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Licensee: Northern States Power Company

Facility: Prairie Island Nuclear Generating Plant

Location: 1717 Wakonade Drive East
Welch, MN 55089

Dates: April 12 - May 24, 1996

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EXECUTIVE SUMMARY

Prairie Island Nuclear Generating Plant, Units 1 & 2
NRC Inspection Report 50-282/96-06, 50-306/96-06, 72-10/96-06

This integrated inspection included aspects of licensee operations, engineering, maintenance, and plant support. The report covers a 6-week period of resident inspection, including the results of announced inspections by regional specialists of activities associated with security, emergency preparedness, and general engineering. Temporary Instruction (TI) 2515/118, "Service Water System Operational Performance Inspection," was closed as part of this inspection.

Operations

- Operators responded promptly and effectively to the reactor trip and the subsequent control rod testing and reactor startup were conducted properly. (Section 01.2)

- Operations Committee review of manual compensatory actions for a problem discovered with auxiliary feedwater pumps was thorough. (Section 01.3)

- Activities associated with dry cask storage were performed very well. (Section 01.5)

- Operator performance in implementing and carrying out the flood procedure was good. In addition, performance of the flood preparation surveillance by maintenance and quality services was excellent. However, the need for several document revisions or enhancements were identified by the inspectors and licensee personnel. (Section 03.1)

- The inspectors identified weaknesses in the licensee's implementation of a Technical Specification change and the implementation of the Temporary Memo process. (Section 03.2)

- The operations staff performed well in identifying and compensating for an instrumentation condition that would provide a nonconservative calculation of reactor thermal power. (Section 04.1)

Maintenance

- The inspectors found the work activities observed were conducted in a professional and thorough manner. (Section M1.1)

- Post-maintenance testing of a repair to an instrument air dryer included a procedure deviation in the system restoration that included undocumented valve manipulations. The procedure for controlling post-maintenance testing of the air dryer was inadequate. (Section M1.2)

- Except for the control room, the licensee's program for identification of equipment problems by the use of repair tags had some weaknesses. A significant fraction of the repair tags on equipment in the plant had no associated current work orders. (Section M2.1)

- The licensee performed well at identifying, investigating, and correcting steam generator tube sleeve issues. (Section M8.1)

Engineering

- Multiple examples of procedural noncompliance were identified in the area of reload core design performed by the Nuclear Analysis Department (NAD). Although the elements of design control were established, failure to follow the procedures resulted in inadequate design control. (Section E2.1)

- NAD also failed to notify the NRC of a significant error is the emergency core cooling system accident analysis in a timely manner. (Section E3.1)

- Examples of weaknesses in timely actions to correct identified problems in NAD were also noted. (Section E7.2)

- A self-assessment performed in the area of 10 CFR 50.59 safety evaluations was comprehensive and identified several issues. (Section E8.4)

- A significant improvement was noted in the licensee's screening of spent fuel cask design issues to determine if safety evaluations were needed. (Section E8.5)

Plant Support

- Weaknesses were noted in the implementation of the radiation work permit for spent fuel cask loading. (Section R1.1)

- Performance during the 1996 emergency preparedness exercise was very good. Emergency classifications and notifications to offsite authorities were made in a timely manner. Activities to mitigate the postulated accident were excellent. The licensee's self-assessment of the exercise was very good. (Section P4)

- The inspectors noted that security personnel did not log when the security diesel generator was taken out of service and that pepper spray canisters used by security had expired self lives. Other security equipment observed was well maintained and performed its function as designed. (Section S2)

- No performance deficiencies were noted with security personnel observed but a minor adverse trend was noted in security incidents caused by security personnel. (Section S4)

- A deficiency was noted in documenting uncorrected vision testing for security personnel. (Section S5)

- The annual Quality Assurance audit of the security program was excellent in scope and well documented. In addition, security had developed an effective method for tracking self-identified findings. (Section S7)

- The inspectors identified an undocumented cable running through 10 CFR 50, Appendix R rated fire barriers and a related minor modification of a fire door. Weaknesses were also evident in that the licensee's daily inspection of the fire door did not identify the concern. (Section F2.1)

Report Details

Summary of Plant Status

Unit 1 operated at or near full power for the entire inspection period except for a power reduction to about 50% on May 18-19, 1996, to facilitate repairs to a feedwater heater drain valve and to investigate possible fouling of a main feedwater pump heat exchanger.

Unit 2 operated at or near full power until a reactor trip occurred on April 18, 1996. The plant was restarted on April 19 and operated at or near full power for the remainder of the inspection period.

During this period the fourth dry cask was inspected, loaded with spent fuel assemblies, and activities continued to prepare it for transport to the Independent Spent Fuel Storage Installation.

I. Operations

O1 Conduct of Operations

O1.1 General Comments (71707)

Using Inspection Procedure 71707, the inspectors conducted frequent reviews of plant operations. In general, the conduct of operations was acceptable; specific events and noteworthy observations are detailed in the sections below.

O1.2 Reactor Trip on Unit 2 Due to Loss of Instrument Air

a. Inspection Scope (93702)

On April 18, with Unit 2 operating at 100% of rated power, control room operators received alarms and instrument indications of a problem with low instrument air pressure. Operators were directed to go to the instrument air compressors where they noted that the 121 instrument air dryer was blowing down continuously. Operators attempted to correct the problem but were not in time to prevent the air operated feedwater regulating valves from drifting closed, causing a loss of feedwater to both Unit 2 steam generators. The reactor tripped approximately three minutes after the first indication of trouble on 22 Steam Generator Low Low Level.

The inspectors responded to the event and observed the station personnel's response to the trip. The inspectors also monitored the station staff's investigation of the cause of the trip and subsequent repairs.

b. Observations and Findings

The licensee determined that the cause of the event was a stuck open purge valve on one of the drying chambers of the 121 air dryer. An identical event occurred on February 13, 1996, as discussed in Inspection Report 96-02, Section 1.3. In that case operators were able to respond in time to prevent a reactor trip. Details of the specific cause and licensee's corrective actions for the April 18 event are contained in Section M1.2 of this report.

Operators responded properly to the trip. There were no other significant equipment problems associated with the trip and all control rods fully inserted. Steam generator levels were recovered using the auxiliary feedwater pumps and the plant was maintained in hot shutdown until repairs were completed. NRC notification according to 10 CFR 50.72 was completed in a timely manner.

The licensee elected to perform control rod drop time testing according to NRC Bulletin 96-01, "Control Rod Insertion Problems," during the shutdown period. The inspectors observed the testing and noted good communications and positive control of plant status during the testing. Because the reactor needed to be borated to maintain sufficient shutdown margin for the withdrawal of all control rods for testing, reactor startup following the repairs to the instrument air dryer was delayed somewhat due to the time needed to dilute out the boron after the test.

The reactor startup was conducted without incident on April 19 and the unit returned to full power on April 20.

c. Conclusions

Operators responded promptly and effectively to the reactor trip and the subsequent control rod testing and reactor startup were conducted properly.

O1.3 All Auxiliary Feedwater (AFW) Pumps Declared Inoperable

a. Inspection Scope (93702)

On May 20, 1996, the licensee declared all four AFW pumps (two per unit) inoperable. This placed both units in a condition prohibited by Technical Specifications and required that within one hour the licensee initiate actions to place both units in at least hot shutdown within the next six hours and below 350 degrees reactor coolant system average temperature within the following six hours according to Technical Specification 3.0.C. The licensee informed the inspectors of the condition and the inspectors closely monitored the licensee's compensatory actions. The licensee also promptly informed the NRC through the emergency notification system according to 10 CFR 50.72. Actions were completed in time to allow the pumps to be declared operable before any actual power reduction.

b. Observations and Findings

The licensee had recently completed a self-assessment of their implementation of the 10 CFR 50.59 safety evaluation program. One of the findings of the assessment questioned the appropriateness of the auxiliary feedwater pump low discharge pressure trip setpoints. The trips were intended to prevent the pumps from being damaged due to cavitation while pumping at runout conditions due to a secondary system break. However, the manufacture recommended the pumps not be run at less than 10% below the minimum total developed head of about 887 psig but the licensee actually set the low pressure trips at 500 psig for the motor-driven AFW pump and 200 psig for the turbine-driven AFW pump. Those setpoints were apparently established in the early 1980s. The licensee could find no calculation or vendor recommendation to justify those setpoints.

Licensee engineering personnel were concerned that certain intermediate sized secondary system breaks could result in the AFW pumps running for a period with discharge pressure below the point where cavitation would begin but above the trip setpoint. Although the secondary break accident analysis assumed that the faulted steam generator would be isolated within ten minutes, there was no assurance the AFW pumps could run in a cavitating condition for that period without damage.

