perform the compensatory measures until the annunciator system was fully restored the following day at 3:48 a.m. The licensee formed an event review team to investigate this event.

The inspectors found that plant operators performed well during the event. Operators remained focused on operating the plant safely. Management responded promptly at the onset of the event and provided proper direction and technical support.

The inspectors reviewed the licensee's delayed decision in declaring the Unusual Event. The criteria for the Unusual Event was defined in Procedure EIP-ZZ-00101, Attachment 1, "Unplanned loss of most or all alarms (annunciators) for greater than 15 minutes." The applicable indicator for this condition was failure of three of the four field power supplies for greater than 15 minutes.

Shortly after the event began, the shift supervisor decided not to declare an Unusual Event. The reasons were:

- Operators checked the field power supplies as part of the immediate actions. Operators noted that the field power supplies had not failed since they indicated 25 VDC. The operators were unaware at this time that 25 VDC was a degraded voltage;

- At approximately 4:11 p.m. switchyard breakers for one of the three offsite power sources opened and re-closed because of a lightning strike. The annunciator was received in the control room. This provided an indication that not all annunciators were lost; and

- The shift supervisor dispatched equipment operators to test various local annunciator panels shortly after onset of the event. Several tests brought in the corresponding control room annunciators. This also provided indication that not all annunciators were lost.

The inspectors found that the shift supervisor's initial decision not to declare the Unusual Event was in accordance with plant procedures. This was because there was indication to operators that the power supplies had not failed and because there was no conclusive indication that most annunciators had failed.

The inspectors found that the licensee made a more conservative interpretation of plant procedures in declaring the Unusual Event later, after further evaluation and troubleshooting.

The inspectors found weaknesses in operator procedures for describing minimum acceptable annunciator field power supply voltages and in what constitutes a failure of "most or all annunciators." The licensee agreed and had already identified action to clarify the procedures.

The inspectors reviewed the licensee's event review team meeting minutes and found the proposed actions to be comprehensive.

c. Conclusions

The inspectors concluded that operators responded well to this event. Operator actions to carry out compensatory actions until the annunciator system was fully restored were conservative. Management and technical support were very good. There were weaknesses in plant procedures regarding the criteria for describing minimum acceptable annunciator field power supply voltages and in what constituted a failure of "most or all annunciators." The licensee's proposed actions from the event review team meeting were comprehensive.

O2 Operational Status of Facilities and Equipment

O2.1 Review of Equipment Tagouts (71707)

The inspectors walked down the following tagout:

- Workman's Protection Assurance 23497 - Residual Heat Removal Train A.

The inspectors did not identify any discrepancies. All tags were on the correct devices and the devices were in the position prescribed by the tags.

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:

- Work Activity W191446 - Replace Fuse Block in Cubicle NN0111 (Feeder Breaker to Westinghouse Process Protection Set 1 Cabinet);

- Work Activity C575024 - Install New Breaker for Valve EFHV0060 (Essential Service Water Train B from Component Cooling Water Heat Exchanger B);

- Work Activity C600988 - Replace Orifice in Component Cooling Water to Residual Heat Removal Pump A Seal Cooler Flow Element EGFE0089;

- Work Activity P603299 - Motor-Operated Valve Periodic Test of Component Cooling Water to Residual Heat Removal Heat Exchanger A Isolation Valve EGHV0101; and

- Work Activity W193716 - Troubleshoot and Repair Ultimate Heat Sink Cooling Tower Fan.

b. Observations and Findings

The inspectors found no concerns with the maintenance observed. All work observed was performed with the work packages present and in active use. 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-SJ-L1P64, "Containment Isolation Valve Leak Rate Test for Valve SJHV00128 (Post Accident Sampling - Pressurizer and Reactor Coolant System Inner Containment Isolation)," Revision 0;

- Surveillance Procedure OSP-AL-P0002, "Section XI Turbine-Driven Auxiliary Feedwater Pump Operability," Revision 24;

