U.S. NUCLEAR REGULATORY COMMISSION

REGION III

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Report No: 50-282/96008, 50-306/96008

Licensee: Northern States Power Company

Facility: Prairie Island Nuclear Generating Plant

Location: 1717 Wakonade Drive East
Welch, MN 55089

Dates: July 10 - August 29, 1996

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EXECUTIVE SUMMARY

Prairie Island Nuclear Generating Plant, Units 1 & 2
NRC Inspection Report 50-282/306/96008

This inspection included aspects of licensee operations, maintenance, engineering, and plant support. The report covers a 7-week period of resident inspection.

Operations

- The NRC identified that the shift manning requirements in the Technical Specifications for one unit operating did not meet the requirements in 10 CFR 50.54. (Section O3.2)

- The NRC identified that there may be some cases where licensed senior operators who were given credit for performing the functions of a senior operator for the purpose of maintaining an active license, may not have meaningfully directed the licensed activities of licensed operators. (Section O5.1)

- Performance of reviews by the Operations Committee were thorough and in accordance with the Technical Specifications. (O6.1)

Maintenance

- Maintenance and surveillance activities observed were well conducted with good communications, proper pre-job planning, and safe work practices. (Section M1.1)

- The NRC identified that the temperature switches in the shield building special ventilation system were not the type described in the Updated Safety Analysis Report. (Section M1.1)

- Failure to adequately review a surveillance test that had not met its acceptance criteria resulted in licensee placing the plant in a condition prohibited by Technical Specifications and an example of a violation of 10 CFR 50, Appendix B. (Section M1.2)

- The NRC identified a failure to take timely corrective actions for the above event resulting in another example of a violation of 10 CFR 50, Appendix B. (Section M1.2) The NRC also identified that the licensee failed to make a timely Licensee Event Report which resulted in a violation of 10 CFR 50.73. (Section M8.1)

- The NRC identified a concern with the licensee's application of an interpretation of one of the Technical Specifications. (Section M3.1)

- A typographical error in a surveillance procedure resulted in an unanticipated challenge to a safety system. (Section M3.2)

Engineering

- The licensee's evaluation of cable tray separation discrepancies was considered untimely and narrowly focused. The NRC identified additional discrepancies that had not been evaluated. (Section E2.1)

- The licensee's evaluation of the potential for boiling and water hammer in the cooling water system for the containment coolers was considered untimely. (Section E2.2)

Plant Support

- The licensee closely monitored the status of road construction to ensure that adequate evacuation routes were maintained.

Report Details

Summary of Plant Status

Both units operated at full power for most of the inspection period. Power on unit 1 was reduced to about 55 percent briefly on August 2-3 to perform maintenance on a main feedwater pump.

During this period the sixth dry spent fuel storage cask was received on the site and underwent preliminary inspections. In addition, an application for licensing of a second Independent Spent Fuel Storage Installation, located offsite, was submitted to the NRC.

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 other sections of this report.

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 engineered safety feature systems:

- Shield Building Special Ventilation System
- Class 1E Electrical Cable Trays

Inspector concerns resulting from the walkdowns are contained in Sections M1.1.b and E2.1.b of this report.

O3 Operations Procedures and Documentation

O3.1 Technical Specification Interpretations

a. Inspection Scope (92901)

The licensee maintained a book of interpretations approved by its Operations Committee to clarify the meanings of certain Technical Specifications. The inspectors reviewed the interpretations for evidence that the licensee may have relied on the opinions of NRC staff members without NRC management concurrence in making any of the interpretations.

b. Observations and Findings

Some of the interpretations indicated they were at least partially based on informal discussions with NRC staff members. The inspectors informed licensee management that only Technical Specification interpretations issued in writing by NRR are considered binding by the NRC.

c. Conclusions

The inspectors had a concern with one interpretation as discussed in Section M3.1 of this report.

O3.2 Technical Specification Discrepancy

a. Inspection Scope (92901)

During a review of several licensee's Technical Specifications, the NRC Region III Operator Licensing Branch noted a discrepancy in the Prairie Island specifications.

b. Observations and Findings

10 CFR 50.54(m)(2)(i) required that by January 1, 1984, licensees of nuclear power units shall meet the minimum licensed operator staffing requirements of a table included in the regulation. The table included a requirement for two licensed senior operators and <u>three</u> other licensed operators for two units with one control room when one of the two units was operating.

