U.S. NUCLEAR REGULATORY COMMISSION

REGION III

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Licensee:	Northern States Power Company
Facility:	Prairie Island Nuclear Generating Plant
Location:	1717 Wakonade Drive East Welch, MN 55089
Dates:	May 14 - June 25, 1997
Inspectors:	S. Ray, Senior Resident Inspector R. Bywater, Resident Inspector P. Krohn, Resident Inspector
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EXECUTIVE SUMMARY

Prairie Island Nuclear Generating Plant, Units 1 & 2
NRC Inspection Report 50-282/97011(DRP); 50-306/97011(DRP)

This inspection was performed by the resident inspectors and included aspects of licensee operations, maintenance, engineering, and plant support.

Operations

- Conduct of operations was generally good during this inspection period. Both units experienced unplanned shutdowns, including a reactor trip from full power on Unit 1. The shutdown and startup operations were well conducted. The inspectors observed proper control room manning, close attention to control panels, generally good use of communication protocols, proper use of and adherence to procedures, and detailed shift briefs in which all members of the crew contributed. (Sections O1.1, O1.2, and O1.3)

- Several examples of inadequate procedures were identified. (Section O3)

- Operators were not knowledgeable about recently identified steam generator level operating restrictions. (Section O4.1)

- The Operations Committee was not adequately carrying out its required responsibilities in several respects. (Section O7.1)

Maintenance

- The majority of inspector observed maintenance and surveillance activities were well conducted with good communications, job planning, work practices, and coordination between departments.

- Several examples of inadequate procedures were identified. (Section M3.1)

Engineering

- The inspectors reviewed two previously identified inspection followup items and concluded the items were acceptable. (Section E8)

- The inspectors reviewed several other aspects of plant performance that involved adequate levels of engineering support. These were discussed in the other sections of this report.

Plant Support

- The inspectors identified a violation for inadequate emergency lighting of access to equipment necessary for safe shutdown of the plant in the event of a fire. (Section F2.1)

Report Details

Summary of Plant Status

Unit 1 operated at or near full power until it experienced a reactor trip on June 2, 1997. The trip was caused by a wiring problem in a control rod drive as described elsewhere in this report. During the subsequent forced outage, the unit was taken to cold shutdown conditions so that the reactor missile shield could be removed to repair the control rod drive wiring problem. Unit 1 was taken critical on June 16, the generator was placed on-line on June 17, and full power was reached on June 18, 1997.

Unit 2 operated at or near full power until May 17, 1997, when it was shutdown to hot shutdown conditions to repair leaking reactor coolant system isolation valves. The unit was taken critical later the same day. Unit 2 was placed on the grid and reached full power on May 18, 1997.

I. Operations

O1 Conduct of Operations

O1.1 General Comments

a. Inspection Scope (71707)

Using Inspection Procedure 71707, the inspectors conducted frequent reviews of 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.

b. Observations and Findings

The inspectors observed proper control room manning, close attention to control panels, generally good use of communication protocols, proper use of and adherence to procedures, and detailed shift briefs in which all members of the crew contributed. No significant problems with normal plant operations were noted. The inspectors noted improvements in communications, short-term turnovers, and shift meetings during this inspection period. Additional comments regarding shutdown and startup operations are contained elsewhere in this report.

c. Conclusions

All normal operations of the plant were well conducted without significant problems.

O1.2 Unit 2 Shutdown and Startup

a. Inspection Scope (71707)

On May 14, 1997, the licensee noted a slight increase in unidentified leakage from the reactor coolant system (RCS) on Unit 2. Over the next few days leakage reached about 0.1 gallons per minute (gpm) and was identified to be going through two isolation valves in the RCS loop B drain line into the reactor coolant drain tank. On May 17, 1997, the licensee took Unit 2 to hot shutdown conditions to repair the problem. The inspectors observed major portions of the shutdown and subsequent startup.

b. Observations and Findings

The power reduction and shutdown were conducted carefully and deliberately. The evolution was well planned and controlled. The appropriate procedures were properly followed in all cases. Two problems occurred near the end of the shutdown which are discussed further, elsewhere in this report:

- After the turbine was taken off-line, the subsequent first opening of the steam dump valves caused a swell in steam generator levels that resulted in levels reaching about 45 percent with reactor power at about 7 percent. The inspectors determined that the guidance and training to operators for control of steam generator levels when power was above 5 percent was inadequate and could have resulted in the plant being outside the design basis. That issue is discussed in Section O3.2 of this report.

- After the main generator was taken off-line operators noticed that one phase of the reserve auxiliary transformer manual disconnect was glowing. Operators subsequently took actions to de-energize the transformer and transfer buses 21 and 22 to Unit 1 sources. During that evolution the licensee made an unavoidable entry into a condition prohibited by Technical Specifications (TSs) and also experienced an unexpected actuation of the turbine-driven auxiliary feedwater (AFW) pump. Those issues are discussed in Sections O8.1 and M3.1 of this report.

The seat leakage on the loop drain valves was stopped by flushing the seats and torquing the valves closed. Reactor startup was conducted later on May 17 with the generator placed back on-line early on May 18. The reactor startup and subsequent power ascension were well conducted with no significant problems.

c. Conclusions

The shutdown and startup operations were well conducted. Problems experienced were primarily due to inadequate procedures and instructions, as discussed elsewhere in this report.

O1.3 Unit 1 Reactor Trip and Startup

a. Inspection Scope (71707, 93702)

On June 2, 1997, Unit 1 experienced a reactor trip from full power. The first-out annunciator indicated that the trip was due to a high negative flux rate and the initial investigation pointed to a problem with the rod control system. The inspectors observed the initial response to the trip and monitored the subsequent cooldown to cold shutdown conditions and the eventual startup after repairs to the system.

b. Observations and Findings

The operators' initial response to the event was good. Immediate and followup actions of the Emergency Operating Procedures were properly executed. Because of valve seat leakage in both the steam and feed systems, some difficulty was experienced with arresting the immediate cooldown of the RCS and returning RCS temperature and pressure to no-load conditions. Operators identified the problems and corrected the situation by isolating the main feedwater system and later the main steam system.

Troubleshooting of the rod control system indicated that the reactor would have to be taken to cold shutdown to effect repairs. The operators reached cold shutdown on June 5. After repairing the rod control system (see Section M1) and several other preexisting and emergent equipment problems, the reactor was taken to hot shutdown on June 14, made critical on June 16, placed on line on June 17, and returned to full power on June 18. The inspectors observed significant portions of those evolutions and noted no problems with operations. Reactivity and reactor power were rigorously monitored and controlled, communications were usually good, the pace of operations was deliberate and not rushed, and procedures were properly followed.

The inspectors identified that certain procedures used during the evolutions contained problems as discussed elsewhere in this report.

