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REGION II

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Report No: 50-369/96-10, 50-370/96-10

Licensee: Duke Power Company

Facility: McGuire Generating Station, Units 1 & 2

Location: 12700 Hagers Ferry Rd.
Huntersville, NC 28078

Dates: October 20 - November 30, 1996

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ENCLOSURE

EXECUTIVE SUMMARY

McGuire Generating Station, Units 1 & 2
NRC Inspection Report 50-369/96-10, 50-370/96-10

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

Operations

- Control of the dual unit TS required shutdowns due to EVCC battery capacity concerns was considered good. Operators maintained adequate oversight of the units during the shutdown conditions and provided good monitoring of critical parameters during unit restarts. Management oversight of the activities was noted to be strong. (paragraph 02.1)

- Restart activities were accomplished in a professional manner. Management oversight and operational criticality briefings were well implemented (paragraph 02.3)

- Operator and engineering actions taken as a result of a SG overfill event were adequate. The event was caused by a leakage through an AFW isolation valve. (paragraph 02.4)

- Reviews in the area of freeze protection were mixed. Corrective actions for several previously identified programmatic problems were noted to be broad; however, several other specific deficiencies were identified. A Unresolved Item (URI) was identified regarding deficiencies in design control of heaters used to protect safety-related FWST transmitters. Other design control problems were identified on heaters protecting non-safety related transmitters. (paragraph 03.1)

- Reviews of operation training flow loops from the simulator concluded it was a beneficial tool to reduce human error. (paragraph 05.1)

- Inspection observations of a Nuclear Safety Review Board (NSRB) meeting indicated that the NSRB was providing good oversight of the facilities operation. (paragraph 07.1)

- A weakness was identified regarding delayed entry into an Abnormal Operating Procedure for malfunctioning power range instrumentation. (paragraph 08.1)

Maintenance

- Licensee identification of system leakage and weld indications was an example of good attention to detail during implementation of the fluid leak management program. (paragraphs M2.1 and M2.2)

- Vital battery testing evolutions were improved over previous performance. Immediate corrective actions for the identified battery capacity problems were conservative and provided adequate justification for equipment operability. (paragraph M2.2)

- The failure of a Unit 1 MF isolation valve resulted in operators initiating a rapid downpower to avoid a safety system challenge. (paragraph M2.3)

- A Non-Cited Violation was identified regarding a failure to perform Unit 1 TS required containment integrity surveillance testing prior to entering MODE 4. (paragraph M3.1)

- Reviews were performed regarding hydrogen analyzer calibration ranges. The inspectors concluded that the licensee's calibration methods were adequate. (paragraph M3.2)

- Review of licensee's program for risk assessment of maintenance activities concluded it was a valuable tool for identifying potential risks. The process was also considered to be well implemented. (paragraph M4.1)

Engineering

- Reviews of problem investigation reports involving potential water hammer events concluded that the station threshold for identifying these instances was being lowered. Evaluations of the described issues were considered adequate. Inspector walkdowns identified possible indications of a water hammer. The inspectors questioned the documentation and adequacy of 1994 reviews which evaluated the previously identified water hammer deformation on the SG Blowdown system. (paragraph E2.1)

- A review of site engineering indicated that engineering was aggressively tracking backlogs, support to operations and maintenance had improved, and emergent work was well supported. (paragraph E2.2)

- A Non-Cited Violation was identified for failure to meet TS 2.2.1 requirements for RCS Loop A Channel II and III trip setpoints. (paragraph E3.1)

- The licensee's review of activities associated with the Spent Fuel Pool (SFP) area painting project were thorough; however, documentation of the review could have been more formalized. This is identified as an URI pending further inspection in this area. (paragraph E4.1)

- An Unresolved Item was identified regarding operability of the control room pressure envelope to support non-related system testing. (paragraph E4.2)

- Reviews concluded that the engineering department was performing effective self assessments and that their findings were similar to those of the Nuclear Assessment and Issues Division, Regulatory Audit Group. (paragraph E7.1)

Plant Support

- During routine tours of the station, the inspectors noted good radiation protection and security controls for ongoing maintenance activities throughout the station. (paragraph R1.1)

Report Details

Summary of Plant Status

Unit 1 began the inspection period at 100 percent power. On October 30, a TS required shutdown was initiated after 125 VDC Vital Power system battery EVCC failed to meet required capacity criteria during a modified performance test. On November 10, the unit was restarted and returned to full power after satisfactory battery cell replacement and testing was accomplished. The unit operated at 100 percent power until November 27, when power was reduced to address hydraulic control problems on feedwater containment isolation valve 1CF26. At the end of the report period, the unit remained at approximately 28 percent power due to the feedwater valve problems.

Unit 2 began the inspection period at 100 percent power. On October 31, a TS required shutdown was initiated after 125 VDC Vital Power System battery EVCC failed to meet the required capacity criteria during a modified performance test. On November 11, the unit was restarted and returned to approximately 100 percent power. The unit operated at approximately 100 percent power for the remainder of the inspection period.

Review of UFSAR Commitments

While performing inspections discussed in this report, the inspectors reviewed the applicable portions of the UFSAR that were related to the areas inspected. The inspectors verified that the UFSAR wording was consistent with the observed plant practices, procedures, and/or parameters. As addressed in section M3.2, a potential discrepancy involving hydrogen analyzer calibration ranges was identified.

I. Operations

O1 Conduct of Operations

O1.1 General Comments (71707)

Using Inspection Procedure 71707, the inspectors conducted frequent reviews of ongoing plant operations. The overall conduct of operations was professional and safety-conscious; specific events and noteworthy observations are detailed in the sections below.

O2 Operational Status of Facilities and Equipment (71707)

O2.1 TS Required Dual Unit Shutdown

a. Inspection Scope (71707)

Beginning October 30, the licensee commenced a controlled shutdown of both units following failure of 125 VDC Vital Power System Battery EVCC to meet TS required capacity during a modified performance test (further discussed in Section M2.2). The licensee staggered the unit shutdowns

to minimize activities in the control room during the evolution. Both units were brought to cold shutdown. No TS time constraints were exceeded during the shutdown of the units.

b. Observations

The inspectors witnessed portions of the dual unit shutdown to cold shutdown. The inspectors noted several complications due to equipment failures. Two ESF actuations occurred. The auxiliary electric boiler was lost which was providing steam to the Unit 1 feedwater pump turbine. The operators manually actuated the Unit 1 auxiliary feedwater pumps to stabilize steam generator levels. The normal Unit 2 offsite power supply breaker tripped for unknown reasons causing a loss of power to the 2ETB vital bus. An autostart and sequencing of the 2B emergency diesel generator resulted. Both of the ESF actuations will be addressed by the licensee via LERs. Other equipment problems with control rod bank overlap setpoints and reactor coolant pump seal differential pressure transmitter root valves were also experienced. Licensed operators responded to the equipment malfunctions promptly by referring to and/or entering the applicable abnormal procedures. No significant increase in the 1B steam generator tube leakage occurred as a result of the system parameter changes associated with the Unit 1 shutdown and restart.

c. Conclusions

Although the unexpected equipment problems caused complications, overall operator response to equipment problems during the dual unit shutdown was good. Appropriate notifications of the ESF actuations were made. The inspectors concluded that operator's control of the shutdowns was good. Operators demonstrated their ability to conduct safe, orderly unit shutdowns despite equipment and other associated problems. The successful evolutions were also attributed to appropriate management oversight during the planning and execution of the shutdowns.

