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Licensee:	Northern States Power Company
Facility:	Prairie Island Nuclear Generating Plant
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Dates:	October 22 - December 2, 1997
Inspectors:	S. Ray, Senior Resident Inspector P. Krohn, Resident Inspector S. Thomas, Resident Inspector
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EXECUTIVE SUMMARY

Prairie Island Nuclear Generating Plant, Units 1 and 2
NRC Inspection Report No. 50-282/97021(DRP); 50-306/97021(DRP)

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

Operations

- Normal plant operations were conducted without significant problems. Operators were especially prompt and conservative in responding to a missing ladder that was needed for access to several valves in the event of a loss-of-coolant-accident and a faulty breaker for a control room ventilation fan (Section O1.1).

- All operations refueling activities observed were performed well with good pre-evolution briefs, careful execution, and proper procedure adherence. Activities observed included the flooding of the reactor cavity after the reactor head was removed, the unlatching, drag testing, and relatching of control rods, fuel shuffling, and the filling of the reactor vessel after steam generator restoration (Section O1.2).

- All equipment responded as expected to the loss of the 10 bank transformer. Operators properly followed the appropriate annunciator response and abnormal operating procedures to recover from the event. The system engineer led a thorough investigation into the cause (Section O1.3).

Maintenance

- Maintenance, surveillance, and refueling outage activities were performed well with only minor problems. All activities observed were performed safely with proper procedures being used and followed. Activities observed included the disassembly, inspection, and re-assembly of a main steam isolation valve; removal of the reactor internals; and an emergency diesel generator 24-hour load test. System engineer involvement was strong (Section M1.1)

- Unit 1 containment-penetration checklists and procedures contained inconsistencies between procedures and editorial errors within procedures. However, the errors were not significant enough to prevent successful performance of the procedures. Labels specified by one procedure were not all installed and operators who performed the procedure apparently failed to identify or correct the discrepancies (Section M3.1).

- The Unit 2 containment-penetration checklists and procedures demonstrated a much improved quality and consistency as compared to those for Unit 1 (Section M3.2)

Engineering

- System engineers were involved in all aspects of plant operations, refueling, maintenance, and surveillance activities. The engineers rapidly investigated any operational abnormalities, took an active role in maintenance and troubleshooting activities, and closely followed all surveillance testing on their systems (Section E2.2).

- The licensee's discovery that there was no analysis for a dilution accident during shutdown conditions was an excellent finding and indicated a thorough Updated Safety Analysis Report review process (Section E8.6).

Plant Support

- The radiation protection staff performed well in controlling exposures during refueling activities (Section R1).

- The response of the fire brigade, control room operators, and other licensee personnel to a fire in the maintenance shop was good (Section F1.1).

- The licensee's finding that the reactor coolant pump oil collection system was inadequate was a result of a proactive, voluntary review of fire protection issues in preparation for a future NRC pilot inspection. Prompt corrective actions were planned to modify the system to be in compliance with NRC requirements (Section F2.1).

Report Details

Summary of Plant Status

Unit 1 remained shutdown in a refueling outage for the entire inspection period. Unit 2 operated at or near full power for the entire inspection period.

I. Operations

O1 Conduct of Operations

O1.1 General Comments

a. Inspection Scope (71707)

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

b. Observations and Findings

Normal plant operations were conducted without significant problems. The inspectors noted two issues where the response of the operators was especially prompt and conservative.

- On November 3, 1997, the inspectors noted that a ladder, staged for auxiliary building operator manipulation of valves CV-31411, CV-31381, CV-31383, and CV-31384, was missing. Each of those control valves had a mechanical stop designed to limit cooling water flow to the respective component cooling water heat exchanger (CCHX). Throttling cooling water flow through the CCHXs was necessary to manage cooling water loads in case a seismic event caused loss of water level in the plant intake bay. If the emergency core cooling systems needed to be subsequently placed in the recirculation mode of operation during a loss of coolant accident event, the auxiliary building operator was required to remove the mechanical stops and restore full cooling water flow to the CCHXs.

  The ladder provided the auxiliary building operator access to the overhead where CV-31411, CV-31381, CV-31383, and CV-31384 were located. It was normally staged adjacent to CV-31411 with a sign indicating its purpose and instructing workers not to move it. The inspectors found the ladder under the 22 CCHX behind some plywood staged for component cooling system outage work on Unit 1. The inspectors brought the mislaid ladder to the attention of the auxiliary building operator. The auxiliary building operator located the ladder and returned it to the normal storage position. As an additional corrective action, the shift supervisor promptly had temporary scaffolding erected for access to the valves.

The licensee also issued a work order to install permanent platforms for access to the valves and had started work on that project at the end of the inspection period.

- On November 26, 1997, a reactor operator (RO) taking control room logs noted that there was no light indication for the status of the 122 control room cleanup fan. A local check confirmed that the breaker suppling the fan had tripped. The operators reset the breaker and later ran the fan while the system engineer observed it. The system engineer reported that the breaker was defective and initiated a work order for repairs.

  Operators were not sure exactly when the fan breaker had originally tripped, but knew conditions had been normal 6 hours earlier when the previous set of logs was taken. Although it was usually considered acceptable to start a Technical Specification (TS) allowed outage time from when a piece of equipment is first discovered to be inoperable, the operators conservatively decided to consider the fan to have been inoperable since the previous log readings. Plant electricians promptly completed repairs on the breaker and the system was returned to an operable status the next day.

c. Conclusions

All normal operations observed were completed properly. On two occasions, operator actions were especially prompt and conservative.

O1.2 Operations Refueling Activities

a. Inspection Scope (71707)

The inspectors observed significant portions of several operations refueling activities on Unit 1. Major activities observed included flooding the reactor cavity after the reactor head was removed, unlatching control rods, drag testing on selected control rods, fuel shuffling for refueling, control rod relatching, and filling of the reactor vessel after steam generator restoration. Updated Safety Analysis Report Section 10.2.1, "Fuel Storage and Fuel Handling Systems," was reviewed as part of this inspection.

b. Observations and Findings

All operations evolutions observed were carefully performed without significant problems. General comments applicable to the evolutions included good pre-evolution briefings, proper procedures being used and followed, and adequate communications. Noteworthy specific comments are discussed below.

