U.S. NUCLEAR REGULATORY COMMISSION

REGION III

Docket No.	50-263
License No.	DPR-22
Report No.	50-263/97006(DRP)
Licensee:	Northern States Power Company
Facility:	Monticello Nuclear Generating Station
Location:	414 Nicollet Mall Minneapolis, MN 55401
Dates:	April 12 - May 27, 1997
Inspectors:	A. M. Stone, Senior Resident Inspector J. Lara, Resident Inspector
Approved by:	J. McCormick-Barger, Chief Reactor Projects Branch 7

EXECUTIVE SUMMARY

Monticello Nuclear Generating Station, Unit 1
NRC Inspection Report 50-263/97006

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

Operations

- Control and performance of the reactor shutdown activities were excellent. Communication and teamwork during the evolution were good (Section O1.2).

- The inspectors concluded that the anticipated transient without a scram (ATWS) system was tested in accordance with technical specifications (TSs) and as described in the updated safety analysis report (USAR). Operators responded appropriately during simulated ATWS conditions (Section O2.1).

- A violation of a TS-required procedure was identified for operators being unaware that an essential service water pump was unnecessarily operating and had been operating for about 46 hours (Section O8.1).

Maintenance

- The observed maintenance activities were performed in a professional manner and in accordance with applicable TS and USAR requirements. However, isolation tags for two valves were reversed because of inattention-to-detail by system engineering (Section M1.1).

Engineering

- The licensee's actions to address and resolve an inadequate net positive suction head concern were appropriate. The decision to shut down and replace the torus suction strainers was conservative (Section E2.1).

- The inspectors identified a concern regarding the adequacy of a TS which allowed the alignment of two power sources through a common transformer. Discrepancies between the as-design electrical system and USAR were also identified (Section E2.2).

- Licensee corrective actions with respect to the reactor core isolation cooling annunciator circuits were determined to be acceptable. An inspection followup item was identified to review additional systems for fuse and drawing discrepancies (Section E2.3).

Report Details

Summary of Plant Status

The unit operated at power levels up to 100 percent power until May 9, 1997, when operators commenced a reactor shutdown as directed by plant management. A concern regarding the available net positive suction head (NPSH) for the core spray (CS) and residual heat removal (RHR) pumps was identified by plant personnel. After further evaluation, plant management decided to shut down the unit to facilitate replacement of the emergency core cooling system's torus suction strainers. This issue is discussed in Sections O1.2 and E2.1.

I. Operations

O1 Conduct of Operations

O1.1 General Comments

Using Inspection Procedure 71707, the inspectors conducted frequent reviews of ongoing plant operations. These reviews included observations of control room evolutions, shift turnovers, operability decisions, and logkeeping. Updated Safety Analysis Report (USAR) Section 13, "Plant Operations," was reviewed as part of the inspection.

In general, the conduct of operations was acceptable. Operators' performance during routine surveillances was excellent. Command and control during the planned reactor shutdown were excellent.

O1.2 Observations of Shutdown

a. Inspection Scope (71707)

As discussed in Section E2.1, the licensee conservatively decided to shut down the reactor due to concerns with the emergency core cooling system (ECCS) suction strainers. On May 9, 1997, operators commenced the shutdown. The inspectors observed portions of the evolution. Documents reviewed included:

- C.3. Shutdown Procedures
- TS Table 4.1.1 and 4.1.2

b. Observations and Findings

The inspectors observed the infrequent evolution briefings conducted for both operations crews involved in the shutdown. Extra non-licensed, licensed, and senior reactor operators were available to support the operations crews. Nuclear engineering personnel were also present. In each briefing, the shift manager emphasized self-checking practices; maintaining a questioning attitude; roles and

responsibilities, including expectations on communication; and reactivity management controls.

During the shutdown sequence, the licensee conducted individual control rod scram testing. During the test, the inspectors observed control room operators and also accompanied non-licensed operators in the reactor building. The test was conducted in a controlled manner with excellent communication between control room operators, plant operators, and the nuclear engineer. The control room operator continuously monitored core responses to the control rod manipulations. The nuclear engineer provided excellent support to the operators.

The inspectors verified that the activities were performed in accordance with the C.3 procedure and that TS-required surveillances were completed prior to changing modes.

c. Conclusions

Control and performance of the reactor shutdown activities were excellent. Communication and teamwork during the evolution were good.