O2.2 Unit 2 Lower Containment Cleanliness Walkdown

a. Inspection Scope (71707)

Prior to restart of Unit 2 after the forced outage to replace the 125 VDC vital battery EVCC, the inspectors conducted walkdowns of accessible containment areas to evaluate containment cleanliness and housekeeping practices. Since containment access was limited during the forced outage, the inspectors focused primarily on the lower containment pipechase area of Unit 2. Several major activities performed during the forced outage involved equipment located in the pipechase.

b. Observations and Findings

The inspectors examined the areas near letdown orifice isolation valve 2NV458 and the service water to reactor building non-essential header containment inside isolation valve, 2RN276. The inspection survey was conducted to ensure that material controls and containment cleanliness expectations were satisfied following repair activities to correct fluid leaks identified during the shutdown period. The inspectors also verified that seismic supports and hangers were reinstalled following the maintenance activities. The inspectors noted that the areas near the maintenance activities were controlled in accordance with cleanliness requirements of McGuire Site Directive 585. The inspectors noted no loose equipment or materials that could adversely effect safety system operability.

During the observation, the inspectors noted indications of active leakage at the pipe cap downstream of primary system vent valve 2NI453. After exiting the containment, the inspectors verified that the licensee was aware of the condition and had scheduled repair during the upcoming Unit 2 EOC11 outage. No other active system leaks were identified.

c. Conclusions

The inspectors concluded that the identification of nonconforming conditions and the control of materials within containment during the forced outage was good, minimizing the likelihood of adversely impacting safety system performance during expected operational and accident conditions.

O2.3 Control of Criticality Evolutions

a. Inspection Scope (71707)

During the inspection period, the inspector witnessed portions of the Unit 1 and Unit 2 restart evolutions.

b. Observations and Findings

One of the evolutions witnessed included the criticality of Unit 1 on November 10. The inspectors focused on overall control of the evolution, operator awareness of plant parameters, interactions between operators involved in the restart, and reactor engineering personnel monitoring criticality status. The inspectors noted good communications between operators and reactor engineering personnel discussing criticality progress. The inspectors noted a large number of personnel in the common Unit 1 and 2 horseshoe area; however, the inspectors did not consider that this number adversely impacted the restart evolution. During the inspector's observation, operators appeared to be well informed and in control of changing plant parameters.

The inspectors also reviewed the SOER 91-01 pre-job briefing package for the criticality evolution. NSD 304, Reactivity Management, requires that reactor startups be treated as an infrequently performed evolution. The purpose of the briefing was to discuss with operators and other involved personnel how the approach to criticality and withdrawal of the control rods was going to be controlled and performed. Emphasis was placed on what parameters the operator at the controls should monitor and the frequency for monitoring. Additionally, expected values were included in the briefing based on reactor engineering 1/M extrapolation. Command and control functions were well established. In addition, the briefing included discussions on low power events at other stations to heighten operator awareness to these potential problems.

c. Conclusions

The inspectors concluded that the startup evolutions, including Unit 1 criticality, were well controlled and accomplished in a professional manner. Although numerous personnel were in the control room at times, this did not appear to adversely impact the restart evolutions. Briefing packages for the evolutions were detailed and highlighted specific items to help focus operators on conducting safe plant manipulations. Operations and Engineering management oversight of the evolutions was evident, specifically OSM involvement with the operator at the controls.

O2.4 Overfill of Steam Generator during Layup

a. Inspection Scope (71707)

On November 6, 1996, the licensee identified that the 1B SG had been overfilled while in MODE 5 wet layup conditions. The inspector reviewed PIP 1-M96-3185 which documented the circumstances where the 1B SG was apparently overfilled.

b. Observations and Findings

During the unit shutdown, all of the SG's were placed in wet layup on the secondary side which required filling the SG's above the normal 100% wide range level indication. The problem was discovered while attempting to drain the 1B SG. The operator identified that the SG would not drain completely with the PORV open for a vent path. This likely occurred due to a loop seal being formed in the steam line due to the over fill event. Immediate corrective actions for the problem included opening the 1B SG drain lines to the condenser to drain any water overfilled to the steam lines. Shortly after the drains were opened, operators verified the 1B SG could be drained. Investigation into the cause of the apparent overfill determined that during the layup condition, 1CF-127 (mainfeed water supply valve to the upper nozzle) had

leakage through the seat. The licensee considered this condition to be acceptable during operation based on the requirement of the valve to open for AFW initiation. The licensee is evaluating repair of this condition.

Additional corrective actions included opening of additional main steam line drains to drain any water to prevent potential water hammer conditions. Civil engineering also performed reviews and piping walkdown to assess any loading problems on the affected main steam lines. The inspector reviewed the corrective actions for the event with engineering personnel and conducted walkdowns of the affected areas to assess the scope of the licensee's actions. No other problems were identified. The licensee performed heatup of the unit with some lines open to maximize evaporation/draining of condensate in the main steam lines. The unit was restarted without any further adverse effects from the condensate carryover to the main steam lines.

c. Conclusions

The inspectors concluded that the operators took appropriate actions to address the abnormal condition once identified. The licensee was also reviewing procedures to identify future monitoring techniques to prevent similar events.

O3 Operations Procedures and Documentation

O3.1 Freeze Protection Program Review (71714)

a. Inspection Scope (71714)

The inspector conducted a review of the licensees' freeze protection program. The inspection was generally conducted in accordance with inspection module 71714, Cold Weather Preparations, and included reviews of corrective actions for LER 50-370/96-01 and response to Escalated Enforcement Violation 50-369,370/96-02-01 (EA 96-80).

b. Findings and Observations

Review of Current Program and Corrective Actions for Previous Problems

The inspector reviewed the licensees corrective actions for deficiencies identified in the cold weather protection program following an event in February 1996 that resulted in the inoperability of safety and non-safety FWST level transmitters.

Following the February 1996 event, the licensee formed a site task force to develop and implement comprehensive corrective actions and to review cold weather preparation activities at the station. The licensee later upgraded the task force to a multi-station cold weather protection team that was chartered to conduct an extensive and comprehensive self

assessment of the adequacy of the cold weather protection program. The team identified and proposed corrective actions for several significant deficiencies in the stations cold weather protection program.

The inspector noted that the licensee had extensively revised both operations and maintenance cold weather procedures. The procedure changes included revising FWST Level instrument calibration and functional test procedures to ensure that the enclosure heater thermostats were properly set. The inspector also noted that the enclosure heaters were added to the annual maintenance cold weather protection procedure to verify that enclosure thermostats were properly set following FWST transmitter calibration and inspection. The inspectors verified that the both the operations and maintenance cold weather protection procedures were being scheduled as a periodic surveillance to ensure completion prior to the start of the cold weather period.

The inspector conducted walkdowns of the refueling water (FW) and Boron Recycle (NB) systems with the respective system engineers to determine if previous problems had been corrected. The inspectors verified that previously missing portions of heat trace and insulation had been added to safety-related FWST impulse lines. The inspectors verified that portions of the FWST impulse lines heat tracing were no longer supplied power by a temporary cord. The inspector verified that the licensee had performed modifications to the FWST enclosures that included installing an additional thermostat to provide a low temperature alarm. In addition, external temperature indicators were added to the safety enclosures. Degraded insulation in the FWST enclosures was also repaired. The inspector noted that the FWST enclosures had not been replaced. Problem Investigation Report 0-M-95-1891 documented the material condition of the enclosures as degraded and that the enclosures although sealed with RTV were not well suited for the current application possibly allowing moisture and cold air to enter potentially affecting transmitter performance. The licensee has proposed a modification to replace the enclosures.

The inspector noted that the system engineers had performed overall comprehensive cold weather related walkdowns of several systems that included the FW and NB systems. Several discrepancies were identified and corrected. One design deficiency was identified in that both the primary and secondary heat tracing for safety-related FWST level transmitter impulse tubing was powered from a single power source and breaker. The failure of the power source or breaker could have resulted in the freezing of the impulse lines. The immediate corrective action included verifying the breaker position during the implementation of the on-demand and monthly freeze protection procedures. However, at the time of the inspector's review, neither the operations on-demand nor the monthly maintenance procedures had been fully developed and implemented. Another design deficiency identified by the licensee showed that FWST vent lines could potentially be blocked by ice and condensation.