The licensee properly reported the reactor trip and resultant automatic actuation of the AFW system in accordance with 10 CFR 50.72 and intended to issue Licensee Event Report (LER) 50-282/97008 as a written followup. The LER will be considered open when issued pending the inspectors' review of the corrective actions.

c. Conclusions

As with the previous Unit 2 evolutions, the Unit 1 trip recovery, cooldown, heatup, and startup operations were all well conducted. Problems experienced were primarily due to problems with procedures and instructions, as discussed elsewhere in this report.

O2 Operational Status of Facilities and Equipment

O2.1 Spent Fuel Storage Pool Action Plan Issue - Power Supply

a. Inspection Scope (92901)

In a letter to the NRC dated November 14, 1996, the licensee identified an action item to change the power supply for the spent fuel pool cooling system to safeguards sources backed by safeguards diesel generators. The inspectors reviewed the status of the modification.

b. Observations and Findings

The inspectors verified that the modification was completed and implemented on March 27, 1997. The modification transferred the power supply for each of the spent fuel cooling pumps from non-safeguards 480 volt power supplies to "A" train and "B" train safeguards 480 volt power supplies.

c. Conclusions

Although the licensee did have the capability to provide power to the spent fuel pool cooling system with an onsite power source, the source was from non-safeguards diesel generators via the non-safeguards electrical distribution system. The modification provided safeguards sources of power to the system and completed the licensee's spent fuel storage pool action item described in its NRC submittal. The modification provided highly reliable power supplies to the spent fuel pool cooling system.

O3 Operations Procedures and Documentation

In the course of normal inspection activities, the inspectors reviewed operating procedures to determine if they contained adequate and accurate information appropriate to the circumstances. For some cases where adequate information was not in the procedures, the inspectors assessed whether operator training and knowledge was adequate to compensate.

O3.1 Procedures and Training for Operation of Charging Pump Suction Valves

a. Inspection Scope (92901)

During followup of an event involving excessive draining from the RCS discussed in Inspection Report 50-282(306)/97006 the licensee's Error Reduction Task Force (ERTF) determined that one of the complications during the event recovery may have been caused by a suction valve for the charging pumps being in an incorrect position. The inspectors reviewed ERTF Report 97-02 regarding this event and procedures for operation of the charging pump suction valves.

b. Observations and Findings

During the recovery from the March 7, 1997, excessive draining event, refilling of the pressurizer during the evening of March 7 and morning of March 8 was complicated due to an unexpected increase in level with no intentional addition of water. ERTF Report 97-02 reported that the probable cause of the increase was that motor-operated valve MV-32062, "21 Refueling Water Storage Tank (RWST) to Charging Pump Suction," was inadvertently left open when MV-32063, "21 Volume Control Tank (VCT) Outlet to Charging Pump Suction Header," was closed as part of the evolution. The valves were interlocked so that MV-32063 could not be closed unless MV-32062 was first opened. Leaving either the RWST or VCT valves to the charging pumps open during cold shutdown conditions could result in draining from those tanks into the RCS.

The inspectors reviewed Operating Procedure 2C1.3, "Unit 2 Shutdown," Revision 39. Step 5.2.45.B instructed operators to close MV-32063 when RCS pressure was less than 100 pounds per square inch - gauge (psig). The procedure had no instructions for first opening MV-32062 and then reclosing MV-32062 after closing MV-32063. Operations supervisors stated that those instructions were not necessary because that operation was part of the training program and within the "skill-of-the-craft" for licensed operators. However, the inspectors interviewed some licensed operators and determined that at least one on-shift licensed reactor operator was not aware of the interlock.

The inspectors determined that 2C1.3 was inappropriate to the circumstances in that it did not contain adequate instructions for the operation of the charging pump suction valves. This was considered an example of a violation of 10 CFR 50, Appendix B, Criterion V. (50-282(306)/97011-01a(DRP))

c. Conclusions

While the existence of the interlock and need to open and then reclose MV-32062 were within the training and skill of most of the operators, it was not known to all of them. Therefore, it was inappropriate to not have that information in the unit shutdown procedure.

O3.2 Procedures and Training for Implementation of Steam Generator Level Operating Restrictions

a. Inspection Scope (71707, 92901)

The inspectors reviewed the implementation of steam generator level operating restrictions recently identified by the engineering and nuclear analysis departments.

b. Observations and Findings

In response to analytical work performed by the nuclear analysis department regarding a postulated main steam line break (MSLB) accident inside of

containment, the engineering department calculated steam generator water level operating limits. (This work was documented in Safety Evaluation 50-471 and Calculation Numbers ENG-ME-312 and ENG-ME-312, Addendum 1). Operating the plant with steam generator level below the calculated limit provided assurance that MSLB accident assumptions were met. The calculations established the following operating limits to assure that the plant was not operated in a condition outside its design bases:

- Maintain narrow range steam generator level less than 38 percent when RCS average temperature was greater than 350 degrees Fahrenheit (°F) and reactor power was less than 5 percent.

- Maintain narrow range steam generator level less than 5 percent above program level when reactor power was between 5 percent and 40 percent.

During the Unit 2 shutdown on May 17, 1997, the inspectors noted that a swell in steam generator level to approximately 45 percent occurred when reactor power was approximately 7 percent. Control room operators were using Procedure 2C1.3, "Unit 2 Shutdown," Revision 39. The procedure only included the steam generator level limitation for when reactor power was less than 5 percent. The control room operators were not aware of the additional limitation for steam generator level and concluded that the swell in steam generator level was not reportable as operating in a condition outside the design basis because reactor power was greater than 5 percent. The inspectors concluded that Procedure 2C1.3 was inadequate because it did not identify the limitation on steam generator level when reactor power was between 5 percent and 40 percent. If the procedure had contained this limitation, operators may have identified the steam generator level swell condition as a reportable event.

Failing to include the appropriate steam generator level limits in 2C1.3 was considered a violation of 10 CFR 50, Appendix B, Criterion V. (50-282(306)/97011-01b(DRP))

Later, the licensee concluded that the swell in steam generator level was a result of a change in secondary coolant density and not an increase in secondary coolant mass. Therefore, the MSLB accident assumptions were not exceeded and the event was not reportable.

The inspectors had additional discussions with operations department personnel regarding the procedure inadequacy and operator training and knowledge of the steam generator level limitation issue (see also Section O4.1). The licensee had made some additional changes to operating procedures and alarm response procedures, and provided additional training to operators.

c. Conclusions

The inspectors concluded that the steam generator level limits were not properly incorporated into all of the appropriate procedures to ensure the limits would be met or properly reported if not met.