O2 Operational Status of Facilities and Equipment

O2.1 Engineered Safety Feature System Walkdowns

The inspectors used Inspection Procedure 71707 to walk down selected portions of the #11 and #12 emergency diesel generators (EDGs), the #13 diesel generator (non-safety related), reactor core isolation cooling (RCIC), and high pressure coolant injection (HPCI) systems. Minor housekeeping issues identified during the walkdowns were promptly corrected by the licensee. No operability concerns were identified.

O2.2 Anticipated Transient Without a Scram System Review

a. Inspection Scope (71707)

During this inspection period, the inspectors performed a review of the Anticipated Transient Without Scram (ATWS) system. The purpose of this review was to verify various operational and design features including the following:

- Operation of the system in accordance with applicable TS requirements
- Operation of the system in accordance with USAR description
- System alignment in accordance with plant requirements
- Surveillance procedures met the TS requirements

The inspectors reviewed the following documents:

- TS Table 3.2.5
- TS 3/4.13.H
- USAR Sections 7.6.2 and 14.8

- Operations Manual B.5.6, "Plant Protection System"
- Annunciator Response Procedures
- Test 1227, Revision 7, "ATWS-RPT, ATWS-ARI, ASDS Rod Insertion and B/U Scram Valve Functional Test"
- Test 0279, Revision 2, "ATWS Reactor Level and Pressure Transmitter"
- Test 0278a and b, Revision 6, "ATWS Recirculation Trip for Reactor Pressure and Level Trip Unit Test and Calibration"
- Work Order (WO) 9601857, Check/Adjust damping on ATWS transmitters
- WO 9601874, Check/Adjust damping on ATWS transmitters
- WO 9601877 Adjust Time Delay, ATWS Relay K101B
- CR (Condition Report) 95-117, "DVM time delay effects on timing of ATWS pump trip time delay"
- CR 94-280, "Spurious low-low level ATWS trip"
- CR 92000179/399, "Reliability of ATWS mitigation systems"
- CR 94000280, "Spurious low-low level ATWS trip on "B" channel"
- CR 95000117, "ATWS system timer influenced by digital volt meter during test"
- Modification 93Q180, Reduce Sensitivity of RPV High Pressure SCRAM Sensing Lines
- SRI 96-026, "Time Response Adjustment of ATWS Level Transmitters LT-2-3-180A-D and Level Trip Delay"
- Calculation CA-93-082, High Pressure SCRAM Time Delay

b. Observations and Findings

The inspectors verified that periodic surveillance testing was accomplished in accordance with TSs. Logic drawings were reviewed to verify that relays were properly challenged during surveillances. A detailed technical review of the surveillance tests showed that specified acceptance criteria were appropriate and reflected design parameters. The inspectors noted that formal calculations were not available for the acceptance criteria; however, the licensee's effort to document supporting calculations in surveillances was ongoing.

The material condition of the system was acceptable. Valves and electrical equipment were verified to be in the correct positions. Outstanding WOs and CRs did not impact system operability. The inspectors reviewed previous surveillance tests and confirmed that acceptance criteria were met. The inspectors also observed instrument and control technicians perform Test 0279. The procedure was workable and the technicians performed in a professional manner.

The inspectors observed two operations crews respond to ATWS situations during simulator training. The scenarios involved an ATWS with and without a standby liquid control (SBLC) system failure. The scenarios were challenging and required entries into several branches of the emergency operating procedures and abnormal procedures. The significant operator actions from a probabilistic risk assessment perspective included controlling reactor level and injecting SBLC. The crews responded appropriately and demonstrated knowledge of the ATWS system.

c. Conclusions

The inspectors concluded that the ATWS system was tested in accordance with TS and as described in the USAR. Operators responded appropriately during simulated ATWS conditions.

