U.S. NUCLEAR REGULATORY COMMISSION

REGION II

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Facility:	Catawba Nuclear Station, Units 1 and 2
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Inspectors:	R. J. Freudenberger, Senior Resident Inspector P. A. Balmain, Resident Inspector R. L. Watkins, Resident Inspector C. W. Rapp, Senior Reactor Inspector
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Enclosure 2

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EXECUTIVE SUMMARY

Catawba Nuclear Station, Units 1 & 2
NRC Inspection Report 50-413/96-08, 50-414/96-08

This integrated inspection included aspects of licensee operations, maintenance, engineering, and plant support. The report covers a 6-week period of resident inspection; in addition, it includes the results of an announced inspection by a regional reactor inspector.

Operations

- Operators consistently took appropriate actions in response to challenges initiated by equipment failures (Sections O1.1, O1.2, and O1.3).

- An inadvertent loss of Spent Fuel Pool inventory was caused by an inadequate Removal and Restoration Order (tagout) for the circumstances. This issue was identified as Non-Cited Violation 413/96-08-01: Inadequate Removal and Restoration Order for Isolating Fuel Pool Cooling System Piping (Section O1.4).

- An effective operator vigilance aid was used during solid plant operations (Section O1.5).

- Control Room Operators were not aware that a Main Feedwater Isolation Valve (MFIV) had been briefly rendered inoperable and did not aggressively question the cause of a low nitrogen accumulator pressure computer alarm associated with the valves actuator (Section M1.2).

Maintenance

- The work order backlog had been significantly reduced over the past several months (Section M1.1).

- During a check of the nitrogen accumulator pressure associated with a MFIV, the valve was rendered inoperable due to failure to follow a maintenance procedure. This was identified as Violation 413/96-08-02: Failure to Follow Procedure When Adjusting MFIV Nitrogen Accumulator Pressure to Backseat Leaking MFIV (Section M1.2).

- An Inspector Followup Item (50-413/96-08-03) was opened to provide independent review of control rod drop time testing data (Section M8.2).

Engineering

- Controls to prevent a loss of Technical Specification (TS) required Shutdown Margin (SDM) during End of Cycle (EOC) reactor restarts were adequate. Although no formal trending program existed, TS required trending for reactivity anomalies provided adequate monitoring of actual versus predicted core performance (Section E1.1).

- The Operating Experience Assessment program effectively communicated a concern regarding charging pump seal cooling flow orifices which applied to Catawba (Section E2.1).

- Consistent support for timely cause determinations was demonstrated by engineering personnel appropriately utilizing the Failure Investigation Process philosophy (Section O1.3).

Plant Support

- Immediate and interim licensee actions to address a licensee identified design basis deficiency associated with the Standby Shutdown Facility auxiliary feedwater supply volume were appropriate (Section S2.1).

Report Details

Summary of Plant Status

Unit 1 operated at full power until a May 8 shut down to comply with Technical Specifications following a Rod Control System failure. The unit was restarted on May 10 and operated at full power until June 4 when coastdown to the 1EOC9 refueling/steam generator replacement outage began. The unit was shut down for the outage on June 12 and remained shutdown for the remainder of the inspection period.

Unit 2 operated at full power until May 6 when the B Main Generator output breaker opened and caused a Main Turbine runback to approximately 50% power. A power increase was initiated on May 7 and power was returned to 100% on May 8. The unit remained at full power until May 17 when power was decreased to 46% for Main Transformer tap adjustments. The unit was returned to full power on May 19 and operated at or near full power for the remainder of the inspection period.

Review of UFSAR Commitments

A recent discovery of a licensee operating their facility in a manner contrary to the Updated Final Safety Analysis Report (UFSAR) description highlighted the need for a special focus review that compares plant practices, procedures, and/or parameters to the UFSAR descriptions. While performing inspections discussed in this report, the inspectors reviewed the applicable portions of the UFSAR that related to the areas inspected. The inspectors verified that the UFSAR wording was consistent with the observed plant practices, procedures, and/or parameters.

Other Inspections

A maintenance inspection was conducted the week of June 3, 1996. The results of the inspection which focused on Maintenance Self-Assessment will be detailed in a separate inspection report.