O3.3 Auxiliary Feedwater System Operability During Reactor Startup

a. Inspection Scope (71707, 92901)

During the Unit 1 startup on June 17, 1997, the inspectors reviewed the TS requirements for AFW system operability and their implementation by procedure.

b. Observations and Findings

TS 3.4.B.1.c required that when RCS average temperature exceeded 350°F, valves and piping associated with the AFW system pumps must be operable. However, necessary changes may be made in motor-operated valve position during startup operation when made under direct administrative control. Startup operation, as defined in the TS, meant the process of heating up a reactor above 200°F, making it critical, and bring it up to power operation. Power operation was defined in the TS as operation at greater than 2 percent of rated thermal power. Therefore, the TS required that the AFW system be operable, but allowed changes in motor-operated valve position under direct administrative control from 350°F RCS average temperature to 2 percent power. Changes in motor-operated valve position were not allowed at greater than 2 percent power unless the appropriate limiting conditions for operation action requirements and time limits were met and no more than one pump was involved.

After a meeting of the Operations Committee on June 16, 1997, (see Section O7.1), where a proposed change to Procedure 1C1.2, "Unit 1 Startup Procedure," Revision 17, was discussed, the inspectors reviewed 1C1.2 for implementation of AFW TS requirements. The procedure instructed operators to place the main feedwater system in service at step 5.6.6 after reaching the point of adding heat. At step 5.6.9, when levels in the steam generators were being maintained satisfactorily by the main feedwater system (this could potentially be at power levels greater than 2 percent), the operators were instructed to shutdown any operating AFW pump and align the AFW system for safeguards operation. Then, operators were instructed to raise reactor power to approximately 6 percent. The procedure contained no information concerning the need to align the AFW system prior to reaching 2 percent power and could have allowed reactor operation greater than 2 percent power in violation of the TS.

Procedure 1C1.2 was not of a type appropriate to the circumstances because it allowed plant operation greater than 2 percent power with an inoperable AFW system, a condition prohibited by the TS. This was considered an example of a violation of 10 CFR 50, Appendix B, Criterion V. (50-282(306)/97011-01c(DRP))

c. Conclusions

The inspectors determined that the Unit 1 startup on June 16-17, 1997, was performed with AFW system operability consistent with the TS. However, Procedure 1C1.2 was incorrect because it could have allowed a condition of AFW inoperability to exist when the unit changed operating modes from startup operation to power operation.

O3.4 Procedures for Control Room Evacuation in the Event of a Fire

a. Inspection Scope (71707, 92901)

During review of an emergency lighting surveillance test on June 24, 1997, the inspectors reviewed the control room evacuation procedure to determine if the surveillance procedure ensured that adequate lighting was available for operators to perform their duties.

b. Observations and Findings

The inspectors reviewed Plant Safety Procedures F5, Appendix B, "Control Room Evacuation (Fire)," Revision 17, and F5, Appendix C, "Control Room Evacuation (Fire) Diesel Generator Operation," Revision 12. Both procedures directed the Unit 1 lead plant equipment operator (LPEO) to go to the safeguards bus No. 15 room and the D1 emergency diesel generator room. Both procedures also directed the Unit 2 LPEO to go to the safeguards bus No. 25 room and the D5 emergency diesel generator room. Procedure F5, Appendix B, instructed the operators to use stairways and pathways illuminated with emergency lights as shown in its accompanying figures. A path was identified for the Unit 1 LPEO to travel to the D1 diesel generator; however, there was no path identified for travel to the bus No. 15 room. Additionally, a path was not identified for the Unit 2 LPEO to travel to the D5/D6 building (the location of safeguards bus No. 25 and the D5 emergency diesel generator), nor were any emergency lights identified in the D5/D6 building itself. As discussed in Section F2.1, the inspectors also identified that access to the safeguards bus No. 15 room was not provided with an emergency light, as required by NRC regulations.

Procedure F5, Appendix B, was not of a type appropriate to the circumstances because it did not identify illuminated pathways of travel to the bus No. 15 room, the D5/D6 diesel generator building, and inside the D5/D6 building. This was considered an example of a violation of 10 CFR 50, Appendix B, Criterion V. (50-282(306)/97011-01d(DRP))

c. Conclusions

The inspectors concluded that emergency lighting considerations were not always incorporated into plant procedures for safe shutdown in the event of a fire.

O4 Operator Knowledge and Performance

O4.1 Steam Generator Level Operating Limitations

a. Inspection Scope (71707, 92901)

The inspectors observed operator performance and reviewed operator knowledge with respect to recently implemented operating limitations on steam generator level.

b. Observations and Findings

As discussed in Section O3.2, during the Unit 2 shutdown on May 17, 1997, control room operators were not aware of the limitation on steam generator level when reactor power was between 5 percent and 40 percent. The inspectors also observed portions of the Unit 2 restart on May 17. The procedure in effect was 2C1.2, "Unit 2 Startup Procedure," Revision 17. This procedure identified both of the power-dependent limitations on steam generator level. However, the control room operators were not aware of the limitation on steam generator level above 5 percent power and assumed that as reactor power was raised to greater than 5 percent during the startup, then it was no longer necessary to closely monitor steam generator level to ensure it did not exceed 38 percent and exceed the MSLB accident analysis assumptions.

c. Conclusions

Operators were not fully trained on changes made to steam generator level limitations. The inspectors discussed this change management problem with the general superintendent of operations, who stated that the shift managers would receive additional training on the limitations.

O7 Quality Assurance in Operations

O7.1 Performance of Operations Committee

a. Inspection Scope (92901)

The inspectors observed the performance of the Operations Committee (OC) in several meetings to determine if the requirements of TS 6.2.B requirements were fulfilled.

b. Observations and Findings

- On June 11, 1997, the battery system engineer brought a proposed procedure deviation to Surveillance Procedure SP 2336, "22 Battery Semi-Annual Inspection," Revision 3, to the OC for consideration. The procedure deviation was to change the acceptance criteria of the test by deleting the requirement for the average specific gravity to be greater than 1.205 and substituting a statement to the effect that the system engineer determine

test acceptability during the review process. The system engineer informed the committee that the most recent surveillance of the 22 battery had shown that the average specific gravity was 1.204 and, unless the deviation was approved, he would have to consider the results unacceptable and enter an 8-hour TS action requirement after which a plant shutdown would be required.

The OC discussed the proposed deviation for about 30 minutes before one of the members informed the committee that Administrative Work Instruction 5AWI 1.5.1, "Procedure Deviation Process," Revision 11, Step 6.2.6.a, specifically prohibited using the deviation process to change acceptance criteria. After about 90 minutes of discussion, the inspectors asked when the surveillance had been completed and whether the battery was considered operable since it did not meet the currently approved acceptance criteria. The inspectors were informed that the battery data had been taken 19 days previously, but the average specific gravity had just been calculated earlier that day. The OC then decided to declare the battery inoperable until an evaluation could be completed.