O8 Miscellaneous Operations Issues

O8.1 (Closed) Unresolved Item (URI) 50-263/97003-01: Operator Unawareness of an Operating Emergency Service Water (ESW) Pump. The ESW pumps were normally placed in a standby condition and were designed to automatically start upon a transfer of the normal power source to the 4160 V buses to alternate power source 1AR transformer or EDGs. On April 8, 1997, a loss of power to the #15 essential electrical bus occurred. Following the restoration of normal offsite power to bus 15, operators performed walkdowns of control room panels and failed to identify that the #13 ESW pump was still operating. As documented in Inspection Report (IR) 50-263/97003, this condition had existed for about 46 hours prior to identification by the inspectors, on April 10, 1997. During this time, four shift turnovers involving three crews had occurred between shift management, control room operators, and auxiliary plant operators. Pump status lights in the control room provided sufficient indications of an operating pump which should have been identified by the on-shift crews. The licensee documented this issue in CR 97001149.

TS Section 6.5, "Plant Operating Procedures," required that detailed written procedures covering plant operations areas be prepared and followed. TS Section 6.5.A.3 required written procedures covering actions to be taken to correct specific and foreseen potential malfunction of systems or components, including follow-up actions required after plant protective system actions have initiated. Administrative procedure 4 AWI-04.01.01, "General Plant Operating Activities," Revision 17, step 4.3.4.A required that all on-duty operators and the shift supervisor shall be aware of the plant status at all times. The failure of operations personnel from April 8 - 10, 1997, to be aware of plant status, as evidenced by an unnoticed operating safety-related pump, was contrary to the procedure and a violation of TS requirements (VIO 50-263/97006-01(DRP)).

II. Maintenance

M1 Conduct of Maintenance

M1.1 General Comments

a. Inspection Scope (62703 and 61726)

The inspectors observed all or portions of selected maintenance and surveillance activities. Included in the inspection was a review of the surveillance procedures or work orders listed as well as the appropriate USAR sections regarding the activities.

b. Observations and Findings

In general, the inspectors found the work performed under these activities to be professional and thorough. All work observed was performed with the work package present and in active use. Technicians were experienced and knowledgeable of their assigned tasks. The inspectors frequently observed supervisors and system engineers monitoring job progress, and quality control personnel were present whenever required by procedure. When applicable, appropriate radiation control measures were in place.

The following work was observed. Specific concerns or observations are provided where appropriate.

- 0026, APRM-Recirc Flow Instrumentation Calibration, Revision 20
- 0397a, SRV Low-Low Set System Quarterly Tests
- WO 9704562, Collect Core Spray Pump Performance Data

- WO 9703394: PC-14-2 Valve Leaks Through the Seat. The inspectors independently verified the equipment isolation tagout and observed portions of the post-maintenance test (PMT). The shift supervisor rejected the original PMT because guidance on how to "verify no leakage through valve" was not provided. The system engineer subsequently wrote specific steps in an attachment to the test document.

- WOs 9703795, 9703797 and 9703798: Replace Nitrogen Purge Solenoid Valves SV3372, SV3373, and SV3381. The inspectors reviewed the equipment isolation tags and identified that the tags placed on two root valves were switched. Both valves were closed; therefore, this discrepancy did not result in a personnel or equipment safety concern. The shift manager immediately stopped the job until the discrepancy was resolved.

  Prior to hanging the isolation tags, operators had noted that the root valves were not labelled and requested assistance from the system engineer. The system engineer reviewed the piping and instrument diagram and mistook the valves' identities. Based on this inaccurate information, operators subsequently hung the tags on the wrong valves. Failure to hang the isolation valves in accordance with the isolation procedure constituted a violation of minor significance and is being treated as a Non-Cited Violation, consistent with Section IV of the NRC Enforcement Policy (50-263/97006-02).

- WO 9704264: #12 EDG #2 Air Start Pressure Switch Hung Up. The inspectors noted that the work was performed while the 1AR transformer was out of service and the IR transformer was degraded due to low voltage. The licensee responded that because the work did not render the EDG inoperable, the reliability of the EDG would be increased. A probabilistic risk assessment study showed an insignificant effect on risk. The inspectors reviewed the USAR and TS and had no further concerns.

- Test 0013: IRM Scram and Rod Block/SRM Rod Block Calibration. This test was required per TSs 4.1 and 4.2 prior to the unit shutdown on May 9, 1997. The inspectors performed a technical review of the procedure to verify that the acceptance criteria complied with TS-required values, the sequence of steps did not pre-condition the test results, and the test was conducted as described in the USAR. No discrepancies were noted.