- Surveillance Procedure OSP-SF-00001, "Shutdown Margin Calculation," Revision 18;

- Surveillance Procedure OSP-NE-0001B, "Standby Diesel Generator B Periodic Tests," Revision 2; and

- Surveillance Procedure ISF-BB-0P458B, "Analog Channel Operational Test on RCS (Reactor Coolant System) Pressure Channel 4," Revision 9.

b. Observations and Findings

Surveillance testing observed during this inspection period 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 General Comments (62707)

The inspectors made several tours of the plant. Material condition and housekeeping of accessible areas of the auxiliary building, the fuel building, the essential service water pumphouse, and most areas of the turbine building were all very good.

M3 Maintenance Procedures and Documentation

M3.1 Waste Gas Holdup System Explosive Gas Monitoring Instrumentation

a. Inspection Scope (61726)

The inspectors reviewed procedures and processes used to demonstrate operability of the explosive gas monitoring instrumentation in the waste gas holdup system.

The inspectors reviewed the following:

- Procedure ISL-HA-000A1(B1), "Loop Anlzr; Waste Gas Analyzer," Revision 11(14);
- Procedure RSP-HA-0004A(4B), "Standardization of the Gaseous Radwaste System Analyzers on Panel HA-161(162)," Revision 0;
- Vendor Manual M-725-00122, "Instruction Manual for Catalytic Hydrogen Recombiner";
- Brooks Instrument Design Specification Sheet DS-1355;
- Instrument Society of America Recommended Practice ISA-RP16.5, "Installation, Operation, Maintenance Instructions for Glass Tube Variable Area Meters (Rotameters)";
- Final Safety Analysis Report Section 16.11.2.7.1; and
- Request for Resolution 14181 Revisions A and B.

b. Observations and Findings

Indications and controls for the explosive gas monitoring instrumentation in the waste gas holdup system were located mainly on catalytic hydrogen recombiner waste processing system Panels HA-161 and HA-162 in the radwaste building. Operability requirements of the explosive gas monitors were described in Final Safety Analysis Report Section 16.11.2.7.1.

The operability requirements included a channel calibration of both instrument channels at least once per 92 days. The licensee used Procedures ISL-HA-000A1(-B1) and RSP-HA-0004A(-4B) to provide documentation and instructions for calibrating both of the waste gas holdup system explosive gas monitoring channels. Although the calibration requirement is once per 92 days, the licensee has been performing the calibration monthly to coincide with a separate requirement to perform a monthly channel test.

The inspectors noted that Pequest for Resolution 14181, Revision A, identified that the oxygen and hydrogen flow indicators (rotameters) on Panels HA-161 and HA-162 were not routinely calibrated. The request for resolution questioned whether these rotameters should be calibrated, since these rotameters were part of the explosive gas monitoring instrumentation described in Final Safety Analysis Report Section 16.11.2.7.1.

Plant engineering personnel concluded on Request for Resolution 14181, Revision A, that the rotameters could not be calibrated because of lack of any adjustment capability. The inspectors found this evaluation to be weak, since the conclusion was unsupported.

The licensee re-evaluated the issue and provided documentation in Request for Resolution 14181, Revision B. Plant engineers concluded in this revision that calibration of the oxygen and hydrogen rotameters on Panels HA-161 and HA-162 was not required. The following basis was provided:

- According to Manual M-725-00122, the ability of the oxygen analyzer to detect oxygen concentration was independent of the flow rate.

- The licensee contacted a representative for the manufacturer of the rotameters at Brooks Instruments. The representative informed the licensee that calibration of the rotameters was not required for the application used at Callaway.

- The section on recommended maintenance in Recommended Practice ISA-RP16.5 did not include a recommendation to periodically verify the calibration of the rotameters.

However, the licensee found that Manual M-725-00122 did state that the hydrogen analyzers were flow sensitive. The licensee evaluated this and again concluded that the hydrogen rotameter did not require calibration. The licensee provided the following justification.