During the previous investigation of the inoperability of safety and non-safety FWST level transmitters, engineering drawings for thermostat heater settings were determined to be inconsistent. As a result, the licensee implemented a self assessment to review the adequacy of setpoint documentation for non-safety related support systems. The license determined that the documentation for non-safety enclosure heaters was less than adequate. The licensee corrected the identified discrepancies.
The inspector noted that some cold weather protection activities were not implemented in a timely manner. Previous concerns were identified in PIP 1-M96-0643 concerning the importance of curtains in maintaining appropriate temperatures within the steam valve vaults. During the current period, painting in the exterior steam valve vault prevented the cold weather protection curtains from being lowered prior to freezing conditions. Compensatory measures such as lowering of curtains into position during off hours while temperatures were below freezing were not immediately considered. However, the inspector acknowledged that a computer alarm would alert control room operators to low mainsteam valve vault temperatures. Problem Investigation Report 1-M96-0643, showed that on a previous occasion instrumentation located in the mainsteam vault had been significantly affected by cold weather requiring the installation of these curtains.

During the current inspection period, the licensee received alarms indicating a low temperature in an FWST enclosure. Investigations revealed that the alarms were valid indicating the heater thermostat had failed to properly control enclosure temperatures at the desired setting. Further investigation by the licensee revealed that the thermostat setpoint had drifted resulting in the heaters being energized at a lower setting. A problem investigation report reviewed by the inspector revealed that the thermostats had demonstrated some unreliability in the past in maintaining desired setpoints. In addition, complicating the proper setting of the thermostat was that the associated temperature dial indicator did not correlate to the actual temperature setting. The licensee promptly replaced and functionally verified the thermostats.

The inspector reviewed the cold weather procedure implemented by the stations operations group and noted a reference to the Emergency Freeze Protection Kit. The Emergency Freeze Protection Kit contains equipment that can be used to prevent important plant equipment from freezing during a cold weather emergency. The inspector noted some deficiencies in the storage and inventory of associated equipment. In addition, the inspector noted a lack of awareness of the location and content of the Emergency Freeze Protection Kit. The licensee acknowledged and documented the inspectors concerns in the appropriate station corrective action documents.

The inspector noted that station engineering conducted a comprehensive review of instrumentation that is normally exposed to cold weather conditions to ensure adequate freeze protection. Other corrective actions included enhancement of existing operator round sheets, work control backlog review for cold weather protection maintenance, and development and scheduling of additional modifications to replace the existing safety and non-safety enclosures. The inspectors also noted significant management oversite and coordination of cold weather protection activities as evidenced by weekly updates to senior plant management during morning meetings. The inspector noted that the assignment and coordination and review of cold weather preparation activities by designated management and engineering representative was good.

Review of FWST Design Controls

The inspector questioned the licensee concerning the type and capacity of the existing FWST enclosure heaters. Engineering personnel confirmed that the installed heaters neither matched the current drawings nor the bill of materials. As a result, the licensee evaluated the heaters in both the non-safety and safety enclosures and determined that heater capacity was significantly greater that the amount currently documented on engineering drawings. The inspectors further questioned whether the installed heater capacity could damage both safety and non-safety transmitters if the thermostats were to malfunction and cause excessive enclosure temperatures.

The licensee evaluated the inspectors concern and determined that if the thermostats were to malfunction causing the heaters to stay energized, that the transmitter enclosures would significantly exceed design temperature limits for both the safety and non-safety transmitters. In addition, the high temperatures could adversely affect uninsulated sensing lines within the enclosures. The licensee implemented immediate compensatory measures which included monitoring of the FWST enclosure temperatures. In addition, further investigation by the inspector determined that the installed enclosure heaters for the non-safety related RMWST level transmitters were also not consistent with the engineering drawings. These level transmitters are not required to mitigate an accident.

As a result, the licensee promptly implemented a minor modification to replaced the FWST enclosure heaters with ones consistent with the engineering drawings. However, these heaters (50 watts) were later determined to be potentially inadequate under worst case cold weather conditions and were promptly replaced with heaters of a higher capacity.

According to the licensee's PIP, excessive enclosure temperatures could cause the non-safety related and safety related level transmitters to become unreliable or fail inhibiting the performance of TS surveillances and potentially degrade the cold leg recirculation swapover safety function.

c. Conclusions

The inspectors concluded that the licensees performance of actions to correct identified deficiencies in the stations cold weather preparation to date were generally good. A significant number of corrective actions were comprehensive and implemented in a timely manner. The inspector concluded that the licensees self assessment was broad, detailed and comprehensive in identifying and correcting some significant deficiencies with the stations cold weather protection programs.

However, some corrective actions were not implemented in a timely manner. These included the development and implementation of on demand and monthly cold weather protection procedures, replacement of existing thermostats that had previously demonstrated some drift in setpoint, and compensatory measures for some impairments. In addition, some design deficiencies that could impact plant operation due to freezing of FWST instrumentation remained uncorrected at the end of the inspection period.

In addition to the above, the inspectors concluded that despite significant improvements made in the administration of the freeze protection program, deficiencies continued to exist with the FWST enclosure heaters, until NRC reviews focused attention on the issues. Specifically, the installed safety and non-safety enclosure heaters were not consistent with current engineering drawings and were determined to be the wrong type and significantly higher capacity. In addition, it was identified that the original 50 watt heater elements described in the design may not have prevented instrumentation freeze events for worst case conditions.

Based on the design adequacy and implementation problems, the inspector reviewed the safety significance of the issue. A postulated failure of thermostats associated with the heaters could result in overheating of the transmitters to temperatures in excess of the manufacturers rating. The transmitters could become degraded and fail. Additionally, some failure modes may not be detected by the operators. The transmitters are used to document TS required FWST level and temperature surveillance data and to actuate safety features. Pending additional review, this issue will be identified as Unresolved Item (URI) 50-369,370/96-10-01, Failure to Ensure Installation of Correct Heaters in FWST Enclosures.

O5 Operator Training and Qualification

O5.1 Operations Mini Flow Loop Training

a. Inspection Scope 71707

The inspector reviewed the licensees use of the miniflow loop simulator for training of non-licensed operators and other plant personnel on concepts designed to reduce human error in normal and abnormal operating conditions.

b. Observations and Findings

The inspector noted the miniflow loop training simulator consisted of fluid loops that contain components represented in the NC, NV and NB systems. The inspector noted that the simulator was used to reinforce fundamental concepts to reduce human error such as STAR, use of phonetics, procedure usage and 3 way communication. Instructors were required to closely monitor and provide a detailed evaluation of an NLO implementation of time critical evolutions contained in station abnormal and emergency operating procedures. The detailed evaluations conducted by the instructors included monitoring of plant response, STAR and self checking, EP/AP rules usage, correct implementation of procedure steps, teamwork and communications.

c. Conclusions

The inspector concluded that the use of the miniflow to reinforce human performance fundamentals such as STAR, teamwork, communications and procedure usage should contribute to a reduction in human performance errors during operations.

O7 Quality Assurance in Operations (40500)

O7.1 Nuclear Safety Review Board (NSRB) (40500)

a. Observations and Findings

On November 20, 1996, the inspector attended a McGuire NSRB meeting. Site presentations to the board included plant performance, reportable events, violations, trends, areas for improvement and other relative issues. The inspectors considered that the information presented to the NSRB gave a realistic view of overall plant performance. In addition to the above, the NSRB conducted a self assessment review and discussion to identify areas where an increased benefit to the plant may be attainable. Numerous proposals for improved performance were suggested and documented for resolution.

b. Conclusions

The inspector concluded that the NSRB members provided good insights to plant management for potential improvements and conducted a objective self assessment of their function to maximize their benefit to the plant.