- Test 0081: Control Rod Drive Scram Insertion Time Test. The inspectors noted good communication between the plant operators, control room operators, and the nuclear engineer. The evolution was conducted in a controlled manner. Scram times were obtained from 113 fully withdrawn control rods. Of the 113, 8 control rods exceeded the 5 percent insertion time acceptance criteria of 375 milliseconds. The diaphragms for the associated scram solenoid pilot valves will be replaced prior to reactor startup.

- Test 1374: Monthly Operability Test of #13 Diesel Generator. This diesel generator was not safety-related but could be utilized during a station blackout condition. The inspectors performed a technical review of the test and verified that the system performed as described in USAR 8.4.2. The inspectors noted that two indication lines showed signs of fretting caused by loose metal clips. This was promptly resolved by the system engineer.

c. Conclusions

The observed maintenance activities were performed in a professional manner and in accordance with applicable TS and USAR requirements. However, isolation tags for two valves were reversed because of inattention-to-detail by system engineering.

M2 Maintenance and Material Condition of Facilities and Equipment (93702)

M2.1 Current Material Condition and Impact on Operations Personnel

The inspectors conducted control room and plant inspections and interviewed operations personnel to assess the material condition of plant equipment. During this period, a pressure switch (PS 2-3-52-A), which provided an interlock for the opening of the low pressure ECCS discharge valves, failed. The licensee immediately initiated a work order and repaired the switch. The inspectors reviewed TS table 3.2.2 and portions of USAR sections 6.2.2 and 6.2.3 and had no concerns.

M8 Miscellaneous Maintenance Issues (92700)

M8.1 (Closed) URI 50-263/97003-06: Personnel Errors During Undervoltage Relay Testing.

(Closed) Licensee Event Report (LER) 50-263/97006, Revision 0: Emergency Diesel Generators Started By Personnel Error During a Monthly Surveillance.

On April 8, 1997, surveillance Test 0301, "Safeguard Bus Voltage Protection Relay Unit Functional Test," Revision 21, was performed by testing undervoltage relays and verifying that appropriate relay contacts closed upon the relays dropping out. During the test, electricians failed to remove a test meter used to monitor continuity across the relay contacts. As a result, the loss of voltage protection logic was satisfied resulting in the trip of the normal power source to 4160 V bus #15. This resulted in the automatic start of the #11 and #12 EDGs and #11 EDG loading on bus #15.

The method to verify continuity (momentary contact or landing of meter leads) was left up to skill-of-the-craft, and there was no explicit procedural requirement to remove any installed test instruments upon completion of applicable procedure steps. Failure to provide adequate instructions for the performance of surveillance Test 0301 is a violation of TS section 6.5.A.4, which required detailed written procedures covering surveillance and testing requirements that could have an effect on nuclear safety. However, 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 50-263/97006-03).

As required by 10 CFR 50.73, on May 8, 1997, the licensee submitted LER 97-006, which documented the automatic actuation of an engineered safeguard feature. The LER discussed the safety significance of the event, immediate actions, corrective actions, and preventative actions. Corrective actions included meeting with electrical maintenance personnel to discuss the lessons learned from this event. Preventative actions included revising the surveillance procedure to provide instructions for the removal of installed test meters. Additionally, instead of using continuity measurements to monitor the status of relay contacts, voltage measurement requirements were added. The inspectors witnessed the performance of the revised surveillance procedure 0301, "Safeguard Bus Voltage Protection Relay Unit Functional Test," Revision 22, during the subsequent monthly testing of the degraded safeguards bus voltage relays. The test was satisfactorily performed. The inspectors considered the licensee's implementation of the corrective actions discussed in the LER to be acceptable.

M8.2 (Closed) LER 50-263/97005, Revision 0: Failure to Analyze Diesel Fuel Samples Within the Technical Specification Required Surveillance Period. In response to Violation 50-263/96009-14, the licensee committed to review other surveillances for potential timeliness concerns. During this process, the licensee identified that diesel generator fuel oil samples were taken monthly, but not analyzed within the TS surveillance time interval. Although analyzed late, all samples supported diesel

operability. The procedure was revised to require the analysis prior to completing the surveillance. The inspectors considered this example to be acceptable implementation of corrective actions for a previous violation. Failure to analyze the fuel oil samples within the surveillance window is considered another example of a previous violation (VIO 50-263/96009-14).