As an interim compensatory measure, the licensee implemented Special Order SO-241 to require that AFW pump discharge pressure be maintained greater than or equal to 900 psig whenever the pumps are running by throttling the discharge motor valves. This order was supported by a temporary change to Operations Section Work Instruction SWI O-2, "Shift Organization, Operation, and Turnover," Revision 31, to add the responsibility of maintaining AFW pump discharge pressure to one of the minimum complement of licensed operators required to be in the control room. A 50.59 safety evaluation was approved by the Operations Committee which reviewed the interim corrective actions and evaluated the manual operator actions against the criteria contained in NRC Generic Letter 91-18 and the associated portions of NRC Inspection Manual Part 9900, Technical Guidance, Section 6.7, "Use of Manual Actions in Place of Automatic Actions."

c. Conclusions

The inspectors noted that the Operations Committee review of this issue was thorough and training was conducted on the manual operator actions for all operating crews. Interviews with selected licensed operators indicated that they understood their responsibilities detailed in the special order. The licensee was reviewing permanent corrective actions as a high priority. The issue will be considered an Unresolved Item pending a review of the enforcement aspects and the corrective actions by regional engineering specialists. (50-282/96006-01)

O1.4 Operators Attempted to Latch Incorrect Fuel Assembly

a. Inspection Scope (40500)

The inspectors conducted a routine review of licensee non-conformance reports to evaluate the licensee's self-assessment and corrective action program. The inspectors selected one non-conformance report for further followup.

b. Observations and Findings

Non-conformance Report 2010423 discussed an event that occurred on March 28, 1996. While moving fuel assemblies within the spent fuel pool for inspections prior to loading fuel into spent fuel casks, an operator attempted to land a fuel handling tool on an incorrect fuel assembly. In this case the operator knew the correct location of the assembly but had difficulty positioning the fuel handling tool and positioned the tool on an assembly in an adjacent row. The operator attempted to land the tool but was unsuccessful because he was using a special handling tool designed for a older model assembly. As a result of attempting to land the tool on a newer model assembly, it became jammed and caused some minor damage to a locktube on the top of the assembly.

Violation 50-282/96002-01 included one example discussed in Section 1.5 of Inspection Report 96-02 that involved operators removing an incorrect fuel assembly from the reactor. Corrective actions for that event included requiring a second verification of the correct fuel assembly after the assembly had been latched. Corrective actions for the event discussed in the violation were not effective in preventing this event, however, because the second verification did not take place until the tool was latched to the assembly. Although that corrective action was reasonable, and would prevent problems in the vast majority of fuel handling evolutions, it was not enough for this case of handling special models of fuel assemblies.

The licensee developed additional corrective actions that included requiring the second verification of proper fuel assembly to take place before landing the fuel handling tool on the assembly. The inspectors verified that the change had been made to Operations Section Work Instruction SWI O-41, "Duties and Responsibilities of Fuel Handling Personnel," Revision 2.

c. Conclusions

The inspectors will verify completion of the remaining corrective actions for this event as part of the closeout of Violation 50-282/96002-01.

O1.5 Spent Fuel Cask Loading Operations

a. Inspection Scope (60855)

The inspectors observed the loading of spent fuel assemblies into the fourth spent fuel storage cask (No. TN 40-04) on May 18, 1996, to ensure that license conditions were met and that corrective actions for fuel handling problems discussed in section O1.4 were implemented. The inspectors also observed or reviewed lid installation, cask draining, vacuum drying, and helium backfilling operations.

b. Observations and Findings

A thorough pre-job briefing was conducted by the spent fuel pool (SFP) system engineer and senior reactor operator in charge of dry cask storage activities. It included discussion of the revised SWI O-41 instruction regarding verification responsibilities of fuel handling personnel. The instruction specified that prior to insertion of a tool or an element in a location, the SFP operator shall observe indexing indicators and the fuel transfer log and receive verification from the SFP supervisor to ensure correct location. Contingency actions were discussed among the fuel handling personnel, control room shift supervisor, and observers in the event of an abnormal event.

Independent samples of SFP water were obtained and analyses performed to verify boron concentration greater than 1800 ppm within 4 hours prior to initiation of cask loading per the Independent Spent Fuel Storage Installation (ISFSI) Technical Specifications. An approved fuel transfer log had been developed by the nuclear engineering department for cask loading and it was highlighted in different colors to make it easier for the fuel handling personnel to read. (This was another corrective action from the refueling outage fuel handling event).

Cask loading was performed by two crews in alternating shifts. Each crew consisted of an SFP operator and an SFP supervisor. The inspectors observed good communications between the operator and supervisor throughout cask loading of horizontal and vertical indexing indicators. Three-way communication was observed for communication of fuel assembly storage locations from the fuel transfer log. However, there was inconsistency in the implementation of the SWI O-41 verification requirement. One crew consistently implemented the instruction for the operator to observe the fuel transfer log and receive verification from the supervisor prior to inserting the handling tool in an assembly. The other crew started this practice but did not continue after transferring several assemblies. The inspectors noted this and reminded the SFP supervisor of the SWI O-41 verification requirement. At that time, the operator and supervisor resumed the practice of both observing the fuel transfer log for verification of fuel assembly location.

Some assemblies selected for cask loading required use of the thimble grip spent fuel handling tool. This tool is rarely used and neither SFP operator had ever used this tool. Prior to the tool being used by either crew, the engineer conducted a briefing on tool use for the fuel handling personnel and they performed a dry run of tool use. The thimble grip handling tool uses mandrels that insert into fuel assembly thimble tubes in order to lift the assembly. The tool has backup fingers that engage the fuel assembly top nozzle if the mandrels slip from the thimble tubes. Contingency actions were discussed if this were to occur during transport of an assembly.

During transport of assembly No. E23 from its SFP location to the cask using the thimble grip handling tool, the assembly slipped from the tool's mandrels and the top nozzle was caught by the backup fingers. The SFP operator, SFP supervisor, and engineer were immediately aware that this had occurred and the SFP supervisor ordered that the assembly be returned to its SFP location. The engineer and SFP supervisor discussed the situation and decided to disengage the tool and attempt to re-engage the tool and lift again. Upon attempting to lift the assembly, the mandrels slipped from the thimble tubes again. The engineer and SFP supervisor decided to not attempt again and properly amended the fuel transfer log to reflect this change. Cask loading proceeded with the next assembly in the transfer log. When all specified fuel moves were complete, the engineer selected an alternate assembly for cask loading that had already been inspected and reviewed for acceptability per the ISFSI technical specifications. The fuel transfer log and cask loading map was revised accordingly and the alternate assembly was loaded into the cask. The engineer and fuel handling personnel performed well in response to this event. A nonconformance report was issued to initiate an evaluation of assembly No. E23.

Lid installation and cask removal from the SFP occurred on May 20, 1996. During cask cavity drying, removal of the remaining water in the cask took an inordinate amount of time. The licensee determined that approximately 180 gallons of water remained in the cask. The probable root cause was air leakage into the pump suction through the non-safety related drain tube fitting during cask draining, resulting in the cask appearing to be completely drained. The licensee performed a calculation to demonstrate that the auxiliary building crane was not overloaded as a result of the weight of excess water in the cask. Also the licensee developed a safety evaluation per 10 CFR 72.48 to demonstrate the acceptability of draining the remaining water out of the cask while it was in the cask decontamination area. The inspectors reviewed the licensee's safety evaluation and procedural changes and attended a meeting of the onsite safety review committee, where this issue was discussed and had no additional concerns. The draining of the remaining water was completed on May 23 and cask drying and helium backfill was completed on May 24.

c. Conclusions

In general, activities associated with dry cask storage were performed very well. However, a concern was raised about the consistency of the implementation of the SWI O-41 verification requirements. The fuel handling verification issue will be addressed in the inspectors' followup of Violation 50-282/96002-01.

O2 Operational Status of Facilities and Equipment

O2.1 Engineering Safety Feature System Walkdowns

a. Inspection Scope (71707, 92903)

The inspectors used Inspection Procedure 71707 to walk down selected portions of the following ESF systems:

- Cooling Water System
- Auxiliary Feedwater System

b. Observations and Findings

Cooling Water System

During a site visit, the NRC Regional Administrator noted that the bolting from the diesel to the foundation for the 12 and 22 cooling water pump engines were different. Bolts on the 12 engine had single nuts and bolts on the 22 engine had double nuts. The issue was brought to the attention of the shift supervisor and system engineer for resolution. The system engineer determined by review of the construction drawing details that single nuts were acceptable. The double nuts were apparently added to the 22 engine after construction but were not required for operability. The inspectors verified by review of drawing NF-38350-19, Revision H, that single nuts were specified.

The inspectors also questioned whether the cooling water diesel exhaust silencers and the exhaust fan penthouses, located on the roof of the screenhouse, were qualified to withstand design basis tornado missiles. Licensee engineering personnel provided a study which demonstrated that similar exhaust silencers and ventilation systems for the D1 and D2 emergency diesel generators were acceptable. The licensee expected that similar results would be obtained for the cooling water diesels and were working on documenting their conclusions. The final conclusions will be reported to the NRC in the Individual Plant Examination for External Events report where they will receive further NRC review.

Auxiliary Feedwater (AFW) System

The inspectors walked down accessible portions of the Unit 1 and Unit 2 AFW systems using drawings and system checklists. During the inspection, one issue that the inspectors reviewed in additional detail was the steam supply piping configuration to the turbine-driven AFW pumps (TDAFW). The inspectors noted that a portion of the steam supply piping traverses compartments in the auxiliary building located on the 695' elevation. The TDAFW steam supply is considered high energy piping per the licensee's high energy line break (HELB) analysis in the Updated Safety Analysis Report (USAR), Appendix I. Equipment necessary to mitigate the consequences of a HELB must be protected from the "harsh" environment that may be created by a HELB. The 695' elevation has been designated as a "mild" area, that will not be affected by a HELB. The inspectors verified by observation of the piping and drawing review that the steam supply piping located in the 695' elevation compartments was encapsulated within guard piping. The guard piping would direct the fluid from a steam supply pipe break to a harsh environment compartment. Therefore, the 695' elevation would remain a mild environment in the event of a TDAFW steam supply HELB.

c. Conclusions

Equipment operability, material condition, and housekeeping were acceptable in all cases. Minor discrepancies were brought to the licensee's attention.