I. Operations

O1 Conduct of Operations

O1.1 Runback Initiated by a Relay Failure (Unit 2)

a. Inspection Scope (71707)

On May 6, Unit 2 experienced an automatic runback from full power to approximately 56 percent power. The runback occurred during a power supply restoration associated with the 2B main transformer cooling fans and pumps (cooling group 2). The power supply for this cooling group had previously been deenergized from its normal power supply and realigned to an alternate source to support the installation of a plant modification (reverse osmosis water purification system). When operators opened the cooling group's alternate power supply during the restoration process the 2B main generator breaker tripped open and initiated the runback. The inspector reviewed the licensee's efforts to determine the cause of the runback.

b. Observations and Findings

Plant equipment responded properly during the automatic runback. Following the automatic runback, operators took actions to reduce power further to 50% to clear overcurrent alarms. The licensee's subsequent investigation determined that the event was initiated by the 2B main transformer loss of cooler power protective relaying circuit. This circuit prevents the transformer from being damaged due to excessive heating caused by a loss of one or both its two cooling groups. The circuit is designed to send a trip signal to the 2B generator output breaker 3 minutes after a loss of power to both of the transformer's cooling groups or 15 minutes after a loss of power to one cooling group. Realigning power supplies to cooling groups was performed locally at the 2B main transformer which is in close proximity to the generator output breakers. The inspector walked through and discussed the realignment process with the operator who performed the power restoration. The operator noticed that the 2B generator output breaker opened with no time delay immediately after the alternate power supply to cooling group 2 was deenergized.

Utilizing the Failure Investigation Process, the licensee developed a troubleshooting and test plan to determine what component in the protection circuit actuated the 2B generator output breaker trip and subsequent runback. The inspector observed the control room pre-job briefing and a portion of the testing and found that the evolution was well controlled. The licensee's testing was effective and determined that the generator output breaker actuation relay (device XC) in the 3 minute time delay portion of the circuit was closing each time power to cooling group 2 was deenergized. This initiated the trip signal to the generator output breaker. The three minute time delay portion of the circuit is not designed to actuate when power to one cooling group is lost and the licensee's testing confirmed that the timer did not actuate or send a valid electrical signal to the XC relay. The licensee suspected the vibration from the movement of an adjacent relay was causing the XC relay to close. The licensee replaced the XC relay and tested the circuit several times to verify the new relay was not susceptible to the same failure. The licensee also performed a failure analysis on the relay and determined on a mockup that the failed XC relay would close due to vibration induced by an adjacent relay actuation and that the failure was repeatable.

The inspector reviewed additional actions which the licensee had initiated as a result of this failure (PIP 2-C96-1059). The actions included performing additional analysis of the failed relay, reviewing the main power system preventive maintenance procedures to ensure that all components important to plant reliability are tested, and vibration monitoring of the Unit 1 relay cabinets during scheduled protective relay preventive maintenance.

c. Conclusion

Based on the above review, the inspector concluded that the licensees's actions to identify the failed XC relay were timely and effective.

O1.2 Control Rod System Failure Results in Required Shutdown (Unit 1)

a. Inspection Scope (71707)

On May 8, during the performance of the Unit 1 monthly control rod movement surveillance, a circuit card located in the Rod Control System logic circuitry failed. The failure resulted in a control rod misalignment in excess of TSs and required the licensee to shutdown Unit 1 to Mode 3 (Hot Shutdown). While in Mode 3, a Low-Low Tave Engineered Safety Feature actuation occurred. The inspector reviewed the Control Rod System failure and Engineered Safety Feature actuation and observed the repair and testing of the failed circuit card.

b. Observations and Findings

During the monthly control rod movement testing of Control Bank A the rods were inserted 10 steps into the core. When Control Bank A was withdrawn to its original position two rods (H-6 and H-10) in the bank failed to move. Due to the misalignment, the licensee entered abnormal operating procedures and the actions required by TS 3.1.3.1, Movable Control Assemblies, to shutdown the unit when rod alignment could not be restored. The licensee completed a unit shutdown to Mode 3 within 6 hours as required. During the shutdown, control rods H-6 and H-10 continued to malfunction and could not be fully inserted. To compensate for the potential loss of negative reactivity the licensee increased Reactor Coolant System boron concentration following the shutdown. The inspector reviewed shutdown margin calculations performed after the unit entered Mode 3 and verified that the required shutdown margin was maintained and compensated for the two inoperable control rods. The licensee determined that the control rod system malfunction was caused by a failed firing card. The inspector witnessed the repair and testing activities and observed that the replacement card responded normally. A similar firing card failure occurred on Unit 2 in June 1994 (refer to NRC Inspection Reports 413,414/94-19, 413,414/96-05, and LER 414/94-02).

While the unit was in Mode 3, a P-12 (Low-Low Tave) Engineered Safety Feature actuation occurred. The licensee determined that this actuation occurred because Unit 1 was aligned to supply the auxiliary steam system which caused the Reactor Coolant System to cooldown from 557 degrees F to the P-12 setpoint of 553 degrees F. The Engineered Safety Feature actuation functions to automatically close the steam dump valves if they are open and blocks the valves from reopening to prevent additional Reactor Coolant System cooldown. At the time the actuation occurred the valves were already closed. The licensee determined that this actuation occurred due to insufficient procedural guidance for aligning the auxiliary steam system during a unit shutdown. Accordingly, procedure revisions were initiated to ensure adequate transfer of the steam supplies. The licensee submitted a Licensee Event Report because of the Technical Specification required shutdown and Low-Low Tave Engineered Safety Feature actuation. The inspector will verify that planned corrective actions are completed when the Licensee Event Report is reviewed in future inspections.

c. Conclusion

The inspector concluded that the licensee's actions in response to this equipment failure were appropriate.