- The inspectors observed the pre-evolution brief for unlatching the control rod drive shafts and performing drag tests on selected control rods prior to reactor vessel upper internals removal. The inspectors noted that the unlatching portion of the brief was very good and that management personnel presented relevant plant specific and industry wide lessons learned regarding this evolution, a thorough briefing of the control rod unlatching procedure was given, and a topic specific video was shown to all personnel involved with the evolution. However, the

inspectors noted that the briefing for the drag testing portion of the evolution was adequate but not as thorough.

The inspectors observed both the unlatching of all of the control rod drive shafts and the drag testing of selected control rods. The inspectors noted that very good control and verification was demonstrated during the entire evolution. Three-party verification occurred prior to the connection of the unlatching tool to any control rod drive shafts for unlatching and/or drag testing. Also, good coordination occurred between personnel in the containment and in the control room during the drag testing to monitor for changes in source range indications.

- Fuel shuffling operations were observed to be completed in a controlled, deliberate manner. Three-party verification was used to verify that the correct fuel assembly was being grappled before the tool was placed on the assembly and that fuel assemblies were about to be placed in the correct locations at the end of the moves. The licensee relaxed the requirement for a nuclear engineer to be one of the individuals performing the verifications and allowed the control room reactor operator to be the third verifier by performing an independent check of the reported location of the manipulator crane against the refueling log procedure.

- The inspectors observed filling of the reactor coolant system (RCS) from the level of the top of the hot legs to one foot below the reactor vessel flange. The prejob brief was thorough and discussed the procedures to be used, participants' responsibilities, communications, equipment involved in the evolution, expected volume control tank (VCT) pressure transients, potential problems (including RCS dilution and overfilling), the expected volume of water to be added to the RCS, and RCS vent paths. The RO took the additional precaution of having other plant operators locally check the vent path and the status of the charging pump to be used during RCS filling.

  The RCS level was raised in a controlled and careful manner. Control room operators checked diverse indications such as boric acid storage tank levels, VCT levels, hot leg ultrasonic level indicator readings, reactor vessel level indicating system readings, and containment local standpipe levels to ensure that RCS inventory was being increased as expected and that water usage corresponded with the level increase.

  During the briefing, the RO discussed the actions that would be taken to add nitrogen to the VCT vapor space if VCT pressure decreased. The RO noted that, although an automatic nitrogen regulator was available to add nitrogen to the VCT, no procedure existed to add nitrogen to the VCT automatically. The RO instead referenced and decided to use a procedure describing manual addition of nitrogen to the VCT (Operating Procedure C12.4, "VCT Gas Control," Section 5.5, "Raising VCT Nitrogen Pressure," Revision 3). After filling the RCS to 1 foot below the reactor vessel flange was complete, the RO submitted a request for a procedure to automatically add nitrogen to the VCT.

- The inspectors observed the pre-evolution brief for latching of the control rod drive shafts and drag testing of all control rods. The brief thoroughly covered all aspects of the latching procedure, lessons learned from past control rod latching evolutions, control rod drag testing, and specific instructions on the operation of special tools and lifting devices required by this evolution.

  The inspectors observed that the evolution was conducted in a slow and controlled manner. The inspectors noted that three-party verification was utilized prior to placing the latching tool onto any control rod drive shaft and that good coordination was exercised between personnel conducting the evolution in containment and those monitoring the evolution from the control room.

c. Conclusions

All operations refueling activities observed were performed well with good pre-evolution briefs, careful execution, and proper procedure adherence.

O1.3 Loss of One Offsite Power Source Transformer

a. Inspection Scope (93702)

On November 17, 1997, the sudden pressure relay on the 10 bank switchyard transformer actuated, causing the transformer to automatically isolate. At the time, the 10 bank transformer was supplying safeguards bus 26, on Unit 2, through the 12 cooling tower transformer. The inspectors reviewed the circumstances of the event and the licensee's corrective actions. Updated Safety Analysis Report Section 8.3, "Auxiliary Power System," was reviewed as part of this inspection.

b. Observations and Findings

After the loss of the 10 bank transformer, bus 26 was automatically repowered from the alternate offsite power source (transformer 2RY) within a few seconds, as designed. The D6 emergency diesel generator did not receive a start signal, nor was one expected, because of the short time the bus was deenergized. All equipment responded as designed. The operating charging pump had been powered from bus 26, therefore, charging flow was lost and the charging and letdown lines isolated. Operators promptly restored charging and letdown when the bus was reenergized. Containment cooling fans and radiation monitor pumps powered from bus 26 were lost and were also promptly restarted. During this time the operators were properly following the appropriate annunciator response procedure.

When the cause of the trip was determined and it became known that the 10 bank transformer would not be immediately restored, the licensee decided to cross-tie the 11 and 12 cooling tower transformer outputs so that bus 26 could be supplied from the 11 cooling tower transformer. Operators conducted a prejob briefing and completed the cross-tie and power transfer using the appropriate abnormal operating procedure without problems. The inspectors observed portions of the above evolutions.

The system engineer directed a thorough review of the event and attempted to determine the cause of the sudden pressure relay actuation. Oil and gas samples from the transformer were analyzed with no unusual indications. The pressure detector setpoint and relay operation were checked and no problems were found. Only one person was in the switchyard relay house at the time of the event and he was eating lunch in an area not near the relay. When the potential causes of the trip were all examined and eliminated, the licensee carefully reenergized the transformer and gradually applied load while closely monitoring its performance. No problems were identified and the transformer was restored to full service on November 30.

The licensee reported the event to the NRC in accordance with 10 CFR 50.72 and intended to issue a follow up Licensee Event Report (LER). The LER will be considered open, when issued, pending inspector review.

c. Conclusions

All equipment responded to the loss of the 1D bank transformer as expected. Operators properly followed the appropriate annunciator response and abnormal operating procedures to recover from the event. The system engineer led a thorough, but fruitless, investigation into the cause of the event.

O2 Operational Status of Facilities and Equipment

O2.1 Engineered Safety System Walkdown (71707)

The inspectors conducted a walkdown of the containment piping penetrations on Unit 1 in the auxiliary building, shield building, and containment building during this inspection period. Containment penetrations on Unit 2 were inspected from the auxiliary building. The purpose of the inspection was to verify the information in various procedures and documents. The results are discussed in Section M3 of this report.