O3 Operations Procedures and Documentation

O3.1 Flood Protection Procedures and Drawings

a. Inspection Scope (71707)

During the spring flood season the inspectors reviewed the licensee's flood protection procedures and observed implementation of them. The inspectors also reviewed design drawings for the flood protection barriers.

b. Observations and Findings

On April 18, 1996, the licensee entered abnormal procedure AB-4, "Flood," Revision 10, in response to a rising river level with a flood crest predicted to occur about a week later. Normal river level is approximately 674.5 feet above mean sea level. On April 18 the river level was 677.8 feet with a predicted crest at 681.3 feet. Actual crest was at about 679.5 feet on April 23.

According to AB-4 for the predicted crest, the licensee ordered fuel oil tanks to be topped off, delayed scheduled 24-hour runs of the emergency diesel generators to conserve fuel oil usage, and took other actions to prepare for the crest. The operating staff frequently contacted the appropriate officials to obtain the latest crest predictions.

The inspectors had the following observations regarding AB-4:

- Section 1 of the procedure stated that "This procedure deals with the actions to be taken for flood levels of 683 feet and higher." and "Preparations for high water levels must be initiated at levels as low as 680 feet." However, Section 4 of the procedure actually contained actions for predicted levels as low as 674 feet (slightly below normal river level).

- The procedure contained no instructions for exiting the procedure as water levels receded. At the end of the inspection period (five weeks after entry into the procedure) the licensee still considered themselves in the procedure even though water levels had receded to within about one foot of normal.

In addition to AB-4, the licensee had surveillance procedure SP 1293, "Flood Preparation - Flood Control Panel Inspection/Installation," Revision 4. That procedure dealt with installing prebuilt flood protection panels over ground level doors into various buildings. The procedure was designed both to be performed prior to a flood and as an annual surveillance to inspect the installation conditions. The inspectors had the following observations regarding SP 1293:

- The annual walkdown of the procedure was performed on May 9, 1996, well after the flood crest. The original due date for the surveillance was June 5, 1996. The previous performance of the surveillance was June 20, 1995, after the 1995 flood season. The purpose of the walkdown might have been better served if the procedure was performed annually a few weeks before the flood season.

- The maintenance worker performing the walkdown identified and documented numerous procedure enhancements such as additional interferences that would have to be removed to install the flood panels.

The inspectors had the following additional comments regarding flood protection documentation:

- Section 2.4 of the Prairie Island Updated Safety Analysis Report (USAR) discussed flood protection design of the plant. Figure 2.4-7, Revision 0, of the USAR showed the locations and mark numbers of the flood protection panels. The inspectors noted that Figure 2.4-7 did not show the D5/D6 emergency diesel generator building and the three flood protection panels associated with it. This was brought to the attention of the appropriate licensee personnel.

- The inspectors noted that Technical Specification 5.1 stated that fourteen doors were provided with flood protection panels. After the addition of the D5/D6 building there were actually seventeen doors. The inspectors notified a member of the licensing staff who informed the inspectors that they had already been aware of the discrepancy as a result of the licensee's service water self-assessment activities.

- Note 7 on drawing NF-117033 stated that all flood panels and the corresponding doors shall be stenciled with the appropriate mark numbers. The inspectors did not observe any mark numbers on any flood panels or doors.

c. Conclusions

Operator performance in implementing and carrying out AB-4 was good. In addition, performance of SP 1293 by maintenance and quality services was excellent. However, the need for several document revisions or enhancements were identified by the inspectors and licensee personnel. Although existing procedures were adequate to provide plant protection in case of a flood, the need for revisions to those documents is considered an Inspection Followup Item. (50-282/96006-02)

O3.2 Procedural Problems with Implementation of Revised Technical Specification

a. Inspection Scope (71707)

While attending a control room shift briefing on April 29, 1996, the inspectors learned that radiation monitor No. 2R51, "Unit 2 loop A main steam line radiation monitor," had failed its surveillance test on April 25, 1996, and was out-of-service. The inspectors reviewed the impact of this condition on plant operations.

b. Observations and Findings

The inspectors discussed the condition with the shift supervisor and referred to Operations Procedure C11, Rev. 14, "Radiation Monitoring System," for specified contingency actions with this monitor out-of-service. Procedure C11, for monitor 2R51 out-of-service, referred to Technical Specification Table 3.15-2 for required actions. However, this table had been removed from the Technical Specifications via a license amendment that was effective on December 17, 1995. The shift supervisor obtained the required response information from other knowledgeable staff. The radiation monitor was repaired and returned to service on April 29, 1996.

The inspectors discussed the licensee's method of implementing procedures required for Technical Specification changes with the secretary of the Technical Specification Change Review Committee. The individual informed the inspectors that Procedure C11 was not revised to support the Technical Specification change due to an oversight. The individual informed the inspectors of the improvements that were being implemented in the way the Review Committee ensures that the plant is prepared to implement revised Technical Specifications, including an improved audit process for identification of necessary training and procedure revisions. A detailed audit report from the Technical Specification Review Committee will be required for presentation to the onsite safety review committee, certifying that the plant is prepared for implementation of revised Technical Specifications.

The licensee informed the inspectors that a Temporary Memo (TM) to Procedure C11 would be issued to correct the error as an interim measure until the procedure was formally revised. On May 3, 1996, TM No. TMA 19960039 was issued and distributed. The inspectors noted that the TM was issued without the required approval of a senior reactor operator and another member of the unit management staff. The inspectors brought this to the attention of the shift manager, who ordered that the TM be retracted and re-issued with the proper approvals. A miscommunication between the TM preparer and support services staff contributed to the error. The inspectors considered this an isolated case.

c. Conclusions

The inspectors identified weaknesses in the licensee's implementation of a Technical Specification change and the implementation of the Temporary Memo process. Actions were in place to ensure that future technical specification changes were thoroughly reviewed for implementation of procedures, training, etc. prior to the effective date of the revision.

04 Operator Knowledge and Performance

04.1 Steam Generator Blowdown Impact on Thermal Power Calculation

a. Inspection Scope (71707)

On May 15, 1996, the inspectors learned of an event at another facility where the licensed thermal power limit was exceeded because steam generator blowdown flow had been isolated without evaluating impact on the calorimetric calculation of reactor thermal power. Unit 2 steam generator blowdown flow was isolated for maintenance on May 15 and the inspectors reviewed the licensee's calorimetric calculation and operations to ensure that licensed power limit was not exceeded.

b. Observations and Findings

The inspectors reviewed the methods of monitoring reactor power in the control room, including use of the emergency response computer system (ERCS) thermal power monitor (TPM) and control board instrumentation, and surveillance procedures SP 2005, Rev. 25, "Unit 2 NIS Power Range Daily Calibration," and SP 2005B, Rev. 7, "Unit 2 Alternate Calculation of Reactor Thermal Power."

The inspectors determined that steam generator blowdown flow was accounted for in the calculation of thermal power. However, the flow instrument for No. 22 steam generator blowdown flow was indicating a flow of 8-12 gpm when the flow path was known to be isolated. This flow rate was provided as an input to the ERCS TPM calculation, resulting in a nonconservative calculated value of thermal power. The inspectors learned that the control room operators had identified the erroneous blowdown flow rate indication when blowdown flow was isolated and evaluated its impact on the TPM calculation. The Unit 2 reactor operator understood that the TPM calculation indicated a lower power level than actual and he appropriately managed reactor power to account for the offset between calculated and actual power. The operator later inserted a default value of zero flowrate for No. 22 steam generator blowdown into the TPM calculation parameter list to avoid having to manually convert and compensate for the erroneous blowdown flow indication.

c. Conclusions

The operations staff performed well in identifying and compensating for an instrumentation condition that would provide a nonconservative calculation of reactor thermal power. An overpower condition was avoided. The licensee initiated a corrective action document to address the blowdown flowrate indication error and its impact on the TPM calculation.

O8 Miscellaneous Operations Issues (92700, 92901)

O8.1 (Open) Licensee Event Report (LER) 50-306/96001: Reactor Trip Caused by Failure of Feedwater Regulating Valve. This item was discussed in Inspection Report 96-04, Section 1.2. However, the LER had not been issued for the inspectors' review at that time. The LER was issued on April 18, 1996. As discussed in the LER, the licensee planned some long term corrective actions to improve the performance of the feedwater regulating valves. The LER will remain open pending the inspectors review of those corrective actions.

O8.2 (Open) Inspection Followup Item 50-282/96004-01: Concern With Ambient Noise Level in the Control Room. This item was discussed in Inspection Report 96-04, Section 1.1. During this inspection period the licensee attempted to retrieve noise survey data from testing done in 1984 as discussed in its Detailed Control Room Design Review Summary Report. The licensee was unsuccessful in finding the data. Thus it is not clear whether the noise in excess of the NUREG-0700 guidelines during operation of the special ventilation system has always existed or has developed over time.

The licensee has initiated an effort working with a consultant to attempt to reduce the ventilation noise. This item will remain open pending the inspectors review of the results of that effort.

II. Maintenance

M1 Conduct of Maintenance

M1.1 General Comments

a. Inspection Scope (61726, 62703)

The inspectors observed all or portions of the following maintenance and surveillance activities:

- SP 1054 Turbine Stop, Governor, and Intercept Valve Test
- SP 1293 Flood Preparation - Flood Control Panel Inspection/Installation
- SP 2046 Multiple Rod Drop Testing
- SP 2307 D6 Diesel Generator Fast Start Test
- SP 2335 D6 Diesel Generator 24 Hour Load Test (Partial)
- WO 9604070 Investigate and Repair 121 Instrument Air Dryer
- WO 9604279 Repair 15A Feedwater Heater Drain Valve
- WO 9604003 Repair 21 Component Cooling Water Pump
- WO 9604070 Repair 122 Air Dryer Using PM3510-1-121

b. Observations and Findings

- For SP 1054, the inspectors noted extremely good three-way communications protocols being practiced by the control room and field operators performing the surveillance.

