August 19, 1998

Mr. M. Wadley
President, Nuclear Generation
Northern States Power Company
414 Nicollet Mall
Minneapolis, MN 55401

SUBJECT: PRAIRIE ISLAND INSPECTION REPORT 50-282/98009(DRP); 50-306/98009(DRP)

Dear Mr. Wadley:

On July 30, 1998, the NRC completed an inspection at your Prairie Island Nuclear Generating Plant. The enclosed report presents the results of that inspection.

During the 6-week period covered by this inspection, all of the operations, maintenance, surveillance testing, and security activities observed were performed well. Most engineering support activities were also performed well with one exception being the engineering evaluation of whether the Technical Specifications required testing of the automatic bypass of the diesel generator reverse current trip during a safety injection.

No violations of NRC requirements were identified during the inspection.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter and its enclosure will be placed in the NRC Public Document Room.

Sincerely,

Original signed by
Geoffrey E. Grant

Geoffrey E. Grant, Director
Division of Reactor Projects

Docket Nos.: 50-282; 50-306
License Nos.: DPR-42; DPR-60

Enclosure: Inspection Report
50-282/98009(DRP);
50-306/98009(DRP)

cc w/encl: Plant Manager, Prairie Island
State Liaison Officer, State of Minnesota
State Liaison Officer, State of Wisconsin
Tribal Council, Prairie Island Dakota Community

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/s/Geoffrey E. Grant

Geoffrey E. Grant, Director
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Enclosure: Inspection Report
50-282/98009(DRP);
50-306/98009(DRP)

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U.S. NUCLEAR REGULATORY COMMISSION

REGION III

Docket Nos:	50-282; 50-306
License Nos:	DPR-42; DPR-60
Report No:	50-282/98009(DRP); 50-306/98009(DRP)
Licensee:	Northern States Power Company
Facility:	Prairie Island Nuclear Generating Plant
Location:	1717 Wakonade Drive East Welch, MN 55089
Dates:	June 19 through July 30, 1998
Inspectors:	S. Ray, Senior Resident Inspector P. Krohn, Resident Inspector S. Thomas, Resident Inspector
Approved by:	M. Kunowski, Acting Chief Reactor Projects Branch 7

EXECUTIVE SUMMARY

Prairie Island Nuclear Generating Plant, Unit 1 and Unit 2
NRC Inspection Report 50-282/98009(DRP); 50-306/98009(DRP)

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

Operations

- Most operations activities were conduced well with operators being challenged by electrical grid problems on several occasions because of hot weather and severe storms. On one occasion, operators failed to display Unit 1 containment parameter trends, contrary to management expectations. (Section O1.1)

- The planned power reduction and subsequent return to full power on Unit 2 were conducted well with a comprehensive pre-job briefing, good supervisory oversight, good reactivity management, and adequate communications. (Section O1.2)

Maintenance

- All of the 14 maintenance and surveillance testing activities observed were performed well. Thorough pre-job briefings were held for all except the most routine work. The activities were all conducted safely, with proper consideration given to the radiation fields and other environmental conditions at the job sites. Communications between all parties involved in the tasks were good. (Section M1.1)

- Technical Specification requirements were properly included in procedures for all of the 46 requirements checked. Only a few minor editorial or format errors were identified in the 71 procedures reviewed. (Section M3.1)

Engineering

- The licensee's initial engineering evaluation of whether testing of the blocking of the reverse current trip of the emergency diesel generators was required by Technical Specifications, as documented in Licensee Event Report 50-282/98007, Revision 0, was flawed in that the points considered did not support the conclusion reached. Nonetheless, adequate corrective actions were taken for the inadequate testing which had existed for about 20 years. After discussions with NRC inspectors, the licensee subsequently re-evaluated the testing requirement. The results of the re-evaluation were reviewed by the inspectors and were satisfactory. (Section E1.1)

- During operations, maintenance, and surveillance testing activities, the system engineers were closely involved. They were consistently present in the field for all except the most routine activities. The system engineers often led and were usually involved in pre-evolution briefings for activities. The system engineers were also very knowledgeable about the operation of their systems and responsive to questions from both the operators and inspectors. (Section E2.2)

Plant Support

- The security force provided timely and important information to the control room staff concerning severe weather conditions having a potential safety impact. Also, the security force reacted quickly to compensate for damage to perimeter monitoring equipment cause by a lightning strike. (Section S1.1)

Report Details

Summary of Plant Status

Unit 1 operated at full power until the morning of June 25, 1998, when power was reduced to aid in correcting grid stability problems caused by severe weather. Unit 1 returned to full power later the same morning and remained there for the remainder of the inspection period. Unit 2 operated at full power until June 28, 1998, when power was reduced to about 40 percent for routine turbine valve testing and condenser cleaning. Unit 2 returned to full power operation on June 29, 1998, and remained there for the remainder of the inspection period.