O1.3 Reactor Trip in Mode 3 (Unit 1)

a. Inspection Scope (71707)

On June 13, with Unit 1 in Mode 3 (Hot Shutdown), an automatic reactor trip was generated due to a loss of power to vital bus 1EDA when breaker F03C opened. All control rods fully inserted into the core. Coincident with the trip, main feedwater isolated and the operators manually started auxiliary feedwater. The inspector assessed the cause of the trip, the main feedwater isolation, the opening of the inverter output breaker, and operator actions.

b. Observations and Findings

The operators were preparing to perform control rod drop time testing. All control rods were fully withdrawn and reactor coolant system Tave was approximately 551°F at the time of the reactor trip. The reactor tripped per design due to the loss of power to one of two intermediate range nuclear instrument channels. A main feedwater isolation signal on reactor trip coincident with low Tave (553°F) was immediately generated per design. The operators appropriately started auxiliary feedwater to re-establish feedwater to the steam generators. The licensee performed some initial troubleshooting and, in the absence of a clear indicator of the cause of the failure, initiated their Failure Investigation Process. The cause of the inverter output breaker opening was later determined to be valid as the result of a failed capacitor in the associated inverter.

c. Conclusions

As noted in the preceding sections (O1.1 and O1.2) and this section, operators consistently took appropriate actions in response to challenges initiated by equipment failures. These actions included:

completion of an automatic runback to clear generator output breaker overcurrent alarms, performance of a TS required controlled shutdown in response to misaligned control rods, and manual initiation of auxiliary feedwater following a reactor trip and main feedwater isolation. Additionally, consistent support for timely cause determinations was demonstrated by engineering personnel appropriately utilizing the Failure Investigation Process philosophy.

O1.4 Spent Fuel Pool Siphon and Unplanned Drain (Unit 1)

a. Inspection Scope (71707 and 40500)

On May 22, operators were isolating a section of the Unit 1 Spent Fuel Pool Cooling System piping in preparation for repair of a seat leak on the Spent Fuel Pool Cooling purification loop outlet throttle and isolation valve 1KF-36. During the isolation, a siphon was created that caused approximately 5,500 gallons of water to be drained from the Spent Fuel Pool to the Recycle Holdup Tank. Spent Fuel Pool level dropped by 0.7 feet and remained above the minimum level required by Technical Specifications. The inspector assessed the licensee's implementation of the Removal and Restoration program as it applied to this case and reviewed PIP 1-C96-1235, which documented the licensee's root cause evaluation of the event.

b. Observations and Findings

In preparation for maintenance on 1KF-36, a Removal and Restoration order (tagout) was prepared for isolating and draining the piping. Removal and Restoration Order 16-972 directed the operator to open valve 1KF-173, a high-point vent valve, to control siphoning from the Spent Fuel Pool to the Recycle Holdup Tank. Opening the vent valve was sequenced to follow the opening of 1KF-64, drain header to the Recycle Holdup Tank isolation valve. This sequence established a siphon from the Spent Fuel Pool to the Recycle Holdup Tank until the vent valve was opened. The vent valve was located two elevation levels above the drain valve in the auxiliary building. A ladder had to be obtained so that the operator could access the vent valve to remove a pipe cap and open the valve to break the siphon. Roughly 20 minutes elapsed between the opening of the drain and vent valves.

The operators who developed the Removal and Restoration sequence were aware that a siphon would be created when the drain valve was opened. Licensee practice has been to open drain valves before vent valves to avoid spillage through an open vent valve as would occur in a pressurized system. Based on this convention, the Removal and Restoration Order was sequenced for the vent valve to be opened after the drain valve. In this case, the elevation difference and system dynamics would have allowed the vent path to be established first with minimal possibility of spillage from the vent valve.

The operators preparing for the isolation requested and received guidance from Engineering that roughly 100 gallons of water would drain from the piping located between the vent valve and the drain valve. However, the operators did not request that the siphon rate or the amount of water be quantified. The rate of siphoning was neither adequately questioned nor fully appreciated until the control room operators noted a 0.7 foot drop in Spent Fuel Pool level roughly two hours after the Removal and Restoration Order was placed. For this reason, the operator placing the Removal and Restoration Order was not aware of the expected siphon; nor was the operator instructed verbally or procedurally to limit the amount that would be siphoned by opening the vent valve in a timely manner.