- For SP 1293, the inspectors noted an extremely thorough job by the maintenance worker in performing the walkdown. He documented several recommended improvements to the procedure, primarily additional interferences that would have to be removed to install the flood protection panels.

c. Conclusions

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

M1.2 Instrument Air Dryer Corrective Maintenance

a. Inspection Scope (62703)

The inspectors reviewed and observed maintenance activities associated with the repair and restoration of No. 122 Air Dryer. Failure of a purge exhaust valve resulted in the loss of instrument air pressure that caused the April 18, 1996 reactor trip discussed in Section O1.2.

b. Observations and Findings

The licensee conducted an investigation of the loss of instrument air pressure and determined that one of the dryer purge exhaust valves failed to close during a drying cycle. A loss of air header pressure resulted through the purge exhaust line.

The licensee inspected each pilot operated inlet and exhaust purge valve and its associated solenoid valve on No. 122 air dryer per WO 9604070. Each of the purge valves except the exhaust valve that failed open had been replaced in February 1996 following the instrument air transient that occurred on February 13. The purge exhaust valve was not replaced because of part supply limitations. A replacement was ordered in February. The licensee identified in May 1996 that the purge exhaust valve had galling on its stem and a damaged actuator piston. A replacement valve was obtained and installed on April 19, 1996.

The inspectors observed post-maintenance testing of No. 122 air dryer on April 20, 1996. During post-maintenance testing, miscommunication between the system engineer, turbine building operator, lead reactor operator, and shift supervisor resulted in the control room operators assuming that the air dryer was returned to service prior to completion of testing. A log entry that the system was returned to service was made in the Unit 2 reactor log prior to completion of testing.

When maintenance was complete, the system engineer requested that operations personnel clear the isolation per the instructions in the work order package. The turbine building operator returned the cleared equipment control tags to the control room. Then, the engineer requested that the air dryer manual bypass valve be opened and the air dryer outlet valve be closed while the dryer sequenced through its purge cycle to observe purge valve performance. The documentation of this was limited to a procedure deviation in the WO stating, "Restore 122 Air Dryer per directions of System Engineer." No equipment control tags were generated or WO steps written to document these valve manipulations. The air dryer remained out-of-service during this time and the control room operators assumed that the air dryer had been restored.

After several air dryer purge cycles, the engineer concluded that the system was functioning properly and requested that the turbine building operator restore to system configuration to normal. At this time the operator closed the manual bypass valve. However, the dryer outlet valve was still closed, resulting in a instrument air header low pressure condition. A low pressure alarm was received in the control room, the No. 122 instrument air filter/dryer automatic bypass valve automatically opened, and control room operators manually opened the 11/21 instrument air cross-tie motor-operated isolation valve to restore Unit 2 air header pressure. An outplant operator was dispatched to locally investigate the condition and discovered that the air dryer outlet valve was closed. The operator opened the dryer outlet valve and the remainder of the system was restored to normal status. This event was an unexpected transient of the instrument air system.

c. Conclusions

Maintenance of air dryer was performed successfully. However, the post-maintenance testing included a procedure deviation in the system restoration that included undocumented valve manipulations. At the conclusion of testing, an oversight resulted in inadvertently isolating air flow, resulting in an unexpected instrument air system transient. Licensee Administrative Procedure 5AWI 3.12.4, Rev. 4, "Post-Maintenance Testing," required that tests shall be conducted according to written instructions or formal procedures as appropriate. The procedure for controlling post-maintenance testing of No. 122 air dryer was inadequate. The inspectors discussed this event with the general superintendent of plant operations, who recognized that improvements were needed in post-maintenance testing control and documentation and that they would be made. This licensee-identified and corrected violation is being treated as a Non-Cited Violation, consistent with Section VII.B.1 of the NRC Enforcement Policy. (50-306/96006-03)

M2 Maintenance and Material Condition of Facilities and Equipment

M2.1 Licensee Program for In-plant Identification of Material Condition Deficiencies

a. Inspection Scope (92902)

The inspectors reviewed the licensee's program for identifying equipment in need of repair and evaluated the effectiveness of its program for entering equipment deficiencies into the work control program.

b. Observations and Findings

The licensee's Administrative Work Instruction 5AWI 3.2.1, "Work Control Package Initiation," Section 6.2.3 stated, "To indicate the deficient condition has been noted and a work order has been submitted, a properly filled out WORK REQUESTED tag should be attached to the item." The work requested tag is a small sticker, also called a "repair tag" in the licensees' computerized work control system, and is the basic method of identifying deficient equipment and initiating a work order. The inspectors noticed a few repair tag stickers on equipment in the plant that did not seem to have associated work orders and conducted an audit.

About 60 maintenance requested stickers randomly observed in the plant were checked to see if work orders were outstanding. The inspectors found that 15 of the stickers did not have an associated work order. In a few of the cases, a work order had been written and later either canceled or completed, but in most of the cases the inspector could find no record of a work order ever being written. Repair tags on equipment for which there was no associated work order was a problem because staff members might see a tag, falsely believe that equipment was scheduled for repair, and not raise equipment problems to their supervisor's attention.

In contrast, the inspectors audited most of the stickers located in the control room and found that they were all in the work control system. The inspectors were informed that both operators and the scheduling specialist had periodically conducted their own audits of the control room stickers to insure that work orders were written.

As a result of the inspectors' concerns, the General Superintendent Plant Operations directed operators to check maintenance requested stickers in the plant and compare them to a list he provided of associated work orders. Discrepancies were corrected by either removing the sticker if the work was done or no work order existed, or initiating a work order for conditions that still needed repair. The inspectors noted that the operators were quite effective in correcting the discrepancies, finding most of the same "orphan" stickers that the inspectors had identified, as well as numerous additional ones. The inspectors provided a list of deficiencies not corrected by the operators to the scheduling specialist who ensured that the discrepancies were corrected.

c. Conclusions

Based on the sampling done by the inspectors, except for the control room, the licensee's program for identification of equipment problems by the use of repair tags had some weaknesses. A significant fraction of the repair tags on equipment in the plant had no associated current work orders. The discrepancies were apparently the result of at least three causes; repair tags not being removed when work was completed or canceled, repair tags not being associated with the corresponding work order in the work control data base, or (in a small number of cases) work orders never being written to correct problems noted on repair tags.

The inspectors discussed proposed corrective actions with the Scheduling Specialist. Those actions included seeing if the repair tag number could be made a required entry for opening a work order in the data base and seeing if "repair tag removed" could be a required entry for closing a work order. The Scheduling Specialist also informed the inspectors that sweeps of the plant for "orphan" repair tags had been done in the past and future periodic sweeps would be considered.

M2.2 Inoperability of Unit 1 Containment Hydrogen Monitors

a. Inspection Scope (62703, 61726, 71707, 92903)

The licensee performed surveillance procedure SP 1226B, Rev. 7, "Containment Hydrogen Monitor Quarterly Calibration," on May 14, 1996. After the surveillance was performed, the licensee determined that both trains of the Unit 1 containment hydrogen monitoring system had been inoperable since Unit 1 startup in March 1996 following the refueling outage. The inspectors initiated a review of the circumstances of the condition.

b. Observations and Findings

The containment hydrogen monitor system consists of two channels (trains) of sensors, with two sensors per channel. The control room output is the auctioneered high sensor of each channel. The sensors provide indication of containment hydrogen concentration following an accident. They provide indication and alarm functions only; no automatic actions. In addition to routine calibration during power operations, the sensors are required to be calibrated several times following an accident. Technical Specification 3.15.A requires both channels operable during operating Modes 1 and 2. One channel is allowed inoperable for 30 days or a report to the NRC must be submitted within 14 days. Two channels are allowed inoperable for 72 hours or the unit must be in Mode 3 within the next 6 hours.

The licensee identified anomalies in system performance during SP 1226B. Two sensors failed high following calibration. Upon further investigation, the licensee identified that regulator valves that provide hydrogen calibration gas to the sensors inside the containment building were set too low. This caused the equipment necessary to perform sensor calibration to not function properly and resulted in inaccurate sensor performance.

The licensee determined that calibration gas bottle regulator setpoints were changed during the refueling outage but the as-found settings following the May 14 surveillance test were much lower than what was expected due to the change in setpoint during the refueling outage. The licensee restored the regulator setpoints to a higher value and performed SP 1226B successfully on May 15, 1996.

The licensee evaluated this condition for reportability and determined that the hydrogen monitor system was inoperable since startup from the refueling outage in March 1996. Therefore, the requirements of Technical Specification 3.15.A were not met and a report to the NRC per 10 CFR 50.73 was required.

c. Conclusions

The inspectors will continue their review of this issue during the next inspection and it is considered an Unresolved Item. (50-282/96006-04)

M3 Maintenance Procedures and Documentation

M3.1 Alternate Shutdown Panel Inverter Low-Voltage Shutoff

a. Inspection Scope (92903)

Based on a regional request, the inspectors questioned the licensee about use of inverters on alternate shutdown panels. The concern was whether the inverters contained a high or low voltage cutoff that was inappropriately set.

b. Observations and Findings

The licensee verified that the instrument loops, that supplied indications at the hot (alternate) shutdown panels, were powered by instrument inverters. The licensee further verified that the inverters did not contain either a high or low voltage cutoff. Instead, the inverters contained high voltage trips of the alternating current (AC) input breakers that transferred the load to the direct current battery and output undervoltage transfers to the alternate AC source. The inverters were included in the preventive maintenance program which verified the correct trip and transfer setpoints.

