December 12, 2000

Mr. M. Hammer
Site General Manager
Monticello Nuclear Generating Plant
Nuclear Management Company, LLC
2807 West County Road 75
Monticello, MN 55362-9637

SUBJECT: MONTICELLO - NRC INSPECTION REPORT 50-263/00-17(DRS)

Dear Mr. Hammer:

On November 17, 2000, the NRC completed the baseline annual inspection of Evaluations of Changes, Tests, or Experiments (10 CFR 50.59) and the baseline biennial Permanent Plant Modifications inspection at your Monticello Nuclear Generating Plant. The enclosed report presents the results of that inspection which were discussed on November 17, 2000, with Mr. M. Hammer, Mr. B. Day, and other members of your staff.

This inspection was an examination of activities conducted under your license as they relate to changes to facility structures, systems, and components, normal and emergency procedures, and the Updated Safety Analysis Report in accordance with the requirements of 10 CFR 50.59, and changes to the facility via permanent plant modifications to verify compliance with the Commission’s rules and regulations and with the conditions of your license. Within these areas, the inspection consisted of a selected examination of design documents, procedures, and representative records, and interviews with personnel.

Based on the results of this inspection, the NRC identified two issues that were determined to be of very low safety significance. The two issues were considered violations of NRC regulations which involved failures to properly conduct 10 CFR 50.59 screening or evaluations in accordance with station procedures. However, the violations were not cited due to their very low safety significance and because they have been entered into your corrective action program. These issues are discussed in the summary of findings and in the body of the attached report.

If you contest these Non-Cited Violations, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, D.C. 20555-0001, with a copy to the Regional Administrator, Region III; the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington, D.C. 20555-0001, and the NRC Resident Inspector at the Monticello Nuclear Generating Plant.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter and its enclosure will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARS) component of NRC's document system (ADAMS). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/NRC/ADAMS/index.html (the Public Electronic Reading Room).

We will gladly discuss any questions you have concerning this inspection.

Sincerely,

/RA/

Ronald N. Gardner, Chief
Electrical Engineering Branch
Division of Reactor Safety

Docket No. 50-263
License No. DPR-22

Enclosure: Inspection Report 50-263/00-17(DRS)

cc w/encl: Site General Manager, Monticello
Plant Manager, Monticello
M. Wadley, Chief Nuclear Officer
S. Northard, Nuclear Asset Manager
M. Roth, Site Licensing Manager
J. Malcolm, Commissioner, Minnesota
  Department of Health
J. Silberg, Esquire
  Shaw, Pittman, Potts, and Trowbridge
R. Nelson, President
  Minnesota Pollution Control Agency
Commissioner, Minnesota Pollution Control Agency
D. Gruber, Auditor/Treasurer
  Wright County Government Center
Commissioner, Minnesota Department of Commerce
A. Neblett, Assistant Attorney General

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U.S. NUCLEAR REGULATORY COMMISSION

REGION III

Docket No:	50-263
License No:	DPR-22
Report No:	50-263/00-17(DRS)
Licensee:	Northern States Generating Company
Facility:	Monticello Nuclear Power Plant
Location:	2807 West Highway 75 Monticello, MN 55362
Dates:	November 13 - 17, 2000
Inspectors:	R. Daley, Reactor Engineer M. Farber, Reactor Engineer G. O'Dwyer, Reactor Engineer S. Sheldon, Reactor Engineer R. Winter, Reactor Engineer
Approved by:	Ronald N. Gardner, Chief Electrical Engineering Branch Division of Reactor Safety

NRC's REVISED REACTOR OVERSIGHT PROCESS

The federal Nuclear Regulatory Commission (NRC) recently revamped its inspection, assessment, and enforcement programs for commercial nuclear power plants. The new process takes into account improvements in the performance of the nuclear industry over the past 25 years and improved approaches of inspecting and assessing safety performance at NRC licensed plants.