O8 Miscellaneous Operations Issues (92700, 92901)

O8.1 (Closed) LER 50-282/96012 (1-96-12): Loss of Offsite Power to Unit 2 and Degraded Offsite Power to Unit 1 Followed by Reactor Trips of Both Units. This LER was previously discussed in Integrated Inspection Report 50-282/96007; 50-306/96007, Section O1.2, and Inspection Report No. 50-282/96008(DRP); 50-306/96008(DRP), Section O8.1. The inspectors verified that all corrective actions discussed in the LER were completed except one. The remaining action was to consider improvements to increase the assurance that the emergency response computer system would be available following a loss of power event. Despite extensive troubleshooting, licensee computer engineers were unable to determine why the computer system did not perform as designed during the event. Without that knowledge, improvements could not be proposed.

The NRC Office for Analysis and Evaluation of Operational Data conducted a review of the event and determined that it was considered an accident sequence precursor for 1996 with a conditional core damage probability of 5.3×10^{-5} as reported in a letter to the licensee dated October 9, 1997.

O8.2 (Closed) LER 50-282/97007 (1-97-07), Revision 0 and Revision 1: Both Trains of Spent Fuel Special Ventilation Inoperable While Handling Loads Over Spent Fuel. This event was previously discussed in Inspection Report No. 50-282/97011(DRP); 50-306/97011(DRP), Section O8.2. The inspectors verified that the interim actions for activities in the spent fuel pool, as discussed in the LER, were in place until September 15, 1997, when the NRC issued License Amendment 130 (Unit 1); 122 (Unit 2) which modified the TSs for fuel handling to correct the problem. As part of the amendment, a new license condition was imposed which decreased the probability of a fuel damage accident by requiring the use of a single-failure-proof crane or spent fuel pool covers when handling heavy loads over irradiated fuel.

O8.3 (Closed) Violation (VIO) 50-282/97011-03(DRP); 50-306/97011-03(DRP): Three Examples of Failure of the Operations Committee to Meet TSs Requirements. This violation was previously discussed in Inspection Reports No. 50-282/97011(DRP); 50-306/97011(DRP), Section O7.1, and 50-282/97016(DRP); 50-306/97016(DRP), Section O7.1. The licensee responded to the violation in a letter dated August 25, 1997. As discussed in the previous reports, the remaining corrective action was to revise Administrative Work Instruction 5AWI 3.3.0, "Operations Committee," to incorporate management expectation clarifications. The inspectors reviewed Revision 5 to 5AWI 3.3.0 issued on September 15, 1997, which completed the corrective action.

II. Maintenance

M1 Conduct of Maintenance

M1.1 General Comments

a. Inspection Scope (61726, 62707)

The inspectors observed all or major portions of the following maintenance, surveillance, and refueling outage activities. Included in the inspection was a review of the surveillance procedures (SP), work orders (WO), or refueling activity procedures listed as well as the appropriate USAR sections regarding the activities. The inspectors verified that the surveillance procedures observed met the requirements of the TSs.

- SP 1264 Reactor Vessel Level Instruments Calibration, Revision 11
- SP 1334 D1 Diesel Generator 24 Hour Load Test, Revision 5
- SP 2295 D5 Diesel Generator Fast Start Test, Revision 19
- WO 9707342 Remove Reactor Vessel Upper Internals per D58.11.5
- WO 9707432 P3170-1-11, Loop A Main Steam Isolation Valve Refueling Inspection
- WO 9708818 Reactor Vessel ISI [Inservice Inspection] Exams on the Reactor Vessel and Nozzles
- WO 9712183 Replace Loop A Main Steam Safety Valves
- R101240A Reactor Vessel Head Removal

The inspectors also observed minor portions of numerous additional refueling outage activities.

b. Observations and Findings

For all of the work observed, procedures were properly used and followed. Maintenance personnel were experienced and knowledgeable of their tasks. The inspectors observed frequent monitoring of work by system engineers. Noteworthy comments on specific work activities are discussed below.

- The inspectors observed the disassembly, inspection, and reassembly of the A main steam isolation valve during performance of WO 9707432. The inspectors observed performance of a dye penetrant examination on the valve seat. The examination identified five cracks beginning in the valve body and continuing into the valve seating area. The dye penetrant examiner recorded the indications and forwarded the information to the system engineer for evaluation. The system engineer identified that the same indications had been observed during previous refueling inspections and were being monitored for growth. Comparison with previous inspection results showed that no growth of the cracks was taking place. The cracks did not affect the structural integrity or performance characteristics of the valve.

- The inspectors observed the pre-evolution brief for the performance of WO 9707342, "Remove Reactor Vessel Upper Internals per D58.11.5." The work order accomplished the lifting and removal of the upper reactor vessel internals and the movement of the vessel internals to a storage stand located in the reactor vessel cavity. The brief was attended by all personnel involved in the evolution and included representatives from operations, plant management, radiation protection, and maintenance. The inspectors noted the brief was thorough and complete.

  The inspectors observed the performance of the entire evolution. The task was performed in a slow and controlled manner. The inspectors noted that the rigger in charge of the lift demonstrated good knowledge of the procedure and lifting equipment involved. Also, the radiation protection staff exercised good control of personnel by monitoring for increased general area exposure levels and by moving people to lower dose areas when warranted.

  A brief delay was experienced during the initial lifting of the reactor vessel internals when the maximum weight, as specified in the procedure, was about to be exceeded. After a brief investigation, it was determined that a protective ring, which was normally left behind in the vessel while the internals were being removed, was still attached to the lifting rig. The reason the protective ring was left attached was to support later inservice inspections. The lifting procedure had not accounted for the extra weight of the protective ring. A temporary procedure deviation was made to the heavy lift procedure and the evolution was completed without further problems.