The inspectors reviewed a preventive maintenance procedure and verified that the setpoints were appropriately prescribed. The maintenance procedure also included a load test on the inverter from no load through full load.

c. Conclusions

The inspectors concluded that the licensee's actions were sufficient to ensure that the inverters would function, if called upon.

M8 Miscellaneous Maintenance Issues (92700, 92902)

M8.1 (Opened and Closed) LER 50-282/96007: Degraded Steam Generator Tube Sleeves. This item was discussed in Inspection Report 96-04. However, the LER had not been issued for the inspectors' review at that time. The LER was issued on April 12, 1996. The licensee has performed well at identifying, investigating, and correcting steam generator tube sleeve issues. The technical issues regarding degraded steam generator tube sleeves are the subject of correspondence between the licensee and the NRC Office of Nuclear Reactor Regulation (NRR) and will be resolved by future licensing activities. This LER is closed to avoid unnecessary duplicate tracking of issue resolution.

M8.2 (Closed) Inspection Followup Item 50-282/94002-01; 50-306/94002-01: Availability of Spare Parts for D5 and D6 Emergency Diesel Generators (EDGs). The inspectors reviewed the licensee's ability to procure and obtain spare parts for the D5 and D6 EDGs. The inspectors concluded that the licensee has taken actions to ensure availability of an adequate supply of spare parts from the EDG vendor. This item is closed.

M8.3 (Open) LER 50-306/96002: Reactor Trip caused by Loss of Instrument Air Pressure. This item was discussed in Sections O1.2 and M1.2. The LER remains open pending additional review by the inspectors of the licensee's corrective actions.

III. Engineering

E1 Conduct of Engineering

E1.1 Independent Spent Fuel Storage Installation (ISFSI) Safety Evaluation Activities

a. Inspection Scope (60851)

The inspectors reviewed issues relating to the ISFSI to verify that safety evaluations required by 10 CFR 72.48 were performed as necessary and that the conclusions of the safety evaluations were technically justified.

b. Observations and Findings

The licensee identified the need to perform 10 CFR 72.48 safety evaluations for issues involving cask weight and storage pad design, lid fastener design, fuel assembly basket thermal performance design, and sequence of operations for cask cavity draining. The inspectors reviewed the safety evaluations, and when necessary consulted with other NRC engineers in the NRC headquarters and regional offices, for technical adequacy. The content of the safety evaluations supported the licensee's determinations that no unreviewed safety question, significant increase in occupational exposure, or unreviewed environmental impact existed.

c. Conclusions

The inspectors concluded that the licensee performed well at identifying ISFSI issues that required evaluation per 10 CFR 72.48. The safety evaluations that were prepared accurately characterized the issues and provided acceptable technical bases for their conclusions. An additional discussion on this topic is contained in Section E8.5.

E2 Engineering Support of Facilities and Equipment

E2.1 Peak Clad Temperature Not Performed for Reload Safety Evaluation

a. Inspection Scope (37550)

The inspector examined the engineering support by the Nuclear Analysis Department (NAD) in design activities related to performing reload safety evaluations. The inspection covered LER 50-282/96005, "Peak Clad Temperature Not Performed for Reload Safety Evaluation" (RSE). One violation was identified. The violation was for multiple examples where engineers and reviewers did not comply with procedures (see Section E2.1.b).

b. Observations and Findings

The licensee did not calculate or verify peak clad temperature (PCT) for the main steam line break (MSLB) as part of the design requirements. Without calculating PCT, the licensee did not verify the number of failed fuel rods would meet the 10 CFR 100 requirements. This omission to calculate PCT was for three cycles for the two Prairie Island reactors.

NAD "Policies and Procedure" NAP2.102T Rev. 12 and topical report NSPNAD-8102-A, Rev. 6, "Reload Safety Evaluation Methods for Application to PI Units" invoked four acceptance criteria for the MSLB event. The 3rd acceptance criteria stated: "The maximum clad temperature calculated to occur at the core hot spot must not exceed 2750°F."

In December of 1995, NAD noted that for the MSLB event in the past three RSEs, they had not calculated PCT. These RSEs were for: May of 1994, for unit 1 Cycle 17, February of 1995, for unit 2 Cycle 17, and September of 1995, for unit 1 Cycle 18.

Sections 6.2 and 6.3 of NAP1.001A, Rev. 9, required the design reviewers and verifiers to confirm traceability and correctness of inputs and outputs for all analyses. Although the design control process provided measures to review and verify methodology, inputs, and outputs, the NAD staff did not adhere to procedure NAP1.001A for these analyses. Noncompliance to procedural requirements is a violation of 10 CFR 50 Appendix B, Criterion V. (50-282/96006-05)

Since February 8, 1996, NAD took several corrective measures to evaluate the impact of this event. This included improved RSE procedures and a check list for review, a management-independent task force to examine lessons learned, an evaluation of consultant support, and to have the site staff audit the NAD design process.

On May 7, 1966, the licensee performed the analysis and verified the number of failed rods with PCT greater than 2750°F would meet the design requirements for unit 2 Cycle 17 (current cycle) until the end of cycle. For the unit 1 Cycle 18 (current cycle), the licensee performed the analysis and verified the same for up to 13 GWD/MTU (expected to reach in February 1997). The licensee did not perform an analysis for the third cycle, unit 1 Cycle 17, as the cycle ended shortly after this problem was verified.

c. Conclusions

Multiple examples of procedural noncompliance were identified in the area of reload core design. Although the elements of design control were established, failure to follow the procedures resulted in inadequate design control. There appeared a need for improvements in procedural compliance within the NAD organization. The inspector noted this recognition among the NAD management staff.

E3 Engineering Procedures and Documentation

E3.1 Failure to Make Timely Report of Significant Error in Emergency Core Cooling System (ECCS) Evaluation Model

a. Inspection Scope (92903)

The inspectors reviewed a licensee-identified failure to follow an NRC requirement associated with reporting.

b. Observations and Findings

On May 1, 1996, the licensee submitted its annual report of corrections to the ECCS evaluation models for the Prairie Island units according to 10 CFR 50.46(a) (3) (ii). In its submittal the licensee reported that one of the corrections discussed in the report met the criteria of a "significant change" as discussed in 10 CFR 50.46(a) (3) (ii) and should have been reported within 30 days. The error was reported to the licensee by a Westinghouse letter dated February 20, 1996. However, correcting the error resulted in a significant decrease in the calculated peak clad temperature and therefore had no safety significance. As discussed in the licensee's report, corrective action has been taken to ensure timely submittals in the future.

c. Conclusions

Failure to report a significant change to the ECCS evaluation model within 30 days was a violation of 10 CFR 50.46. This failure constitutes a violation of minor significance and is being treated as a Non-Cited Violation, consistent with Section IV of the NRC Enforcement Policy. (50-282/96006-06)

E6 Engineering Organization and Administration

E6.1 Organization Within the Nuclear Analysis Department

a. Inspection Scope (36800)

The inspector evaluated the NAD organization. This evaluation was to further examine the conditions contributing to the above mentioned procedural noncompliance (see Section E2.1.b).

b. Observations and Findings

The inspector noted that there were two process groups, Monticello, and Prairie Island, each having a manager. Each manager had about eight engineers and associates as direct reports. However, the group separation was mainly for performance evaluation. The technical work was performed by any engineer reporting to either of the two managers. Design documents for Monticello plant could be generated by an engineer reporting to the manager of the Prairie Island group and approved by either manager.

c. Conclusions

The inspector noted that a more clear distinction in areas of responsibilities between the two reactor groups could further reduce the potential for errors in the design analyses of the reload cores.

E7 Quality Assurance in Engineering Activities

E7.1 Quality Assurance Within the Nuclear Analysis Department

a. Inspection Scope (35740)

The inspection examined the quality assurance program within the NAD organization. This evaluation was to examine its effectiveness in mitigating events and scenarios similar to the events related to the LER 96005 (see Section E2.1.b).

b. Observations and Findings

The inspector noted that a member of the NAD staff had the line function QA responsibility. However, the individual directly reported to a process manager, made recommendations to that manager, and his findings were subject to the manager's approval. In the case of the event identified in the LER 96005, it was designated as a level 2 in the Assessment Form. The QA specialist indicated a level 1 could have been more appropriate. (The higher deficiency potentially could receive greater scrutiny in evaluation process.)

The inspector also found that audits by site QA organization were performed on a frequency of once every two years. The inspector reviewed several audit reports. The inspector discussed some of the findings and the rigor of the audits with the site QA personnel.

c. Conclusion

Although having a QA person to support a line organization is a good practice, the inspector considered that having an independent QA staff within the NAD organization would be more effective.

E7.2 Corrective Actions Within the NAD Organization

a. Inspection Scope (35741)

This inspection evaluated the corrective action program within the NAD to assess timely implementation of the program and its effectiveness (see Section E2.1.b).

b. Observations and Findings

The inspector noted the Follow-on Item FOI-A0760 was generated on January 21, 1993. This item pertained to failure to document the justification for not calculating the PCT (see Section E2.1.b). However, this item was still open as of May 24, 1996. The attempt to close this item led to the discovery of missing the calculation for the PCT in the MSLB analyses. The inspector considered this as one example of untimely corrective action.

The second example pertained to another event. The Assessment Number 95.010 dated April 26, 1995, documented a need to correct an error in calculation of weight of the uranium fuel in the N3P version of N3P93252 code. This concern was first identified in 1991 and as of May 24, 1996, the code was still in error. However, the error was manually adjusted in prior cycle analyses because of the awareness of the design engineer of this error.

c. Conclusions

The inspector considered the above two examples as weaknesses in timely correcting identified problems. In the second example, the inspector considered that due to lack of documentation for the adjustment of the weight of the uranium, and not having corrected the code, another design engineer could potentially overlook this adjustment in future analyses.