O8 Miscellaneous Operations Issues (92700)

O8.1 (CLOSED) URI 50-369,370/96-08-01: Operator Abnormal Procedure Usage

On September 30, 1996, the Unit 2 nuclear power range channel NIS41 was declared inoperable due to the failure of a -25 VDC low voltage power supply. The inspectors reviewed log entries and noted that the control room operators received several Power Range HI Voltage Failure annunciator alarms. The inspector noted that the control room operators did not enter the AP-16, Malfunction of Nuclear Instrumentation, Case III, Power Range Malfunction, since the power range instrumentation indication was not erratic and the apparent failure of the power supply appeared to have no impact on normal plant operations. The inspectors noted that operations management procedures stated that abnormal procedures should be implemented upon recognition of entry conditions listed in the appropriate abnormal procedure. However, Operations management considered that entry should also be based on an evaluation of symptoms referenced in the respective procedure.

Based on the inspectors observations, the decision not to enter abnormal procedures delayed the execution of operator actions and complicated repair efforts to the nuclear power range instrumentation. This contributed to the plant being placed within several minutes of a TS required shutdown. Additional research by the licensee indicated the failure of the nuclear instrumentation low voltage power supply can cause instances where the reactor trip bistable setpoints exceed TS limits.
The inspector concluded after additional discussions with station operations management and NRC regional inspectors that not entering the abnormal procedure under these conditions was a weakness. Operations management indicated that management expectations were not met in this case. The low voltage power supply failure is being reviewed by the licensee to evaluate power range instrumentation operability. The inspectors are evaluating this condition to determine generic implications. The URI is closed.

O8.2 (Closed) Violation 50-369, 370/95-23-03: Hydrogen Recombiner Procedures not Adequately Maintained.

This violation addressed two issues with the hydrogen recombiner system. (1) The temperature indications were failing and accurate readings could not be obtained. (2) The test procedure PT/1/A/44550/04A did not verify the combiner reference junction temperature in the required band.

Regarding issue (1) the licensee implemented Minor Modifications (MM-7807 and 7808). These Minor Modifications replaced the obsolete temperature indications with new instrumentation which will allow remote reference temperature compensation. The inspector verified the new instrumentation was installed and operable.

Regarding issue (2) the licensee revised PT/1,2/A/4450/04A Hydrogen Recombiner Operability Test, and EP1,2/A/500/G-1, Placing Hydrogen Recombiners in Service to remove the steps concerning the reference junction temperature. As noted in issue (1) above the reference junction temperature is compensated for by the new instrumentation. The inspector reviewed the above procedures dated 2/8/96 change 8, and Rev. 2 and verified that the above noted changes had been made. This item is closed.

O8.3 (CLOSED) LER 370/96-01: RWST Level Instrumentation Past Inoperable (Frozen Instrument Lines). This LER is closed. Corrective actions associated with this LER will be tracked under violation EA-96-80.

II. Maintenance

M1 Conduct of Maintenance

M1.1 General Comments (61726 and 62707)

The inspectors witnessed selected surveillance tests to verify that approved procedures were available and in use, test equipment in use was calibrated, test prerequisites were met, system restoration was completed, and acceptance criteria were met. In addition, the resident inspectors reviewed and/or witnessed routine maintenance activities to verify, where applicable, that approved procedures were available and in use, prerequisites were met, equipment restoration was completed, and maintenance results were adequate.

a. Inspection Scope

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

Work Order/Procedure	Title
• PT/0/A/4600/078	RCCA Drop Timing Using Rod Position Grey Code
• IP/1/A/3000/22B	Reactor Coolant System Flow Calibration Loop A, Protection Channel II
• PT/2/A/4150/13	Reactor Coolant Flow Calibration
• PT/0/A/4700/62	Daily Surveillance of Reactor Building Entries

M2 Maintenance and Material Condition of Facilities and Equipment

M2.1 Weld Indications At 2RN276 and 2NV458

a. Inspection Scope (62703)

During the forced dual unit outage, the licensee identified and corrected components exhibiting leakage. The inspector monitored the repairs made and evaluated the significance of the individual problems as follows.

b. Observations and Findings

Weld Indication at Valve 2RN-276

On November 5, with Unit 2 in MODE 5 (cold shutdown), the licensee identified a potential crack in a weld at valve 2RN-276A. The valve is the inside containment isolation valve for penetration 2M-315 for the RN non-essential header return from the RCP motor air coolers. The indication was identified via walkdowns being performed as part of a fluid leak management program during the forced Unit 2 outage. The indication (approximately 2 inches in length) was on the valve body (schedule 10 stainless) to pipe (schedule 40 carbon steel) weld on the containment side of the valve and was exhibiting slight leakage. Upon discovery, a Type C, as-found, local leak rate test was satisfactorily performed (zero penetration leakage) as well as re-verification that no previous leakage had been detected during previous outage tests. Radiographic testing indicated an unsatisfactory weld at the Class B Schedule 40 to Schedule 10 interface on the containment side of the valve. Ultrasonic testing was also performed and no other cracks were identified prior to removal of the valve for repair.

Licensee valve history research concluded that although the valve was specified to be procured as Schedule 40, it was manufactured schedule 10 and accepted as such. A review of other associated RN system valves was performed to identify similar material applications; however, no similar applications were identified. The licensee concluded that the as-found installation, although acceptable, was not common and therefore was not a generic concern. The affected weld area was removed and metallurgically analyzed; however, a specific failure mechanism could not be identified. The most likely failure mechanism was determined to be stress corrosion cracking.

The inspectors monitored repairs to the valve which included the addition of a schedule 40 stainless steel spool piece being shop welded to the original schedule 10 valve body (Class B). This was changed to give added assurance that containment penetration integrity will be maintained. The Class F boundary side of 2RN-276A was ground out and field welded back as originally installed. The licensee concluded that this repair approach was satisfactory based on the many years it took

the crack to develop and that the weld will be re-inspected and evaluated during the 2EOC11 refueling outage.

Repair of Weld Indications At Letdown Orifice Isolation Valve 2NV458

Observations and Findings

On October 31, the licensee identified a leaking welded joint upstream of letdown orifice isolation valve 2NV458. The licensee determined that the crack initiated at the top of a socket weld and migrated into the adjacent piping. Unit 2 was in Mode 5 at the time of discovery. The licensee had noted increased leakage during the operating cycle of approximately 15 gallons per minute (gpm). Although water hammer events had previously occurred in this portion of the chemical volume and control (NV) system, no damage to hangers or weld cracks indicative of a water hammer were identified.

The licensee repaired the affected area, and replaced similar socket welded joints located on the 75 gpm orifice letdown line. The letdown orifice isolation valve was also removed for repairs. Upon inspection of the valve, the licensee noted steam cutting of the plug and seat ring. Metallurgical analysis of the weld and piping confirmed that the weld crack was due to vibration induced fatigue. The licensee reviewed operational data and determined that higher than expected vibration (approximately 3 times the generally accepted limit) levels are incurred when the 75 gpm orifice is in service.

The licensee performed additional inspections of other Unit 1 and Unit 2 socket welds with no defects identified. Prior to startup of either unit, the licensee performed vibration monitoring of the letdown orifice isolation lines to obtain additional data to better understand the operating conditions. Following the vibration testing, licensee engineering recommended operating Unit 2 with the 75 gpm orifice isolated to reduce vibration levels thereby reducing the likelihood of additional vibration induced weld failures.