- The inspectors observed portions of SP 1334, "D1 Diesel Generator 24 Hour Load Test." While the engine was running, operators reported that a fitting (approximately 1.5" by 4.0"), that was part of the oil fill equipment, had been dropped into the diesel's oil sump. The problem occurred during a routine oil addition which was required during extended diesel operation. The operators

discussed the situation with the D1 diesel generator system engineer and it was determined that the fitting posed no immediate threat to the engine or its lube oil system and that the surveillance procedure could be completed. Subsequent discussions between the inspectors and the D1 system engineer revealed that the proposed corrective actions were to remove the fitting from the oil sump and place a coarse screen in the area of the oil fill location. The actions were expected to be completed after the Unit 1 refueling outage.

c. Conclusions

Maintenance, surveillance, and refueling outage activities were performed well with only minor problems. All activities were performed safely with proper procedures being used and followed. System engineer involvement was strong.

M1.2 Reactor Vessel Inservice Inspection (ISI) Weld Examinations

a. Inspection Scope (73753)

The inspectors observed portions of WO 9708818, "Reactor Vessel ISI Exams on the Reactor Vessel and Nozzles." The activity included an examination of Unit 1 reactor vessel welds 7 and 10, the nozzle to safe end welds, the safe end to pipe welds and the inner radius region of both outlet nozzles. The inspectors reviewed the ultrasonic data for all indications recorded during the inspection.

b. Observations and Findings

For the strongest reportable indication in the outlet nozzles, the inspectors observed the vendor's analysis of the ultrasonic data. The inspectors agreed with the licensee consultant's analysis methods and interpretation of the indication as a manufacturing, non-service related flaw.

The inspectors also reviewed procedure ISI-LTS-1, "Limitations to Nondestructive Examination," Revision 0. The procedure provided instructions for the identifying, quantifying, and recording of limitations encountered while performing ISI examinations. When limited ISI examinations were encountered, ISI-LTS-1 ensured that the examiner forwarded a description of the specific limitations encountered to the licensee field supervisor for review. If after supervisory review, alternate inspection methods were unable to provide additional coverage and the maximum examination coverage was less than 90 percent, ISI-LTS-1 required a relief request to be submitted to the NRC.

c. Conclusions

The inspectors agreed with the licensee's interpretation of the indications reported in the Unit 1 reactor vessel outlet nozzles. The indications reported were original construction, manufacturing flaws and not the result of inservice conditions. Procedure ISI-LTS-1 adequately covered situations where less than 100 percent examination coverage was achievable.

M3 Maintenance Procedures and Documentation

M3.1 Unit 1 Containment Penetrations and Related Procedures

a. Inspection Scope (92902)

As a followup of previously identified errors in the documentation regarding some containment penetrations, the inspectors reviewed various checklists, procedures, and documents for the Unit 1 containment penetrations. The inspectors reviewed the following:

- Integrated Checklist C1.1.19-1, "Containment Integrity Checklist-Unit 1," Revision 27
- Integrated Checklist C1.1.19-1, "Containment Integrity Checklist-Unit 1," Revision 26
- Integrated Checklist C1-3, "Containment Integrity Checklist-Unit ____," Revision 6
- H19, "Containment Leakage Rate Testing," Revision 2
- Updated Safety Analysis Report (USAR), Section 5, Table 5.2-1-(Part A), "Unit 1 Containment Vessel Penetrations," Revision 13,
- SP 1072.1, "Local Leakage Rate Test of Penetration 1," Revision 13
- SP 1072.2, "Local Leakage Rate Test of Penetration 2," Revision 12
- SP 1072.13A, "Local Leakage Rate Test of Penetration 13A," Revision 15
- SP 1072.18, "Local Leakage Rate Test of Penetration 18," Revision 12
- SP 1072.21, "Local Leakage Rate Test of Penetration 21," Revision 13
- SP 1072.25A, "Local Leakage Rate Test of Penetration 25A," Revision 12
- SP 1072.25B, "Local Leakage Rate Test of Penetration 25B," Revision 12
- SP 1072.26, "Local Leakage Rate Test of Penetration 26," Revision 18
- SP 1072.41A, "Local Leakage Rate Test of Penetration 41A," Revision 10
- SP 1072.41B, "Local Leakage Rate Test of Penetration 41B," Revision 10
- SP 1072.42A, "Local Leakage Rate Test of Penetration 42A," Revision 12
- SP 1072.42F, "Local Leakage Rate Test of Penetration 42F," Revision 10
- SP 1072.45, "Local Leakage Rate Test of Penetration 45," Revision 12
- SP 1072.49B, "Local Leakage Rate Test of Penetration 49B," Revision 10
- SP 1072.50, "Local Leakage Rate Test of Penetration 50," Revision 11

b. Observations and Findings

- In Integrated Checklist C1.1.19-1 the inspectors identified the following discrepancies:

  1. The schematic associated with penetration 19 did not label valve SA-3-5.
  2. On page 9 of 102, for penetration 38A, the valve listed with MV-32139 should have been CL-22-3, not CL-22-1.
  3. On page 9 of 102, for penetration 38B, the valve listed with MV-32133 should have been CL-22-1, not CL-22-3.
  4. Valve AF-30-1, which was associated with penetration 46A, was listed twice, on two different elevations.
  5. Valve AF-30-2, which was associated with penetration 46B, was listed twice, on two different elevations.

- Special Instruction (note 5) of Integrated Checklist C1.[illegible].19-1 stated, "Containment integrity tags (blue tags) should be installed/verified in [illegible]led, during performance of this checklist. Containment integrity tags should be [illegible]lled on all components in this checklist." The inspectors identified the discrepan[illegible] listed below during a spot check. Brown tags were formerly used for marking components.

Components not labeled with blue tags	Components labeled with brown tags	Components, not required by the checklist, labeled with brown tags
CV-31741	MV-32065	CV-31447
CV-31634	MV-32066	SI-16-4
CV-31311	MV-32199	SI-16-5
MV-32141	MV-32273	SI-16-6
MV-32135	MV-32024	SI-20-30
CL-57-5	CV-31444	SI-21-1
CL-57-3	RC-3-1*	CV-31449
MV-32242	Penetration 49B inner blind flange*	MV-32069
		CV-31456
		MV-32068
		CV-31450
	* labeled with both brown and blue tags	VC-16-18

The above discrepancies indicated a lack of attention-to-detail on the part of the operators who last performed the checklist.