E8 Miscellaneous Engineering Issues (92700, 92903)

E8.1 (Closed) Unresolved Item 50-282/96002-06: Safety Injection (SI) Flow Test Criteria Not Met. This item was discussed in Inspection Report 96-02 and was open pending review of LER 50-282/96004. The inspectors reviewed the LER, as documented below.

E8.2 (Closed) LER 50-282/96004: High Head Safety Injection Water Flow Rates Outside Technical Specification Limits. As discussed in Inspection Report 96-02, Section 2.6, the licensee discovered during a routine surveillance that the SI differential flows were not within Technical Specification (TS) requirements. The licensee determined that the root cause of this event was the method used to adjust the throttle valves; the discharge pressure and flow readings fluctuated significantly resulting in inaccurate values being used to establish throttle valve positions. In order to correct this problem, the licensee was modifying the procedure to require stabilization of flows following valve manipulations and use of one-minute average flow and pressure computer readings.

The inspectors discussed the method used to obtain the one-minute average flow with the responsible engineer. The inspectors verified that the discharge pressure and flow computer points used were scanned every second, and that a 60-second value was averaged to obtain the flow used. The inspectors deemed that this method would result in accurate readings.

The inspectors also reviewed the licensee's safety consequence analysis of the differential flows being outside the TS-required values. The licensee determined that the actual safety consequences were low because the TS value allowed for some pump degradation, which did not actually exist. The inspectors concurred with the licensee's assessment. The failure to maintain SI differential flows within the values required by TS 4.5.B.3.h.1 is a violation. However, this licensee-identified and corrected violation is being treated as a Non-Cited Violation, consistent with Section VII.B.1 of the NRC Enforcement Policy. (50-282/96006-07)

E8.3 <u>(Closed) Unresolved Item 50-282/96002-08</u>: Inconsistencies in Pipe Rupture Analysis. This issue was discussed in Inspection Report 96-02, Section 3.1, and involved questions concerning the amount of water assumed by the licensee in the internal flooding analysis in Section I.4.4 of the Updated Safety Analysis Report (USAR). The inspectors were informed by the licensee that the issue had previously been identified by the licensee during design basis documentation reviews and was considered a follow-on item for which they had an action plan in place.

During this inspection period the inspectors were updated on the results of that action. The licensee had completed a reevaluation of internal flooding using conservative bounding values of 152,000 gallons of water instantly released for a feedwater line break and 7000 gallons per minute with an unlimited supply for a fire protection line break.

The results of the analysis demonstrated that the conclusions in the USAR regarding protection of equipment required for safe shutdown during those events were still valid. The licensee intended to submit the analysis as a safety evaluation to the Operations Committee and update the USAR in the next regular revision. The inspectors had no additional questions regarding this issue.

E8.4 <u>(Closed) Violation 72-10/95014-04</u>: Safety Evaluation Not Performed for a Design Change to Dry Cask Bolting. This violation concerned the failure to perform a safety evaluation on a design change to the bolts on an independent spent fuel storage installation cask. The corrective actions to this violation are identical to the violation above. As discussed in Section E1.1 of this report, during this inspection period the licensee identified several new issues associated with dry cask design that required safety evaluations. A significant improvement was noted in the licensee's screening of cask design issues.

E8.5 (Closed) Temporary Instruction (TI) 2515/118: Service Water System Operational Performance Inspection. This TI was left open pending resolution of items identified by the licensee's self-assessment performed in accordance with this TI. Of the 43 items that remained open at the end of the self-assessment, the licensee has completed all actions on 24 of them. Five items dealt with the flow model. These items were preliminarily reviewed when the first revision of the flow model was run; a final flow model run using the second revision was in progress. Six items required modifications; these modifications were all in progress. The remaining eight items were in various stages of completion. The inspectors reviewed the status of the items remaining open and determined that none of them involved any safety concerns.

E8.6 (Closed) Unresolved Item 50-282/96002-09; and (Closed) LER 50-282/96005: Peak Clad Temperature Not Performed for Reload Safety Evaluation. On February 8, 1996, the licensee notified the NRC on the issue of not calculating peak clad temperature for the MSLB event (see Section E2). On March 8, 1996, the licensee issued the LER 96005. Based on this inspection, the corrective actions taken and in progress, this LER is closed.

IV. Plant Support

R1 Radiological Protection and Chemistry (RP&C) Controls

R1.1 Radiation Work Permit (RWP) Review and Implementation Problem

a. Inspection Scope (71750)

The inspectors reviewed the radiation protection aspects of spent fuel cask loading.

b. Observations and Findings

The inspectors attended the pre-job briefing for spent fuel cask loading on May 18, 1996. At the briefing, personnel were informed that the RWP for the work activities was RWP No. 9. The inspectors reviewed the RWP prior to entering the auxiliary building and noted that the RWP specified that radiation protection (RP) technician coverage was required during cask loading. The only RP person onsite was the duty shift chemist; which was normal for a Saturday during routine plant operation. No extra RP technician was scheduled to provide coverage of the cask loading activities. The inspectors discussed this with the spent fuel pool (SFP) system engineer prior to the initiation of cask loading. The engineer was unaware of the RWP requirement for RP technician coverage and informed the inspectors that RP technician coverage during cask loading was intended only for the removal of any objects from the SFP. Radiation Protection Department management approval was obtained for the RWP change and the RWP was revised prior to cask loading. The inspectors found the change acceptable. All other radiation monitors and alarms were operable.

c. Conclusions

The inspectors concluded that the RWP as originally written was acceptable, but the RP Department did not ensure that personnel were scheduled to meet the RWP requirements or that the RWP was revised accordingly. Additionally, the inspectors concluded that only a cursory review of the RWP was performed by the plant personnel involved in cask loading, because the requirement for RP technician coverage during cask loading was not recognized by them.

P3 Emergency Preparedness (EP) Procedures and Documentation

P3.1 Review of Exercise Objectives and Scenario (82302)

The inspectors reviewed the 1996 exercise objectives and scenario which arrived in sufficient time before the exercise to permit NRC review. The scenario provided an adequate framework for the exercise and the objectives were appropriately demonstrated in the facilities evaluated by the inspectors.

P4 Staff Knowledge and Performance in EP

P4.1 1996 Evaluated Biennial Emergency Exercise

a. Inspection Scope (82301)

The inspectors evaluated licensee performance in the following emergency response facilities during the 1996 evaluated emergency exercise:

- Control Room Simulator
- Technical Support Center
- Operational Support Center
- Emergency Operations Facility

b. Observations and Findings

b.1 Control Room Simulator (CRS) crew response to the indications and annunciators was timely and correct. The crew rapidly diagnosed plant conditions, used appropriate emergency procedures, classified the events properly, and properly set priorities for their actions.

Operators closely monitored conditions and attempted to anticipate trends and maintain control of the plant rather than simply responding to malfunctions.

Crew teamwork was generally very good; operators made frequent and appropriate recommendations to the shift supervisor, such as the suggestion to use Technical Support Center (TSC) personnel to calculate cold shutdown boron concentration.

CRS log keeping was poor. Upon termination of the exercise at 12:25 p.m., the inspectors observed that the last entry in the reactor operators' log was for 9:30 a.m. and the last entry in the supervisors' log was at 10:13 a.m..

b.2 The Technical Support Center (TSC) was rapidly and efficiently activated. The Emergency Director (ED) provided periodic briefings on facility activation, plant status, and current emergency issues which kept personnel well informed. An informative initial plant public address announcement was made which included the cause of the emergency and instructions for station personnel. Alert notifications from the TSC to offsite authorities and the NRC were made in a timely manner.

Following the scenario earthquake, the ED and principal staff made excellent decisions to perform system walkdowns and send nonessential personnel home. The ED's concern for plant personnel safety was demonstrated by the decision to evacuate the auxiliary building due to rising radiation levels.

The ED and principal staff's emergency response team task determination and prioritization were excellent. Priorities were adjusted as the situation warranted.

The TSC status board writer did a good job of maintaining the event status board and calling attention to changes in significant parameters. However, The emergency work status board did not display needed information such as time of emergency team dispatch and task completion status.

Security performed accountability rapidly and efficiently. The ED properly cautioned TSC personnel not to use the accountability card reader until it had been activated.

The ED and principal staff proactively and continuously reviewed emergency action levels (EALs) and recommendations were made to the Emergency Operations Facility (EOF) for the General Emergency (GE) classification.

b.3 The Operational Support Center (OSC) was activated in a timely manner and functioned well. The OSC coordinator and department coordinators conducted facility briefings at appropriate times. The informational content of the briefings was good.

On two occasions, the OSC Coordinator was unable to ascertain the status of emergency response teams and requested the emergency work status board keeper for their status. The emergency work status board did not always include needed information, such as appropriate task descriptions and the status of ongoing and completed emergency work. Demonstration of appropriate team status and task descriptions on TSC and OSC emergency work status boards will be tracked as Inspection Followup Item. (50-282/96006-09)

Radiation protection personnel provided good control of dosimetry issuance and dose reporting. The OSC contamination control boundary was adequately maintained, except in one case when an individual crossed the step-off pad without frisking. This individual was promptly stopped before any potential contamination could be spread and properly frisked for contamination.

b.4 The EOF was promptly staffed and assumed responsibility for offsite portions of the emergency response within one hour of the Alert declaration. The Emergency Manager (EM) managed the facility well and kept noise to an acceptable level. The EM held facility briefings every 30 minutes and provided an opportunity for support leads to comment on emergency conditions.