Prairie Island Technical Specification Table 6.1-1 only required two licensed senior operators and <u>two</u> other licensed operators (four total licensed operators) for that condition. However, review of licensee administrative procedures and observed practices for shift manning disclosed that more than the number of licensed operators required by either the regulation or the specification were assigned to each shift. The Technical Specification table met or exceeded the manning requirements of 10 CFR 50.54 for all other reactor conditions.

The general superintendent plant operations indicated that the licensee would submit a Technical Specification amendment request to bring the specification in line with the regulation.

c. Conclusions

This issue is considered an Inspection Followup Item pending submittal of the Technical Specification amendment request. (50-282/96008-01)

05 Operator Training and Qualification

05.1 Maintaining Operator Licenses in an Active Status

a. Inspection Scope (92901)

The inspectors became aware that the licensee had occasionally credited duty working in the work control center (WCC) as "actively performing the functions of an operator or senior operator" for the purposes of maintaining operator licenses in an active status in accordance with 10 CFR 55.53(e). The inspectors reviewed the requirements for maintaining an active license and the licensee's practices.

b. Observations and Findings

Clarification and guidance in implementing 10 CFR 55 were published in NUREG-1262, "Answers to Questions at Public Meetings Regarding Implementation of Title 10, Code of Federal Regulation, Part 55 on Operators' Licenses." Credit for duty in the WCC appeared to conflict with that guidance. After discussions with the Region III Operator Licensing Branch, the general superintendent plant operations agreed to discontinued the practice of crediting work in the WCC. No current active licensees are involved.

In addition, the licensee normally credited the shift manager (SM) with actively performing the functions of a senior operator. Although licensee administrative procedures required that the SM hold a senior reactor operator license, neither the Technical Specifications nor 10 CFR 50.54 required that the SM be licensed. Those documents only required two senior licensed individuals per shift and those positions were normally filled by the shift supervisors. In reviewing the duties of the SM, as described in licensee administrative procedures, it was not clear to the inspectors whether the SM meaningfully and fully maintained license proficiency by directing the licensed activities of licensed operators as discussed in NUREG-1262.

The general superintendent of plant operations stated that the licensee would submit a letter to the Operator Licensing Branch detailing which duties and shift positions were considered to meet the active performance of license duties criteria.

c. Conclusions

These issues are considered an Inspection Followup Item pending review of the licensee's letter detailing duties and shift positions to be considered for meeting license duties criteria. (50-282/96008-02)

O6 Operations Organization and Administration

O6.1 Performance of the Operations Committee

a. Inspection Scope (40500)

Over the course of the inspection period the inspectors attended several meetings of the Operations Committee and evaluated its effectiveness.

b. Observations and Findings

The Operations Committee was required by Section 6.2.B of the Technical Specifications to meet at least monthly to review plant operations and advise the plant manager. The inspectors noted that meetings were normally scheduled every two weeks and usually held more often as required.

The inspectors noted that the meetings were controlled by an agenda and that the agenda was followed. A committee quorum was verified for each meeting and attendance was monitored during the meeting to insure the quorum was maintained. Alternate members in attendance were identified.

Each issue to be reviewed was presented to the committee by a sponsor who was knowledgeable of the background and details of the issue. Candid and thorough technical discussions of each issue were conducted with the chairman giving each member's opinion due consideration.

c. Conclusions

The inspectors concluded that the Operations Committee conducted activities in a thorough manner and in accordance with the Technical Specifications. No discrepancies were noted.

O8 Miscellaneous Operations Issues (92700)

O8.1 (Open) Licensee Event Report (LER) 50-282(306)/96-12: Loss of Offsite Power to Unit 2 and Degraded Offsite Power to Unit 1 Followed by Reactor Trips of Both Units. This event was extensively discussed in Inspection Report 50-282(306)/96-07, Section O1.2. At the time of that report the LER had not yet been issued. The inspectors reviewed the LER when issued and found it to be complete and accurate except that the event date in Section 5 of the LER was listed as June 30, 1996, when the actual event, as discussed in the text of the LER, was June 29, 1996. The inspectors informed the licensing engineer of the date discrepancy.