- The inspectors performed a spot check of the physical condition and labeling of the containment vessel penetrations. The condition and labeling of the inspected penetrations was good, with no observed discrepancies.

- In USAR Table 5.2-1-(Part A), the following discrepancies were noted:

  1. On page 1 of 6, referring to penetration 6A, the valve listed as CV-31908 should have been CV-31098.
  2. On page 3 of 6, referring to penetration 29A, the valves listed (CS-18, MV-32103, and CS-11) should have been associated with penetration 29B. The valves that should have been listed with penetration 29A were CS-19, MV-32105, and CS-12. After further discussion with the licensee and review of planned corrective documentation, the inspectors learned that this discrepancy had already been identified by the licensee during the USAR update program and that corrective actions would be taken.
  3. On page 3 of 6, referring to penetration 29B, the valves listed (CS-19, MV-32105, and CS-12) should have been associated with penetration 29A. The valves that should have been listed with penetration 29B were CS-18, MV-32103, and CS-11. After further discussion with the licensee

and review of planned corrective documentation, the inspectors learned that this discrepancy had already been identified by the licensee during the USAR update program and that corrective actions would be taken.

- During a cross-reference check of Integrated Checklist C1.1.19-1 and the penetration-specific local leakage rate surveillance procedures (SP 1072 series), the inspectors identified the following inconsistencies in the given location of the containment vessel penetrations:

Penetration Number	Location per Integrated Checklist C1.1.19-1		Location per penetration-specific procedure	
1	728'6"	323°	724'	39°
2	723'	39°	725'	336°
13A	729'6"	330°	720'	274°
18	715'	240°	716'	240°
21	720'	271°	720'	269°
25A	770'	316°	770'	345°
25B	770'	345°	770'	316°
26	720'	226°	720'	266°
41A	783'	57°	806'	57°
41B	806'	57°	760'	57°
42A	723'	262°	724'	37°
42F	723'	37°	723'	350°
45	719'	40°	738'	278°
49B	719'	43°	720'	43°
50	723'	41°	720'	43°

c. Conclusions

The Unit 1 containment penetration checklists and procedures contained inconsistencies between procedures and editorial errors within procedures. Although many errors were corrected by Revision 27 to the Containment Integrity Checklist-Unit 1, there were still many remaining. Most of the component labeling discrepancies and editorial errors should have been identified and addressed during performance of the checklist and incorporated into Revision 27. That failure indicated lack of attention-to-detail on the part of the operators who last performed the checklist. However, the errors were not significant enough to prevent successful performance of the checklist. After being informed of the above errors, the licensee actively pursued correcting the discrepancies. The USAR discrepancies also appeared to be editorial in nature and were of minor safety significance.

M3.2 Unit 2 Containment Penetrations and Related Procedures

a. Inspection Scope (92902)

The inspectors reviewed various checklists, procedures, and other documents regarding all of the Unit 2 containment penetrations. The inspectors reviewed the following:

- Integrated Checklist C1.1.19-4, "Containment Integrity Checklist-Unit 2," Revision 24
- Integrated Checklist C1-3, "Containment Integrity Checklist-Unit ___," Revision 6
- Updated Safety Analysis Report, Section 5, Table 5.2-1-(Part B), "Unit 2 Containment Vessel Penetrations," Revision 13
- Updated Safety Analysis Report, Section 5, Table 5.2-1-(Part A), "Unit 1 Containment Vessel Penetrations," Revision 13
- SP 2072.1, "Local Leakage Rate Test of Penetration 1," Revision 8
- SP 2072.2, "Local Leakage Rate Test of Penetration 2," Revision 8
- SP 2072.11, "Local Leakage Rate Test of Penetration 11," Revision 7
- SP 2072.16, "Local Leakage Rate Test of Penetration 16," Revision 10
- SP 2072.20, "Local Leakage Rate Test of Penetration 20," Revision 16
- SP 2072.23, "Local Leakage Rate Test of Penetration 23," Revision 10
- SP 2072.31, "Local Leakage Rate Test of Penetration 31," Revision 10
- SP 2072.44, "Local Leakage Rate Test of Penetration 44," Revision 8
- SP 2072.52, "Local Leakage Rate Test of Penetration 52," Revision 13
- SP 2072.55, "Local Leakage Rate Test of Penetration 55," Revision 8

b. Observations and Findings

- The inspector identified that the "containment integrity tag" for valve CV-31209 was hung on the piping adjacent to the valve and not on the component itself.

- In Integrated Checklist C1.1.19-4 the inspectors identified the following:

  1. On the schematic for penetration 28B, the valve labeled CV-31516 should have been labeled CV-31518.
  2. Valve 2CS-24-2, which was associated with penetration 29B, was labeled with a containment integrity tag even though not required by the checklist.

- The inspectors performed a spot check of the physical condition and labeling of the auxiliary building penetrations. The condition and labeling of the inspected penetrations were good, with no observed discrepancies.

- The inspectors compared the Containment Integrity Checklist-Unit 2 (C1.1.19-4) and the Unit 2 specific local leak rate surveillance procedures. No inconsistencies were noted in the given locations for any of the listed penetrations.

- In USAR Table 5.2-1-(Part B), for penetration 12, the table listed 2VC-8-2 as the required isolation valve. The correct valve was 2VC-8-1. After discussion with the licensee, and review of planned corrective documentation, the inspectors learned that this discrepancy had already been identified by the ongoing USAR update program and that corrective actions would be taken.

c. Conclusions

The discrepancies with the Unit 2 checklists and procedures were generally editorial or minor in nature. No unlabeled components were found and the corresponding auxiliary building penetrations were clearly labeled. The Unit 2 checklists and procedures demonstrated a much improved attention to detail and consistency compared to Unit 1.

After being informed of the above errors, the licensee actively pursued correcting the discrepancies. The one USAR discrepancy was licensee identified and appeared to be editorial in nature and of minor safety significance.