Calibration of the system involved sending a waste gas sample through the hydrogen analyzer and recording the flow rate. A standard gas sample with a known concentration of hydrogen was then sent through the hydrogen analyzers at that same flow rate. The hydrogen analyzers were then adjusted to read the true hydrogen concentration. The system was then rechecked and set up to process future waste gasses using this same flow rate until the next calibration. The licensee found that, although it was not critical that the rotameters display the true hydrogen flow rate passing through the system, it was important that the rotameters have good repeatability.

The licensee did some research on repeatability of the rotameters. According to Specification Sheet DS-1355, the repeatability of the rotameters was very good, to within 0.5 percent of full scale. The licensee concluded that this would have no significant impact on hydrogen recombiner rotameter readings.

Although there was no recommendation by the Brooks representative and in Recommended Practice ISA-RP16.5 to calibrate the rotameters, there were recommendations to check for leaks, deterioration, and wear. The licensee stated that the monthly calibrations and channel tests were sufficient to identify these problems.

The inspectors agreed with the licensee's assessment that the rotameters on Panels HA-161 and HA-162 did not require periodic calibration given the method of processing waste gasses at a constant flow rate.

The inspectors reviewed the test and calibration records for the waste gas holdup system explosive gas monitoring channels performed within the last 2 years. The inspectors found that the licensee performed the surveillance testing satisfactorily in accordance with the Final Safety Analysis Report Section 16.11.2.7.1.

c. <u>Conclusions</u>

The inspectors concluded that:

- The licensee has been performing testing and calibration of the waste gas system explosive gas monitoring channels in accordance with requirements;
- The rotameters on Panels HA-161 and HA-162 did not require periodic calibration given the method the licensee uses to process waste gasses at a constant flow rate; and,

- The monthly channel calibrations and channel tests were sufficient to check for leaks, deterioration, and wear of the rotameters.

M8 Miscellaneous Maintenance Issues (92902)

M8.1 (Open) Licensee Event Report (LER) 50-483/96007: missed surveillance and literal Technical Specification compliance violations due to personnel oversight.

The missed surveillance, identified by the licensee, involved a loop calibration for Component Cooling Water Flow Transmitter EGFT0062 that had not been performed during the April 1995 refueling outage. This activity had initially been scheduled every 18 months as a preventive maintenance activity as opposed to a surveillance. The frequency was subsequently changed to 36 months based on performance results, which would be allowed for a preventive maintenance activity. Because this calibration was a Technical Specification required surveillance, the frequency change was inappropriate and resulted in a violation of Technical Specification 4.7.3.b.1. Additional reviews by the licensee determined that portions of other surveillances required to be performed "at least once per 18 months during shutdown" were actually being performed in Mode 1 and had been performed in this mode since plant startup.

The inspector reviewed the corrective actions for the missed loop calibration and determined that the surveillance had been appropriately captured in the surveillance data base with the correct frequency and plant conditions. Licensee reviews identified several other Technical Specification required surveillances that were being performed as preventive maintenance tasks and relocated them into the surveillance data base. None of those had exceeded their scheduled dates.

While reviewing the corrective actions for the surveillances performed in Mode 1 vice shutdown, the inspector determined that, although the surveillances were currently scheduled to be performed in the next refueling outage, the surveillance data base had not been changed to reflect that they were to be performed while shutdown. The inspectors will review the licensee's actions further as they are completed.

M8.2 (Closed) LER 50-483/96008: missed Technical Specification surveillance of 125-VDC batteries due to cognitive personnel error.