The LER discussed some long-term corrective actions that remained outstanding. The LER will remain open pending the inspectors' review of the completed corrective actions.

II. Maintenance

M1 Conduct of Maintenance

M1.1 General Comments

a. Inspection Scope (61726, 62707)

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

- SP 1032A Safeguards Logic Test At Power-Train A
- SP 1032C Safeguards Boric Acid Logic Test (Trains A and B)
- SP 1035A Reactor Protection Logic Test At Power-Train A
- SP 1106A 12 Diesel Cooling Water Pump Test
- SP 1115 Monthly Power Distribution Map, Unit 1
- WO 9604475 Calibrate 22 Shield Building Special Ventilation System Temperature Switches
- WO 9606961 Installation of Stop Logs in Cooling Water System
- WO 9607089 Transfer 2R Transformer from Switchyard Bus 1 to 2
- WO 9607323 Measure Time Required for Containment Fan Coil Unit Fans to Stop

b. Observations and Findings

- For SP 1032A, 1032C, and 1035A the surveillances were conducted by two experienced instrument and control (I&C) specialists in the safeguards relay room and a recently licensed reactor operator (RO) in the main control room.

  The inspectors observed that the surveillances were well conducted and that the personnel involved were professional and paid close attention to procedure detail. The I&C specialists reviewed the steps of the procedures with the RO before beginning each surveillance to ensure the RO understood the procedures and the expected equipment responses. Communications during the surveillances were good. All equipment responded as expected.

- For WO 9604475 one of the temperature switches would not reset after tripping. The I&C specialist determined that a stiff wire inside the switch enclosure was interfering with the mechanism's ability to reset. It did not appear to affect the trip function which was the required action of the switch. The specialist repaired the switch.

  The inspectors, while observing the work activity, identified to the specialist that neither the work order nor the calibration sheets used by the specialists listed the acceptance band for the temperature switches. Due to the inspector questioning, the specialists contacted the system engineer to find out the correct

band. One of the switches was found to be outside the acceptance band but was returned to within the band for the as left condition.

The inspectors reviewed the Updated Safety Analysis Report (USAR) for the shield building special ventilation system. Section 5.3.2.2.4.3 of the USAR stated that the charcoal filter water deluge valves were actuated by U.L. approved fusible link switches manufactured to melt at 250°F. Actually the deluge valves were actuated by the temperature bulb type switches calibrated under WO 9604475. There had never been fusible links in the system so the USAR description has always been incorrect. The inspectors notified the system engineer. The discrepancy was considered an Inspection Followup Item pending the inspectors verifying that a change has been submitted to the USAR. (50-282/96008-03)

c. Conclusions

The observed maintenance and surveillance activities were well conducted with good communications, proper pre-job planning, and safe work practices.

M1.2 Failure to Promptly Identify and Correct an Adverse Condition

a. Inspection Scope (92902)

On July 26, 1996, the inspectors were informed that the licensee had discovered they had operated for a short period of time in a condition prohibited by Technical Specifications. The inspectors reviewed the details of the event.

b. Observations and Findings

On July 25, 1996, the licensee performed SP 1106A, "12 Diesel Cooling Water Pump Test," revision 48, a quarterly surveillance test on the 12 diesel-driven cooling water pump (DDCLP). As part of the test, the stroke time of valve CV-31423, the DDCLP heat exchanger outlet control valve, was measured. The stroke time was recorded as 0.32 seconds, which was outside the acceptance criteria of 0.5 to 1.5 seconds. The test was performed by a licensed reactor operator and the results were reviewed by the shift supervisor and shift manager who all failed to notice the unacceptable stroke time. Step 1.5.1 of SP 1106A required that for valve stroke times outside the acceptable range and less than the maximum time, the valve be immediately retested and if still outside the acceptable range, a work order be issued for an engineering evaluation within 96 hours. Since corrective action was not taken to immediately retest or evaluate the valve, the unacceptable stroke time rendered the 12 DDCLP inoperable.