I. Operations

O1 Conduct of Operations

O1.1 General Comments

a. Inspection Scope (Inspection Procedures (IPs) 71707, 92901)

The inspectors conducted frequent reviews of plant operations. The reviews included observations of control room evolutions, shift turnovers, pre-job briefings, communications, control room access management, logkeeping, control board monitoring, and general control room decorum. Updated Safety Analysis Report (USAR), Section 13, "Plant Operations," Revision 15, was reviewed as part of the inspection.

b. Observations and Findings

- On June 23, 1998, the inspectors noted that Unit 1 containment radiation monitor and humidity trends were not being displayed as discussed in station Temporary Instruction (TI) 98-08. The TI had been issued on February 10, 1998, because of the discovery of leakage from a Unit 2 part length control rod housing, and established practices for increased monitoring of various Unit 1 parameters for indications of reactor coolant system leakage. Although the TI did not specifically require continuous display of the containment and radiation trends, it was a management expectation that the trends be displayed on a control board screen unless the screen needed to be used for a higher operational priority.

  The inspectors brought the lack of monitoring to the attention of the shift manager and the Unit 1 lead reactor operator who subsequently restored the display to a control board screen. The Unit 1 shift supervisor determined that the trends had been reviewed and found to be normal about seven hours previously. The inspectors considered the failure to display the containment parameters to be an attention-to-detail weakness on the part of the operating crew but not a violation of regulatory requirements.

- On June 24, 1998, the inspectors noted that the Unit 2 source range nuclear instrument meters (2NI-51C and 2NI-51B) at the train A and B hot shutdown

panels were reading downscale low at 10^{-1} counts per second. Since Unit 2 was at 100 percent power, the normal reading should have been greater than 10^{5} counts per second. The inspectors brought the discrepancy to the attention of the Unit 2 shift supervisor who contacted the system engineer. The system engineer subsequently issued Work Order (WO) 9806956 to direct the investigation and repair the instruments, which were repaired and returned to service on July 9, 1998. The problem was determined to be a bad test generator card in the circuit.

The inspectors had identified a similar problem with the Unit 1 source range nuclear instrument indications at the hot shutdown panels on January 13, 1998. This was documented in Inspection Report 50-282/98003(DRP); 50-306/98003(DRP), Section M1.1. One of the corrective actions for that problem was to revise monthly Surveillance Procedures (SP) 1222 (Unit 1)/2222 (Unit 2), "Event Monitoring Instrument Channel Check," to include checks of the hot shutdown panel source range nuclear instrument indications. The inspectors verified that these surveillance procedures had been revised to include source range nuclear instrument channel checks at the hot shutdown panels.

Channel checks in accordance with SP 2222 had last been performed on June 15, 1998, at which time it was determined that the Unit 2 source range nuclear instrument indications at the hot shutdown panels were normal. Thus, the indications could have been reading downscale low for as long as nine days prior to the inspectors finding. There was no Technical Specification (TS) operability requirement for hot shutdown panel instrumentation.

Abnormal Operating Procedure (AOP) 2C1.3 AOP1, "Shutdown From Outside the Control Room - Unit 2," Revision 3, listed the actions necessary to place Unit 2 in a hot shutdown condition following a control room evacuation. Step 2.4.28 of the procedure directed the reactor operators to monitor the source range nuclear instruments at the hot shutdown panels to verify shutdown reactor conditions. For the condition identified by the inspectors, the Unit 2 source range meters at the hot shutdown panels would not have provided the reactor operators with the ability to monitor shutdown reactor power in the source range. The safety significance of not being able to read actual source range neutron power level was mitigated by Step 2.4.30 of 2C1.3 AOP1 which instructed the reactor operators to add boron to the reactor coolant system prior to the xenon level decreasing below the pre-trip condition. That action ensured that sufficient negative reactivity would be present in the core to maintain subcritical reactor conditions at hot shutdown.

- The operations staff was challenged many times during this inspection period by grid instabilities caused by severe weather, high winds, downed power lines, and record power usage because of abnormally hot outside temperatures. On separate occasions, the Blue Lake and Byron offsite power lines were lost because of weather-related causes. A lightning strike on site caused minor damage to some security equipment (discussed in Section S1.1). During one period of unusually severe weather, operators were requested by the utility's corporate electrical distribution center to reduce the site output. Operators

rapidly reduced power on Unit 1 and controlled the turbine and control rods in manual. For each of these challenges, the inspectors observed that the operators took effective action to ensure the plant was maintained in a safe condition.

c. Conclusions

Most operations activities were conduced well with operators being challenged by electrical grid problems on several occasions because of hot weather and severe storms. On one occasion, operators failed to display Unit 1 containment parameter trends contrary to management expectations.

O1.2 Unit 2 Power Reduction and Ascension

a. Inspection Scope (IP 71707)

The inspectors observed major portions of the control room activities surrounding a Unit 2 power reduction to about 40 percent on June 28, 1998, and subsequent return to full power on June 29, 1998. The power reduction was performed to support turbine stop, governor, and intercept valve surveillance testing, cleaning of the inner-pass condenser tubes, and inspection and repair of the inner-pass Amertap (condenser tube cleaning system) screens.

b. Observations and Findings

The inspectors observed the shift manager and Unit 2 shift supervisor conduct a pre-evolution briefing for the planned power reduction on June 28, 1998. The briefing discussed precautions and limitations, expected xenon transients, control rod positions and plans to maintain axial flux difference in the target band, rod deviation alarms, personnel assignments, and a general outline of the power reduction procedure, 2C1.4, “Unit 2 Power Operation,” Revision 15. The reactor power decrease was performed in a slow and controlled manner. The inspectors noted an instance of good supervisory oversight when the Unit 2 shift supervisor counseled the lead reactor operator on maintaining a focus on the primary plant parameters vice the expected transients occurring on balance-of-plant equipment. The inspectors also observed good reactivity management by the reactor operator who maintained reactor coolant average temperature closely matched to the reference value, understood and compensated for the xenon transients occurring in the core, and carefully controlled boric acid additions and control rod positions to maintain axial flux difference in the middle of the target band during the power reduction.