The EOF communicators did not initially transmit a protective action recommendation (PAR) change notification form to required locations. Both the Assistant Radiation Protection Support Supervisor (RPSS) and the EOF Communicators had responsibility to fax PAR notifications forms to the State emergency operations centers and other emergency response facilities. The EOF communicators were told a PAR change notification form had been telecopied to offsite agencies by the Assistant RPSS, and they did not transmit the notification form. Clear identification of responsibilities for transmittal of PAR notification forms and demonstration of offsite PAR notifications will be tracked as Inspection Followup Item. (50-282/96006-10)

The EM appropriately coordinated with the Technical Support Supervisor and his staff to evaluate the EALs and classify the GE within ten minutes. However, the EOF staff did not recognize that the EAL for the GE classification was met without needing a safety injection (SI) to occur. This delayed the GE classification by approximately two minutes until the SI occurred. Review of the licensee's actions to evaluate the EAL procedures for clarity and appropriate classification demonstration will be tracked as Inspection Followup Item. (50-282/96006-11)

The RPSS and staff proactively performed dose assessments to evaluate any potential radiological release impact on the public. Also, the staff continuously monitored the weather forecast and wind direction to determine if a change in PARs was needed.

c. Conclusions

The exercise was successful and demonstrated that the onsite emergency plans are adequate and the licensee is capable of implementing them. Overall exercise performance was very good. Emergency classifications and associated notifications to the State, local government, and NRC were made in a timely manner. Post exercise facility critiques involved exercise controllers and participants and were generally very good.

S2 Status of Security Facilities and Equipment

a. Inspection Scope (81700)

The inspector reviewed the condition of security equipment, systems, and facilities.

b. Observation and Findings

During a preventive maintenance surveillance performed on the security diesel on February 14, 1996, the security diesel was placed out-of-service for approximately seven days. The security plan requires the security system to have a continuous backup power supply system for security components. The security department was not advised of the security diesel being placed out-of-service and therefore the incident was not logged in the security event log as required by Section II(B) of Appendix G to 10 CFR Part 73 which requires an act with the potential for reducing the effectiveness of the safeguards system below that committed to in a licensed physical security plan to be recorded within 24 hours after occurrence.

After being advised of this concern, the security department's analysis determined that an alternate source of power for the security equipment could have been provided if emergency power was needed during the period the security diesel was out-of-service.

Shelf life for pepper spray canisters used by the licensee required more effective monitoring. The manufacturer stated that the serviceability of the canisters could not be guaranteed if retained beyond four years from date of manufacture. The majority of the pepper spray canisters used by the security force were manufactured in 1990 and therefore may not have been serviceable. The security staff is ordering replacement pepper spray canisters and will monitor shelf life for the canisters in the future. Resolution of this issue is an Inspection Followup Item. (50-282/96006-12)

Except as noted above, other security equipment observed during the inspection was well maintained and performed its functions as designed. Maintenance support was timely, especially if compensatory measures were required because of security equipment malfunctions.

c. Conclusions

The failure to log the out-of-service condition in the security event log is a violation of Section II(B) of Appendix G to 10 CFR Part 73. The preventive maintenance surveillance procedure will be modified to have steps added to advise the on duty security shift supervisor when the surveillance is being initiated, and for the security supervisor to log the security diesel out-of-service time into the security event log when the surveillance is initiated. This licensee-identified and corrected violation is being treated as a Non-Cited Violation, consistent with Section VII.B.1 of the NRC Enforcement Policy. (50-282/96006-13)

S4 Security and Safeguards Staff Knowledge and Performance

a. Inspection Scope (81700)

The inspector toured various security posts and observed work in progress. Interviews with security officers were conducted to determine if the officers were knowledgeable of post requirements.

b. Observation and Findings

No performance deficiencies were noted during visits to the security posts. Personnel interviewed and observed on post were knowledgeable of the post responsibilities and procedures and performed the tasks according to the procedure requirements.

Security incident logs showed that during the first quarter of 1996, ten security loggable incidents were caused by the security force. Six of the ten incidents related to security force members performance. To a certain degree, each of the incidents involved inattention to detail. The trend was a bit higher than previous quarters.

c. Conclusions

The minor adverse trend in security force performance will be monitored as an Inspection Followup Item to determine if the trend is a precursor to an adverse performance trend or just isolated performance problems that have been corrected. (50-282/96006-14)

S5 Security and Safeguards Staff Training and Qualification

a. Inspection Scope (81700)

The inspector reviewed training and qualification records for newly hired security force personnel.

b. Observation and Findings

The training and qualification records reviewed were complete and accurate except for several cases involving uncorrected vision testing. In several cases, the uncorrected vision testing was not performed or if completed, the results of the testing were not documented. The uncorrected vision testing is performed to determine if a second pair of glasses is required to be maintained onsite as required by the Security Force Training and Qualification Plan. Additionally, the vision testing protocol provided to the off site medical clinic requires testing of uncorrected vision. The error appeared to have been committed by the same physician. It should be noted that in all cases where the uncorrected test results were not recorded, a second pair of glasses were maintained on site for the personnel involved. The security staff is addressing this concern with the medical clinic.

S7 Quality Assurance in Security and Safeguards Activities

a. Inspection Scope (81700)

The inspector reviewed the most recent Quality Assurance audit of the security program, the security performance trend data for the past two quarters, and self assessment findings tracking system.

b. Observation and Findings

The annual Quality Assurance audit of the security program was excellent in scope and well documented. Additionally, the security section had developed an effective method for tracking self-identified findings through use of the security issue log. The log identified the issues, actions taken to address the issues and the completion date. Most issues monitored were resolved in a timely manner.

S8 Miscellaneous Security and Safeguards Issues (92904)

S8.1 (Closed) Inspection Followup Item 50-282/95007-02; 50-306/95007-02: Security Supervisors Failed to Take Proper Action. This issue was addressed in Section 3.b of Inspection Report 97-07 and pertained to an incident in which two security supervisors failed to take timely action when advised that a threat had been made against a security supervisor. Another incident of a security supervisor allegedly making a threat against a security officer prevented closure of the item during the previous inspection. Investigation of both issues have been completed. The investigations concluded that neither of the incidents constituted a significant threat to the personnel involved and both issues were instances of poor judgement rather than lack of trustworthiness and reliability for the individuals involved. The inspector's review of the investigations concluded that they were adequate.

S8.2 (Open) Inspection Followup Item 50-282/95003-01; 50-306/95003-01: Loss of Two Onsite Security Support Positions. This issue was addressed in Section 5.b of Inspection Report 95-03 and pertained to monitoring and evaluating the potential impact of the loss of three contract security support positions to an offsite location. The new personnel assigned to the three positions have just recently assumed these responsibilities. Additionally, a task analysis to objectively determine staffing needs which was initially scheduled to be completed in April 1996 will not be completed until July 1996. Finally, a new concept of the training program is in the formulation stage. For these reasons, this item will remain open.

S8.3 Drug Testing of Licensee Personnel

The licensee recently advised NRC Region III that extensive drug testing of licensee personnel had been conducted because of an allegation the licensee received that some licensee personnel may have abused drugs some years ago. The testing process has been completed. Eighty-two licensee personnel were tested for illegal drugs. Of all those tested, one test results was positive for a prohibited drug.

The licensee's quality assurance staff conducted a surveillance of the drug testing process and initially concluded that some actions required by the NRC Fitness-For-Duty program (10 CFR Part 26) had not been completed. Further review and evaluation by the licensee subsequently concluded that the drug testing was conducted and documented according to the licensee's Company Drug Testing Policy rather than the NRC Fitness-For-Duty program. However, the one individual that was positive for a prohibited drug was tested under the criteria of 10 CFR Part 26. This conclusion appears to be justified. The NRC is continuing to review the issue and the review results, when completed, will be addressed by separate correspondence.

S8.4 Independent Spent Fuel Storage Installation (ISFSI) Visitor Control Records

NRC Region III was advised by the licensee that some security related documents pertaining to visitor control for the ISFSI may have been altered by two security personnel. The licensee's investigation of the alleged record alteration incident and their corrective actions to address the issue were well documented. Conversely, the licensee's investigation pertaining to the actions that lead up to the alleged record alteration (incomplete visitor authorizations) was not as well documented. The NRC is independently reviewing the incident. The review results, when completed, will be addressed by separate correspondence.

F2 Status of Fire Protection Facilities and Equipment

F2.1 Undocumented Cables Running Through Fire Rated Barriers

a. Inspection Scope (71707)

While on a tour of the control room on April 15, 1996, the inspectors observed an improperly installed antenna cable traversing the doorway between the control room and the Operations Support Center (OSC) (fire door No. 128).

b. Observations and Findings

Fire door No. 128 is a 1.5-hour rated fire door and is required to be maintained operable per Procedure F5 Appendix K, Rev. 2, "Fire Detection and Protection Systems." The inspectors observed that the fire door was dented to allow the antenna cable to pass from the OSC to the control room without being pinched and cut by the door and door jam. The inspectors informed the shift manager of the condition and followed the path of the cable. The licensee declared the fire barrier inoperable pending a review by the fire protection engineer to evaluate door integrity and repair. The engineer determined that the door was not inoperable but initiated a work order to remove the cable and repair the dent.

The cable was from a stereo receiver located in the OSC. The cable entered the control room and exited the control room through a cable penetration into the turbine building, where it continued to an antenna located on the roof. The inspectors observed other similar cables in the control room and one passed through a cable penetration from the control room to another stereo receiver located in the operator's lounge.