III. Engineering

E2 Engineering Support of Facilities and Equipment

E2.1 Concerns with Net Positive Suction Head for Low Pressure ECCS Pumps

a. Inspection Scope (37551)

In February 1997, the licensee initiated a CR to evaluate the applicability of a NPSH concern identified at another nuclear utility. On April 15, 1997, the licensee completed the evaluation and concluded that the concern was applicable. The inspectors reviewed the licensee's evaluation, prompt operability determination, and corrective actions. Several conference calls between cognizant NRC and licensee staff members were held throughout this period. Also, the following documents were reviewed:

- LER 50-263/97007, "Inadequate NPSH for the ECCS Pumps for Certain Single Failures During Loss of Coolant Events"
- CR 97001188, "Higher ECCS Suction Strainer Head Losses Calculated"

b. Observations and Findings

The current NPSH calculation assumed a 1-foot head loss per 10,000 gallons per minute (gpm) through the suction strainers. The licensee determined that the actual calculated head loss was about 11.7 feet. The licensee identified that under certain conditions, the available NPSH for the CS pumps was insufficient and would result in pump cavitation. Specifically, during a design bases loss of coolant accident, a failure of the low pressure coolant injection (LPCI) loop select logic could cause all four RHR pumps to inject into the broken recirculation pipe. This water would suppress the drywell pressure and subsequently decrease available NPSH. The CS pumps would be the only pumps available to recover reactor vessel level. About 3 minutes into the accident, the available NPSH for the CS pumps would be less than the required NPSH and would result in pump cavitation.

The licensee concluded that the CS pumps would remain operable and would supply sufficient flow to the reactor vessel under deficient NPSH conditions. This conclusion was based on previous vendor testing performed on a similar pump that operated for several hours with NPSH deficits with no observable damage. The test pump also supplied 90 percent flow for 30 minutes without damage. The licensee also performed calculations to show that the degraded flow was sufficient to re-flood and maintain level in the vessel.

The inspectors requested support from the NRC Office of Nuclear Reactor Regulation staff to review the licensee's basis for operability. The staff had several technical questions and requested additional information with respect to (1) the material similarity of the vendor-tested pump and that installed at the plant; (2) the assumed containment overpressure available and required; and (3) how potential strainer plugging from debris (as discussed in NRC Bulletin 96-03, "Potential Plugging of Emergency Core Cooling Suction Strainers by Debris in Boiling Water Reactors," was addressed.

The licensee performed strainer clogging calculations and concluded that debris from insulation material had the potential to further degrade the available NPSH for the CS pumps. On May 9, 1997, the licensee conservatively decided to shut down the reactor to replace the ECCS torus suction strainers.

c. Conclusions

The licensee's actions to address and resolve the inadequate NPSH concern were appropriate. The decision to shut down and replace the strainers was conservative.

E2.2 USAR Chapter 8 Electrical Systems Concerns

a. Inspection Scope (37551)

The inspectors reviewed the licensee's offsite power supplies to evaluate whether the plant electrical configurations were as described in the USAR and TS.

b. Observations and Findings

A simplified electrical one-line diagram is shown in Attachment 1 of this report. The USAR described three transformers to supply the site with offsite power from substations and all three sources can provide adequate power for the plant's safety-related loads. These included primary station auxiliary transformer 2R, reserve transformer 1R, and reserve auxiliary transformer 1AR. Transformers 2R and 1AR were considered one offsite source when 1AR was supplied from 345-kilovolt (kV) bus #1 since numerous common mode failures existed which could cause simultaneous de-energization of both transformers.

In order to maintain 1R transformer operable, a minimum of 119 kV was required at the high voltage side of the transformer. The inspectors noted that on two recent occasions, less than 119 kV was experienced at the high side of 1R transformer when the #10 transformer was taken out-of-service. These two instances indicated that 1R transformer was dependent on the 345-kV lines through the #10 transformer. The inspectors were concerned that system load growth had increased to where the plant would be outside the licensing basis whenever the #10 transformer provided power to both the 1AR and 1R transformers. Although the USAR was not clear on the dependence of the 119-kV and 345-kV lines on the #10 transformer, the licensee stated that the current design was within the licensing basis. The inspectors were provided with a safety evaluation from a 1984

modification (84MO41) which acknowledged the dependence on the #10 transformer and possible low 115 kV bus voltage.