Performance of the OC was weak in this case because it initially failed to identify that the proposed procedure deviation was prohibited by 5AWI 1.5.1. In addition, the OC failed to promptly consider operability of the battery. The battery should have been declared inoperable as soon as the OC identified that the surveillance had not met the established acceptance criteria.

An additional discussion regarding the surveillance procedure is contained in Section M3.1 of this report.

- As discussed above, 5AWI 1.5.1, "Procedure Deviation Process," Revision 11, Step 6.2.6.a, specifically prohibited using the deviation process to change the scope or intent of a procedure including changes to acceptance criteria. Similarly, 5AWI 1.5.4, "Temporary Memos," Revision 5, Step 6.1.1, prohibited using the temporary memo process to change the scope or intent of a procedure. However, the OC had initiated a practice of allowing the procedure deviation or temporary memo process to be used to change the scope or intent of a procedure as long as the changes were reviewed by the OC prior to implementation.

  Examples of the use of this practice included Temporary Memos TMA 1997-0097, 0098, 0099, 0101, and 0102, all reviewed by the OC on June 9, 1997, and TMA 1997-0103 reviewed on June 11, 1997. All of the above temporary memos included changes to acceptance criteria for surveillance activities. As noted in the item above, the OC recognized on June 11, 1997, that it could not use the procedure deviation process for changes to acceptance criteria, but it apparently did not identify that the temporary memo process could not be used either since the OC reviewed such a temporary memo later the same day.

Although the practice was in conflict with 5AWI 1.5.1 and 1.5.4, OC review was accomplished before implementation of procedure changes which changed the scope or intent of the procedure so TS requirements were met. However, the finding indicated a weakness in the OC members' understanding of the provisions of the administrative work instructions.

- TS 6.2.B.4.c required OC review of proposals which would effect permanent changes to normal and emergency operating procedures and any other proposed changes or procedures that will affect nuclear safety as determined by the plant manager. The inspectors noted that the OC review of most procedure revisions was extremely superficial. For almost all revised procedures, the committee members did not actually see the procedures. The responsible supervisor brought one copy of the procedure to the meeting and discussed the changes that were being made. Those discussions were usually very brief.

  In one meeting during the inspection period, the supervisor simply said, "We made some changes to the system so we had to change the procedure." No questions were asked by the rest of the committee and that completed the entire extent of OC review of that particular revision. For the June 11, 1997, OC meeting a total of 34 procedure revisions were reviewed in less than 10 minutes. For only one of the revisions was there any questions or discussion by committee members and none of the procedure revisions were seen by the committee as a whole.

  At one point during those procedure reviews, the acting chairman had to ask two of the members to stop an unrelated discussion so they could listen to the proceedings and the inspectors noted that another of the members appeared to be reading other agenda material during the procedure reviews and not paying attention. The inspectors' observations indicated that it was a typical meeting. The superficial review of the 34 procedure revisions by the OC was considered an example of a violation of TS 6.2.B. (50-282(306)/97011-03a(DRP))

  In other sections of this report, several examples of problems with procedures were discussed. Most of those procedures had received OC review, indicating that this review had not been detailed enough to identify the problems.

- The inspectors noted that the committee did not always complete periodic reviews of procedures required by TS 6.2.B.4.h in a timely manner. For example, the agenda for the May 21, 1997, meeting included a notice that biannual reviews for 24 procedures were overdue. The inspectors noted that the notice was typical of the number of procedures overdue. The inspectors did not identify a case where a procedure overdue for biannual review was actually used (many of them were infrequently performed

surveillances), but included on the list were some abnormal operating procedures and annunciator response procedures that could have been needed at any time.

The inspectors had discussed this issue with the OC chairman approximately a year earlier and, although some improvements were made in timeliness, the situation was not adequately corrected. On the agenda for the June 25, 1997, OC meeting at the end of the inspection period, 10 procedures were listed as overdue for review, but the list also included procedures that would become overdue before the end of the month so timely action could be taken.

Failure of the OC to review the 24 procedures within the biannual review period was considered an example of a violation of TS 6.2.B. (50-282(306)/97011-03b(DRP))

- TS 6.2.B.1 specified that no more than two alternate members shall participate as voting members of the OC at any one time. For the April 26, 1997, meeting three alternates were in attendance. Only two of the alternates were necessary to meet the quorum requirements, but there was no indication at the meeting or in the minutes regarding which two of the three alternates were considered voting members.

  In reality, formal votes of the committee were extremely rare and decisions were reached by consensus with all members (and alternates) present encouraged to participate. "Voting" was accomplished by the chairman asking if anyone had any objections to the proposed action. While the participation of as many members as possible was encouraged, the TS requirements for no more than two alternates voting should have been clearly implemented for "official" OC actions. For example, in the April 26 meeting the OC reviewed Temporary Memo TMA 1997-0063 which implemented a procedure change to Surveillance Procedure SP 1102, "11 Turbine Driven Auxiliary Feedwater Pump Test."

  Failure of the OC to insure that no more than two alternates participated as voting members is considered an example of a violation of TS 6.2.B. (50-282(306)/97011-03(DRP))

- At an OC meeting on June 16, 1997, the inspectors observed discussion of a proposed Temporary Memo (TM) identifying a change to operating procedure 1C1.2, "Unit 1 Startup Procedure," Revision 17. The purpose of the TM was to minimize the time spent at very low power after a main feedwater pump was placed in service to avoid exceeding maximum steam generator level limitations at low power. The change to the procedure would have directed operators to increase reactor power to approximately 6 percent prior to placing the AFW system in its safeguards alignment. No OC member objected to the TM and it was considered acceptable pending completion of a 10 CFR 50.59 screening at the request of the OC chairman.

Following the meeting, the inspectors discussed a concern with the engineer who prepared the TM. During the engineer's 10 CFR 50.59 screening preparation, followup of the inspector's concern, and followup of additional concerns raised by the general superintendent of engineering (the acting OC chairman), he identified that the proposed TM would have resulted in a violation of TS 3.4.B.1.c. The AFW discharge valves were throttled to maintain steam generator level, but the TS required those valves to be operable at greater than or equal to 2 percent power. The licensee did not implement the TM and used the procedure as written.

In this case the OC review of the procedure revision was not actually completed because the chairman asked the engineer to conduct a 10 CFR 50.59 screening. However, the performance of the OC was weak in that none of the members identified that the proposed change would have resulted in violating TS.

c. Conclusions

The inspectors identified examples where OC review of procedures were superficial and not timely, and where the OC did not always ensure that the number of voting members met the TS requirements. Other weaknesses were identified in that the OC did not always follow the administrative work instructions for temporary procedure changes, did not recognize that one proposed change would have resulted in a violation of TS, and did not aggressively pursue an operability issue in one case. At the end of the inspection period, the plant manager (OC chairman) was in the process of developing expectations and improvements in those areas.