M8 Miscellaneous Maintenance Activities (92700, 92902)

M8.1 (Closed) LER 50-282/97005 (1-97-05): Surveillance Interval Discrepancies with Diesel Oil Sample Analyses. This event was previously discussed in Inspection Report No. 50-282/97011(DRP); 50-306/97011(DRP), Section M8.2. It was considered a Non-Cited Violation (NCV 50-282/97011-04(DRP); 50-306/97011-04(DRP)). The inspectors verified that the corrective actions discussed in the LER were completed. The licensee determined that numerous other surveillances, such as battery and charcoal efficiency tests, which required laboratory or other analyses, were also being counted as complete before the analyses were done. The surveillances were revised to insure that all required analyses were completed before the surveillances were counted as complete.

M8.2 (Closed) VIO 50-306/97002-04(DRP): Inadequate Procedure for Control of Heavy Loads. This issue was previously discussed in Inspection Reports No. 50-282/97002(DRP); 50-306/97002(DRP), Section M3.1, and 50-282/97005(DRP); 50-306/97005(DRP), Section M3.1. The licensee also discussed the issue in LER 2-97-01. The licensee responded to the violation and discussed the corrective actions in a letter to the NRC dated March 26, 1997. On March 18, 1997, a predecisional enforcement conference was held for another violation involving control of heavy loads, and a Notice of Violation for that event was issued on April 30, 1997 (Enforcement Action (EA) 97-073/01013(DRP)). Since the licensee's corrective actions for both violation 50-306/97002-04 and the enforcement action were closely related in that the entire heavy loads program was undergoing a comprehensive review, the complete set of corrective actions will be reviewed when EA 97-073 is closed. This violation was closed to avoid duplication.

III. Engineering

E2 Engineering Support of Facilities and Equipment

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

While performing the inspections discussed in this report, the inspectors reviewed the applicable portions of the USAR that related to the areas inspected and used the USAR as an engineering/technical support basis document. The inspectors compared plant practices, procedures, and/or parameters to the USAR descriptions as discussed in each section. The inspectors verified that the USAR wording was consistent with the observed plant practices, procedures, and parameters. Some minor discrepancies were noted regarding Table 5.2-1 for containment penetrations as discussed in Section M3 of this report.

E2.2 General Comments (37551)

Throughout the inspection period, the inspectors noted frequent involvement by system engineers in all aspects of plant operations, refueling, maintenance, and surveillance activities. The engineers rapidly investigated any operational abnormalities, took an

active role in maintenance and troubleshooting activities, and closely followed all surveillance testing on their systems.

E8 Miscellaneous Engineering Issues (92700, 92903)

E8.1 (Closed) LER 50-306/96001 (2-96-01): Reactor Trip Caused by Failure of Feedwater Regulating Valve. This event was previously discussed in Integrated Inspection Reports No. 50-282/96004; 50-306/96004, Sections 1.2 and 3.2, 50-282/96006; 50-306/96006, Section O8.1, and 50-282/96007; 50-306/96007, Section E2.2. The licensee completed a Spare Parts Change Evaluation to modify the feedwater regulating valve stem joint configuration and increase the joint resistance to torsional moments. The modification was made to the Unit 2 valves in the 1997 spring outage and to the Unit 1 valves in the 1997 fall outage. Prior to the modification, the valves were instrumented and stem torsional stresses were measured in various flow regimes. Stresses were found not to be excessive in any position for the valve design.

Although the licensee determined that operating procedures did not need to be changed to expedite transition through the low flow region during plant startup, the inspectors' observations in several recent startups indicated that operators were sensitive to avoiding long-term operation with the feedwater regulating valves in low flow regions. For a different reason, there was already a precaution in Operating Procedure 1(2)C1.2, "Unit 1(2) Startup Procedure," Revision 17, to minimize the time spent between hot standby and 15 percent power. In addition, Operating Procedure 1(2)C28.2, "Unit 1(2) Feedwater System," Revision 11(8), contained precautions, notes, and steps, intended to help minimize the time the feedwater regulating valves were operated in low flow conditions.

E8.2 (Closed) Inspection Followup Item (IFI) 50-282/96007-03(DRP): Resolution of Incorrectly Stated Hydrogen Monitor System Component Locations in USAR, Section 5.4.2.2.3. This issue was previously discussed in Integrated Inspection Report No. 50-282/96007; 50-306/96007, Sections E2.3 and E8.1. It involved an inaccurate statement in the USAR regarding the location of the containment hydrogen monitor system data acquisition and control assemblies. The inspectors verified that the most recent revision of the USAR deleted the inaccurate information.

E8.3 (Closed) IFI 50-282/96010-02(DRP): Drawing Error in USAR. This issue was previously discussed in Inspection Report No. 50-282/96010; 50-306/96010, Sections M3.1 and E2.1. It involved an error in USAR Figure 11.1-5, Revision 13. The inspectors verified that the USAR figure had been corrected in Revision 14 (Drawing NF-39222, Revision AW).

E8.4 (Closed) LER 50-282/96019 (1-96-19): Spent Fuel Storage Racks Outside Design Basis due to Boroflex Degradation. This LER was previously discussed in Inspection Report No. 50-282/96016; 50-306/96016, Section E8.1. On June 12, 1997, the NRC issued License Amendment 129 (Unit 1); 121 (Unit 2) which established the TSs necessary to take credit for soluble boron in the spent fuel pool instead of Boroflex. The inspectors verified that the TS changes granted by the amendment had been implemented.

E8.5 (Open) LER 50-282/97011 (1-97-11): Failure to Test the Low Pressure Auto-start Function of 121 Motor Driven Cooling Water Pump and Inadequate Separation Between

Trains A and B Low Pressure Auto-start Switches. This LER was previously discussed in Inspection Report No. 50-282/97018(DRP); 50-306/97018(DRP), Section E1.1. The LER was left open pending a review of the enforcement aspects of the findings.

As reported in the LER, the licensee-identified findings involved both a violation of the TS 4.5.A.5.a surveillance requirements and reflected a failure to assure that the design basis for switch separation was properly translated into specifications, drawings, procedures, and instructions in accordance with 10 CFR 50, Appendix B, Criterion III. However, the issues were licensee-identified as part of the actions to address NRC Generic Letter 96-01. Concerning the low pressure auto-start function, although it was not regularly tested, the low pressure detector setpoint was routinely calibrated. In addition, several actual starts of the pump on low pressure have been documented as a result of operational events. Concerning the switch separation issue, the 121 cooling water pump was not usually considered a safety-related pump and inadequate separation would only be a condition outside the design basis on the relatively rare occasions when the 121 pump was aligned to replace the 22 cooling water pump. In those cases, a single fault might affect both the remaining 121 and 12 cooling water pumps.