The inspector reviewed PIP 1-C96-1235 which included documentation of the licensee's root cause evaluation of the issue. The root cause evaluation concluded that a knowledge deficiency existed regarding the rate at which water inventories can be transferred by siphons.

Although a siphon which inadvertently drained the Spent Fuel Pool by 0.7 feet was created, the inspector concluded the direct safety impact of this issue was minimal. The initial Spent Fuel Pool level was slightly high and the Low Spent Fuel Pool alarm level was not reached. Also, a passive siphon breaker in the Spent Fuel Pool suction piping would have terminated the siphon above the minimum level required by TS.

c. Conclusions

The loss of Spent Fuel Pool inventory was identified by a control room operator, and the impact to safety was minimal. Nonetheless, the issue was of regulatory significance since these barriers were potentially challenged. Technical Specification 6.8.1 requires that procedures be established for activities important to safety including removal and restoration of equipment for maintenance. Removal and Restoration Order 16-972 was inadequate in that it was sequenced to induce a siphon from the Spent Fuel Pool to the Recycle Holdup Tank without ensuring that adequate guidance and controls were in place to monitor the rate of drain and establish limits based on Spent Fuel Pool level. This licensee-identified and corrected violation is being treated as a Non-Cited Violation, consistent with Section VII.B.1 of the NRC Enforcement Policy. NCV 413/96-08-01: Inadequate Removal and Restoration Order for Isolating Fuel Pool Cooling System Piping.

O1.5 Solid Plant Operations - Unit 1 (71707)

On June 15, the inspector was monitoring control room activities associated with solid plant operations. A lap-top computer had been installed to continuously monitor Reactor Coolant System pressure and temperature to assist control room operators in maintaining Reactor Coolant System parameters below pressurizer Power Operated Relief Valve (PORV) lift setpoints. The computer displayed color-coded graphical and digital data as well as audible alarms to ensure that operators were informed of changes in monitored parameters and could provide a timely response based on rate of change to avoid lifting a pressurizer PORV. The display was well designed and the system was an effective operator vigilance aid.

O8 Miscellaneous Operations Issues (92901)

O8.1 (Closed) LER 50-414/94-03: Manual Reactor Trip on Loss of Normal Feedwater Due to System Performance

The Licensee Event Report addressed a manual reactor trip following an unsuccessful runback from full power due to the loss of one main feedwater pump during troubleshooting. Inspector review of the cause of this event and short-term corrective actions is documented in NRC Inspection Report 50-413,414/94-22. Long-term actions were reviewed during this report period.

Long-term actions focused primarily on evaluation of the digital feedwater control system response and any potential for improving system performance. The only change implemented was a change to the main feedwater pump differential pressure program to change the normal operating position of the feedwater control valves from 55% to 65% at full power to improve efficiency of the main feedwater pumps. The engineering evaluation concluded that the leading cause of the failure to survive the runback was the operator's manual action. No other actions to improve digital feedwater control system performance were necessary. The inspector reviewed the corrective action status as described in PIP 2-C94-0993, observed main feedwater control valve position at full power, and considered the licensee evaluation acceptable. This item is closed.

II. Maintenance

M1 Conduct of Maintenance

M1.1 General Comments

Between the period from December 1995 through June 1996, the facility operated in what the licensee terms an "innage" period. There were no scheduled refueling outages on either unit during this time. At the beginning of the innage, the work request inventory was the highest among the three Duke facilities. Nonetheless, it was improved over past performance for this site. By the end of the innage period through improved scheduling efficiency, targeted use of overtime, and strong site focus, the total work request inventory and the non-outage corrective work order backlog were significantly reduced. As mentioned in NRC Inspection Report 50-413,414/96-05, plant tours revealed that plant material condition had improved consistent with the reduction in the backlog.

a. Inspection Scope (62703)

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

- WO 96045817 Replacement of solenoid 1CASV1500 associated with 1CA-150, which failed because of steam leak on 1CF-42
- WO 94062382 Component Cooling Water Pump 1B2 IWP (Model WO)
- WO 96011269 Steam Pressure Loop D Channel 4 Calibration
- WO 96035546 Analog Channel Operational Test (ACOT) Channel IV
- WO 96037800 ACOT for Refueling Water Storage Tank Level Channel IV

b. Observations and Findings

The inspectors observed portions of the work performed under these activities to verify that applicable procedures were available and being used by plant personnel. Where required, second person verification was obtained, and appropriate protective clothing and gear were used. Engineering personnel were present for portions of certain tests, demonstrating a responsive support function. Operations and maintenance staff were knowledgeable of the scope of the work tasks and were generally familiar with the procedures in use. Minor procedural changes caused short delays in work completion for one task.

The inspector's reviews and observations of additional maintenance and surveillance activities are documented in M1.2.