O8 Miscellaneous Operations Issues (92700)

O8.1 (Open) LER 50-306/97003: Auto-start of 22 Turbine Driven Auxiliary Feedwater Pump on Undervoltage Signal and Entry into Technical Specification (TS) 3.0.C for the Inoperability of Reactor Coolant Pumps When Buses 21 and 22 Were De-energized. This LER discussed two separate events resulting from the need to temporarily de-energize buses 21 and 22 to transfer their power sources on May 17, 1997. The auto-start of the auxiliary feedwater pumps was discussed in Section M3.1 of this report and was considered an example of a violation for an instruction of a type inappropriate for the circumstances.

The entry into TS 3.0.C when both reactor coolant pumps were secured was unavoidable and properly reported. However, operations management at first questioned whether there was an entry into TS 3.0.C since the pumps were only without power for 34 minutes and TS 3.1.A.1.b.(1) allowed both reactor coolant pumps to be shutdown for up to one hour under certain conditions. The inspectors pointed out that the basis for a similar specification in the Improved Standardized Technical Specifications (ISTS) clearly indicated that the one-hour allowance was only to permit performance of designated special tests and that there was a distinction between the pumps being shutdown but available, and being inoperable due to loss of electrical power. Licensee management then agreed that the plant

had unavoidably been in a condition prohibited by TS and under the TS 3.0.C action requirements.

The LER will remain open pending licensee completion and inspector review of the corrective actions described therein. A further discussion of the instruction which resulted in the TS 3.0.C entry is found in Section M3.1 of this report.

O8.2 (Open) LER 50-282(306)/97007(DRP): Both Trains of Spent Fuel Pool Ventilation Inoperable While Handling Load Over Spent Fuel. While reviewing the implications of a Quality Services report, Prairie Island management determined that when fuel handling operations were in progress in the spent fuel pool, both trains of the spent fuel special ventilation system had been momentarily inoperable whenever plant personnel opened one of the personnel doors to gain entry into the spent fuel pool enclosure. The ventilation system became operable almost immediately when the access door was shut. As reported in the LER, there were numerous occasions where spent fuel or other loads were being handled while the doors were opened. This condition was in violation of TS 3.8.D.3 which stated: "With both trains of the spent fuel pool special ventilation system inoperable, suspend all fuel handling operations and crane operations with loads over spent fuel (inside the spent fuel pool enclosure)."

As an interim corrective action, the licensee intended to put procedure changes in place and conduct special training to require that all doors be closed when handling fuel or crane operations with loads over spent fuel occur. As a long-term permanent corrective action, TS 3.8.D will be revised to recognize the fact that personnel can momentarily have the personnel door open and the ventilation system can be inoperable if the requirements of NUREG-0612 for handling heavy loads are applied. The licensee submitted a License Amendment Request for that purpose on May 7, 1997. The LER will remain open pending completion of the corrective actions.

When fuel handling or handling of loads over spent fuel has occurred in the past and a personnel door was opened, a violation of TS 3.8.D had occurred. 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(306)/97011-02(DRP))

II. Maintenance

M1 Conduct of Maintenance

M1.1 General Comments

a. Inspection Scope (61726, 62707, 92902)

The inspectors observed all or portions of the following maintenance and surveillance activities. Included in the inspection was a review of the surveillance

procedures (SPs), test procedures (TPs), or work orders (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 except where noted.

•	SP 1101	12 Motor Driven Auxiliary Feedwater Pump Once Every Refueling Outage, Revision 28
•	SP 1103	11 Turbine Driven Auxiliary Feedwater Pump Once Every Refueling Shutdown, Revision 26
•	SP 1116	Monthly Power Distribution Map Unit 1, Revision 23
•	SP 1198	Nuclear Instrumentation System Power Range Startup Test, Revision 12
•	SP 1250	Test of the Reactor Trip Breakers Using the Main Control Board Switches, Revision 9
•	SP 1319	Rod Position Verification - Unit 1, Revision 4
•	SP 1332	Safe Shutdown Emergency Light Verification, Revision 2
•	SP 1376	Auxiliary Feedwater Pump Technical Specification Flow Verification Test, Revision 2
•	SP 2319	Rod Position Verification - Unit 2, Revision 5
•	TP 1546	Control Rod Drive Mechanism Timing Test, Revision 13
•	WO 9704346	Investigate Unit 2 Reactor Coolant Drain Tank Level Increase
•	WO 9704408	Rod Drop Testing on Unit 2
•	WO 9704425	Transfer Buses 21 and 22 from 2RX to 1RX Transformer
•	WO 9704458	Cycle CV-31084 (11 Steam Generator Power Operated Relief Valve)
•	WO 9704505	Diagnostic Testing on CV-31084
•	WO 9704634	Troubleshoot and Repair Cause of Rod Control Power Cabinet Power Supply Failure
•	WO 9704730	Remove Missile Shield Instrumentation
•	WO 9704741	Meggar all Rod Control Coils from Rod Drive Room
•	WO 9704670	Investigate Stationary Gripper Coil Cable to Rod H8

b. Observations and Findings

For all of the work observed procedures were properly used and followed. Good prejob briefs were conducted for all except one item discussed below. Proper safety precautions were followed for all except one job discussed below. Work was properly planned and scheduled, and proper permission was received prior to its execution. The inspectors determined that several of the procedures observed were inappropriate for the circumstances or had other weaknesses as discussed in Section M3.1 of this report.

- For SP 1103, "11 Turbine Driven Auxiliary Feedwater Pump Once Every Refueling Shutdown," Revision 26, on June 17, 1997, the inspectors noted that the prejob brief did not outline the broad sequence of the surveillance

procedure or specific steam generator level restrictions below 40 percent power.

During the performance of the SP with the reactor critical at 33 percent power, the water level in steam generator A increased beyond the main steam line break analyzed limit of 49 percent narrow range (actual level increased to 49.9 percent). This resulted in the licensee making a one hour notification to the NRC in accordance with 10 CFR 50.72 for operating in a condition outside of the design basis. Had the prejob brief discussed the level limits more thoroughly the operators might have decided to perform the test above 40 percent power where the level restrictions were not applicable. That is what they eventually decided to do when they reperformed the test later.

The licensee retracted the 10 CFR 50.72 notification report on June 18, 1997, when the MSLB analysis for the specific plant conditions at the time of the event showed that Unit 1 had not been operating outside of design basis conditions.