On June 29, 1998, the inspectors observed the power restoration to 100 percent. The inspectors noted that reactivity control and operator attentiveness were adequate. Communications were informal at times, but adequate. During the power ascension, an increasing negative trend for axial flux difference was observed. Although no target band or TS limit for axial flux difference was exceeded, and good operator action was taken once the trend was identified, the power ascension was delayed, at 89 percent reactor power and for approximately three hours, while the concern was addressed.

c. Conclusions

The planned power reduction and subsequent return to full power on Unit 2 were conducted well with a comprehensive pre-job briefing, good supervisory oversight, good reactivity management, and adequate communications.

O7 Quality Assurance in Operations

O7.1 Change in Membership on the Safety Audit Committee (SAC)

The licensee announced that an individual, who was consultant and former employee of Northern States Power Company, was named as the chairman of the SAC starting with the meetings the week of July 6, 1998. In addition, the operation of the SAC was changed so that senior plant managers would still participate in the meeting but no longer be considered voting members of the SAC.

The new SASC chairman met with the inspectors to discuss the membership changes and his plans for the SAC. He stated that one of his goals was that the SAC members would spend more time meeting with members of the plant staff and inspecting the plant in preparation for the formal SAC meetings.

II. Maintenance

M1 Conduct of Maintenance

M1.1 General Comments

a. Inspection Scope (IPs 61726, 62707)

The inspectors observed all or portions of the maintenance and surveillance test activities conducted in accordance with the listed procedures. Included in the inspection was a review of the surveillance procedures (SPs) or WOs listed as well as the appropriate USAR sections regarding the activities. The inspectors verified that the surveillance procedures observed met the requirements of the TSs.

- SP 1095, “Bus 16 Load Sequencer Test,” Revision 12;
- SP 1100, “12 Motor-Driven AFW [Auxiliary Feedwater Pump] Monthly Test,” Revision 53;
- SP 1112, “Steam Exclusion Damper Test,” Revision 34;
- SP 1194, “Bus 15 Load Sequencer Test,” Revision 11;
- SP 1323, “11 Station Battery Monthly Test,” Revision 3;
- SP 1324, “12 Station Battery Monthly Test,” Revision 3;

- SP 2089, “Residual Heat Removal Pumps and Suction Valves From The Refueling Water Storage Tank,” Revision 52;

- SP 2267, “Steam Generator Blowdown Isolation Motor Valve Cycling,” Revision 12;

- SP 2297, “Cycling of CRDM [Control Rod Drive Mechanism] Cooling Valves,” Revision 5;

- SP 2323, “21 Station Battery Monthly Test,” Revision 3;

- SP 2324, “22 Station Battery Monthly Test,” Revision 3;

- WO 9804962, “Adjust/Repair Excess Seal Leak on 1B Pump Seal [for 12 Motor-Driven AFW Pump]”;

- WO 9807518, “Investigate Transmitter 1LT-471 Condensing Pot Fitting Leak,” and

- WO 9807591, “Replace Cracked Fitting on 1LT-471 Condensing Pot.”

b. Observations and Findings

- The inspectors observed the pre-evolution briefing for and the performance of testing in accordance with SP 1100, “12 Motor-Driven AFW Pump Monthly Test,” Revision 53. The discussion in the briefing included the portions of the surveillance test procedure to be completed, precautions and limitations, personnel assignments, acceptance criteria, communication methods, control of personnel-in-training observing the surveillance test, and past problems. The individual conducting the briefing also mentioned that work in accordance with WO 9804962 would be accomplished during the test. The WO contained instructions for maintenance personnel to tighten a packing gland on the 12 motor-driven AFW pump while the pump was running. The inspectors noted, however, that no maintenance personnel were present at the pre-evolution briefing.

  During performance of the surveillance test, the inspectors noted that the system engineer was present to observe the running of the 12 motor-driven AFW pump as well as the tightening of the packing gland. The system engineer ensured that the packing leakage was reduced but that the packing gland was not over tightened. The inspectors also observed that the operator performing the test completed a thorough walkthrough of the steps prior to performing the test and maintained good control of the equipment during the test. An operations department trainee also observed the test.

- The inspectors observed the pre-evolution briefing for and the performance of testing in accordance with SP 2089, “Residual Heat Removal Pumps and Suction Valves From The Refueling Water Storage Tank,” Revision 52. The surveillance

test consisted of a 15-minute run, on recirculation, of each residual heat removal pump, during which time pump performance and vibration data were obtained. The test included a partial stroke-open of residual heat removal (RHR) pump discharge check valves and a closure of the RHR pump suction check valves.