This event involved an electrician performing a weekly battery surveillance on the wrong train of batteries resulting in Train B of the station batteries being without a current surveillance for approximately 17 hours. The licensee identified this condition during a surveillance data review and immediately took actions to collect and verify data that demonstrated the operability of the Train B battery. Additional corrective actions included counseling of the individual and revisions to the battery surveillance data sheets to include the specific train of battery being surveilled. Additionally, the license reviewed other cases in which a generic procedure could allow the incorrect train to be surveilled because of lack of specificity in the procedure regarding component identification. The inspector reviewed the completed corrective actions and concluded that they were adequate to preclude recurrence. This event constituted a noncompliance with Technical Specification 4.8.2.1.a, which requires that the batteries be surveillance tested once every 7 days. This nonrepetitive, licensee-identified, and corrected violation is being treated as a noncited violation consistent with Section VII.B.1 of the NRC Enforcement Policy (50-483/9714-01).

M8.3 (Closed) Licensee Event Report 50-483/97003: error in the Technical Specification description for turbine-driven auxiliary feedwater pump start.

The licensee identified that, during submittal, approval, and implementation of Amendment 43 to the Operating License in 1989, an error was introduced into the Technical Specifications regarding the conditions required to cause an automatic start of the turbine-driven auxiliary feedwater pump. The error went unnoticed until identified by the licensee on March 31, 1997. The NRC was informed. In addition to verifying that the turbine-driven auxiliary feedwater pump start logic was being operated as designed, the licensee initiated an emergency Technical Specification change to correct the error. The change was subsequently approved by NRC and incorporated by the licensee. The licensee's license amendment process was revised to provide additional review and scrutiny of license amendment requests. The plant was never operated outside of the design basis as a result of this administrative error, and the licensee's corrective actions were appropriate.

M8.4 (Closed) LER 50-483/97004: missed surveillance per Technical Specification 4.3.2.1 on the turbine and feedwater pump trip slave relay due to cognitive personnel error.

This LER involved a missed surveillance for Slave Relay K620 that was required to be performed each cold shutdown exceeding 24 hours. On one occasion, in October 1995, the surveillance was not performed following a cold shutdown condition exceeding 24 hours. During additional reviews of this event by the licensee, they discovered that slave relay surveillance tests for Relays K620 and K630 had, on several occasions, been performed at power instead of "during refueling" as required by Technical Specifications.

Corrective actions included verifying that the surveillance tests were technically accurate and verifying that testing at power provided the same assurance of operability as testing during refueling. The task sheets for the subject surveillances were revised to specifically note the exact condition requirements of the Technical Specifications. The responsible individuals were counseled on the importance of reviewing Technical Specifications when revising surveillance procedures to ensure all applicable requirements are met.

These occurrences represent noncompliances with ths Technical Specifications regarding the addressed surveillance requirements. This nonrepetitive, licensee-identified, and corrected violation is being treated as a noncited violation in accordance with Section VII.B.1 of the NRC Enforcement Policy (50-483/9714-02).

III. Engineering

E1 Conduct of Engineering

E1.1 Axial Offset Anomaly

a. Inspection Scope (37551)

The inspectors reviewed actions the licensee was taking to address an axial offset anomaly. This included review of the licensee's compliance with requirements for maintaining various reactor core parameters within specified values.

The inspectors reviewed:

- Cycle 9 Reactor Engineering Trend Reports;
- Procedure ESP-ZZ-00014 "Heat Flux Hot Channel Factor," Revision 20;
- Technical Specifications 3/4.1.1.5; 3/4 1.3.6, 3/4.2.1, and 3/4.2.2;
- Cycle 9 Core Operating Limits Report;
- Suggestion-Occurrence-Solution Report 96-1821; and
- Procedure OSP-SF-00001, "Shutdown Margin Calculation," Revision 18.

b. Observations and Findings

During this operating cycle the licensee experienced a substantial deviation from the predicted behavior of core axial offset. This deviation was a gradual unexpected power shift toward the bottom of the core and began to occur around 4,000 megawatt days per metric ton of uranium of core burnup. The power shift would continue until enough fuel had been used in the bottom of the core that power shifted back to the top of the core. This shift would happen later in core life.

The licensee identified that the most likely cause of the axial offset deviation was a buildup of crud on the outside of the fuel pins in the upper spans of the core. This would create the presence of negative reactivity in the upper regions of the core, forcing the power shape to skew toward the bottom of the core.