The inspectors performed visual inspection of the repaired piping and valve. Seismic supports were also inspected. The inspectors did not identify any indications of hangar or seismic support damage indicative of a significant water hammer event. The licensee has also identified long term repair actions to replace letdown orifice isolation valves 2NV458, 1NV457, and 2NV457 during the upcoming Unit 1 and Unit 2 outages. A modification will be completed to replace the currently installed valve design.

c. Conclusions

The inspectors concluded that the licensee's identification of the system leaks was an example of good attention to detail during implementation of the fluid leak management program. The inspectors

reviewed the repair process and post-maintenance testing, which included type C local leak rate testing of 2RN276 and valve stroke timing of both 2RN276 and 2NV458, and concluded that it was adequate.

M2.2 125 VDC Vital Battery Modified Performance Test

a. Inspection Scope (61726)

On October 28, the licensee conducted a TS 4.8.1.2e required 60 month surveillance test of the 125 VDC Vital Power System Battery Bank EVCC. The batteries being tested were AT&T high specific gravity lead-acid round cell type batteries. In an effort to minimize the number of battery discharges for testing, the licensee opted to perform a modified performance test. The modified test was developed and implemented to meet the TS 18 month service test requirements as well as the 60 month performance test requirements.

b. Observations and Findings

During the test, EVCC battery bank failed to meet the TS minimum capacity limit of 80%. Actual battery capacity was calculated at approximately 78.5%. The inoperability of the battery bank resulted in a shutdown of both nuclear units in accordance with the TS action statement. On October 30 and 31, both units were taken offline respectively. The units were subsequently brought to cold shutdown in a controlled manner while replacement cells could be installed and tested.

The licensee used on-site spare cells and additional cells purchased from the Palo Verde Power Station. All 59 cells of the inoperable EVCC battery bank were replaced. As an additional measure to assure operability of the remaining vital battery banks, the licensee performed testing of a representative samples from each of the remaining three banks; EVCA, EVCB, and EVCD. The sample cells were replaced with either McGuire spare cells or new replacement cells from Palo Verde. The sample cells were performance tested in accordance with the station's standard performance test procedure to provide assurance that the remaining vital battery bank performances had not degraded below TS minimum. The inspectors witnessed portions of the sample cell testing. The test results demonstrated that the battery performance of the remaining three vital batteries should met TS minimum requirements. The replacement EVCC battery bank was also service tested and met TS acceptance criteria. No additional performance testing was necessary since the battery bank had been recently successfully tested by the vendor. Following the service test, the EVCC battery was recharged and subsequently declared operable.

To provide additional assurance that the installed battery banks performance would not degrade over time, the licensee committed to perform additional testing of the sample cells from the EVCB and EVCD batteries. The licensee has also initiated actions to replace all high specific gravity AT&T round cell batteries with conventional rectangular cell batteries by the end of the Unit 2 EOC11 outage.

c. Conclusion

The inspectors noted that the battery testing evolution as well as the subsequent testing of the replacement cells and sample cells was improved over previous tests, well planned, and executed in accordance with applicable procedures. Procedural discrepancies were not identified. The licensee's immediate actions to correct the battery inoperability were prompt and provided adequate assurance of equipment operability.

M2.3 Inoperability of Feedwater Isolation Valve 1CF26

a. Inspection Scope (62703)

On November 27, control room operators received indications of low nitrogen pressure coincident with excessive operation of the hydraulic pump for the D Steam Generator Containment Isolation Valve, 1CF26. Valve 1CF26 is the first isolation valve outside containment from the D steam generator designed to autoclose on a feedwater isolation signal. Operators were dispatched immediately to investigate the cause for the control room alarms. The operating crew was notified of a significant leak of hydraulic fluid from the valve actuator. The control room operators responded without delay in entering the rapid downpower abnormal procedure and realigned feedwater flow to the D steam generator through the upper feedwater nozzle (auxiliary feedwater). The unit was stabilized at approximately 20% power. The alignment through the upper nozzle was appropriate action to ensure that an unexpected closure of 1CF26 would not result in a loss of feedwater to the D steam generator causing a LO-LO S/G level automatic reactor trip signal to be generated.

b. Observations and Findings

Subsequent investigations by maintenance technicians identified that the hydraulic fluid was leaking past a blown O-Ring seal on a Schraeder valve. An emergency work order, WO 96094697, was initiated to repair the valve. Following the valve repair, the licensee performed functional testing of the hydraulic pump. The pump failed to meet the acceptance criteria and was replaced. The pump was tested and satisfied the established acceptance criteria.

During the repair effort, the licensee identified other concerns with leaking manifold solenoid valves 1SV1 and 1SV2. Additional WOs were written to repair these valves. At the close of the reporting period, Unit 1 remained at approximately 28% power while component repair and testing were being completed.

c. Conclusion

The inspectors concluded that the operator response to reduce reactor power and establish feed flow through the upper nozzle in accordance

with station procedures was conservative and effective in preventing a potential safety system challenge. The initial Maintenance response and Engineering support to the equipment failure was prompt.

M3 Maintenance Procedures and Documentation

M3.1 Failure to Perform TS Required Containment Integrity Verification

a. Inspection Scope (61726)

During the restart of Unit 1 on November 8, the licensee identified that a TS required surveillance for cold shutdown containment integrity was not performed prior to the unit proceeding from MODE 5 to MODE 4. The inspectors were informed of the missed TS surveillance shortly after it was identified and reviewed the licensee's corrective actions and safety significance of the problem. Unit 1 was in MODE 4 when the problem was identified. PIP 1-M96-3218 was identified to document the problem.

b. Observations and Findings

PT/1/A/4200/02B, Cold Shutdown Containment Integrity Verification, implements TS SR 4.6.1.1 by verifying that all applicable penetrations are in the closed position during each cold shutdown except that such verification need not be performed more often than once per 92 days. The PT was last performed for Unit 1 in January 1996. The licensee determined that the TS SR had been erroneously signed off as being performed within the required frequency based on operations personnel review of the MODE 4 checklist procedure. Surveillance program guidance documents had listed the frequency of the surveillance based on the refueling outage performance frequency (18 months) rather than the conditional frequency of at least every 92 days during cold shutdown conditions. Personnel mistakenly concluded that the PT was within the required frequency prior to entering MODE 4.

Immediate corrective actions included satisfactory performance of the required surveillance and review of other MODE change surveillance requirements on both units. No additional problems were immediately identified. Based on this event and other surveillance program area problems, the licensee initiated a Quality Improvement Team (QIT) to evaluate the existing Surveillance program regarding scheduling and performance of TS surveillances and the adverse trend developing in those areas. The inspector reviewed the proposed scope of the QIT with lead personnel and concluded that the scope was broad enough to identify other potential problem areas.

c. Conclusions

The inspectors concluded that based on the successful completion of the PT after the problem was identified, the significance of the missed surveillance was minimal. However, based on the number of other

problems previously identified in the area of TS surveillances, the inspectors concluded that initiation of the QIT review was warranted. The failure to perform the surveillance in accordance with TS SR 4.6.1.1 is identified as a non-cited violation (NCV 369/96-10-02). 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. However, increased management attention to this area and the QIT evaluation is warranted.