The inspectors noted that the pre-evolution briefing was thorough and that there was a good discussion between the operators concerning expected plant responses during the performance of the surveillance test. The inspectors observed the conduct of the RHR pump suction valve timing tests and the performance of 22 RHR pump test from the control room. Good control of the evolution was observed and all equipment being tested functioned within the guidelines of the SP. The inspectors observed the performance of the 21 RHR pump test locally in the auxiliary building. The inspectors noted that the 21 RHR heat exchanger coolant inlet valve opened smoothly subsequent to the 21 RHR pump being started, formal communications were being utilized between the control room and the operators in the control room, the operators were performing system walkdowns during the time the RHR pump was operating, and the system engineer was present throughout the performance of the surveillance test procedure.

- The inspectors witnessed portions of testing in accordance with SP 1112, “Steam Exclusion Damper Test,” Revision 34, from the control room and locally at the applicable instrument test racks located in the Unit 1 rod control drive room, Bus 15 room, and Bus 16 room. The control room operator made good use of the “point then operate” technique when manipulating control switches during the performance of the surveillance test. Also, good communication and coordination were observed between the control room operator and the instrument and control specialist during the test. The inspectors noted that the proper action was taken when an improperly adjusted limit switch caused dual open/shut indication. Operators verified that all the dampers were in the correct position and documented the limit switch problem in the comments section of the SP.

- The inspectors observed the performance of testing in accordance with SP 2267, “Steam Generator Blowdown Isolation Motor Valve Cycling,” Revision 12, from the control room. During the performance of this testing, the steam generator blowdown isolation valves were cycled, acceptable stroke times were verified, local valve position verifications were performed, and remote valve position indication lights were checked. The inspectors noted that positive control of the evolution was maintained from the control room and that effective communications were maintained throughout the performance of the test. During one portion of the surveillance, the inspectors noted good attention to radiation dose control practices when actions were taken to limit an operator’s exposure to a hot-spot to the very minimum time necessary to perform that valve cycling evolution.

- The inspectors observed the performance of testing in accordance with SP 2297, “Cycling OF CRDM Cooling Valves,” Revision 5, locally in the Unit 2 auxiliary building and at the containment chilled water system control panel. During the

performance of the test, containment chilled water isolation valves for the shroud cooling units were cycled, acceptable stroke times for the isolation valves were verified, local valve position verifications were performed, and remote valve position indication lights were checked. The inspectors noted good coordination of the operators in the shield building, auxiliary building, turbine building, and the control room during the performance of the test. Although there was an adequate discussion of how the valve stroking times would be obtained, an error by one of the operators who was timing the valve stroke necessitated the reperformance of one set of the CRDM shroud cooling coil chill water isolation valve stroke tests.

- The inspectors attended the pre-job briefing for the containment entry at power to conduct the investigation and repair attempt of a leaking steam generator level detector (1LT-471) in accordance with WO 9807518. The discussions in the briefing included the appropriate concerns with both the radiation dose and high temperature in the work area. The actual repair attempt was unsuccessful because it was discovered that the leak was from a crack in a welded fitting rather than simply a loose fitting. The mechanics and system engineer determined that welding of a new fitting would be required. Since the work was beyond the scope of WO 9807518, new WO 9807591 was issued for that task.

- The inspectors attended the pre-job briefing for the performance of work in accordance with WO 9807591, “Replace Cracked Fitting on 1LT-471 Condensing Pot.” The briefing included an overview of the repair procedure, foreign material exclusion, and post-maintenance testing of the repair. Since the repair was to be done at power, cool areas, low dose areas, and environmental and radiological stay times were also discussed. The brief was attended by representatives from the radiation protection, instrument and control, operations, maintenance, and engineering groups.

  During the briefing, concerns were raised about the effect that the welding machine would have on the other narrow-range level instruments for the 12 steam generator. Since the protective relays associated with 1LT-471 had been placed in the tripped condition, as required by TSs, a spike on one of the remaining two steam generator narrow range level channels, or on the steam flow channels, could have caused a reactor trip and/or an engineered safety feature system actuation. The engineers decided to evaluate further the effect of welding on the level instruments prior to performing the repair on the cracked fitting.

  A shop test was performed utilizing a low frequency welding machine similar to the machine used for the actual weld repair; a Foxboro transmitter, similar to the transmitter used for steam flow detection; and a Rosemont transmitter, which was similar to the transmitter used for steam generator narrow range level detection. Welding was performed within varying distances from to the transmitters. The superintendent of instrument and control engineering informed the inspectors that no interference from the welding was noted on the Rosemont

transmitter, and only minor interference was noted on the Foxboro transmitter. In addition to the shop testing, the engineering staff researched industry reports pertaining to welding inside containment while at power. This search yielded references to only interference, due to welding, associated with source range nuclear instruments and containment radiation monitors. The engineers also reviewed a report from the Ginna Nuclear Generating Station, at which a similar repair had been accomplished with no reported interference.

The inspectors attended the rebriefing for the performance of the repair of the cracked fitting on the 1LT-471 condensing pot. The briefing was thorough and was attended by representatives of the same departments that attended the initial briefing. Additional topics covered were the location of the welding machine and routing of welding leads to minimize electrical interference, additional communications between the welders and the control room operators, and the conduct of a test-strike of the welding arc to monitor control room parameters for interference or spiking. The inspectors observed the performance of the test-strike and no abnormal instrument operations were noted in the control room. The repair was successfully completed and the steam generator narrow-range level detector was placed back in service.