M1.2 Feedwater Isolation Valve Steam Leak (62703)

a. Inspection Scope

On May 24, the licensee identified a packing leak on the 1B steam generator Main Feedwater Isolation Valve (MFIV), 1CF-42. On June 5, during a check of 1CF-42 accumulator nitrogen pressure, maintenance personnel bled accumulator nitrogen pressure from 2775 psig to roughly 1700 psig, rendering the MFIV inoperable. Control Room Operators were not aware that the valve was inoperable and did not aggressively question the cause of a low nitrogen accumulator pressure computer alarm generated during the maintenance.

b. Observations and Findings

After the 1CF-42 packing leak was identified, operators increased the valve's surveillance frequency and consulted with Engineering for an operability determination. The licensee determined that there were no immediate operability concerns associated with 1CF-42 or other equipment in the vicinity of the packing leak. Over a period of roughly one week, the packing leak worsened and caused solenoid valve 1CASV1500, the A-train solenoid valve associated with the 1B steam generator main feedwater to auxiliary feedwater bypass valve (1CA-150), to fail (see section M1.1). Because 1CA-150 was needed for shutdown (below 15% power), the licensee began to develop a strategy for replacing the 1CA-150 solenoid as soon as practical.

On June 4, the inspector attended a Plant Operation Review Committee (PORC) Meeting which was held to approve engineering recommendations for actions regarding the packing leak on MFIV 1CF-42. The PORC concluded, among other things, that a check of the valve's actuator nitrogen accumulator pressure was desired.

On June 5, procedure IP/0/A/3010/009B, Nitrogen Charging for Main Feedwater Isolation Valve Actuators, was implemented to check the nitrogen pressure in the accumulator and adjust it if necessary.

After being notified that valve 1CF-42 had been backseated and the leakage was stopped, the inspector obtained a copy of IP/0/A/3010/09B, reviewed work order 95024430, and discussed the work that was performed with the technicians and operators involved.

During discussions with the inspector, the maintenance technician indicated that accumulator nitrogen pressure was checked and determined to be 2775 psig. IP/0/A/3010/09B, Enclosure 11.1, Procedure Sign-off and Calibration Data Sheet, states that the allowable range is the desired pressure +50 psig. The desired pressure at 90°F was 2725 psig. In accordance with step 10.2.24 of the procedure, nitrogen pressure was adjusted toward the desired pressure, and the steam leak decreased. The technician continued to bleed nitrogen pressure to approximately 1700 psig to backseat the valve. A computer alarm was received when pressure dropped to the setpoint range of 2050 to 2150 psig decreasing. (Step 10.2.24 of the procedure only allows adjustment of the pressure toward the desired pressure; reduction of the nitrogen pressure to backseat the valve was not directed by the procedure.) Nitrogen was recharged to the desired pressure of 2725 psig and the nitrogen charging procedure was completed.

During the pre-job discussion among the Work Control Center SRO, the Control Room SRO, and the maintenance technicians, the operators understood that the scope of the work to be performed by the maintenance crew was a check of nitrogen pressure and charging if needed; there were no plans to bleed nitrogen to backseat the valve and no control room alarms were to be expected. However, a computer alarm for low accumulator nitrogen pressure was received at 10:05 a.m. The maintenance technician working on MFIV 1CF-42 contacted the control room shortly after the alarm to inform operators that the maintenance activity had caused the alarm. The Control Room Operators did not exhibit a questioning attitude to assess if the scope of the work had changed or if nitrogen pressure had actually been lowered to less than the minimum pressure required for operability as indicated by the alarm.

The inspector questioned whether valve 1CF-42 was operable during the nitrogen bleed. The operators indicated that a pressure of less than 2050 psig rendered the valve inoperable. The inspector also questioned if the valve was operable during pressure testing. The control room operators indicated that they thought it was operable. Inoperability of the valve had not been entered into the Technical Specification Action Item Log. The inspector subsequently reviewed the maintenance procedure in detail. Step 10.2.7 of IP/0/A/3010/09B states that Step 10.2.8, removing the cap on the actuator fill valve to obtain a nitrogen pressure reading using the test rig, renders the valve inoperable. The valve is inoperable in this condition because the pressure test rig is not seismically qualified. Step 10.2.7 directs the technician performing the procedure to notify Operations that the MFIV will be inoperable with the performance of Step 10.2.8. The Work Control Center SRO was informed that MFIV 1CF-42 would be inoperable during the performance of the pressure check. However, this information was not communicated to the control room operators, who assumed that the valve was operable and available during the entire procedure.