The new process monitors licensee performance in three broad areas (called strategic performance areas) reactor safety (avoiding accidents and reducing the consequences of accidents if they occur), radiation safety (protecting plant employees and the public during routine operations), and safeguards (protecting the plant against sabotage or other security threats). The process focuses on licensee performance within each of seven cornerstones of safety in the three areas:

Reactor Safety

- Initiating Events
- Mitigating Systems
- Barrier Integrity
- Emergency Preparedness

Radiation Safety

- Occupational
- Public

Safeguards

- Physical Protection

To monitor these seven cornerstones of safety, the NRC uses two processes that generate information about the safety significance of plant operations: inspections and performance indicators. Inspection findings will be evaluated according to their potential significance for safety, using the Significance Determination Process, and assigned colors of GREEN, WHITE, YELLOW or RED. GREEN findings are indicative of issues that, while they may not be desirable, represent very low safety significance. WHITE findings indicate issues that are of low to moderate safety significance. YELLOW findings are issues that are of substantial safety significance. RED findings represent issues that are of high safety significance with a significant reduction in safety margin.

Performance indicator data will be compared to established criteria for measuring licensee performance in terms of potential safety. Based on prescribed thresholds, the indicators will be classified by color representing varying levels of performance and incremental degradation in safety: GREEN, WHITE, YELLOW, and RED. GREEN indicators represent performance at a level requiring no additional NRC oversight beyond the baseline inspections. WHITE corresponds to performance that may result in increased NRC oversight. YELLOW represents performance that minimally reduces safety margin and requires even more NRC oversight. And RED indicates performance that represents a significant reduction in safety margin but still provides adequate protection to public health and safety.

The assessment process integrates performance indicators and inspection so the agency can reach objective conclusions regarding overall plant performance. The agency will use an Action Matrix to determine in a systematic, predictable manner which regulatory actions should be taken based on a licensee's performance. The NRC's actions in response to the significance (as represented by the color) of issues will be the same for performance indicators as for inspection findings. As a licensee's safety performance degrades, the NRC will take more and increasingly significant action, which can include shutting down a plant, as described in the Action Matrix.

More information can be found at: http://www.nrc.gov/NRR/OVERSIGHT/index.html

SUMMARY OF FINDINGS

IR 50-263/00-17(DRS), on November 13 - 17, 2000, Nuclear Management Company, LLC, Monticello Nuclear Generating Plant. Permanent Plant Modifications, and the Evaluations of Changes, Tests, or Experiments in accordance with 10 CFR Part 50.59.

The inspection was conducted by reactor engineers from the Division of Reactor Safety. There were two no-color findings identified during this inspection; both were considered Non-Cited Violations. The significance of findings is indicated by their color (Green, White, Yellow, Red) using Inspection Manual Chapter 0609 "Significance Determination Process," (SDP). Findings for which the SDP does not apply are indicated by "no color."

Cornerstone: Mitigating Systems

- No color. The licensee failed to follow the station procedure requirements for preparing a 10 CFR 50.59 screening or evaluation for Updated Safety Analysis Report (USAR) changes that resulted from implementation of criticality accident controls in accordance with 10 CFR 50.68. This is considered a Non-Cited Violation of 10 CFR 50, Appendix B, Criterion V. This violation was identified by the NRC and promptly entered by the licensee into the corrective action program as Condition Report 20004536.

  There was no significant impact to the cornerstone because the licensee's Safety Review Item evaluation ensured that criticality accident monitoring requirements were met by demonstrating full compliance with 10 CFR 50.68. Changes to the USAR necessitated by the Safety Review Item were required to be made in accordance with 10 CFR 50.59, but were not (Section 1R02).

- No color. The licensee failed to follow station procedure requirements for preparing a 10 CFR 50.59 evaluation that resulted from a modification that bypassed the Emergency Core Cooling System load shed trip/lockout signal to the Residual Heat Removal Service Water (RHRSW) pumps following a design basis Loss of Coolant Accident. The evaluation failed to address the appropriateness of bypassing the interlock and the acceptability of deleting the USAR wording which described the interlock. This is considered a Non-Cited Violation of 10 CFR 50, Appendix B, Criterion V. This violation was identified by the NRC and promptly entered by the licensee into the corrective action program as Condition Report 20004494.