- The inspectors observed the performance monthly battery testing in accordance with SP 1323, SP 1324, SP 2323, and SP 2324. The testing of the station safeguards batteries included measuring the voltage of each cell, measuring the temperature and density of the pilot cell in each battery, and assessed the general condition and appearance of the cells in each battery. The inspectors noted that procedure precautions were followed, that all steps in the surveillances were completed adequately, and that no abnormal conditions or parameters were observed on any of the batteries.

c. Conclusions

All of the 14 maintenance and surveillance testing activities observed were performed well. Thorough pre-job briefings were held for all except the most routine work. The activities were all conducted safely, with proper consideration given to the radiation fields and other environmental conditions at the job sites. Communications between all parties involved in the tasks were good.

M3 Maintenance Procedures and Documentation

M3.1 Review of TS Requirements and Actual Testing Methods

a. Inspection Scope (IP 92902)

The inspectors reviewed TS Sections 3 and 4 to identify instances where testing of specific contacts, components, trains, or systems were required. Those requirements were then compared to plant operating, surveillance, and preventive maintenance procedures to determine if the TS requirements were being adequately met. During the course of this review, the inspectors examined 46 TS requirements and the subsequent

implementation of these requirements in 71 operating, surveillance, or preventive maintenance procedures.

b. Observations and Findings

The inspectors identified minor editorial or format errors in some procedures as follows:

- Technical Specification 4.6.B.5 required that the integrity of the station battery fuses be checked once each day when the battery charger was running. Operating Procedure C20.9, "Station Battery and DC [Direct Current] Distribution System," Revision 21, Section 5.4, stated that this requirement was implemented every six hours when the turbine building operator (TBO) took log readings. The TBO recorded the battery charger voltage and the associated battery voltage. If the battery charger voltage was five or more volts higher than the battery voltage, then the fuses inside the battery disconnect may have been open-circuited and C20.9, Section 5.4, referenced 1(2)C20.9 AOP5, 1(2)C20.9 AOP6, "Battery Fuse Failure," for this condition.
  The TBO log readings were taken using Prairie Island Nuclear Generating Plant (PINGP) forms PINGP 195, "Turbine Building Data - Unit 1," Revision 48, and PINGP 196, "Turbine Building Data - Unit 2," Revision 66. The Unit 1 TBO logs correctly required the recording of battery charger voltage and battery voltage on page 3 for the 11 and 12 125-volt station batteries. The logs instructed the operator to subtract the battery voltage from the battery charger voltage reading and referred to Note 10 at the bottom of the same page. Note 10 stated, in part, that if the battery charger voltage was five or more volts higher than the battery voltage, the operator was to notify the shift supervisor and consult abnormal operating procedure 1C20.9 AOP3/4. Abnormal operating procedures 1C20.9 AOP3, "Failure of 11 Battery Charger," Revision 2 and 1C20.9 AOP4, "Failure of 12 Battery Charger," Revision 2, however, discussed battery charger failures and were the incorrect procedures for a battery fuse failure. Abnormal operating procedures 1C20.9 AOP5/6 were the correct references as discussed in C20.9, Section 5.4.

  Likewise, the Unit 2 TBO logs contained the same error on page 3 where Note 6 was referenced when battery voltage was subtracted from battery charging voltage for the 21 and 22 125-volt station batteries. Note 6 referenced 2C20.9 AOP 3, "Failure of 21 Battery Charger," Revision 2 and 2C20.9 AOP 4, "Failure of 22 Battery Charger," Revision 2. The correct references were 2C20.9 AOP5, "Failure of 21 Battery Fuse," Revision 0 and 2C20.9 AOP6, "Failure of 22 Battery Fuse," Revision 0.

  The inspectors discussed this finding with the superintendent of electrical systems engineering on July 1, 1998. The superintendent acknowledged the error and said the Unit 1 and Unit 2 turbine building operator logs would be revised to correct the abnormal operating procedure references. At the inspection exit meeting, he further stated that the log sheet revisions had been initiated.

- Writers guide procedure H14.4, “Surveillance & Periodic Test Procedure Guideline,” Revision 10, Section 8.2, stated “The acceptance criteria for the specific test will be described in this [Purpose and General Discussion] section.” Section 8.2 also stated, “If specific steps need to be completed to meet the acceptance criteria, those steps will be marked with an asterisk (*) placed in the left margin beside the step number. A statement in the text for the general discussion section will be included, saying ‘Those steps marked with an asterisk (*) must be completed satisfactorily in order for the test to be acceptable.’” The inspectors found some examples where asterisks were not used for steps that appeared to contain acceptance criteria.

  In the first example, TS Table 4.1-1A, Functional Unit 6, Source Range, Neutron Flux, Shutdown, Functional Test, Note 10, stated that “Quarterly surveillances in modes 3, 4 and 5 shall also include verification that permissives P-6 and P-10 are in their required state for existing plant conditions by observation of the permissive annunciator window.” That requirement was met by SP 1011 [2011], “Nuclear Source Range Functional Test,” Revision 17, Steps 7.39.7, 7.39.9, and 7.39.10. for Modes 3, 4, and 5. Precautions and Limitations Step 3.3 stated that “Criteria designated ACCEPTABLE VALUES are directly related to TSs.” However, SP 1011[2011] did not designate Steps 7.39.7, 7.39.9, and 7.39.10 as being related to TS requirements or as an acceptable value. Also, Steps 7.39.7, 7.39.9, and 7.39.10 did not have an asterisk in the left margin to designate that those specific steps needed to meet the acceptance criteria of TS Table 4.1-1A.