The axial offset deviation resulted in deviation from expected values for several core parameters. These included: (1) the limit for the heat flux hot channel factor F(q); (2) the limit for axial flux difference; and (3) the limit on shutdown margin. The licensee confirmed that the deviation from expected values did not represent a safety hazard and began a detailed investigation.

Heat Flux Hot Channel Factor F(q)

The inspectors reviewed Technical Specification 3.2.2 for requirements associated with the heat flux hot channel factor F(q). The limit on this parameter ensured that fuel damage did not occur due to overpowering any given location in the core.

On July 7, 1997, calculations from an incore flux map indicated that the value of the heat flux hot channel factor F(q) exceeded the surveillance limit defined in Technical Specification 4.2.2.2.c. Although the more restrictive limit defined by the Limiting Condition for Operation of Technical Specification 3.2.2 was not exceeded, there were actions that the licensee was required to take after exceeding the surveillance limit. The licensee properly followed the required actions, by placing tighter limits on the upper and lower limits for axial flux difference and by resetting the associated alarm setpoints.

A subsequent flux map the following day demonstrated that the surveillance limit had not been exceeded; however, the proximity to the heat flux hot channel factor F(q) surveillance limit was still so close that the new limits on axial flux difference were left in place while further evaluations were conducted.

Determination of the heat flux hot channel factor F(q) is normally performed every 31 effective full power days with an incore flux map. However, because of an increasing trend on several occasions, the licensee was required to perform the heat flux hot channel factor F(q) measurement every 7 effective full power days. Performance of the weekly measurements was required by Technical Specification 4.2.2.2.e. The inspectors reviewed the weekly measurements and identified no concerns.

The inspectors reviewed trend reports for the heat flux hot channel factor F(q) for the present operating cycle and had discussions with reactor engineering personnel. Although the surveillance limit had been exceeded during the operating cycle, the inspectors determined that the licensee had never exceeded the Limiting Condition for Operation value for the heat flux hot channel factor F(q) during the current operating cycle. Exceeding the Limiting Condition for Operation value for the heat flux hot channel factor F(q) would have required the licensee to reduce reactor power.

The inspectors reviewed recent performances of the heat flux hot channel factor F(q) determination per Procedure ESP-ZZ-00014. The inspectors found that the licensee was properly performing the procedure and documenting and reviewing the results. The inspectors identified no concerns.

Axial Flux Difference

Axial flux difference involves a comparison of the power produced in the top of the core to the power produced in the bottom of the core. A positive axial flux difference implies that more power is being produced in the top half of the core than the bottom half of the core. A negative value implies more power is being produced at the bottom half of the core than the top half of the core. The limits for axial flux difference were specified in Technical Specification 3.2.1 and in the Core Operating Limits Report.

As mentioned previously, the incore flux map performed on July 7, 1997, resulted in the licensee placing tighter restrictions on the upper and lower limits for the axial flux difference than originally specified in the Core Operating Limits Report. For the negative limit, the axial flux difference value at 100 percent power was adjusted from -17.0 percent to -15.5 percent. Exceeding the upper or lower limits would have required the licensee to reduce reactor power.

The inspectors reviewed trend reports for axial offset during the present operating cycle. Axial offset is the axial flux difference divided by the fraction of reactor power. Axial offset was trended since it provided a more direct indicator of the actual axial flux distribution in the core at different power levels.

Just prior to reducing reactor power to 95 percent, axial offset was at the lowest value for the cycle, at approximately -14.0 percent. This was well below the predicted value of -1.0 percent, but within the lower administrative limit of -15.5 percent. After the power reduction to 95 percent reactor power, axial offset was approximately -11.5 percent. The power reduction allowed more margin to the lower limit; at 95 percent reactor power the lower limit was approximately -18.0 percent.

The axial offset anomaly has caused the licensee difficulty in controlling the amount of available shutdown margin. Other problems related to the anomaly included the adverse effects of xenon transients and potentially higher than expected outage dose rates.