M3.2 Containment Hydrogen Analyzer Procedure Review (62703)

The inspector observed the performance of IP/0/A/3250/39, Containment Hydrogen Analyzer monthly calibration. The procedure provides a means of performing analog channel operational tests and calibrating the stations hydrogen analyzers. The hydrogen monitoring system consists of two redundant Teledyne analyzer systems with a dual range of 0% to 10% and 0% to 30% hydrogen by volume. The inspector reviewed the procedure and noted that the instruments are calibrated at TS required values 0% and 9% hydrogen concentrations. The inspector identified that the hydrogen analyzer capability may be inconsistent with the FSAR required ranges (0 to 30 %) since concentrations greater than 9% cannot be accurately detected and indicated. The licensee identified correspondence to the NRC which provided justification for the current calibration practices. Based on additional review and discussion with NRC staff, the inspectors concluded that the current hydrogen analyzer calibration method was adequate. However, the licensee was evaluating whether FSAR revisions were required in this area. This item will be listed in URI 50-369,370/96-04-02, FSAR Discrepancies.

M4 Maintenance Staff Knowledge and Performance

M4.1 Maintenance Risk Assessment

a. Inspection Scope (62703)

The inspector reviewed the licensee's program for risk assessment of maintenance activities.

b. Observations and Findings

The inspector reviewed Maintenance Directive 3.25, "Maintenance Risk Assessment," which provides the guidance for assessing the risks associated with performing maintenance on plant systems, structures, and components. The inspector attended licensee meetings where the risks associated with upcoming maintenance activities were assessed. Factors considered to determine the overall risk of performing an activity included: work taking place on other plant equipment, work on energized/pressurized systems, non-routine evolutions, complex evolutions, activities involving hazardous materials, high dose activities, activities involving first time procedure use, activities

involving multiple disciplines, activities involving shift changes, etc. As the risk of the activity increases, the level of attention/oversight was increased. A checklist is completed by the job supervisor which identifies the challenges to be discussed at the pre-job brief/post-job critique. If the activity is determined to be a Medium/High risk activity, contingency plans are developed and increased supervisory involvement is required. The checklists and contingency plans are included in the work packages carried in the field by the maintenance workers. In addition, each of the Medium/High risk activities are discussed at the morning Maintenance supervisors meeting for a final review prior to the activity being performed.

The inspector observed a Medium/High risk maintenance activity being performed in the field. The work, which included the calibration and testing of Unit 1 degraded voltage and undervoltage relays, was being conducted in accordance with Work Order 96088497-01. This activity was considered Medium/High risk because it involved auto start circuitry for the Emergency Diesel Generators as well as personnel safety considerations. The inspector verified the pre-job brief/post-job critique checklist was in the work package as well as the associated contingency plans. Discussions with the individuals performing the work indicated that they were completely familiar with the risks and contingencies associated with the activity.

Discussions with licensee management indicated that since this process was instituted maintenance rework, lost time accidents, and work order backlog have all been substantially reduced because of this process.

c. Conclusions

The inspector concluded that this process was a valuable tool for identifying potential risks associated with maintenance activities and that the licensee has implemented the process effectively at all levels of the Maintenance organization.

M8 Miscellaneous Maintenance Issues (92902)

M8.1 (CLOSED) URI 50-369/96-08-03: RCS Low Flow Trip Setpoints

This item is closed. (Reference Paragraph E3.1)

III. Engineering

E1 Conduct of Engineering

E1.1 Letdown Orifice Line Repair

a. Inspection Scope (37550)

During a forced outage, an inspection by maintenance and engineering identified water leaking from the upstream weld of valve 2NV458 (75 GPM letdown line isolation valve). Closer observation detected a spray of water from the socket weld area of the valve.

b. Observations and Findings

During a forced outage in November 1996, a walkdown performed by maintenance and engineering identified a crack in the upstream socket weld of 2NV458. Engineering performed an evaluation of the crack and had the failure site examined by a metallurgy laboratory to determine the cause of the failure. The cause of the failure was identified by the laboratory as being high cycle fatigue. The damaged area of the pipe was removed and replaced with new schedule 160 pipe. All socket welds in the 75 GPM flow path were rewelded. The replacement work and rewelding were accomplished with a work plan and work orders instead of by a modification as there were no new welds added to the piping and drawing changes were not required.

c. Conclusions

The inspector determined that the work plan and cause determination were well prepared in a short time frame to support the outage in progress. The inspector concluded this timely identification and response to an emerging problem was an engineering department strength.

E2 Engineering Support of Facilities and Equipment

E2.1 Identification of Potential Water Hammer Related Events

a. Inspection Scope (37551)

The inspector reviewed several PIPs associated with potential water hammer evidence identified by operators. NRC walkdowns were also performed on the secondary side to identify other potential problems.

b. Observations and Findings

The inspector reviewed recently identified PIPs concerning potential water hammer events. PIP 1-M96-3229 documented that due to leakage past 1CF-124, feedwater system recirculation to upper surge tank flush valve, a potential water hammer noise was occurring. PIP 0-M96-3325 documented

water hammer type noise occurring in the main steam line loops in the main steam valve vaults during plant heatup. The inspector discussed the details of the PIP evaluations with civil engineering personnel as well as other activities associated with the walkdown and identification of potential water hammer threats. Walkdowns of the identified areas did not identify any long term degradation or other evidence of water hammer, beyond the PIP described problems. Based on the discussions, the inspectors concluded that the licensee was taking appropriate measures to evaluate the specific concerns, which included performing walkdowns of the affected areas, review of operational evolutions creating the conditions, and reviewing operating procedures to determine any enhancements should be incorporated to prevent the conditions.

The inspector also conducted several independent walkdowns of a variety of secondary systems to identify other evidence of previous water hammer conditions. The inspector identified one particular area on the Unit 1 SG blowdown piping which exhibited evidence of deformation of the line and associated supports. In addition, some recent excessive pipe movement was evidenced by handrail paint on the subject piping, where the piping had impacted the nearby handrails. The inspector reported the issue to the licensee. The licensee researched the conditioned and determined that the problem had been previously identified in 1994. The inspector reviewed calculations provided which discussed the effect of the water hammer on the piping supports; however, no documentation could be readily identified which evaluated the deformation of the piping itself. At the end of the inspection period, the licensee was continuing to review the existing piping configuration for acceptability. In addition, the licensee was reviewing the piping and SG blowdown operation to identify methods to reduce pipe movement or further restrain the affected area.

c. Conclusions

The inspectors concluded that the recent identification of several potential water hammer events by operators was prudent and conservative. Engineering evaluations into the cause of the conditions was considered responsive, although still ongoing at the end of the inspection period. However, the inspectors also concluded that documentation for the acceptability of a previously identified water hammer event could have been more detailed. At the end of the inspection period, the licensee continued to evaluate the as-left condition of the SG blow down piping deformation questioned by the inspector.

E2.2 Engineering Backlogs

a. Inspection Scope (37550)

The inspector reviewed engineering's efforts to control backlogs in the areas of Temporary Modifications (TM), Control Rod Indication Problems (CRIPS), and operator workarounds.

b. Observations and Findings

The engineering department was active in the identification of backlogs in their own work as well as those items effecting operation of the facility. These other items included operator workarounds, open TMs, and CRIPS. The inspectors reviewed the outstanding lists of these items.

c. Conclusions

The inspectors reviewed the licensee's listings of operator workarounds, TMs, and CRIPS Discussions were held with members of the maintenance, modifications and operations staffs to determine the adequacy of engineering support to those organizations. The inspector observed that the number of operator workarounds was high, 60 for both units. This was caused by the implementation of a new program for workaround identification, which formalized the process but lowered the threshold for workaround identification. Several of the listed workarounds were scheduled to be eliminated during the next refueling outages. The inspector concluded that the engineering department was providing aggressive and effective support to the operations, maintenance and modification departments. This support was resulting in the low number of engineering open items.