  In the second example, TS 4.4.B.3.b required that inlet heaters and the associated controls for each train of the shield building and auxiliary building special ventilation systems be determined to be operable. That requirement was met with SP 1074, “Auxiliary Building Special Vent System Functional Test,” Revision 23, Steps 7.14.1 and 7.14.2. The reference section in SP 1074 listed the surveillance procedure as meeting the requirements of TS 4.4.B.3. Steps 7.14.1 and 7.14.2, however, do not include asterisks to designate them as specific items requiring satisfactory completion to meet TS requirements.

  In the third example, TS 4.4.I.b for the containment hydrogen recombiners required that a thorough visual examination show no evidence of abnormal conditions such as loose wiring or foreign material within the recombiner enclosures. In addition, TS 4.4.I.c required that the integrity of all heater electrical circuits be verified by performing a resistance-to-ground test. The acceptance criterion for the test was a resistance reading of greater than or equal to 10,000 ohms. Preventative maintenance (PM) procedures, PM 3150-1, “Electrical Testing & Inspection of 11 & 12 Containment Hydrogen Recombiners,” Revision 7, and PM 3150-2, “Electrical Testing & Inspection of 21 & 22 Containment Hydrogen Recombiner,” Revision 7, Steps 7.3.2 and 7.3.4, met those TS requirements but did not include asterisks in the left hand margin of the steps.

  Also for the third example, the inspectors noted that the procedure in question was a PM not an SP. Writers guide procedure H14.5, “Maintenance Procedure Guidelines,” Revision 5, was the applicable guide for that type of procedure.

Procedure H14.5 did not contain instructions for the use of asterisks in TS acceptance criteria steps. The inspectors discussed the issue with the author of H14.5 who stated that very few PM procedures were used to meet TS requirements but that asterisks should be used on the applicable steps for consistency with SP procedures. At the inspection exit meeting, the writers guide author indicated that he had initiated procedure revisions to correct the problem.

c. Conclusions

Technical Specification requirements were properly included in procedures for all of the 46 requirements checked. Only a few minor editorial or format errors were identified in the 71 procedures reviewed.

M8 Miscellaneous Maintenance Issues (IPs 92700, 92902)

M8.1 (Closed) Enforcement Action (EA) 97-073 01013: Inadequate Control of Heavy Loads Over the Reactor. This issue was previously discussed in Inspection Report 50-282/97002(DRP); 50-306/97002(DRP), Section M1.2; a public Enforcement Conference held on March 18, 1997; an associated NRC letter containing a Notice of Violation, dated April 30, 1997; the NSP response to the Notice, dated May 26, 1997; the NSP response to an earlier non-escalated Notice of Violation, dated March 26, 1997; Licensee Event Report (LER) 2-97-01, dated March 17, 1997, and Inspection Report 50-282/97005(DRP); 50-306/97005(DRP), Sections M3.1 and M8.1.

As discussed in the documents referenced above, the licensee took extensive corrective actions for this violation. The inspectors verified the completion of the corrective actions and noted a significant improvement in the attention given to heavy load evolutions and in the quality and quantity of heavy loads procedures and training. Some procedures remained to be revised, specifically, some of those for Unit 2 refueling operations, but the inspectors verified that the licensee's procedure revision tracking program would ensure that those revisions would be completed prior to the next scheduled Unit 2 refueling outage.

M8.2 (Closed) Unresolved Item 50-282/98007-04(DRP): Possible Failure to Perform TS Required Testing of the Unit 1 Emergency Diesel Generators (EDGs);

(Closed) LER 50-282/98007, Revision 0 (1-98-07-00): Diesel Generator Logic Testing; and

(Closed) LER 50-282/98007, Revision 1 (1-98-07-01): Diesel Generator Logic Testing.

This issue is discussed in Section E2.1 of this report. It was considered a violation of TS requirements. The unresolved item and LER are closed and the remaining corrective actions will be reviewed when the violation is closed.

III. Engineering

E1 Conduct of Engineering

E1.1 Failure to Verify Bypassing of Trips on EDGs

a. Inspection Scope (IPs 92700, 92903)

The inspectors and a regional electrical specialist reviewed the unresolved item previously discussed in Inspection Report 50-282/98007(DRP); 50-306/98007(DRP), Section M3.2; LER 50-282/98007, Revision 0 (1-98-07-00), "Diesel Generator Logic Testing," issued on May 29, 1998; and LER 50-282/98007, Revision 1 (1-98-07-01), "Diesel Generator Logic Testing," issued on July 30, 1998.

b. Observations and Findings

The licensee originally considered the LER to be a voluntary submittal for information only. In the original LER, the licensee stated its conclusion that the issue was not a violation of regulatory requirements. The primary points supporting the licensee's conclusion were that the testing of other contact pairs in the maximum credible accident (MCA) relay was sufficient to prove that the reverse current trip blocking contact pair was operable, that the reverse current trip could not actuate in a design basis event, that the possibility of a spurious actuation of the reverse current trip was insignificant, and that the reverse current trip was not considered a diesel generator system trip.