Additional reviews indicated that TS 3.9.A.1 allowed for continued plant operation with transformers 1R and 1AR serving as the two offsite sources. However, there were no restrictions on both transformers being powered from the #10 transformer. Therefore, a loss of the #10 transformer would cause the unavailability of both offsite sources; the de-energization of the 1AR transformer and less than 119 kV at the 1R transformer. The licensee stated that although allowed by TS, the 1R and 1AR transformers would not both be energized from the #10 transformer, whenever possible. This operational restriction was not described in the TS or USAR, but was in Operations Manual B.9.3-05, "345 kV Substation," section A.2.c. The inspectors concluded that the current design of transformers 1R and 1AR being dependent on #10 transformer was not described in the licensee's USAR and required further NRC review. This issue is an Inspection Followup Item (IFI 50-263/97006-04(DRP)) pending review by the Office of Nuclear Reactor Regulation.

Additional review by the inspectors indicated a discrepancy in USAR 8.3.3, Performance Analysis. The USAR stated that "provisions are made for automatic, fast transfer of the auxiliary load from the primary station transformer to the reserve transformer or the auxiliary reserve transformer"; (i.e. 2R, 1R, and 1AR, respectively). The inspectors noted that the electrical design included a fast transfer feature from transformer 2R to 1R upon protective relaying actuation. However, there was no fast transfer feature of the auxiliary loads from transformer 1R to 1AR. The licensee stated that there was never a fast transfer scheme for all auxiliary loads among the three transformers. The inspectors noted that USAR section 8.2.1 stated that transformers 2R and 1R were of adequate size to provide the plant's full auxiliary load requirements. However, transformer 1AR was sized to provide only the plant's essential 4 kV buses and connected loads. The licensee was asked to evaluate the USAR discrepancy. The results of that evaluation will be reviewed during a future inspection (IFI 50-263//97006-05(DRP)).

c. Conclusions

The inspectors identified a concern regarding the adequacy of a TS which allowed the alignment of two power sources through a common transformer. Discrepancies between the as-design electrical system and USAR were also identified.

E2.3 As-Built Discrepancies in RCIC 250 Vdc Motor Control Center (MCC)

a. Inspection Scope (37551)

The inspectors performed a review of the licensee's 50.72 event notifications associated with the RCIC system.

b. Observations and Findings

On May 6, 1997, the licensee reported to the NRC that the RCIC system was declared inoperable due to an undervoltage alarm on the 250 volts-direct current (Vdc) MCC. The following day, the RCIC was returned to service and declared operable after replacement of internal components on the MCC undervoltage monitoring system. Later that day, the licensee again notified the NRC regarding a loss of the undervoltage alarm monitoring system for the RCIC 250 Vdc MCC. In this case, the ability to monitor the available power to primary containment group 5 isolation valves was lost. A loss of power could have gone undetected and; therefore, could have resulted in a failure of the Group 5 valves to close on demand.

Corrective action for the first event included replacement of internal electronic components followed by a burn-in test of the annunciator monitor circuit. For the second event, additional electronic components were replaced which were not replaced for the first event. The inspectors reviewed the licensee's corrective actions and post-maintenance testing and determined that they were acceptable.

During the review of ongoing work activities, the inspectors identified discrepancies regarding as-built configuration. These included:

- as-built wiring connections for the reflash annunciator monitor were not accurately reflected in drawing NF-36969-1;
- fuses sized at 1 ampere (A) and 1.5 A were found installed whereas drawing NF-36969-1 required 0.5 A fuses;
- in MCC cubicle D31114 (RCIC test return valve MO-3502), a 6¼ A fuse was installed where a 6 A fuse was required (also, different fuse type)
- undervoltage relay isolation fuses were shown on individual MCC bucket drawings but not on Class 1 elementary drawings. Class 1 drawings were defined as drawings which were considered essential to safe and reliable plant operation.