Shutdown Margin

The axial offset anomaly has caused shutdown margin to reduce at a faster rate than predicted. This was due primarily to the combination of two factors:

- The formation of crud deposits at the upper elevations of the fuel occurred at a rate greater than anticipated. Boron concentration was enhanced within the crud, concentrating a neutron poison and inserting negative reactivity at the upper elevations of the fuel. The licensee's calculational model assumed that, upon a reactor trip, all this boron would be released, resulting in a positive reactivity insertion at the upper elevations of the core. Therefore, more negative reactivity would have to be inserted to make and keep the reactor subcritical.

- The axial offset anomaly caused less fuel to be depleted near the top of the core. As happens upon a reactor trip, the temperature of the coolant at the top of the core would decrease with an attendant increase in coolant density. With more available fuel and denser coolant at the top of the core, more positive reactivity would exist at the top of the core than would normally exist. Therefore, more negative reactivity would have to be inserted to overcome the positive reactivity.

Technical Specifications 3.1.3.6 and 4.1.1.5.2 require that the shutdown margin in Modes 1 and 2 be greater than or equal to 1300 pcm. As of July 28, 1997, shutdown margin had reduced to a value of 1511 pcm. The licensee's calculation showed shutdown margin reaching 1300 pcm on August 15, 1997. At that time, in order to preserve the shutdown margin at 1300 pcm, a power reduction would be required. The inspectors reviewed the licensee's current performance of shutdown margin calculations in accordance with Procedure OSP-SF-00001 and identified no concerns.

Procedure OSP-SF-00001 assumed a 10 percent uncertainty for the available rod worth. The inspectors noted that the licensee had an approved safety evaluation to assume a 3 percent rod worth uncertainty. This would extend the date when shutdown margin would be at 1300 pcm. The licensee has not yet assumed the 3 percent rod worth uncertainty in the shutdown margin calculations because shutdown margin trend curves generated by Westinghouse and the licensee differ. The licensee was working to resolve those differences prior to assuming the 3 percent rod worth uncertainty.

The licensee was taking the following actions as a result of the axial offset anomaly.

- On June 10, 1997, the licensee began performing weekly shutdown margin calculations for more accurate tracking and trending. In addition, the licensee was developing a program to provide a real time computer display of the shutdown margin. The program would gather the necessary data from the plant computer, perform the calculation, and display the results. The licensee would still perform a separate calculation once per week.

- The licensee began to evaluate acceptability of an increase to the control rod insertion limits from 161 steps to 181 steps on Control Rod Bank D. This would allow approximately 10 extra days until the shutdown margin limit was reached.

- The licensee revised the Core Operating Limits Report to incorporate a change to the upper and lower limits for the axial flux difference. The new limits specified new full power upper and lower limits of +6.0 percent and -17.0 percent, respectively. The purpose of the revision was to generate additional margin to the heat flux hot channel factor F(q) limit.

- The licensee, in concert with Westinghouse, performed a safety evaluation to address the impact of the crud deposition on the various postulated design basis accidents. The safety evaluation concluded that the axial offset anomaly did not represent an unreviewed safety question and, hence, would not adversely affect plant operation. The licensee planned to re-evaluate the safety evaluation in August 1997 and perform another safety evaluation if necessary.

- The licensee has been holding weekly conferences with Westinghouse personnel since the beginning of June 1997. The purpose of the conferences was to discuss current concerns regarding the anomaly and set priorities for resolving those concerns.

The inspectors will continue to follow the licensee's actions. Pending further review of the acceptability of using a 3 percent rod worth uncertainty in shutdown margin calculations, and the other actions the licensee is taking, this is considered an inspection followup item (50-483/9714-03).

c. Conclusions

The inspectors concluded that the licensee was in compliance with the Technical Specifications for the heat flux hot channel factor F(q), axial flux difference, and shutdown margin. The licensee was aggressively monitoring shutdown margin and the other plant parameters to ensure the plant remained within the operating limits.