The inspectors concluded that the points did not support the conclusion. Although the probability was small, a spurious actuation of the reverse current trip during EDG operation was possible and the MCA relay contacts were designed to block such a trip. Technical Specification 4.6.A.3.e required verification that diesel generator system trips which were automatically bypassed during a safety injection were actually bypassed. The reverse current trip was considered to be a diesel generator system trip in accordance with current NRC guidance, although the interpretation in effect at the time the TS was written was unclear.

Although the licensee incorrectly concluded that the testing of the block feature was not required by TS, it did complete successful testing of the feature on both Unit 1 EDGs shortly after the issue was identified. In addition, as discussed in the LER, the EDG test procedures were being modified to include the testing. The inspectors determined that the safety significance of the previous failure of the licensee to test the trip blocking feature was low because of the low probability of spurious trips and the fact that the block features performed as designed when finally tested.

The inspectors also determined that the licensee missed previous opportunities to identify the issue, as discussed in Inspection Report 50-282/98007(DRP); 50-306/98007(DRP). Most recently, the licensee considered the blocking feature to be outside of the scope of the reviews conducted in response to NRC Generic Letter No. 96-01, "Testing of Safety-Related Logic Circuits." Although the inspectors agreed that the reverse current trip itself could be considered outside the scope, automatic blocking of the trip during a safety injection should have been included in the scope because its failure had a small possibility of affecting a safety function. On July 30,

1998, the licensee issued a revision to the LER which stated that testing of the block of the reverse current trip was required by TS.

On January 18, 1978, the NRC issued Amendment 25 (Unit 1)/19 (Unit 2) to the TS which added the requirement for verification of the automatic bypassing of the trips currently contained in TS 4.6.A.3.e. The licensee did not verify that the generator reverse current trips were automatically bypassed until April 30 and May 7, 1998, for the D2 and D1 EDGs, respectively. This non-repetitive, licensee-identified and corrected violation is being treated as a Non-Cited Violation, consistent with Section VII.B.1 of the NRC Enforcement Policy (50-282/98009-01(DRP)). The LERs and unresolved item associated with this issue are closed in Section M8.2 of this report.

c. Conclusions

The licensee's initial engineering evaluation of whether testing of the blocking of the reverse current trip of the emergency diesel generators was required by Technical Specifications, as documented in Licensee Event Report 50-282/98007, Revision 0, was flawed in that the points considered did not support the conclusion reached. Nonetheless, adequate corrective actions were taken for the inadequate testing which had existed for about 20 years. After discussions with NRC inspectors, the licensee subsequently re-evaluated the testing requirement. The results of the re-evaluation were reviewed by the inspectors and were satisfactory.

E2 Engineering Support of Facilities and Equipment

E2.1 Review of USAR Commitments (IPs 37551, 92903)

While performing the inspections discussed in this report, the inspectors reviewed the applicable portions of the USAR that related to the areas inspected and used the USAR as an engineering/technical support basis document. The inspectors compared plant practices, procedures, and/or parameters to the USAR descriptions as discussed in each section. The inspectors verified that the USAR wording was consistent with the observed plant practices, procedures, and parameters. No discrepancies were noted.

E2.2 System Engineer Involvement in Plant Activities

As discussed in the Operations and Maintenance sections of this report, when observing operations, maintenance, and surveillance testing activities, the inspectors noted close and frequent involvement of the system engineers. They were consistently present in the field for all except the most routine activities and often led and were usually involved in pre-evolution briefings for activities. The system engineers were also very knowledgeable about the operation of their systems and responsive to questions from both the operators and inspectors.

E8 Miscellaneous Engineering Issues (IP 92903)

E8.1 (Closed) Violation 50-306/98007-01(DRP): Failure to Promptly Identify or Correct a Flooding Concern for a Unit 2 Main Steam Isolation Valve Junction Box. This violation was previously discussed in Inspection Report 50-282/98007(DRP); 50-306/98007(DRP), Section O2.1, and in the NSP response, dated July 6, 1998, to the Notice of Violation. The licensee determined in calculation ENG-ME-370 that the flow out of the room containing the junction box would be higher than the flow in during a feedwater break event. Thus, the junction box would not become flooded. The inspectors reviewed the calculation and had no additional concerns.

E8.2 (Closed) Violation 50-282/98007-03(DRP): Inadequate Procedure for Electrical Testing of the D2 EDG. This violation was previously discussed in Inspection Report 50-282/98007(DRP); 50-306/98007(DRP), Section M3.1, and in the NSP response, dated July 6, 1998, to the Notice of Violation. The inspectors verified that the appropriate procedures had been quarantined to prevent their use until the required revisions were completed.