After the inspector's questions about the use of IP/0/A/3010/09B to backseat the MFIV and stop the packing leak, the licensee initiated PIP 1-C96-1341 to document the issue and initiated a root cause investigation to identify the factors that contributed to the occurrence. The inspector reviewed the PIP and discussed the status of the root cause analysis with responsible personnel. The root cause evaluation was not complete at the end of the report period.

c. Conclusions

The reduction of the actuator nitrogen accumulator pressure below the desired pressure was not directed by the procedure. This issue was NRC-identified, and the inoperability of a MFIV without Control Room Operator knowledge with the unit at power is of more than minor safety significance. Therefore, this failure to follow procedure IP/0/A/3010/09B, Nitrogen Charging for Main Feedwater Isolation Valve Actuators, step 10.2.24 is characterized as a violation. VIO 413/96-08-02: Failure to Follow Procedure When Adjusting MFIV Nitrogen Accumulator Pressure to Backseat Leaking MFIV. In addition, this example revealed a lack of aggressive questioning on the part of control room operators to understand the cause and consequences of the low MFIV nitrogen accumulator pressure computer alarm.

M8 Miscellaneous Operations Issues (92902)

M8.1 (Closed) Violation 50-414/94-11-01: Procedural Errors and Inadequate Reviews of Power Range ACOTs

This violation was issued due to mathematical errors that occurred during the performance of power range ACOTs in April 1994 which were undetected by independent verification and supervisory review of the activity. The inspector reviewed NRC Inspection Report 50-413,414/94-11, which documented the violation, the licensee's response to the violation dated 6/30/94, and PIP 2-C94-0504. The inspector verified that the stated corrective actions were implemented. In addition, licensee programs to improve human performance have been implemented since the occurrence of this violation. Similar occurrences of inadequate independent verification and supervisory review have not been identified by the inspector. This violation is closed.

M8.2 <u>End of Cycle Control Rod Drop Testing</u>

a. <u>Inspection Scope (92902)</u>

NRC Bulletin 96-01, Control Rod Insertion Problems, reported that control rods had failed to fully insert in fuel assemblies with greater than 30,000 Megawatt Days/Metric Tonne Uranium (MWD/MTU) exposure and directed EOC control rod drop timing testing. The data provided by the licensee indicated that about 50% of the Unit 1 Cycle 9 fuel bundles would exceed 30,000 MWD/MTU exposure.

b. <u>Observations and Findings</u>

The inspector attended a Plant Operations Review Committee (PORC) meeting conducted to approve changes to the test procedure. Initially, the licensee planned to conduct this test while in Mode 2; however, plant configuration would not support this test in Mode 2 because a Feedwater Isolation (FWI) would occur. Consequently, the FWI would cause a loss of steam generator inventory and a potential for Auxiliary Feedwater (AFW) initiation if the FWI was not reset and main feedwater reestablished, complicating recovery from this test. The licensee planned to defeat the FWI signal by holding the FWI reset switches in the 'RESET' position while conducting the test. However, as identified by an earlier PORC review, FWI was required to be operable in Mode 2; therefore, holding the FWI reset switches in the 'RESET' position would result in defeating a valid ESF actuation signal. Because the FWI signal was not required in Mode 3, the licensee decided to conduct this test in Mode 3 by borating the control rods to fully withdrawn and establishing a sufficient shutdown margin.

Normally, the licensee conducted rod drop time testing by fully withdrawing a single control rod bank and opening the reactor trip breakers. Installed measurement and test equipment (M&TE) was capable of recording up to eight control rods simultaneously. However, for the EOC timing testing, the licensee used prototype equipment that permitted the simultaneous monitoring of all control rods. This prototype equipment was scheduled to be permanently installed at the McGuire Nuclear Station as a replacement for the Digital Rod Position Indication (DRPI) system. The inspector witnessed equipment installation and checkout including removal of DRPI channel 'A' from service. Channel 'A' provided raw rod position information to the test equipment. After the equipment checkout was satisfactorily completed, the DRPI 'A' channel was returned to service. To perform the timing test, reactor power was reduced to about 1%. Boron concentration was then increased and control rods withdrawn to maintain a constant reactor power level. After all control rods were fully withdrawn, boron concentration was increased to maintain the reactor in Mode 3 (k_{eff} < 0.99). Once Mode 3 conditions were established, DRPI channel 'A' was removed from service to provide raw rod position information. A digital input from the P-4 contact on the reactor trip breakers would trigger the prototype equipment to capture raw rod position information when the reactor was tripped. The raw rod position information would then be used to calculate individual control rod drop times and velocity profiles. However, due to the failure of the Channel 1 Vital Power Supply, the reactor tripped on June 13, 1996, before the prototype equipment was placed into service. Consequently, control rod drop time testing was performed utilizing the normal method described above.

c. Conclusions

The control rod drop timing data was not available at the conclusion of the inspection; therefore, the inspector will conduct an independent review of the data when the data is made available. Conduct of this review is identified as IFI 50-413/96-08-03: Review of EOC Control Rod Drop Timing Data.