The licensee's corrective actions, as discussed in the LER, were adequate to resolve the issues. The low pressure auto-start feature was successfully tested during this inspection period. A design change to move the switch was being developed and was anticipated to be completed in early December 1997. The LER will remain open until the completion of the design change.

These non-repetitive, licensee-identified and corrected violations are being treated as Non-Cited Violations, consistent with Section VII.B.1 of the NRC Enforcement Policy (50-282/97021-01(DRP); 50-306/97021-01(DRP)).

E8.6 (Open) LER 50-282/97012 (1-97-12): Chemical and Volume Control System Malfunction Unanalyzed for Boron Dilution During Shutdown Modes of Operation. As part of its ongoing USAR review project, the licensee discovered and reported that there was no analysis for a potential boron dilution accident occurring in shutdown conditions. The results of such an accident might be more severe than the analyzed accident at power because operators might have less time to recognize and correct the condition due to decreased effective reactor coolant system volume. The licensee also reported that a boron dilution event, when the residual heat removal system was returning flow via the vessel injection path instead of the normal cold leg path, could result in a different type of dilution transient because thorough mixing of the water in the reactor vessel would not occur.

The discovery was an excellent finding and indicated a thorough USAR review process. Initial licensee actions to specify higher shutdown boron concentrations and assure adequate mixing when returning residual heat removal flow through the vessel injection path (the licensee intended to run at least one reactor coolant pump when in that condition) were adequate to compensate for the lack of final analysis results. The LER will remain open pending an NRC specialist's review of the long-term corrective action results.

IV. Plant Support

R1 Radiological Protection and Chemistry Controls (71750)

During normal resident inspection activities, routine observations were conducted in the areas of radiological protection and chemistry controls using Inspection Procedure 71750. No discrepancies were noted. The inspectors noted good performance of the radiation protection staff in controlling exposures during refueling activities.

P1 Conduct of Emergency Preparedness Activities (71750)

During normal resident inspection activities, routine observations were conducted in the area of emergency preparedness using Inspection Procedure 71750. No discrepancies were noted.

S1 Conduct of Security and Safeguards Activities (71750)

During normal resident inspection activities, routine observations were conducted in the areas of security and safeguards activities using Inspection Procedure 71750. No discrepancies were noted. The Superintendent Security and Team Leader Security Services met with the inspectors during the inspection period to provide a briefing of current security activities and plans.

F1 Control of Fire Protection Activities

F1.1 Fire in the Maintenance Shop

a. Inspection Scope (93702)

On October 28, 1997, a fire was reported in the maintenance shop which was located in the service building off of the 735 foot elevation of the turbine building. The inspectors observed the response to the event from the control room and Technical Support Center.

b. Observations and Findings

The licensee's fire brigade responded rapidly to the event with sufficient personnel and equipment. The fire brigade leader clearly established his authority and properly followed response procedures. The fire was located in an air filtration unit serving a cutting and welding table. Although there was apparently only a small fire in the unit, smoke discharge was very heavy due to a service air supply blowing through the unit. It took about 20 minutes to identify the air supply and isolate it.

Personnel were immediately evacuated from the adjacent maintenance and computer areas and eventually the entire turbine building was evacuated. The Technical Support Center was manned in order to assist in personnel accountability. Other actions taken included calling the city of Red Wing fire department, starting the control room cleanup fans, isolating the control room outside air dampers, starting the turbine building roof exhaust fans, and isolating the service building computer room halon system.

The Red Wing fire department responded to the site but the fire was out and smoke was being cleared by the time they arrived on the scene. The fire department ambulance service briefly treated 11 employees for smoke inhalation and one additional employee

was administered oxygen by the licensee's safety department personnel. All employees returned to work after treatment.

The cause of the fire was believed to be a spark or hot particle drawn into the air filtration unit, igniting the paper filter. The unit apparently contained a large amount of dust and slag which contributed to the fire. Smoke was released primarily from a duct which was blown off by a series of small dust explosions in the unit.

At the time of the event, the licensee made a Notification of Unusual Event due to a fire lasting more than 10 minutes and notified the NRC in accordance with 10 CFR 50.72. On November 24, 1997, the licensee also voluntarily issued LER 1-97-14 for the event. The LER is closed in Section F8.1 of this report.

c. Conclusions

The response of the fire brigade, control room operators, and other licensee personnel to the fire was good. No significant discrepancies were noted.

F2 Status of Fire Protection Facilities and Equipment

F2.1 Inadequate Lube Oil Collection System for Reactor Coolant Pumps

a. Inspection Scope (92904)

On November 11, 1997, the licensee reported in accordance with 10 CFR 50.72 a condition outside the design basis of the plant whereby the lube oil collection system for reactor coolant pumps on both units did not meet all the requirements of 10 CFR 50, Appendix R, Section III.O. The inspectors reviewed the circumstances of the finding and the licensee's corrective actions.

b. Observations and Findings

During a walkdown of the lube oil collection system by a consultant, the licensee identified that the system was not adequate to contain a pressurized oil leak from the oil lift pump system piping. The oil collection system was adequate to collect leakage from the low pressure portions of the lube oil system.

The licensee could not immediately determine why a system capable of collecting pressurized oil leakage had not been originally installed. The licensee had requested and received an exemption from another of the requirements of Section III.O of Appendix R, that the oil be collected and drained to a vented closed container, as discussed in a letter from J. R. Miller (NRC) dated July 31, 1984, but no exemption from the requirement to have the ability to collect pressurized leakage was identified. The NRC had conducted an inspection of the implementation of Appendix R requirements as discussed in Inspection Report 50-282/87004(DRS); 50-306/87004(DRS) and determined that the collection system was in conformance with Section III.O of Appendix R. However, it was not clear from that report whether the inspectors were aware that pressurized leakage might not be collected. The inspection apparently was primarily focused on verifying the licensee's justification for the exemption request.