- For WO 9704730, "Remove Missile Shield Instrumentation," conducted on June 6, 1997, which implemented portions of Special Operations Procedure D3, "Reactor Vessel Head Removal," Revision 36, the inspectors observed electricians working on a very narrow section of the refueling cavity ledge without wearing safety harnesses. The inspectors pointed this out to the electrical supervisor who took actions to correct the situation.

- For WO 9704634, "Troubleshoot and Repair Cause of Rod Control Power cabinet Power Supply Failure," the licensee traced the problem that had caused the Unit 1 trip to a grounded cable connector at the top of the H8 control rod drive mechanism. The ground had apparently caused at least one of the control rod drive power cabinets to lose power resulting in a drop of the control rods powered from that cabinet. That quickly led to the negative rate trip of all control rods. Troubleshooting and repair of the problem was methodical, as found conditions were carefully documented, and detailed post repair tests were conducted to identify any other potential problems.

c. Conclusions

The majority of inspector-observed maintenance and surveillance activities were well conducted with good communications, job planning, work practices, and coordination between departments. Two work problems were noted above and several procedure problems are discussed in Section M3.1 of this report.

M3 Maintenance Procedures and Documentation

M3.1 Adequacy and Accuracy of Plant Procedures

a. Inspection Scope (92902)

In the course of monitoring plant operations, maintenance, and surveillance activities, the inspectors reviewed the associated procedures to determine if they were adequately detailed and accurate.

b. Observations and Findings

The following procedures were determined to be of a type inappropriate for the circumstances:

- The inspectors noted a procedural inadequacy during execution of SP 1116, "Monthly Power Distribution Map Unit 1," Revision 23, on June 19, 1997. Step 6.3 under "Prerequisites and Initial Considerations" instructed the performer to remove seal table radiation monitor (R7) fuses during flux mapping. R7 is one of three radiation monitors (at least one of which must be operable) required by TS 3.1.C.1. There was no procedural step in SP 1116 instructing the performer to reinstall the fuses and then verify satisfactory operation of the seal table radiation monitor.

  Although the performers actually made sure the fuses were reinstalled, SP 1116 was not appropriate to the circumstances in that it contained no instructions to reinstall the fuses and insure that the R7 monitor was operable at the end of the test. This was considered an example of a violation of 10 CFR 50, Appendix B, Criterion V. (50-282(306)/97011-01e(DRP))

- The inspectors noted a procedural problem during execution of SP 1319, "Rod Position Verification - Unit 1," Revision 4, on June 17, 1997. The SP was performed when a misalignment was noted between individual rod position indications (RPIs) and the rod group positions for two control rods in the D control bank. The purpose of the procedure was to determine if the misalignment was actual or merely a problem with the RPIs. Step 5 under "Special Considerations" stated that if the rod position verification proved that the rods are not misaligned, then the rod shall not be considered misaligned per TS 3.10.E.2 and the RPI shall be considered operable provided the RPI follows rod motion. That was the case with the June 17 performance of the surveillance.

  The inspectors agreed that successful completion of the surveillance could prove that the rods were not misaligned, but questioned whether the RPIs could be considered operable since the very purpose of the surveillance was to prove that the RPIs were reading inaccurately. The inspectors discussed the concern with licensee management who had Nonconformance Report

2010776 initiated to evaluate the situation. The investigation determined that the individual RPIs could be considered operable, but the fact that they were reading inaccurately made the rod deviation monitor inoperable because the licensee would not be assured of being able to detect certain subsequent actual rod position deviations. Thus the procedure should have implemented the requirements of TS 3.10.I.

In this case the licensee conducted a calibration of the RPIs and adjusted their output to be within the alignment specification.

SP 1319 was considered inappropriate to the circumstances because with proven inaccuracies in RPIs, an input to the rod deviation monitor, it did not contain instructions to consider the rod deviation monitor inoperable. This was considered an example of a violation of 10 CFR 50, Appendix B, Criterion V. (50-282(306)/97011-01f(DRP))

- The inspectors noted a procedural problem during execution of SP 1376, "Auxiliary Feedwater Pump Technical Specification Flow Verification Test," Revision 2, on June 16, 1997. Step 1.1 under "Purpose and General Discussion" stated that the test is performed after each cold shutdown and prior to exceeding 10 percent power. Other steps in the procedure result in the discharge motor-operated valves for both of the AFW pumps to both of the steam generators being either throttled or closed simultaneously and under direct administrative control.

  TS 3.4.B 1.c allowed operations while making necessary changes to motor-operated valve positions under direct administrative control but only during startup operation. Startup operation was defined in TS 1.0 as the process of heating up a reactor above 200°F, making it critical, and bringing it up to power operation. Power operation was defined in TS Table TS. 1-1 as any critical operation greater than 2 percent rated thermal power.

  During power operation TS 3.4.B.2 allowed only one AFW pump, system valve, or piping to be inoperable for 72 hours. There was no provision for two pumps or sets of system valves to be inoperable. Thus SP 1376 was not appropriate to the circumstances in that it would have allowed the surveillance to be done at up to 10 percent power but conducting it above 2 percent power would have resulted in a violation of TS. This was considered an example of a violation of 10 CFR 50, Appendix B, Criterion V. (50-282(306)/97011-01g(DRP))

  The test observed was actually conducted at less than 2 percent power so no actual violation of the TS occurred in this case.

- During performance of WO 9704425, "Transfer Buses 21 and 22 from 2RX to 1RX Transformer," on May 17, 1997, an unexpected start of the turbine-driven AFW pump occurred as discussed in Section O1.2 of this report. The licensee determined that the work order instructions were inadequate in that

the writer had failed to recognize that de-energizing both the 21 and 22 buses would result in an automatic start of the AFW pump on undervoltage, by design, if the control switch was in "Shutdown Auto." As discussed in LER 50-306/97003, standing plant operating procedures were consulted during the development of the work order but those procedures were applicable to different plant conditions.

Maintenance work order WO 9704425 instructions were inappropriate to the circumstances in that it failed to prevent an unnecessary actuation of an engineered safety system. This was considered an example of a violation of 10 CFR 50, Appendix B, Criterion V. (50-282(306)/97011-01h(DRP))

- On June 24, 1997, the inspectors reviewed the licensee's performance of Surveillance Procedure SP 1332, "Safe Shutdown Emergency Light Verification," Revision 2, and identified procedural inadequacies. The procedure did not verify that an emergency light was available to illuminate access to the safeguards bus No. 15 room, did not verify that emergency light No. 15 was available to illuminate the safeguards bus No. 15 room, did not verify that emergency light No. 1 was available to illuminate access to the D1 emergency diesel generator room, and did not verify that emergency light No. 95 was available to illuminate access to the D5/D6 emergency diesel generator building. This was considered an example of a violation of 10 CFR 50, Appendix B, Criterion V. (50-282(306)/97011-01i(DRP))

The following additional procedures were determined to contain weaknesses, but not to the extent that they were considered violations of 10 CFR 50, Appendix B, Criterion V:

- The inspectors noted that SP 1103, "11 Turbine Driven Auxiliary Feedwater Pump Once Every Refueling Shutdown," Revision 26, on June 17, 1997, contained a procedural weakness. Step 5.1 (Special Considerations) stated that a minimum steam generator pressure of 825 psig was required for performance of the surveillance. Step 6.1 (Prerequisites and Initial Conditions) stated reactor power must be greater than 15 percent power and less than 98 percent to perform the surveillance. Actually, steam generator pressure normally decreased to less than 825 psig as power was increased above approximately 45 percent reactor power. Thus, Steps 5.1 and 6.1 become inconsistent above approximately 45 percent power.