The licensee initiated a nonconformance report and investigation to inspect the integrity of the cable penetration fire barriers and review the history of the antenna cables. The cable fire barriers met 10CFR 50 Appendix R criteria and were not impaired. The cables passing through the cable penetration fire barriers had apparently been installed many years ago and the licensee identified no record of their installation. It was unknown at what time fire door No. 128 was damaged. The inspectors considered that the licensee had opportunity to identify and correct the degraded condition. Fire door No. 128 is included on a daily inspection that operators perform and log to verify that the door is closed and free of obstructions.

c. Conclusions

Licensee administrative procedure 5AWI 3.13.0, Rev. 1, "Fire Preventive Practices," requires a documented fire protection review of modifications that have the potential to interfere with installed fire protection equipment. The damage to the fire door, although later evaluated to not render the barrier inoperable, was considered to have the potential to interfere with the ability of the door to serve as a fire barrier. This violation constitutes a violation of minor significance and is being treated as a Non-Cited Violation, consistent with Section IV of the NRC Enforcement Policy. (50-282/96006-15)

V. Review of USAR Commitments

A recent discovery of a licensee operating their facility in a manner contrary to the Updated Safety Analysis Report (USAR) description highlighted the need for a special focused review that compares plant practices, procedures, and parameters to the USAR descriptions. While performing the inspections discussed in this report, the inspectors reviewed the applicable portions of the UFSAR that related to the areas inspected. The following inconsistency was noted between the wording of the USAR and the plant practices, procedures, and parameters observed by the inspectors:

As discussed in Section O3.1 c. of this report, Section 2.4 of the Prairie Island USAR discussed flood protection design of the plant. Figure 2.4-7, Revision 0, of the USAR showed the locations and mark numbers of the flood protection panels. The inspectors noted that

Figure 2.4-7 did not show the D5/D6 emergency diesel generator building and the three flood protection panels associated with it. This was brought to the attention of the appropriate licensee personnel. This was a minor editorial problem and was not considered a violation.

VI. 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 May 24, 1996. The licensee acknowledged the findings presented.

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

X3 Management Meeting Summary

On April 17, 1996, Mr. H. Miller, NRC Regional Administrator, and others of his staff, and Mr. J. Howard, Chairman and Chief Executive Officer of Northern States Power Company (NSP), an others of his staff, met at the Prairie Island site for a public management meeting to discuss the latest Systematic Assessment of Licensee Performance (SALP) Report 50-282(306)/96-01. Frank and open discussions regarding licensee performance in each functional area were conducted. The visit also included plant tours by Mr. Miller and other NRC managers as well as meetings with individual NSP employees.

PARTIAL LIST OF PERSONS CONTACTED

Licensee

M. Wadley, Plant Manager
M. Agen, Senior Consultant, Emergency Planning
K. Albrecht, General Superintendent, Engineering
J. Goldsmith, General Superintendent, Design Engineering
J. Hill, Manager, Quality Services
G. Lenertz, General Superintendent, Maintenance
H. Nelson, Process Manager, Nuclear Analysis Department
D. Schuelke, General Superintendent, Radiation Protection and Chemistry
M. Sleigh, Superintendent, Security
J. Sorensen, General Superintendent, Plant Operations

INSPECTION PROCEDURES USED

IP 35740: QA Program (Administration)
IP 35741: QA Program (Audits)
IP 36800: Organization
IP 37550: Engineering
IP 37551: Onsite Engineering
IP 40500: Effectiveness of Licensee Controls in Identifying, Resolving, and Preventing Problems
IP 60851: Design Control of ISFSI Components
IP 60855: Operation of an ISFSI
IP 61726: Surveillance Observations
IP 62703: Maintenance Observations
IP 71707: Plant Operations
IP 71750: Plant Support Activities
IP 81700: Physical Security Program for Power Reactors
IP 82301: Evaluation of Exercises for Power Reactors
IP 82302: Review of Exercise Objectives and Scenarios for Power Reactors
IP 92700: Onsite Followup of Written Reports of Nonroutine Events at Power Reactor Facilities
IP 92901: Followup - Plant Operations
IP 92902: Followup - Maintenance
IP 92903: Followup - Engineering
IP 92904: Followup - Plant Support
IP 93702: Prompt Onsite Response to Events At Operating Power Reactors

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

50-282/96006-01	URI	Auxiliary Feedwater Pump Discharge Pressure Trip Setpoints
50-282/96006-02	IFI	Revisions or Enhancements Needed for Flood Procedures
50-306/96006-03	NCV	Inadequate Control of Post Maintenance Testing
50-282/96006-04	URI	Inoperability of Containment Hydrogen Monitors
50-282/96006-05	VIO	Failure to Follow Procedures in Reload Safety Evaluations
50-282/96006-06	NCV	Failure to Report Significant Error in Accident Analysis
50-282/96006-07	NCV	Safety Injection Flow Test Criteria Not Met
50-282/96006-08	IFI	Safety Evaluation Issues Discovered During Self-Assessment
50-282/96006-09	IFI	Team Status and Task Descriptions on the TSC and OSC Emergency Work Status Boards
50-282/96006-10	IFI	Responsibilities for Transmitting PAR Notification Forms
50-282/96006-11	IFI	Evaluation of EAL Procedures and Classification
50-282/96006-12	IFI	Monitoring Shelf Life of Pepper Spray Canisters
50-282/96006-13	NCV	Failure to Log Security Diesel Out-of-Service
50-282/96006-14	IFI	Minor Adverse Trend in Security Force Performance
50-282/96006-15	NCV	Failure to Document a Fire Protection Review of a Modification
50-306/96001	LER	Reactor Trip Caused by Failure of Feedwater Regulating Valve
50-306/96002	LER	Reactor Trip Caused by Loss of Instrument Air Pressure
50-282/96007	LER	Degraded Steam Generator Tube Sleeves

Closed

50-282/94002-01	IFI	Availability of Spare Parts for D5 and D6 Emergency Diesel Generators
50-306/94002-01	IFI	Availability of Spare Parts for D5 and D6 Emergency Diesel Generators
50-282/95007-02	IFI	Security Supervisors Failed to Take Proper Actions
50-306/95007-02	IFI	Security Supervisors Failed to Take Proper Actions
50-282/96002-06	URI	Safety Injection Flow Test Criteria Not Met
50-282/96002-08	URI	Inconsistencies in Pipe Rupture Analysis
50-282/96002-09	URI	Peak Clad Temperature Analysis Not Performed for Core Reload Design
50-306/96006-03	NCV	Inadequate Control of Post Maintenance Testing
50-282/96006-06	NCV	Failure to Report Significant Error in Accident Analysis
50-282/96006-07	NCV	Safety Injection Flow Test Criteria Not Met
50-282/96006-13	NCV	Failure to Log Security Diesel Out-of-Service
50-282/96006-15	NCV	Failure to Document a Fire Protection Review of a Modification
50-282/96004	LER	High Head Safety Injection Water Flow Rates Outside Technical Specification Limits

50-282/96005	LER	Peak Clad Temperature Analysis Not Performed for Core Reload Design
50-282/96007	LER	Degraded Steam Generator Tube Sleeves
72-10/95014-04	VIO	Safety Evaluation Not Performed for Design Change to Dry Cask Bolting
TI 2515/118	TI	Service Water System Operational Performance Inspection

Discussed

50-282/95003-01	IFI	Loss of Two Onsite Security Support Positions
50-306/95003-01	IFI	Loss of Two Onsite Security Support Positions
50-282/96002-01	VIO	Examples of Failure to Follow Procedures
50-282/96004-01	IFI	Concern With Ambient Noise Level in the Control Room

LIST OF ACRONYMS USED

AC	Alternating Current
AFW	Auxiliary Feedwater
AWI	Administrative Work Instruction
CFR	Code of Federal Regulations
CRS	Control Room Simulator
EAL	Emergency Action Level
ECCS	Emergency Core Cooling System
ED	Emergency Director
EDG	Emergency Diesel Generator
EM	Emergency Manager
EOF	Emergency Operations Facility
EP	Emergency Preparedness
ERCS	Emergency Response Computer System
°F	Degrees Fahrenheit
FOI	Follow-on Item
GE	General Emergency
GPM	Gallons Per Minute
GWD/MTU	Gigawatt Days Per Megawatt Ton
HELB	High Energy Line Break
IFI	Inspection Followup Item
IP	Inspection Procedure
ISFSI	Independent Spent Fuel Storage Installation
LER	Licensee Event Report
MSLB	Main Steam Line Break
NAD	Nuclear Analysis Department
NCV	Non-Cited Violation
NIS	Nuclear Instrumentation System
NRC	Nuclear Regulatory Commission
NRR	Nuclear Reactor Regulation
NSP	Northern States Power Company
OSC	Operational Support Center
PAR	Protective Action Recommendation
PCT	Peak Clad Temperature
PDR	Public Document Room

PR&C	Radiological Protection and Chemistry
PSIG	Pounds Per Square Inch Gauge
QA	Quality Assurance
RP	Radiation Protection
RPSS	Radiation Protection Support Supervisor
RSE	Reload Safety Evaluation
RWP	Radiation Work Permit
SALP	Systematic Assessment of Licensee Performance
SFP	Spent Fuel Pool
SI	Safety Injection
SP	Surveillance Procedure
SWI	Section Work Instruction
TDAFW	Turbine Driven Auxiliary Feedwater
TI	Temporary Instruction
TM	Temporary Memo
TN	Transnuclear, Inc.
TPM	Thermal Power Monitor
TS	Technical Specification
TSC	Technical Support Center
URI	Unresolved Item
USAR	Updated Safety Analysis Report
VIO	Violation
WO	Work Order