On July 26, SP 1106B, a quarterly surveillance test on the 22 DDCLP was performed. Results of the test were satisfactory. During the test the 22 pump was logged out-of-service from 4:23 AM to 4:33 AM during the

engine cooldown prior to shutdown. During the entire period, the 121 motor-driven cooling water pump was not aligned for safeguards operation and thus could not be considered an operable cooling water pump to substitute for one of the DDCLPs. Thus for a ten minute period on July 26, all three of the safety-related cooling water pumps were technically inoperable, a condition not covered by Technical Specification 3.3.D limiting conditions for operation and action requirements.

Later in the morning on July 26, the superintendent systems engineering informed the shift manager of the situation and they determined that the plant had been in a condition prohibited by Technical Specifications and had unknowingly entered Technical Specification 3.0.C requiring a plant shutdown. The inspectors were notified and a operating log entry was made. As discussed in Section M3.1 of this report, the test on the 12 DDCLP was then reperformed and the stroke time of valve CV-31423 was found to be acceptable.

c. <u>Conclusions</u>

Technical Specification 3.3.D.1 for the cooling water system stated that a reactor shall not be made or maintained critical nor shall reactor coolant system average temperature exceed 200°F, unless the certain conditions are satisfied. The conditions specified included the limiting conditions and action requirements for either one or two of the five cooling water pumps inoperable. There was no allowed condition for three of the cooling water pumps inoperable.

The condition, which only existed for ten minutes, was not cited because later testing indicated that the stroke time of valve CV-31423 may have been mistimed and the system would have performed as designed even with the stroke time slightly faster than specified. Therefore the system operability specification was actually met. In addition, the 121 pump was available throughout the entire time and the 22 pump would have been available after it finished its cooldown cycle.

However, a reactor operator and two senior operators failed to promptly identify the original unacceptable results for the surveillance on the 12 pump and as a result it was not noticed and corrective actions were not taken for over 24 hours. Moreover, in this case the inspectors identified that the licensee did not properly followup the event and failed to report it per 10 CFR 50.73 or take corrective actions to ensure that similar surveillance failures would be promptly identified. Prairie Island Administrative Work Instruction 5 AWI 3.6.0, revision 4, step 6.8.1, required that if an event is determined to be a reportable event, Licensing and Management Issues department personnel shall complete Form 264, "Report Identification," which assures assignment of an individual(s) to prepare an investigative report. 5AWI 3.6.1, revision 5, step 6.1.2, required that the content of investigative reports shall be prepared in accordance with the guidance provided in the attached Table 2 which included, as Section 9, corrective action taken to prevent repetition of the event and of similar events.

Corrective action was not initiated until it was brought to their attention by the inspectors over 30 days later. The failure to report the event is discussed further in Section M8.1 of this report.

The failure of the operators to properly review the results of the surveillance test and identify the fact that valve CV-31423 had not met its test acceptance criteria for over 24 hours was an example of a violation of 10 CFR 50, Appendix B, Criteria XVI, which required that measures shall be established to assure that conditions adverse to quality are promptly identified and corrected. (50-282/96008-04)

Failure of the licensee to institute their investigative and corrective action processes to promptly correct the deficiencies which led to the failure to properly review the surveillance was a second example of a violation of 10 CFR 50, Appendix B, Criteria XVI. (50-282/96008-05)

An additional concern of the inspectors as a result of this event is discussed in Section M3.1 of this report.

M3 Maintenance Procedures and Documentation

M3.1 Inspectors' Concern With Technical Specification Interpretation

a. Inspection Scope (92902)

As a result of the discovery that valve CV-31423 had failed to meet its acceptance criteria during testing, as discussed in Section M1.2, SP 1106A was reperformed on July 26, 1996. The inspectors observed the test.

b. Observations and Findings

During that test the stroke time of valve CV-31423 was measured as 0.9 seconds, within the allowed band. The system engineer determined that the original stroke time may have been mistimed due to the extremely short time required to operate the stopwatch.

Also, during the test on July 26, the engine governor for the 12 DDCLP was observed to be hunting so the pump was considered inoperable and a work order was written to repair the governor. The governor was repaired and the surveillance test was satisfactorily performed on July 27. The 12 DDCLP had been considered inoperable for about a 24 hour period due to the governor problem.