  In addition, during actual performance of the test at about 33 percent power as discussed in Section M1.1 of this report, operators were not able to maintain steam generator levels within procedure limitations. The limitations were not applicable above 40 percent so the actual power band that the procedure could be successfully performed without steam generator limitations in was about 40-45 percent not 15-98 percent as listed.

- As discussed in Section O8.1 of this report, during performance of WO 9704425, "Transfer Buses 21 and 22 from 2RX to 1RX Transformer," on May 17, 1997, both reactor coolant pumps (RCP) on Unit 2 needed to be secured for a short time. The WO instructions did not correctly identify that the evolution would result in placing the plant in a condition prohibited by TS and entry into the TS 3.0.C general action requirements. Although the condition was unavoidable due to plant equipment problems, the WO should have contained the correct evaluation of the TS requirements.

- On June 11, 1997, the battery system engineer analyzing the results of Surveillance Procedure SP 2336, "22 Battery Semi-Annual Inspection," Revision 3, determined that the average specific gravity of the cells did not meet the acceptance criteria of the procedure. The inspectors determined that SP 2336 had been recorded as successfully completed on May 23, 1997, when the individual cell data was taken. However, TS 4.6.B.3 required that all measurements be recorded and compared with previous data to detect signs of deterioration or need of equalizing charge according to the manufacturer's recommendation.

  The required analysis of the data for signs of deterioration was not completed until June 11, 19 days after the surveillance was recorded as "complete." In this case the analysis was completed within the required surveillance interval grace period so TSs were not violated. At the end of the inspection period a procedure revision was being processed to change the surveillance procedure to insure that the surveillance would not be recorded as complete until the analysis was completed.

  Additional information regarding how the fact that the surveillance did not meet its acceptance criteria was handled by the system engineer and OC is discussed in Section O7.1 of this report.

c. Conclusions

The inspectors identified several examples of procedure problems for maintenance and surveillance activities. Similar findings were discussed in Section O3 of this report for operating procedures. The problems were primarily that procedures were missing relevant information regarding limitations or TS requirements or contained information that was inaccurate or misleading. The limited sampling done during this inspection indicated that the problems were fairly widespread. Several other recent inspection reports also discussed inadequate procedures and it was one of the subjects of the management meeting discussed in Section X2 of this report. The fact that the OC seldom conducted detailed reviews of proposed procedure revisions as discussed in Section O7.1 of this report may have contributed to the problem.

M8 Miscellaneous Maintenance Issues (92700, 92902)

M8.1 (Closed) Inspection Followup Item (IFI) 50-282(306)/97009-02(DRP): Lubricating Oil Sampling and Usage Practices. As discussed in Inspection Report 50-282(306)/97009, the inspectors had a concern regarding the usage of oil by different work groups at the site. The maintenance department filtered oil from barrels in the lubricating oil storage room using a portable filter cart when changing oil in equipment but operations department personnel obtained oil from the barrels without filtering it when adding oil to equipment.

The inspectors discussed this with engineering, operations, and maintenance personnel and were informed that a project was initiated to improve oil cleanliness controls prior to use. The first phase included the procurement of oil filtration equipment, the second phase included the cleanup of the oil storage room, and the third phase was the purchase of several 100-gallon stainless steel tanks which were to be used for storage of filtered oil. In the interim, temporary guidance was provided to operations personnel regarding oil addition practices. The licensee's actions were appropriate in addressing the issue.

M8.2 (Open) LER 50-282(306)/97005: Surveillance Interval Discrepancies with Diesel Fuel Oil Sample Analysis. This LER was issued on May 8, 1997, and described a situation, first identified at the Monticello plant, where surveillance requirements for sampling of diesel fuel oil storage tanks were being recorded as completed when taken but before the results of the analysis were returned from the laboratory. The inspectors reviewed the licensee's corrective action for that finding.

As described in the LER, the licensee identified other surveillances which required offsite analysis and were being recorded as complete before the analysis results were obtained. However, the licensee did not identify all those which needed onsite analysis. As discussed in Section M3.1 of this report, the licensee and inspectors identified that the semi-annual battery inspection was also recorded as complete before the required analysis of trends for degradation was completed. The licensee took additional corrective actions for that problem.

The non-repetitive, licensee identified and corrected violation as described in the LER is being treated as a Non-Cited Violation, consistent with Section VII.B.1 of the NRC Enforcement Policy. (50-282(306)/97011-04(DRP))

III. Engineering

E2 Engineering Support of Facilities and Equipment

E2.1 Review of USAR Commitments (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.

E8 Miscellaneous Engineering Issues (92903)

E8.1 (Closed) IFI 50-282(306)/95013-01(DRP): Discrepancy Between As-Operated Plant and Assumption in the Individual Plant Examination (IPE). As discussed in Inspection Reports 50-282(306)/95013(DRP) and 50-282(306)/96004(DRP), the inspectors identified that the instrument tube hatches leading to the reactor vessel cavity were not being maintained blocked partially open as recommended by the IPE to provide ex-vessel cooling during a severe accident.

During an inspection of the Unit 2 containment during this inspection period, the inspectors identified that the hatches were in a condition were they could potentially become unblocked and closed. The licensee installed improved blocking devices, controlled by safeguards hold tags, on both units to provide assurance that the hatches would remain open during a severe accident. Additionally, the licensee completed a calculation to demonstrate that the hatch configuration provided sufficient water to the reactor vessel cavity. The inspectors reviewed the calculation and the configuration control of the hatches and had no additional concerns.

E8.2 (Closed) IFI 50-282/96004-03(DRP): Discrepancies in Spent Fuel Cooling Operation and Current Licensing Basis. As discussed in Inspection Report 50-282(306)/96004, the inspectors identified that operating procedure C16, "Spent Fuel Cooling System," Revision 21, did not contain administrative controls for spent fuel pool (SFP) cooling equipment configuration that were consistent with the USAR. The inspectors reviewed the current revision of C16 (Revision 24) and verified that appropriate limitations, notes, and instructions had been written to control equipment configuration consistent with the USAR.