It should be noted that the wiring and fuses within the reflash annunciator enclosure (first 2 items above) were categorized as nonsafety-related. At the end of the inspection period, the licensee was reviewing past modifications and design documents to determine if the discrepancies were the result of past design weaknesses or current design and fuse replacement practices. This issue will be an Inspection Followup Item pending further NRC inspections of additional MCC circuits to determine if the incorrect fuse and drawing omissions were examples of broad deficiencies (IFI 50-263/97006-06(DRP)).

c. Conclusions

Licensee corrective actions with respect to the RCIC annunciator circuits were determined to be acceptable. An IFI was identified to review additional fuse configurations and drawings.

E2.4 Core Spray Test Return Valve Position Limit Switch

a. Inspection Scope (37551)

The inspectors reviewed the licensee's corrective actions regarding the potential for the CS test return valve to remain partially open following an ECCS signal.

b. Observations and Findings

CR 97001001, "Potential Limit Switch Closure of CS Test Return MOVs on ECCS Initiation," documented that modification 90Z071 changed the control logic for the CS Test return motor-operated valves (MO-1749 and MO-1750). The change was intended to bypass the close torque switch on an ECCS auto initiation signal, thereby closing on the limit switch. However, this logic change resulted in the valves closing to the position where the close indication light limit switch was set (approximately 98 percent). Therefore, the valves could remain partially open and divert flow from the reactor to the torus.

At the time that the licensee identified this issue, the test return valves had been manually seated closed with the torque switch; therefore, no operability concerns existed. The licensee performed WOs 9704064 and 9704065, CS Valves MO-1749/1750, "Limit Switch Setting Determination," to determine the true valve position at the close limit switch indication.

The inspectors reviewed the operability determination and observed the implementation of the WOs to evaluate the effectiveness of the corrective actions. The test return valves were stroked open and manually closed to determine the valves' position based on the limit switch indication light. The licensee's evaluation concluded that the valves were essentially closed upon limit switch actuation. Additionally, motor contactor dropout time and motor inertia would provide an additional seating force. The inspectors did not identify any deficiencies during the review of the licensee's operability and reportability determinations.

c. Conclusions

The licensee's evaluation of the conditions and corrective actions were determined to be acceptable.

E8 Miscellaneous Engineering Issues (37551)

E8.1 (Closed) IFI 50-263/96002-01: This item pertained to a discrepancy in the USAR regarding the performance of surveillances. The inspectors observed that instrument technicians lifted covers off of instrumentation during surveillances. However, USAR section 7.6.3.3.1 stated that operations personnel must remove the cover plate, access plug, or sealing device from instruments. The licensee has revised the USAR section (revision 14) to allow any authorized personnel to remove the cover plate, access plug, or sealing device from instruments.

E8.2 (Closed) IFI 50-263/96002-02: This item pertained to a discrepancy in the USAR regarding the required fuel pool cooling temperatures. USAR 10.2.2.3 contained two maximum spent fuel storage pool temperatures (125 degrees Fahrenheit (°F) and 140°F). This issue was subsequently reviewed by the NRC as documented in IR 50-263/96003, paragraph 3.4. Using more realistic decay heat loads, the licensee determined a maximum spent fuel pool temperature of 140°F. The licensee has revised USAR 10.2.2.3 (revision 14) to reflect the results of the analysis.

E8.3 (Closed) IFI 50-263/96005-05: This item pertained to a discrepancy in USAR section 8.5.2.2 which stated that each 125 Vdc battery was rated for 96 As at a 4-hour rate. However, the actual battery capacity as indicated on the battery nameplate was 95 As at a 4-hour rate. The licensee had evaluated the actual battery capacity and determined that the battery size was sufficient to carry design loads. The licensee has revised the USAR section (revision 14) to reflect the actual battery capacity.

IV. Plant Support

R1 Conduct of Radiological Protection and Chemistry Controls (71750)

During normal resident inspection activities, routine observations were conducted in the areas of radiological protection and chemistry controls. No discrepancies were noted.

P1 Conduct of Emergency Preparedness Activities (71750)

During normal resident inspection activities, routine observations were conducted in the area of emergency preparedness. No discrepancies were noted. Three notifications were made to the NRC pursuant to 10 CFR 50.72. The licensee later retracted two notifications involving a loss of a power monitoring system for the RCIC motor control center. The inspectors agreed these events were not reportable.