Technical Specification 3.3.D.1.a, for the plant conditions at the time, stated that if one diesel-driven cooling water pump is inoperable, the 121 (motor-driven) cooling water pump shall (inspectors' emphasis) be aligned as a replacement for the inoperable DDCLP. Alignment of the 121 pump consisted of aligning valves to line the pump discharge to the appropriate cooling water header and aligning the pump's power supply to the appropriate safeguards bus. However, the licensee maintained a Technical Specification Interpretation (TSI 3.3-15, Revision 0) that

stated that they may elect not to satisfy Technical Specification 3.3.D.1.a and instead apply the requirements for two cooling water pumps inoperable contained in Technical Specification 3.3.D.2.a. That specification included a 7-day allowed outage time for the condition of a DDCLP inoperable and the 121 pump inoperable because it was not aligned for safeguards operation.

On July 26, the licensee elected to apply the interpretation and entered the 7-day allowed outage time instead of realigning the 121 pump. The condition existed for about 24 hours.

c. Conclusions

The inspectors were concerned that the interpretation gave no guidance of the conditions under which the licensee would expect to apply it. It basically allowed establishment of a less conservative (but allowed by Technical Specifications) condition to be elected by the operating shift. The inspectors were concerned that the interpretation did not require the consideration of risk or establish time limits within which the 121 pump should be realigned. During the 24-hour time period between the discovery of the problem with the 12 DDCLP governor and its subsequent repair, there was no overriding reason for not aligning the 121 pump as a replacement.

At the end of the inspection period the licensee was in the process of drafting a Technical Specification amendment request which would formalize its interpretation. Submittal of amendment request would allow resolution of the inspectors' concern because the NRC will conduct a safety evaluation of the practice as part of its review of the amendment. This issue is considered an Inspection Followup Item pending resolution of the technical specifications changes (50-282/96008-06).

M3.2 Typographical Error in Surveillance Procedure

a. Inspection Scope (93702)

On August 22, 1996, the inspectors were informed that the safety-related DDCLP #12 had started unexpectedly during a surveillance test of the other DDCLP. The inspectors reviewed the circumstances of the event.

b. Observations and Findings

During performance of surveillance SP 1106A, "12 Diesel Cooling Water Pump Test," Revision 48, the operator read step 7.8 which stated, "If ony [sic] one CL pump (11, 21, or 121) is running, then N/A Steps 7.9.1 through 7.9.4 and 7.58.1 through 7.58.4." The step contained a typographical error in the word "ony" which could easily be interpreted to have meant "any" or "only." Two pumps were running at the time but the operator apparently interpreted the word to be "any" and he wrote "N/A" in the indicated steps. The word was apparently intended to be "only" so the operator should have performed the steps. The steps not

performed included separating the two cooling water headers and establishing adequate flow in the A header for the pump being tested.

Later in the surveillance the operator was unable to establish adequate flow in the A header and determined that he should have performed the steps to separate the headers. He discussed the situation with the shift supervisor and they decided to perform the missed steps. After the operator closed an isolation valve between the two cooling water headers, flow in the B header increased and pressure dropped low enough that the 22 DDCLP automatically started to supply the B header. At that point the test was abandoned and the system lineup returned to normal.

The event was reported as an automatic actuation of an engineered safety feature system as required by 10 CFR 50.72. Action was initiated by the licensee to correct the typographical error and the licensee intended to issue a followup written report in accordance with 10 CFR 50.73.

c. Conclusions

10 CFR 50, Appendix B, Criteria V, required in part that activities affecting quality shall be prescribed by documented instructions, procedures, or drawings, of a type appropriate to the circumstances. Procedure SP 1106A, Revision 48, was not appropriate to the circumstances because it contained a word with a typographical error that could be interpreted to mean two different words of substantially different meaning. This resulted in an unnecessary challenge to a safety system when operators attempted to recover from a misinterpretation of the word.

The event had minor safety significance and the 22 DDCLP performed as designed to recover pressure in the B cooling water header. Both cooling water headers remained operable throughout the event. Thus this failure to meet the requirements of Criteria V constituted a violation of minor significance and is being treated as a Non-Cited Violation, consistent with Section IV of the NRC Enforcement Policy. (50-282/96008-07)

M8 Miscellaneous Maintenance Issues

M8.1 Failure to Report Operation Prohibited by Technical Specifications

a. Inspection Scope (92902)

While reviewing the events associated with the operation in a condition prohibited by Technical Specifications discussed in Section M1.2 of this report, the inspectors attempted to review the LER that should have been submitted within 30 days.

b. Observations and Findings

Thirty-three days after the event the inspectors asked licensee representatives why the NRC had not received the LER. The inspectors

were informed that the LER had not been submitted because licensee personnel failed to initiate actions to draft the LER after the event. As soon as the inspectors made licensee personnel aware of the failure to report, actions were initiated to draft the LER.