E8 Miscellaneous Engineering Issues (92903)

E8.1 (Closed) Unresolved Item 50-483/9707-05: Response time discrepancy with control room ventilation radioactivity monitors.

The inspectors identified a discrepancy between the Final Safety Analysis Report and plant test practices. The discrepancy involved the response time of the control room ventilation isolation system.

The inspectors reviewed:

- Suggestion-Occurrence-Solution Report 97-357;
- NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants";
- Final Safety Analysis Report Sections 15.6.5.4 and 15A.3; and
- Final Safety Analysis Report Tables 7.3-7 and 15.6-8.

Final Safety Analysis Report Table 7.3-7 stated that the response time of the control room ventilation radioactivity Monitors GKRE04 and GKRE05 was less than 3.0 seconds. Monitors GKRE04 and GKRE05 continuously monitor the supply of air of the normal heating, ventilation, and air conditioning system for particulate, iodine, and gaseous radioactivity. The monitors isolate control room ventilation from the outside environment in the event high airborne radioactivity is introduced into the control room heating, ventilation, and air conditioning supply duct.

The only response time test performed on the monitors was in April 1984, during preoperational testing. The response time reported for Monitor GKRE04 was 4.1 seconds. The response time reported for Monitor GKRE05 was 3.8 seconds. The licensee initiated action to resolve the discrepancy between the Final Safety Analysis Report and the preoperational test results.

The licensee could not identify a reason for the discrepancy and could not identify any documentation that would indicate an effort to validate the 3.0 second response time listed in the Final Safety Analysis Report Table 7.3-7.

The inspectors considered the acceptance of the as-is condition of the response time for Monitors GKRE04 and GKRE05 to be a de facto change to the facility as described in the Final Safety Analysis Report. As such, the licensee was required by 10 CFR 50.59 to evaluate and document the acceptability of the change to ensure that an unreviewed safety question did not exist. The inspectors considered the acceptance of this de facto change a violation of 10 CFR 50.59.

The licensee had already identified the need to review the system containing Monitors GKRE04 and GKRE05. This was described in the licensee's letter of February 5, 1997, which committed them to review the Final Safety Analysis Report. In the letter, the licensee stated that a task team was formed in March 1996 to determine the scope of review required to provide assurance that the Callaway Plant is operated according to the Final Safety Analysis Report.

The task team completed the review in July 1995 and identified actions and prioritized various Final Safety Analysis Report sections for review. One section identified for review was the control building heating, ventilation, and air conditioning system containing Monitors GKRE04 and GKRE05. The licensee's action plan was to perform the review from August 31 through September 18, 1998. The licensee had identified a review team and estimated 520 man-hours to complete the review.

The Commission recently approved modifications to the NRC Enforcement Policy (NUREG 1600) to address departures from the Final Safety Analysis Report. Section VII.B.3 of the Policy, "Violations Involving Old Design Issues," addresses enforcement discretion when licensees identify the problem. Although NRC inspectors identified the error in Final Safety Analysis Report Table 7.3-7, it is apparent that the licensee would have identified those errors in light of the defined scope, thoroughness, and schedule of their review plan.

The failure to evaluate and document the acceptability of the Final Safety Analysis Report response time discrepancy for Monitors GKRE04 and GKRE05 represented a noncompliance with 10 CFR 50.59. This is being treated as a noncited violation in accordance with Section VII.B.3 of the NRC Enforcement Policy (50-483/9714-04).

The licensee completed a preliminary investigation and did not identify any operability issues. The licensee found that Final Safety Analysis Report Section 15A.3 stated that only radiation dose due to a postulated loss-of-coolant accident was discussed in Final Safety Analysis Report Chapter 15. This was because a study of the radiological consequences in the control room due to various postulated accidents indicated that the loss-of-coolant accident was the limiting case. As such, the ventilation path containing Monitors GKRE04 and GKRE05 would isolate on a safety injection signal before these monitors would detect sufficient activity to initiate the control room ventilation isolation signal.