III. Engineering

E1 Conduct of Engineering

E1.1 Monitoring and Trending of Shutdown Margin

a. Inspection Scope (61710)

The inspector reviewed the results of control rod worth testing conducted during Unit 2 Zero Power Physics Testing (ZPPT) on November 29, 1995. The licensee determined that the total reactivity worth of the reference bank, Shutdown Bank (SDB) 'B,' was 955 pcm. As reported in the licensee's report dated February 12, 1996, this was 11% greater than predicted by the core design report which exceeded the review criteria of 10%. Exceeding the review criteria did not represent a substantial deviation that would have resulted in core design reanalysis. However, because the reference bank total reactivity worth did exceed the review criteria, the inspector reviewed the trending of actual core reactivity against core reactivity predicted by the core design report. Additionally, because the remaining SDBs had total reactivity worths substantially less than SDB 'B,' the inspector questioned the effect on core performance of large SDB reactivity worth, specifically EOC shutdown margin (SDM) when boron concentration (C_B) was low.

b. Observations and Findings

The licensee recognized that EOC SDM requirements could prevent the withdrawal of SDB control rods prior to reactor restart. In this case, the licensee implemented additional procedural controls to prevent withdrawal of the SDB control rods until all prerequisite conditions for reactor restart were met. SDB control rods were then withdrawn as part of the normal control rod withdrawal sequence for reactor criticality.

To determine if the licensee was taking action to ensure adequate EOC SDM, the inspector reviewed the licensee's core physics trending program. The licensee indirectly monitored SDM by trending the results of the reactivity anomaly surveillance performed every 31 Effective Full Power Days (EFPD) as required by Technical Specifications (TS). Monitoring for reactivity anomaly would identify any potential for core reactivity to be greater than predicted at EOC resulting in reduced SDM. The inspector reviewed the core performance data and noted the trend of actual C_B was consistently lower than predicted C_B, indicating the core was less reactive than predicted. The same data was sent to corporate reactor design engineers for review. This provided for a second review of the data and allowed for earlier identification of potential problems that may indicate a need for core design reanalysis.

The core performance trending program was primarily based on TS requirements; however, there was no formal procedure for data collection or analysis of the data or the trend. The licensee was in the process of developing guidance that defined the parameters to be trended. This guidance only described data collection as a duty of a particular job position and did not provide formal controls over data collection and analysis of the data or the trend.

c. Conclusions

The inspector concluded that adequate controls existed to prevent a loss of TS required SDM during EOC reactor restarts. Although no formal trending program existed, the inspector also concluded that TS required trending for reactivity anomalies provided adequate monitoring of actual versus predicted core performance.

E2 Engineering Support of Facilities and Equipment

E2.1 Charging Pump Seal Cooling Flow Orifices

a. Inspection Scope (37551)

On March 20, 1996, McGuire Nuclear Station personnel discovered that the outboard flush adapter on the 1B charging pump was installed backwards. The concern was that this incorrect installation could adversely affect the thrust loading on the pump and outboard seal. The licensee Operation Experience Assessment group in the Duke Power Company General

Office determined that the charging pumps at Catawba had a similar seal design that incorporates flush lines. Since the concern was applicable to Catawba, PIP 0-C96-0695 was initiated to investigate and resolve the concern. The inspector reviewed the PIP and discussed the issue with licensee engineering personnel.

b. Observations and Findings

Catawba engineering personnel determined that the discharge-end adapter was mounted backwards in charging pump 1A. In charging pump 1B the discharge and suction-end adapters were switched, and the suction-end spacer was installed backwards. The cause for this improper adapter installation was attributed to inadequate procedural guidance for installing the flow adapters in their correct position and orientation.

Corrective action included a proposed procedure enhancement to MP/C/A/7150/16A, Centrifugal Charging Pump Corrective Maintenance, to provide adapter installation guidance. In addition, the licensee initiated and completed WOs 96025307-01 and 96025308-01 to correct the orientation of the improperly installed flow adapters. These corrective actions were documented in PIP 0-C96-0695.

The pump vendor, Ingersoll Dresser Pump Company, had been contacted and performed an operability evaluation. The inspector reviewed the report that the vendor provided, "Seal Injection and Balance Line Adaptor Flow Evaluation." The report indicates that both the 1A and the 1B charging pump seals had adequate cooling flow with the seal flow adapters installed as they were initially found. Therefore, the pumps had been capable of performing their intended function in the past.

c. Conclusions

The inspector concluded that the licensee's Operating Experience Assessment program effectively communicated this concern with generic applicability to Catawba. The operability evaluation was acceptable, and the proposed corrective actions were appropriate.