S1 Conduct of Security and Safeguards Activities (71750)

During normal resident inspection activities, routine observations were conducted in the areas of security and safeguards activities. No discrepancies were noted. On May 14, 1997, the inspectors met with the security supervisor to discuss current security issues. Topics included personnel performance trending, new training initiatives, status of corrective actions to previously identified concerns, and material condition of security-related equipment.

V. Management Meetings

X1 Exit Meeting Summary

On June 3, 1997, the inspectors presented the inspection results to the Plant Manager and the Manager, Quality Services. 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

M. Wadley, Vice President, Nuclear Generation
W. Hill, Plant Manager
M. Hammer, General Superintendent, Maintenance
K. Jepson, Superintendent, Chemistry & Environmental Protection
L. Nolan, General Superintendent, Safety Assessment
M. Onnen, General Superintendent, Operations
E. Reilly, Superintendent, Plant Scheduling
C. Schibonski, General Superintendent, Engineering
A. Ward, Manager, Quality Services
J. Windschill, General Superintendent, Radiation Protection
L. Wilkerson, Superintendent, Security
B. Day, Training Manager

INSPECTION PROCEDURES USED

IP 37551:	Onsite Engineering
IP 61726:	Surveillance Observations
IP 62703:	Maintenance Observations
IP 71707:	Plant Operations
IP 71750:	Plant Support
IP 92700:	Onsite Followup of Written Reports of Nonroutine Events at Power Reactor Facilities
IP 93702:	Prompt Onsite Response to Events at Operating Power Reactors

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

50-263/97006-01	VIO	Failure to Follow Procedure: Operators Did Not Know Status of ESW Pump
50-263/97006-02	NCV	Failure to Hang Isolation Tags on Correct Valves
50-263/97006-03	NCV	Failure to Provide Adequate Instructions for Performance of Surveillance Test
50-263/97006-04	IFI	Restrictions on Operation with #10 Transformer
50-263/97006-05	IFI	USAR Discrepancy Regarding Fast Transfer Capability
50-263/97006-06	IFI	As-Built Discrepancies in RCIC Motor Control Center

Closed

50-263/96002-01	IFI	Discrepancy Between USAR Description and Conduct of Surveillances
50-263/96002-02	IFI	Discrepancy in USAR Regarding Fuel Pool Temperatures
50-263/96005-05	IFI	Discrepancy in USAR Regarding Battery Capacity
50-263/97003-01	URI	Operator Unawareness of Operating ESW Pump
50-263/97003-06	URI	Personnel Errors During Undervoltage Relay Testing
50-263/97005-00	LER	Failure to Analyze Diesel Fuel Samples Within TS Required Surveillance Period
50-263/97006-00	LER	Emergency Diesel Generators Started By Personnel Error During a Monthly Surveillance
50-263/97006-02	NCV	Failure to Hang Isolation Tags on Correct Valves
50-263/97006-03	NCV	Failure to Provide Adequate Instructions for Performance of Surveillance Test

Discussed

50-263/96009-14	VIO	Test Results Not Evaluated Prior to Return of Equipment to Service
50-263/97007-00	LER	Inadequate NPSH for the ECCS Pumps for Certain Single Failures During Loss of Coolant Events

LIST OF ACRONYMS USED

A	Ampere
ATWS	Anticipated Transient Without Scram
CR	Condition Report
CS	Core Spray
°F	Degrees Fahrenheit
DRP	Division of Reactor Projects
ECCS	Emergency Core Cooling System
EDG	Emergency Diesel Generators
ESW	Emergency Service Water
gpm	Gallons Per Minute
HPCI	High Pressure Coolant Injection
IFI	Inspection Followup Item
IR	Inspection Report
kV	Kilovolt
LER	Licensee Event Report
LPCI	Low Pressure Coolant Injection
MCC	Motor Control Center
NCV	Non-Cited Violation
NPSH	Net Positive Suction Head
PMT	Post-Maintenance Test
RCIC	Reactor Core Isolation Cooling
RHR	Residual Heat Removal
SBLC	Standby Liquid Control
TS	Technical Specification
USAR	Updated Safety Analysis Report
Vdc	Volt-Direct Current
VIO	Violation
V	Volt
WO	Work Order

ATTACHMENT 1
Simplified Electrical Diagram