E3 Engineering Procedures and Documentation

E3.1 Unit 1 RCS Loop A Low Flow Trip Setpoint TS Violation

a. Inspection Scope (37551)

On October 10, Engineering personnel reviewing the effect of planned S/G replacement activities on the NC system discovered that the 2 of 3 trip setpoints specified in the Maintenance procedure used to adjust the RCS Loop A Loss of Flow Reactor Trip bistables on the 1A RCS loop were not in compliance with TS 2.2.1. The licensee initiated actions to evaluate and correct the condition as necessary.

b. Observations and Findings

Technical Specification 2.2.1 requires that the Low Reactor Coolant Flow Reactor Trip System Instrumentation Trip Setpoint be set at ≥ 91% of minimum measured flow per loop with an allowable lower limit of > 90% of the minimum measured flow per loop. The TS defines minimum measured flow per loop as 95,500 gallons per minute (gpm).

The inspectors reviewed the Unit 1 RCS Flow Calibration Loop A Protection Channel II, IP/1/A/3000/22B. The inspectors determined that the procedure did not provide specific guidance for calibration of RCS Flow pursuant to the requirements of TS 2.2.1 and as a result did not meet the requirements of TS 3.3.1. The procedures did not confirm that

individual loop flows were in compliance with TS values prior to establishing limiting safety system setpoints. Calibration procedures did not require validation of reactor coolant loop flow values prior to verification of trip setpoints.

The RCS Flow trip setpoints were established to ensure that the reactor core and RCS do not exceed the safety limits during normal operation and design basis anticipated operational occurrences and to assist the Engineered Safeguards Features Actuation System (ESFAS) in mitigating the consequences of accidents. The low flow reactor trip provides core protection to prevent DNB by mitigating the consequences of a loss of flow resulting from the loss of one or more reactor coolant pumps. The Reactor Protection System (RPS) is designed to ensure that pump operation is within the assumptions used for loss of coolant flow analysis, which also assured that adequate cooling is provided to permit an orderly reduction in power if flow from a coolant pump is lost during operation. The licensee failed to ensure individual loop flow trip setpoints met TS requirements. The reduction in actual loop flow over a long period of time had not been recognized until October 10, 1996. Although actual total NC system flow remained above the TS minimum value, the actual Loop A flow was less than the 95,500 gpm specified in TS Table 2.2.1. The Loop A channel I setpoint had been established with enough margin to meet the TS minimum value requirements. An Engineering review revealed that the bistables associated with Unit 1 NC system Loop A channels II and III flow transmitters were set to values in violation of allowable TS limits. The values used for Loop 1A channel I and all channels of Loop 1B, 1C, and 1D bistables were determined to be in compliance with TS. Unit 2 bistables were evaluated and determined to be in compliance with TS. The misadjusted bistable setpoints were adjusted to values allowed by TS. The licensee failed to maintain the Unit 1 Loop A Low Reactor Coolant Flow Reactor Trip System Instrumentation Limiting Safety System Setting above 90% of minimum measured flow. The Loop A channels II and III trip setpoints were not in compliance with the Technical Specifications.

c. Conclusion

The inspectors concluded that the failure to establish setpoints in accordance with the requirements of TS 2.2.1 is a violation. 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 (50-369/96-10-03: RCS Loop A Low Flow Trip Setpoints). URI 50-369/96-08-03: RCS Low Flow Trip Setpoints is therefore closed.

E4 Engineering Staff Knowledge and Performance

E4.1 Review of SFP Area Painting Project

a. Inspection Scope (37551)

The inspectors reviewed available documentation to support a work order written to allow painting in the area around and above the Unit 2 spent fuel pool (SFP).

b. Observations and Findings

To facilitate painting in the area of the SFP, the licensee installed a temporary cover over the pool and important equipment to prevent paint overspray from affecting the safe operation of the pool and related support systems. The inspector toured the Unit 2 SFP area during painting evolutions and raised a number of questions concerning potential adverse impacts. Some of the areas in question involved SFP level monitoring, the affect of securing SFP skimmer operation, accessibility of equipment used for emergency or abnormal event response, and the affect of paint fumes on SFP area charcoal filter trains. The inspector reviewed the established work order regarding the evolution and noted that a 10 CFR 50.59 evaluation screening as described in Nuclear System Directive 209, had not been performed to document that the covering of the SFP did not introduce any USQ. The screening was not performed due to the evolution being completed under a work order and not as a plant modification. The inspector discussed the specific concerns with modifications and operations personnel and determined that details reviews had been performed to support the safe implementation of work order; however, the review was not well documented. The licensee agreed that the performance of a 10 CFR 50.59 evaluation screening would have formalized the documentation to support this infrequently performed evolution.

c. Conclusions

The inspectors concluded that although no 10 CFR 50.59 evaluation screening was performed for the Unit 2 SFP area painting evolution, the licensee had adequately evaluated the evolution for potential adverse impacts to the SFP. Based on the inspectors questions, the licensee modified existing controls and oversight of the Unit 1 SFP area painting which was still in progress at the end of the inspection period. However, based on the number of potential ways the evolution could have impacted the SFP operation and that the covering of the pool was a non-routine evolution, the inspectors considered that the performance of a 10 CFR 50.59 screening evolution as described in Nuclear Safety Directive 209 may have been more appropriate. The inspectors also

concluded that the threshold for performing 50.59 screening evolutions for complex work order type evolutions should be reviewed to ensure adequate attention in this area. This item is identified as URI 50-369,370/96-10-04, 50.59 Evaluation for SFP Area Painting Project

E4.2 Breaching of the Control Room Pressure Boundary to Support other System Testing

a. Inspection Scope (37551)

The inspectors questioned the operability impact of breaching the control room pressure boundary to install temporary instrumentation cables for surveillance testing.

b. Observations and Findings

During the inspection period, the inspector identified a condition where a rod timing testing procedure allowed the breaching of the control room envelope for the running of cables through one of the existing pressure doors. The cables were installed through the door to support testing via PT/0/A/4600/078, RCCA Drop Timing Using Rod Position Grey Code (from the CR area to the reactor trip breaker area). Specifically, the procedure allowed the breaching of the CR boundary doors if security and the work control group were notified and compensatory actions were in place. The compensatory actions; however, were not specifically defined within the procedure. The inspector was informed that in the event of a control room isolation event, the posted security guard would cut or disconnect the cables and close the door to ensure operability of the VC system. The inspectors could not identify these prescribed actions in the procedure or instructions to operators that the VC system was breached and timely compensatory actions were required to ensure system operability and limit operator dose during a LOCA or other event.

The licensee's procedure justified the VC system breach based on use of a "3 minute rule" or interpretation. Through engineering review, this rule allowed for the VC system to be breached as long as contingency measures were in place that assures the system could be sealed or restored within 3 minutes of an ESF actuation. The source of the 3 minute rule was MCC-1227-00-00-0048, Dose Consequence Impact of Mark BW Fuel Reload for Accident Analyzed in Chapter 15 of McGuire FSAR. The 3 minute criteria was based on the amount of time it would take to seal a given VC system breach and allow the CR pressurization fans to pressurize the CR to ensure radiological doses to operators would not be exceeded. The analysis used conservative assumptions for dose modeling when determining the amount of time the breach could exist following the ESF actuation.

The inspectors noted that the licensee considered the VC system degraded, but operable, when breached based on the planned compensatory measures to remove the cabling and close the pressure door within 3

minutes. The licensee did not enter the TS LCO for the CR ventilation system during the evolution. It should be noted that if the 3 minute interpretation/compensatory measures were not valid, both trains of the VC system would be considered inoperable and the unit would be in TS 3.0.3.

A previous related issue was identified in IR 369, 370/96-07, where a modification to the VC system was questioned by NRC inspectors based on the proposed use of the 3 minute interpretation. Reviews concerning that issue and maintenance practices of the VC system were still ongoing during the current inspection review.

c. Conclusions

Based on the observed testing activities, the inspector questioned the use of compensatory actions to ensure operability of the VC system and also the adequacy of the established compensatory measures, as they relate to NRC guidance in these areas. This issue is identified as URI 369, 370/96-10-05, Use of Compensatory Measures to Ensure VC System Operability.