IV. Plant Support

S2 Status of Security Facilities and Equipment

S2.1 Standby Shutdown System Outside Design Basis

a. Inspection Scope (71750)

On May 6, the licensee declared the Standby Shutdown System inoperable and entered TS 3.7.13, Standby Shutdown System. The system was considered inoperable as the result of an evaluation of its design basis which indicated that the volume of non-safety, non-condensate quality feedwater available to supply the steam generators during security events was outside the design basis. The inspector reviewed the licensee's initial actions in response to the condition. In addition, the adequacy of the licensee's compensatory actions were assessed by a security specialist inspector as documented in NRC Inspection Report 50-413,414/96-09.

b. Observations and Findings

The Standby Shutdown System is utilized to combat Appendix R (remote shutdown), security, and loss of all AC power scenarios. The system provides alternate means for 1) limited reactor coolant system inventory makeup, 2) steam generator auxiliary feedwater, and 3) steam generator pressure control for heat removal. The licensee identified that the assumed available volume of water used for an alternate means of steam generator auxiliary feedwater was based on a calculation with assumptions which were not valid for that purpose. Upon this discovery, the licensee performed rough calculations which indicated that a sufficient volume for operation of auxiliary feedwater for the full duration stated in the design basis response to some security scenarios was not available. The Standby Shutdown System was declared inoperable and TS 3.7.13 was appropriately entered. The TS action was exited following implementation of actions to compensate for the identified vulnerability. The licensee submitted an LER regarding this issue. Long-term actions will be assessed by the NRC during a closeout inspection of the LER.

c. Conclusions

Immediate and interim licensee actions to address this design basis deficiency were appropriate. Long-term corrective actions will be assessed during a closeout inspection of the Licensee Event Report.

V. Management Meetings

X1 Exit Meeting Summary

The inspectors presented the inspection results to members of licensee management at the conclusion of the inspection on June 20, 1996. The licensee acknowledged the findings presented. No proprietary information was identified.

PARTIAL LIST OF PERSONS CONTACTED

Licensee

Addis, B., Training Manager
Bhatnager, A., Operations Superintendent
Coy, S., Radiation Protection Manager
Forbes, J., Engineering Manager
Funderburk, W., Work Control Superintendent
Harrall, T., IAE Maintenance Superintendent
Kelly, C., Maintenance Manager
Kimball, D., Safety Review Group Manager
Kitlan, M., Regulatory Compliance Manager
Lowery, J., Compliance Specialist
McCollum, W., Catawba Site Vice-President
Nicholson, K., Compliance Specialist
Patrick, M., Safety Assurance Manager
Peterson, G., Station Manager
Propst, R., Chemistry Manager
Rogers, D., Mechanical Maintenance Manager
Tower, D., Compliance Engineer

INSPECTION PROCEDURES USED

IP 37551:	Onsite Engineering
IP 40500:	Effectiveness of Licensee Controls in Identifying, Resolving, and Preventing Problems
IP 61710:	Control Rod Worth Measurements for Pressurized Water Reactors
IP 62703:	Maintenance Observation
IP 71707:	Plant Operations
IP 92901:	Followup - Operations
IP 92902:	Followup - Maintenance

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

50-413/96-08-01	NCV	Inadequate Removal and Restoration Procedure for Draining Fuel Pool Cooling System Piping (Section O1.4)
50-413/96-08-02	VIO	Failure to Follow Procedure when Adjusting MFIV Accumulator Nitrogen Pressure (Section M1.2)
50-413/96-08-03	IFI	Review of EOC Control Rod Drop Timing Data (Section M8.2)

Closed

50-413/94-03	LER	Manual Reactor Trip on Loss of Normal Feedwater Due to System Performance (Section O8.1)
50-414/94-11-01	VIO	Procedural Errors and Inadequate Reviews of Power Range ACOTs (Section M8.1)

LIST OF ACRONYMS USED

ACOT - Analog Channel Operational Test
AFW - Auxiliary Feedwater
CB - Boron concentration
CFR - Code of Federal Regulations
CNS - Catawba Nuclear Station
DG - Diesel Generator
DRPI - Digital Rod Position Indication
EFPD - Effective Full Power Days
EOC - End-of-Cycle
ESF - Engineered Safety Feature
FSAR - Final Safety Analysis Report
FWI - Feedwater Isolation
I&C - Instrument and Control
IAE - Instrument and Electrical
IFI - Inspector Followup Item
LER - Licensee Event Report
MFIV - Main Feedwater Isolation Valve
M&TE - Measurement and Test Equipment
MWD/MTU - Megawatt Days/Metric Tonne Uranium
pcm - percent millirho
PIP - Problem Investigation Process
PORC - Plant Operations Review Committee
PORV - Power Operated Relief Valve
psig - pounds per square inch gauge
SDB - Shutdown Bank
SDM - Shutdown Margin
SRO - Senior Reactor Operator
Tave - Average Reactor Coolant System Temperature
TS - Technical Specifications
WO - Work Order
ZPPT - Zero Power Physics Testing