The licensee's administrative actions for ensuring timely submittal of an LER usually were initiated by the use of NRC Form 361, "Event Notification Worksheet," for events reported pursuant to 10 CFR 50.72. However, for operations prohibited by Technical Specifications, a report in accordance with 10 CFR 50.72 was only required if a shutdown was actually initiated. In this case, a shutdown was not initiated because the condition no longer existed by the time it was discovered. Thus no NRC Form 361 was initiated.

On July 26, 1996, upon discovery that the plant had been in a condition prohibited by Technical Specifications, an entry was made in the operations log summarizing the event and noting that it was reportable as an LER. However, no further action was apparently taken to ensure the report would be written.

c. Conclusions

10 CFR 50.73(a)(2)(i)(B) stated that the licensee shall report, within 30 days after discovery, any operation or condition prohibited by the plant's Technical Specifications. However, a condition prohibited by Technical Specifications involving the cooling water pumps was identified by the licensee on July 26, 1996, but was not reported in accordance with 10 CFR 50.73 within 30 days. This was a Violation. (50-282/96008-08) The LER had not been submitted at the end of the inspection period and will be reviewed by the inspectors when issued.

There have been other recent issues, including another NRC-identified failure to submit a report required by Technical Specifications discussed in Inspection Report 50-282(306)/96-07 and NRC-identified weaknesses in tracking biannual procedure reviews that indicated a need for additional licensee attention in implementing the administrative requirements of Section 6 of the Technical Specifications.

III. Engineering

E2 Engineering Support of Facilities and Equipment

E2.1 Cable Trays Did Not Meet Separation Criteria

a. Inspection Scope (37551, 92903)

On July 31, 1996, the licensee reported in accordance with 10 CFR 50.72 that they had been operating outside the design basis because they had discovered several cases of cable trays that didn't meet the separation criteria in the Updated Safety Analysis Report (USAR). The inspectors reviewed the circumstances surrounding the finding, reviewed the operability determination, and conducted walkdowns of selected areas of

the plant to determine if the licensee had properly bounded the scope of the condition.

b. Observations and Findings

In February 1992, licensee personnel first noted that some cable tray installations did not appear to meet the separation criteria of Section 8.7.2 of the USAR. Quality Control (QC) surveillance S0248 documented several cases. The issue was entered into the licensee's Follow On Item (FOI) tracking system as FOI A0688. The FOI system was in use at the time to resolve issues generated as part of the design basis reconstitution program.

On May 4, 1993, an initial assessment of the FOI was completed. The assessment included the results of an engineering walkdown of the original QC findings, a preliminary operability determination, a preliminary reportability determination, and proposed corrective actions to resolve the discrepancies. At that time the issues were determined to be valid but not reportable. The installations were determined to be operable, but not in accordance with the USAR. The corrective actions proposed included the following: install cable tray barriers where required; conduct a detailed evaluation against the requirements of current electrical standards; revise the USAR with the results of that evaluation; and evaluate the discrepancies for potential seismic interaction concerns. This was then assigned a priority in the FOI system to be followed up at a later date.

On June 26, 1996, Safety Evaluation (SE) 371 of the cable tray discrepancies was brought before the Operation Committee (OC) for review. The SE was not approved at that meeting but the issue was again determined not to be reportable. The SE was presented again to the OC on July 24 and July 31. At the July 31 OC meeting the issue was determined to be reportable as operation outside the design basis.

In their review of the issue the inspectors noted that in over four years since the original discovery, the licensee had apparently made no effort to determine if the scope of the problem went beyond the cable trays identified in the original QC surveillance. The inspectors were informed by engineering personnel that they thought the discrepancies were limited to trays added as part of the event monitoring instrumentation added as a Three Mile Island action requirement in the early 1980s. The inspectors noted several trays listed in the safety evaluation that appeared not to be a part of that modification.