E8.3 (Closed) Violation 50-306/98007-05(DRP): Inadequate Procedure for Installing Auxiliary Feedwater (AFW) Flow Element Orifices. This violation was previously discussed in Inspection Report 50-282/98007(DRP); 50-306/98007(DRP), Section M3.3, and in the NSP response, dated July 6, 1998, to the Notice of Violation. The inspectors verified that the corrective actions discussed in the response had been completed. The inspectors also noted that maintenance personnel have been making good use of the new checklist for review of maintenance work orders and have rejected several work orders until the requested procedure revisions were made. In addition, the inspectors verified that WO 9804014 and WO 9804016 had been written with instructions for reorienting the orifice plates and had been properly coded for completion in the next Unit 2 refueling outage.

E8.4 (Closed) Violation 50-306/98007-06(DRP): Failure to Promptly Correct Incorrect Acceptance Criteria in AFW Surveillance Test Procedure. This violation was previously discussed in Inspection Report 50-282/98007(DRP); 50-306/98007(DRP), Section M3.4, and in the NSP response, dated July 6, 1998, to the Notice of Violation. The inspectors verified that the appropriate procedure changes discussed in the response were completed. In addition, the inspectors noted that the newly instituted procedure quarantine process was being used effectively in cases where procedure revisions were determined to be needed prior to the next use of the procedure.

IV. Plant Support

S1 Conduct of Security and Safeguards Activities

S1.1 Lightning Strike in Protected Area Resulting in Partial Loss of Perimeter Monitoring Functions

a. Inspection Scope (IP 71750)

A lightning strike occurred in the protected area on the evening of June 27, 1998, when severe weather crossed the region. The inspectors, who were onsite for the Unit 2 power reduction discussed in Section O1.2 of this report, interviewed the duty security

lieutenant to determine the extent of the damage and the compensatory actions being taken.

b. Observations and Findings

The inspectors were told that the lightning strike disabled two remote security multiplexing units. Immediate compensatory actions were taken in accordance with the security plan. Security personnel had also been monitoring the local weather reports and informed the control room when a tornado touchdown within eight miles of the plant was reported.

Work orders 98004 and 98005 were issued for the repair of the multiplexing units. Instrument and control technicians responded to repair the security instrumentation on June 28, 1998, and all security monitoring systems were returned to normal status by June 29.

c. Conclusions

The security force provided timely and important information to the control room staff concerning severe weather conditions having a potential safety impact. Also, the security force reacted quickly to compensate for and repair damages to perimeter monitoring equipment cause by a lightning strike.

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 July 30, 1998. 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

J. Sorensen, Plant Manager
K. Albrecht, General Superintendent Engineering, Electrical/Instrumentation & Controls
T. Amundson, General Superintendent Engineering, Mechanical
J. Goldsmith, General Superintendent Engineering, Generation Services
J. Hill, Manager Quality Services
G. Lenertz, General Superintendent Plant Maintenance
R. Lindsey, General Superintendent Safety Assessment
D. Schuelke, General Superintendent Radiation Protection and Chemistry
T. Silverberg, General Superintendent Plant Operations
M. Sleigh, Superintendent Security

INSPECTION PROCEDURES USED

IP 37551:	Engineering
IP 61726:	Surveillance Observations
IP 62707:	Maintenance Observations
IP 71707:	Plant Operations
IP 71750:	Plant Support Activities
IP 92700:	Onsite Follow-up of Written Reports of Non-routine Events at Power Reactor Facilities
IP 92901:	Follow up - Operations
IP 92902:	Follow up - Maintenance
IP 92903:	Follow up - Engineering

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

50-282/98009-01(DRP)	NCV	Failure to Verify Bypassing of Trips on EDGs

Closed

97-073 01013	EA	Inadequate Control of Heavy Loads Over the Reactor
50-282/98007-04(DRP)	URI	Possible Failure to Perform TS Required Testing of the Unit 1 EDGs
50-282/98007, Rev. 0 (1-98-07-00)	LER	Diesel Generator Logic Testing
50-282/98007, Rev. 1 (1-98-07-01)	LER	Diesel Generator Logic Testing
50-306/98007-01(DRP)	VIO	Failure to Promptly Identify or Correct a Flooding Concern for a Unit 2 Main Steam Isolation Valve Junction Box
50-282/98007-03(DRP)	VIO	Inadequate Procedure for Electrical Testing of the D2 EDG
50-306/98007-05(DRP)	VIO	Inadequate Procedure for Installing AFW Flow Element Orifices
50-306/98007-06(DRP)	VIO	Failure to Promptly Correct Incorrect Acceptance Criteria in AFW Surveillance Test Procedure

Discussed

None. .

LIST OF ACRONYMS USED

AFW	Auxiliary Feedwater
AOP	Abnormal Operating Procedure
CFR	Code of Federal Regulations
CRDM	Control Rod Drive Mechanism
DC	Direct Current
DRP	Division of Reactor Projects
EA	Enforcement Action
EDG	Emergency Diesel Generator
IP	Inspection Procedure
LER	Licensee Event Report
MCA	Maximum Credible Accident
NRC	Nuclear Regulatory Commission
NSP	Northern States Power Company
PDR	Public Document Room
PINGP	Prairie Island Nuclear Generating Plant
PM	Preventive Maintenance
RHR	Residual Heat Removal
SAC	Safety Audit Committee
SP	Surveillance Procedure
TBO	Turbine Building Operator
TI	Temporary Instruction
TS	Technical Specification
URI	Unresolved Item
USAR	Updated Safety Analysis Report
VIO	Violation
WO	Work Order