E7 Quality Assurance in Engineering

E7.1 Quality Assurance and Self Assessments

a. Inspection Scope (37550)

The inspector reviewed several recent self assessments performed by the engineering department and the Regulatory Audit Group.

b. Observations and Findings

The inspector reviewed selected engineering department self assessments. These included SA-96-29(MC)(ENG) - Engineering/Maintenance Benchmark, SA-96-20(MC)(ENG) - Engineering Desktop, Forced Outage Critique, and SA-96-07(MC)(RA) - Consolidated Performance Assessment.

c. Conclusions

The inspector concluded that the engineering department was performing effective self assessments and that their findings were similar to those of the Nuclear Assessment and Issues Division, Regulatory Audit Group.

E8 Miscellaneous Engineering Issues (92902)

E8.1 (CLOSED) VIO 50-369/96-01-01: Failure to Adequately Evaluate the Cumulative Effects of a Design Change for all Operational Modes

On February 2, 1996, while conducting slave relay testing of the A train component cooling water (KC) system valves, a significant transient occurred on the Unit 1 KC system. The transient was of sufficient complexity, magnitude and quickness to produce a failure of one of the four valves which modulate flow to the reactor coolant pump motor upper bearing oil coolers. The loss of cooling flow to the 1A NC pump motor eventually forced control room personnel to manually trip the reactor.

The normal component cooling water system alignment had been modified due to valve 1KC1A performance problems. The valve had been placed in the closed position because of performance problems identified during testing. The abnormal system alignment did not receive an adequate review to ensure that a problem would not be created during expected operational conditions.

Following the manual reactor trip, the unit was returned to operation. Corrective maintenance was subsequently performed online and the system was tested and returned to service. To prevent similar oversights, the licensee developed and issued required training packages to engineering and operations personnel to ensure that essential station personnel were cognizant of the issue and understand station management expectations when performing plant design changes.

The inspectors reviewed the station training package and verified that station personnel were aware of the event and the importance performing a detailed review of the plant changes for all operational conditions. The inspectors also verified current system alignment and component performance to be within station guidelines. This item is closed.

E8.2 (CLOSED) LER 50-369/96-01: Unit 1 Manual Reactor Trip Initiated as a Result of Equipment Failure.

See paragraph E8.1.

IV. Plant Support

R1 Radiological Protection and Chemistry Controls

R1.1 General Comments (71750)

Using Inspection Procedure 71750, plant support activities were observed and reviewed to ensure that programs were implemented in conformance with facility policies and procedures and in compliance with regulatory requirements. Activities reviewed included radiological controls, physical security, emergency preparedness, and fire protection. In general, the conduct of plant support activities was professional and safety-conscious.

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 December 2, 1996. The licensee acknowledged the findings presented.

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

PARTIAL LIST OF PERSONS CONTACTED

Licensee

Byrum, W., Manager, Radiation Protection
Boyle, J., Mechanical/Nuclear Systems Engineering
Cross, R., Regulatory Compliance Specialist
Dolan, B., Safety Assurance Manager
Geddie, E., Manager, McGuire Nuclear Station
Herran, P., Manager, Engineering
Jones, R., Superintendent, Operations
Loucks L., Radiation Protection Manager (Acting)
McMeekin, T., Vice President, McGuire Nuclear Station
Michael R., Chemistry Manager
Nazar, M., Superintendent, Maintenance
Pierce, B., Engineering
Sample, M., Manager, Steam Generator Maintenance Group
Snyder, J., Manager, Regulatory Compliance
Thomas, K., Superintendent, Work Control
Thrasher, J., Modifications Engineering Manager
Travis, B., Manager, Mechanical/Civil Equipment Engineering
Tuckman, M., Senior Vice President, Duke Power Company

NRC

S. Shaeffer, Senior Resident Inspector, McGuire
M. Sykes, Resident Inspector, McGuire
G. Harris, Resident Inspector, McGuire
S. Rudisail, Project Engineer, RII
P. Kellogg, Regional Inspector

INSPECTION PROCEDURES USED

IP 92700:	Onsite Followup of Written Reports of Non-routine Events at Power Reactor Facilities
IP 92902:	Maintenance Followup
IP 71707:	Conduct of Operations
IP 71714:	Cold Weather Preparations
IP 71750:	Plant Support
IP 62703:	Maintenance Observations
IP 61726:	Surveillance Observations
IP 40500:	Self Assessment
IP 37551:	Onsite Engineering
IP 37550:	Engineering

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

50-369,370/96-10-01	URI	Failure to Ensure Installation of Correct Heaters in FWST Enclosures (paragraph O3.1)
50-369/96-10-02	NCV	Failure to Perform TS Required Containment Integrity Verification (paragraph M3.1)
50-369/96-10-03	NCV	RCS Loop A Low Flow Trip Setpoints (paragraph E3.1)
50-369,370/96-10-04	URI	50.59 Evaluation for SFP Area Painting Project (paragraph E4.1)
50-369,370/96-10-05	URI	Use of Compensatory Measures to Ensure VC System Operability (paragraph E4.2)

Closed

50-369,370/96-08-01	URI	Operator Abnormal Procedure Usage (paragraph O8.1)
50-369,370/95-23-03	VIO	Hydrogen Recombiner Procedures Not Adequately Maintained (paragraph O8.2)
50-370/96-01	LER	RWST Level Instrumentation Past Inoperable (paragraph O8.3)
50-369/96-08-03	URI	RCS Low Flow Trip Setpoints (paragraphs M8.1 and E3.1)

50-369/96-01-01	VIO	Failure to Adequately Evaluate the Cumulative Effects of a Design Change for all Operational Modes (paragraph E8.1)
50-369/96-01	LER	Unit 1 Manual Reactor Trip Initiated as a Result of Equipment Failure (paragraph E8.2)

Discussed

URI 50-369.370/96-04-02 FSAR Discrepancies (paragraph M3.2)

LIST OF ACRONYMS USED

AFW	Auxiliary Feedwater
CR	Control Room
CRIP	Control Room Indication Problem
DNB	Departure from Nucleate Boiling
EP/AP	Emergency Procedure/Abnormal Procedure
ESF	Engineered Safety Features
ESFAS	Engineered Safeguards Features Actuation System
FSAR	Final Safety Analysis Report
FW	Refueling Water System
FWST	Refueling Water Storage Tank
GPM	Gallons Per Minute
IR	Inspection Report
KC	Component Cooling Water System
LCO	Limiting Conditions Operating
LER	Licensee Event Report
LOCA	Loss of Coolant Accident
NB	Boron Recycle System
NC	Reactor Coolant System
NCV	Non-Cited Violation
NLO	Non-Licensed Operator
NSD	Nuclear Safety Directive
NSRB	Nuclear Safety Review Board
NV	Chemical and Volume Control System
OSM	Operations Shift Manager
PDR	Public Document Room
PIP	Problem Investigation Process
PORV	Pressure Operated Relief Valve
psia	per square inch absolute
PT	Performance Test
QIT	Quality Improvement Team
RCCA	Rod Control Cluster Assembly
RCS	Reactor Coolant System
RN	Nuclear Service Water System
RWST	Refueling Water Storage Tank
SFP	Spent Fuel Pool
SG	Steam Generator
SOER	Significant Operating Event Report (INPO)
STAR	Stop-Think-Act-Review
TM	Temporary Modification
TS	Technical Specifications
URI	Unresolved Item
USFAR	Updated Final Safety Analysis Report
USQ	Unreviewed Safety Question
VC	Control Area Ventilation System
VIO	Violation
WO	Work Order