Over the next few days the inspectors conducted independent walkdowns of selected areas of the plant and noted several additional cable tray installations that did not appear to conform to the USAR requirements. These trays included ones from original plant construction and later modifications not associated with event monitoring instruments. The inspectors informed plant engineering personnel of the findings so they

could be evaluated for operability. In two of the cases identified by the inspectors, plant drawings showed fire barriers should have been installed between the trays but apparently never were.

One of the installations found by the inspectors was determined not to be able to meet the operability criteria of SE 371 and was reported to the NRC in accordance with 10 CFR 50.72 on August 9, 1996. The tray interactions involved redundant pressurizer heater circuits. The licensee took immediate corrective actions to administratively maintain one set of cables deenergized until fire barriers were installed on August 13.

As a result of the inspectors' findings, the licensee established a project team to expand the inspections for cable tray interactions and develop further corrective actions. The licensee made a followup written report of details of the findings and their planned corrective actions as LER 50-282(306)/96-13 on August 30, 1996. The LER is considered open pending further review by the NRC Region III Division of Reactor Safety.

c. Conclusions

The inspectors concluded that the licensee's evaluation of this issue was untimely and narrowly focused. It took over four years to complete the safety evaluation and to determine that the configurations were outside the plant's design basis and therefore reportable. Until prompted by additional NRC findings, the licensee's investigation of the issue involved only those tray interactions listed in the original QC surveillance despite evidence in the original list that the interactions might not be limited to event monitoring instrument cables.

This issue is considered an Unresolved Item pending further review by the Region III Division of Reactor Safety. (50-282/96008-09)

Because of the way the licensee handled this issue, the NRC was concerned that other open FOIs may not have originally been properly evaluated for operability and reportability. As part of the corrective actions discussed in LER 50-282(306)/96-13, the licensee committed to reviewing a sampling of the 93 remaining open FOIs by October 31, 1996.

E2.2 Concerns With Containment Fan Cooler Operability

a. Inspection Scope (37551, 92903)

On June 25, 1996, Westinghouse Electric Corporation issued Nuclear Safety Advisory Letter (NSAL) 96-003 discussing a potential problem with boiling and subsequent water hammer in the cooling water piping associated with containment fan coolers during a loss of coolant accident coincident with a loss of offsite power. On August 12, 1996, the NRC issued Information Notice (IN) 96-45, "Potential Common-Mode Post-Accident Failure of Containment Coolers," on the same subject. The inspectors monitored the licensee's evaluation of the issue.

b. Observations and Findings

Several conference calls were held between the licensee and various NRC offices during the course of the evaluation of the issue. The licensee initially stated that they believed the plant was not susceptible to the common mode failure described in the NSAL and IN because the diesel-driven cooling water pumps (DDCLPs) would start and provide adequate cooling water flow before boiling would occur in the containment coolers.

However, on August 27, 1996, the licensee completed a preliminary operability evaluation which included a description of a potential case where one of the DDCLPs was out-of-service and the motor-driven 121 cooling water pump was supplying the cooling water header. In that case initial calculations indicated boiling could occur in the system before flow was re-established from the motor-driven pump. The licensee's evaluation concluded that the system would still perform its safety function when certain conservative assumptions in the calculation were reevaluated. At the end of the inspection period, the licensee was continuing to evaluate the issue with more detailed calculations.

c. Conclusions

The inspectors were concerned that the licensee's preliminary operability determination appeared to be untimely. The licensee was informed of the issue when they received NSAL 96-003 in late June and didn't complete the operability determination until about two months later.

This issue is considered an Unresolved Item pending further review by the Region III Division of Reactor Safety. (50-282/96008-10)

E2.3 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/or parameters to the USAR descriptions. While performing the inspections discussed in this report, the inspectors reviewed the applicable portions of the USAR 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:

The inspectors noted that Section 5.3.2.2.4.3 of the USAR stated that the charcoal filter water deluge valves in the shield building special ventilation system were actuated by U.L. approved fusible link switches manufactured to melt at 250°F. Actually the deluge valves were actuated by the temperature bulb type switches. There had never been fusible links in the system so the USAR description has always been incorrect. The issue was considered an Inspection Followup Item as discussed in Section M1.1.b of this report.