The licensee evaluated other accident scenarios to assure that the loss-of-coolant accident was still the bounding control room dose accident assuming the delay in the response time of Monitors GKRE04 and GKRE05. The licensee found that the loss-of-coolant accident was still the bounding control room dose accident.

The licensee also performed research regarding monitor response time and did not identify any significant degradation over time. The licensee identified that the only real contributor to monitor response time was from the monitor's counter and signal conditioning circuit. The licensee found that the response time of the counter and conditioning circuit was not a limitation of the installed hardware but a design feature to reject spurious inputs.

The inspectors agreed with the licensee's preliminary evaluation. The inspectors found that the difference between the preoperational test measured response times of 4.1 and 3.8 seconds and the Final Safety Analysis Report stated response time of 3.0 seconds would not significantly affect the control room dose consequences from a loss-of-coolant accident.

Pending the inspectors' review of the licensee's completed actions to resolve the discrepancy in Final Safety Analysis Report Table 7.3-7, this is considered an Inspection Followup Item (50-483/9714-05).

IV. Plant Support

R1 Radiological Protection and Chemistry (RP&C) 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. 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.

V. Management Meetings

X1 Exit Meeting Summary

The exit meeting was conducted on August 1, 1997. 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
H. D. Bono, Supervising Engineer, Regulatory Support, Quality Assurance
D. G. Cornwell, General Supervisor - Electrical, Maintenance
M. S. Evans, Superintendent, Health Physics
D. T. Fitzgerald, Superintendent, Security
D. W. Griffith, Engineer, Independent Safety Engineering Group, Quality Assurance
R. T. Lamb, Superintendent, Operations
J. V. Laux, Manager, Quality Assurance
A. C. Passwater, Manager, Licensing and Fuels
G. L. Randolph, Vice President, Chief Nuclear Officer
R. R. Roselius, Superintendent, Chemistry and Radwaste
M. E. Taylor, Assistant Manager, Work Control
W. A. Witt, Superintendent, Systems Engineering

INSPECTION PROCEDURES USED

37551	Onsite Engineering
61726	Surveillance Observations
62707	Maintenance Observation
71707	Plant Operations
71750	Plant Support Activities
92902	Followup - Maintenance
92903	Followup - Engineering
93702	Prompt Onsite Response to Events at Operating Power Reactors

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

9714-01	NCV	Missed Technical Specification surveillance of 125 VDC batteries (Section M8.2)

9714-02	NCV	Missed Technical Specification surveillance of turbine and feedwater pump trip slave relays (Section M8.4)
9714-03	IFI	Axial offset anomaly issue (Section E1.1)
9714-04	NCV	Final Safety Analysis Report response time discrepancy for Monitors GKRE04 and GKRE05 (Section E8.1)
9714-05	IFI	Resolve discrepancy in Final Safety Analysis Report table for Monitors GKRE04 and GKRE05 (Section E8.1)

Closed

96008	LER	Missed Technical Specification surveillance of 125 VDC batteries (Section M8.2)
9714-01	NCV	Missed Technical Specification surveillance of 125 VDC batteries (Section M8.2)
97003	LER	Technical Specification error in description of turbine driven auxiliary feedwater pump start (Section M8.3)
97004	LER	Missed Technical Specification surveillance on turbine and feedwater pump trip slave relays due to personnel error (Section M8.4)
97014-02	NCV	Missed Technical Specification surveillance of turbine and feedwater pump trip slave relays (Section M8.4)
9707-05	UNR	Response time discrepancy with control room ventilation radioactivity monitors (Section E8.1)
9714-04	NCV	Final Safety Analysis Report response time discrepancy for Monitors GKRE04 and GKRE05 (Section E8.1)

Discussed

96007	LER	Missed surveillance and literal Technical Specification compliance due to personnel oversight (Section M8.1)