As an interim corrective action, the licensee issued temporary changes to the reactor coolant pump operating and annunciator response procedures to specify an inspection of the reactor coolant pump area after the last start of the oil lift pump prior to full power operations and after a low oil level alarm. The licensee discussed the issue with the NRC Office of Nuclear Reactor Regulation by telephone on November 21, 1997. After the discussion, the licensee agreed to modify the collection system on Unit 1 prior to startup from the current refueling outage and on Unit 2 at the first shutdown of sufficient duration. The Operations Committee reviewed Design Change 97FP02 on November 25, 1997, which was to correct the problem.

The inadequate oil collection system was only of moderate safety significance because the oil lift system was normally only pressurized for short times and plant conditions at those times we not normally expected to be such that an oil leak would ignite. In addition, there was no safe shutdown equipment in the immediate area of the reactor coolant pumps. Although there have been fires at other facilities due to pressurized leakage from the oil lift pumps, the fires usually involved cases where the lift pumps were run for extended periods and/or fibrous insulation in the area was oil soaked. For the Prairie Island facility, the insulation in the area is metal reflective type and is not subject to oil soaking.

Section III.O of Appendix R, of 10 CFR Part 50 required, in part, that reactor coolant pumps be equipped with an oil collection system, if the containment is not inerted during normal operation, and that such collection systems shall be capable of collecting lube oil from all potential pressurized and unpressurized leakage sites in the reactor coolant pump lube oil systems. However, the licensee identified that the collection systems for both reactor coolant pumps on both units were not capable of collecting lube oil from all potential pressurized sites in the oil lift pump system. This non-repetitive, licensee-identified and corrected violation is being treated as a Non-Cited Violation, consistent with Section VII.B.1 of the NRC Enforcement Policy (50-282/97021-02(DRP); 50-306/97021-02(DRP)).

The licensee intended to issue an LER for the finding. The LER will be considered open, when issued, pending the inspectors' review.

c. Conclusions

The licensee's finding was the result of a proactive, voluntary review of fire protection issues in preparation for a future NRC pilot inspection. Prompt corrective actions were planned to modify the system to be in compliance with NRC requirements.

F8 Miscellaneous Fire Protection Issues (92700)

F8.1 (Closed) LER 50-282/97014 (1-97-14): Maintenance Shop Fire in the Service Building. This event was discussed in Section F1.1 of this report. The report was submitted as voluntary supplemental information for the Notification of Unusual Event. The inspectors verified that the report adequately discussed the cause and contributing factors for the event and that corrective actions would be tracked through the licensee's Error Reduction Task Force.

V. Management Meetings

X1 Exit Meeting Summary

The inspectors presented the inspection results to members of licensee management at the conclusion of the inspection on December 2, 1997. The licensee acknowledged the findings presented. The inspectors asked the licensee whether any materials examined during the inspection should be considered proprietary. No proprietary information was identified.

PARTIAL LIST OF PERSONS CONTACTED

Licensee

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

INSPECTION PROCEDURES USED

IP 37551:	Engineering
IP 61726:	Surveillance Observations
IP 62707:	Maintenance Observations
IP 71707:	Plant Operations
IP 71750:	Plant Support Activities
IP 73753:	Inservice Inspection
IP 92700:	Onsite Follow-up of Written Reports of Nonroutine Events at Power Reactor Facilities
IP 92901:	Follow up - Operations
IP 92902:	Follow up - Maintenance
IP 92903:	Follow up - Engineering
IP 92904:	Follow up - Plant Support
IP 93702:	Prompt Onsite Follow up of Events

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

50-282/97012	LER	Chemical and Volume Control System Malfunction Unanalyzed for Boron Dilution During Shutdown Modes of Operation
50-282/97021-01(DRP) 50-306/97021-01(DRP)	NCV	Failure to Test the Low Pressure Auto-start Function of the 121 Motor Driven Cooling Water Pump and Inadequate Separation Between Trains A and B Low Pressure Auto-start Switches
50-282/97021-02(DRP) 50-306/97021-02(DRP)	NCV	Inadequate Lube Oil Collection System for Reactor Coolant Pumps

Closed

50-306/96001	LER	Reactor Trip Caused by Failure of Feedwater Regulating Valve
50-282/96012	LER	Loss of Offsite Power to Unit 2 and Degraded Offsite Power to Unit 1 Followed by Reactor Trips of Both Units
50-282/96019	LER	Spent Fuel Storage Racks Outside Design Basis due to Boroflex Degradation
50-282/97005	LER	Surveillance Interval Discrepancies with Diesel Oil Sample Analysis
50-282/97007	LER	Both Trains of Spent Fuel Special Ventilation Inoperable While Handling Loads Over Spent Fuel
50-282/97014	LER	Maintenance Shop Fire in the Service Building
50-282/96007-03(DRP)	IFI	Resolution of Incorrectly Stated Hydrogen Monitor System Data Acquisition Components in Updated Safety Analysis Report, Section 5.4.2.2.3
50-282/96010-02(DRP)	IFI	Drawing Error in the Updated Safety Analysis Report
50-306/97002-04(DRP)	VIO	Inadequate Procedure for the Control of Heavy Loads
50-282/97011-03(DRP) 50-306/97011-03(DRP)	VIO	Three Examples of Failure of the Operations Committee to Meet Technical Specifications Requirements

Discussed

EA 97-073/01013(DRP)	VIO	Inadequate Control of Heavy Load Over the Reactor
50-282/97011	LER	Failure to Test the Low Pressure Auto-start Function of the 121 Motor Driven Cooling Water Pump and Inadequate Separation Between Trains A and B Low Pressure Auto-start Switches

LIST OF ACRONYMS USED

CCHX	Component Cooling Heat Exchanger
CFR	Code of Federal Regulations
DRP	Division of Reactor Projects
DRS	Division of Reactor Safety
EA	Enforcement Action
ISI	Inservice Inspection
IP	Inspection Procedure
LER	Licensee Event Report
NCV	Non-Cited Violation
NRC	Nuclear Regulatory Commission
PDR	Public Document Room
RCS	Reactor Coolant System
RO	Reactor Operator
SP	Surveillance Procedure
USAR	Updated Safety Analysis Report
VCT	Volume Control Tank
WO	Work Order