The inspectors found several cases in addition to those identified by the licensee where cable tray separation did not meet the criteria listed in Section 8.7.2 of the USAR. The issue was considered an Unresolved Item as discussed in Section E2.1 of this report.

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. A routine inspection by a regional specialist took place during the inspection period and the results will be documented in a separate report.

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. The inspectors noted that the licensee was closely monitoring the status of road construction in the area of the plant to ensure that adequate evacuation routes were maintained.

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.

F1 Control of Fire Protection Activities (71750)

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

V. Management Meetings

X1 Exit Meeting Summary

The inspectors presented the inspection results to members of the licensee management at the conclusion of the inspection on August 29, 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.

PARTIAL LIST OF PERSONS CONTACTED

<u>Licensee</u>

M. Wadley, Plant Manager
K. Albrecht, General Superintendent Engineering
J. Goldsmith, General Superintendent Design Engineering
J. Hill, Manager Quality Services
G. Lenertz, General Superintendent Plant Maintenance
J. Leveille, Licensing Engineer
R. Peterson, Design Standards Principal Engineer
D. Schuelke, General Superintendent Radiation Protection and Chemistry
M. Sleigh, Superintendent Security
J. Sorensen, General Superintendent Plant Operations

INSPECTION PROCEDURES USED

IP 37551:	Onsite Engineering
IP 40500:	Effectiveness of Licensee Controls in Identifying, Resolving, and Preventing Problems
IP 61726:	Surveillance Observations
IP 62707:	Maintenance Observations
IP 71707:	Plant Operations
IP 71750:	Plant Support Activities
IP 92700:	Onsite Followup of Written Reports of Nonroutine Events at Power Reactor Facilities
IP 92901:	Followup - Operations
IP 92902:	Followup - Maintenance
IP 92903:	Followup - Engineering
IP 93702:	Prompt Onsite Response to Events At Operating Power Reactors

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

50-282/96008-01	IFI	Shift Manning Technical Specification Doesn't Meet 10 CFR 50.54 Requirements
50-282/96008-02	IFI	Questions Regarding Which Duties can be Credited for Maintaining Operator Licenses in an Active Status
50-282(306)/96-12	LER	Loss of Offsite Power to Unit 2 and Degraded Offsite Power to Unit 1 Followed by Reactor Trips of Both Units
50-282/96008-03	IFI	Error in Updated Safety Analysis Report Description of Fire Protection Actuation Switches
50-282/96008-04	VIO	Failure to promptly identify a condition adverse to quality
50-282/96008-05	VIO	Failure to take timely corrective actions for a condition adverse to quality
50-282/96008-06	IFI	Voluntarily entry into Technical Specification for inoperable motor driven cooling water pump
50-282/96008-07	NCV	Surveillance Procedure Inappropriate for the Circumstances due to Typographical Error
50-282/96008-08	VIO	Failure to Submit Required Licensee Event Report Within 30 Days
50-282/96008-09	URI	Cable Tray Installations Which Didn't Meet Separation Criteria in the Updated Safety Analysis Report
50-282(306)/96-13	LER	Cable Tray Separation Discrepancies
50-282/96008-10	URI	Operability of Containment Fan Coolers

LIST OF ACRONYMS USED

CFR	Code of Federal Regulations
CL	Cooling Water
DDCLP	Diesel-Driven Cooling Water Pump
°F	Degrees Fahrenheit
FOI	Follow On Item
I&C	Instrument and Controls
IFI	Inspection Followup Item
IN	Information Notice
IP	Inspection Procedure
LER	Licensee Event Report
N/A	Not Applicable
NCV	Non-Cited Violation
NRC	Nuclear Regulatory Commission
NRR	Nuclear Reactor Regulation
NSAL	Nuclear Safety Advisory Letter
OC	Operations Committee
PDR	Public Document Room
%	Percent
QC	Quality Control
RO	Reactor Operator
SE	Safety Evaluation
SP	Surveillance Procedure
SM	Shift Manager
U.L.	Underwriters Laboratory
URI	Unresolved Item
USAR	Updated Safety Analysis Report
VIO	Violation
WCC	Work Control Center
WO	Work Order