The inspectors had identified an issue regarding a step in Emergency Operating Procedures (EOPs) requiring isolation of component cooling water (CCW) to the SFP heat exchangers to prevent exceeding CCW pump design flow rates. The licensee had independently identified this as a follow-on-item (FOI) in its configuration management program and determined that the CCW pumps would perform satisfactorily even if flow to the SFP exchangers was not isolated. However, the licensee identified that the CCW heat exchanger design flow rate was exceeded if the SFP heat exchanger was not isolated.

The CCW heat exchanger vendor was contacted and confirmed that CCW heat exchanger would not be damaged due to operation at the high flow rate if the flow rate was reduced within a couple of hours. The step in the EOPs to isolate the spent fuel heat exchanger accomplished this. The inspectors reviewed the FOI resolution package and had no additional concerns. The above discussion was also consistent with system operations information in the USAR that SFP cooling could be isolated or transferred to the CCW system of the unaffected unit.

IV. Plant Support

R1 Radiological Protection and Chemistry Controls (71750)

During normal resident inspection activities, routine observations were conducted in the areas of radiological protection and chemistry controls using Inspection Procedure 71750. 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 using Inspection Procedure 71750. No discrepancies were noted.

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 using Inspection Procedure 71750. No discrepancies were noted.

F2 Status of Fire Protection Facilities and Equipment

F2.1 Emergency Lighting

a. Inspection Scope (71750, 92904)

During review of a surveillance procedure, the inspectors identified a violation of NRC requirements for emergency lighting.

b. Observations and Findings

On June 24, 1997, the inspectors identified that 8-hour emergency lighting was not provided for access and egress routes to the safeguards bus No. 15 room. Safeguards bus No. 15 was needed for operation of safe shutdown equipment. This was in violation of 10 CFR 50, Appendix R, Section III.J, which required emergency lighting units with at least an 8-hour battery power supply in all areas needed for operation of safe shutdown equipment and in access and egress routes thereto. (50-282/97011-05(DRP))

c. Conclusions

The inspectors concluded that emergency lighting required by NRC regulations was not available for access to an area needed for operation of safe shutdown equipment.

V. Management Meetings

X1 Exit Meeting Summary

The inspectors presented the inspection results to members of the licensee management at the conclusion of the inspection on June 25, 1997. 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.

X2 Management Meeting Summary

NRC and Northern State Power (NSP) management met in Region III on May 20, 1997, to discuss recent operating events, including engineered safety features actuations, intentional entry into TS 3.0.C LCO statements, excessive draining of reactor coolant, unintentional dilution or positive reactivity additions, and procedural adequacy and adherence. NRC management described those events as possible precursors to future problems and NRC enforcement actions.

NSP management described past performance improvement initiatives including pre-job briefs (1994), communications (1995), self-checking (1996), and procedural compliance (1997) and their success. NSP management stated they would continue with Error Reduction Task Force efforts and implementation of recent outside consultant findings. NRC and NSP management agreed to meet again to discuss the status and results of those efforts.

PARTIAL LIST OF PERSONS CONTACTED

Licensee

J. Sorensen, Plant Manager
K. Albrecht, General Superintendent Engineering, Electrical/I&C
T. Amundson, General Superintendent Engineering, Mechanical
J. Goldsmith, General Superintendent Engineering, Generation Services
J. Hill, Manager Quality Services
G. Lenertz, General Superintendent Plant Maintenance
J. Maki, Outage Manager
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 Nonroutine Events at Power Reactor Facilities
IP 92901:	Followup - Operations
IP 92902:	Followup - Maintenance
IP 92903:	Followup - Engineering
IP 92904:	Followup - Plant Support
IP 93702:	Prompt Onsite Followup of Events

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

50-282(306)/97011-01	VIO	Nine Examples of Procedures of a Type not Appropriate to the Circumstances
50-282(306)/97011-02	NCV	Both Trains of Spent Fuel Pool Ventilation Inoperable While Handling Load Over Spent Fuel
50-282(306)/97011-03	VIO	Three Examples of Failure of Operations Committee to Meet TS Requirements
50-282(306)/97011-04	NCV	Surveillance Interval Discrepancies with Diesel Fuel Oil Sample Analysis
50-282/97011-05	VIO	Failure to Provide Safe Shutdown Emergency Lighting for Access and Egress Routes to the Safeguards Bus No. 15 Room

50-282(306)/97005	LER	Surveillance Interval Discrepancies with Diesel Fuel Oil Sample Analysis
50-282(306)/97007	LER	Both Trains of Spent Fuel Pool Ventilation Inoperable While Handling Load Over Spent Fuel
50-282/97008	LER	Unit 1 Reactor Trip Due to Loss of Rod Control System Power
50-306/97003	LER	Auto-start of 22 Turbine-Driven AFW Pump on Undervoltage Signal and Entry into TS 3.0.C for the Inoperability of RCPs When Buses 21 and 22 Were De-energized

Closed

50-282(306)/95013-01	IFI	Discrepancy Between As-Operated Plant and Assumptions in the Individual Plant Examination
50-282/96004-03	IFI	Discrepancies in Spent Fuel Cooling Operation and Current Licensing Basis
50-282(306)/97009-02	IFI	Lubricating Oil Sampling and Usage Practices

LIST OF ACRONYMS USED

AFW	Auxiliary Feedwater
AWI	Administrative Work Instruction
CCW	Component Cooling Water
CFR	Code of Federal Regulations
CV	Control Valve
°F	Degrees Fahrenheit
DRP	Division of Reactor Projects
ERTF	Error Reduction Task Force
FOI	Follow-on-item
gpm	Gallons per Minute
IFI	Inspection Followup Item
IP	Inspection Procedure
IPE	Individual Plant Examination
ISTS	Improved Standardized Technical Specifications
LCO	Limiting Condition for Operation
LER	Licensee Event Report
LPEO	Lead Plant Equipment Operator
MSLB	Main Steam Line Break
MV	Motor-Operated Valve
NRC	Nuclear Regulatory Commission
NSP	Northern States Power Company
OC	Operating Committee
PDR	Public Document Room
psig	Pounds per Square Inch - Gauge
RCS	Reactor Coolant System
RPI	Rod Position Indication
RWST	Refueling Water Storage Tank
SI	Safety Injection
SFP	Spent Fuel Pool
SP	Surveillance Procedure
TM	Temporary Memo
TP	Test Procedure
USAR	Updated Safety Analysis Report
VCT	Volume Control tank
TS	Technical Specification
USAR	Updated Safety Analysis Report
VIO	Violation
WO	Work Order