  There was no significant impact to the cornerstone because it hypothesized the extremely low probability, simultaneous occurrence of a Loss of Coolant Accident and Loss of Offsite Power. Loss of offsite power, in conjunction with a loss of coolant accident would require load shedding and sequencing, which would trip running RHRSW pumps and necessitate clearing the interlock in order to restart them (Section 1R17).

Report Details

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems and Barrier Integrity

1R02 Evaluations of Changes, Tests or Experiments (IP 71111, Attachment 2)

.1 Review of 10 CFR 50.59 Evaluations and Screenings

a. Inspection Scope

The team reviewed 16 evaluations performed pursuant to 10 CFR 50.59. The evaluations related to permanent plant modifications, Safety Review Items (SRI) setpoint changes, procedure changes, and changes to the Updated Safety Analysis Report (USAR). The team also reviewed 27 screenings where the licensee had determined that a 10 CFR 50.59 evaluation was not necessary.

b. Findings

Criticality Accident Monitoring Provisions

The licensee prepared and implemented Safety Review Item 99-003, Revision 0, to show that the plant met the requirements defined in 10 CFR 50.68, "Criticality Accident Requirements." Issued in 1998, 10 CFR 50.68 allowed licensees to meet either the requirements in 10CFR 70.24 or the newly established requirements contained within 10 CFR 50.68 for criticality accident monitoring.

This SRI was issued to demonstrate and document that controls were in place in accordance with 10 CFR 50.68, to ensure continued compliance, and involved changes to several pages of the Monticello plant USAR. While reviewing SRI 99-003, Revision.0, the inspector identified that a 10 CFR 50.59 Applicability Screening was not performed. This omission is contrary to the procedural requirements contained within Monticello procedure 4 AWI-05.06.01. Procedural step 4.3.3 requires performance of a 10 CFR 50.59 Applicability Screening for all SRI packages.

Additionally, a 10 CFR 50.59 evaluation in accordance with 4AWI-05.06.03, 10 CFR 50.59 evaluations, was not performed for the SRI changes to the Monticello USAR. While the SRI ensured that criticality accident monitoring requirements were met by demonstrating full compliance with the requirements contained within 10 CFR 50.68, changes to the USAR necessitated by the SRI were required to be made in accordance with 10 CFR 50.59.

The failure to perform a screening or an evaluation in accordance with station procedures is a violation of 10 CFR 50, Appendix B, Criterion V. This violation is considered a Non-Cited Violation, (50-263/00-17-01(DRS)), consistent with the General Statement of Policy and Procedure for NRC Enforcement Actions (NUREG 1600) (Enforcement Policy), Section VI.A.1. This violation was identified by the NRC and promptly entered into the corrective action program by the licensee as Condition Report (CR) 20004536.

1R17 Permanent Plant Modifications (IP 71111, Attachment 17)

.1 Review of Recent Permanent Plant Modifications

a. Inspection Scope

The team reviewed 14 permanent plant modifications that were installed in the last several years. The modifications were chosen based upon their affecting systems that had high risk significance in the licensee's Individual Plant Evaluation or high maintenance rule safety significance. Most of the modifications involved changes to mitigating systems. The team reviewed the modifications to verify that the completed design changes were in accordance with the specified design requirements and the licensing bases and to confirm that the changes did not affect any systems' safety function. Design and post-modification testing aspects were verified to ensure the functionality of the modification, its associated system, and any support systems. The team also verified that the modifications performed did not place the plant in an increased risk configuration.

b. Findings

Residual Heat Removal Service Water (RHRSW) Pump Emergency Core Cooling System (ECCS) Load Shed Bypass

Design change 98Q140, “RHRSW Pump ECCS Load Shed Bypass,” was performed to allow for an easier method for bypassing the ECCS load shed trip/lockout signal to the RHRSW pumps following a design basis loss of coolant accident. This interlock prevented starting an RHRSW pump unless reactor vessel level was greater than 48 inches and RHR pumps for the associated AC bus were secured. Design Change 98Q140 eliminated the need for an operator to install jumpers and contact boots to get the RHRSW pumps started (and torus cooling on-line) within the time assumed in the Monticello accident analysis.

As a result of the modification, wording in the USAR was required to be changed to reflect the new design. Specifically, the following USAR wording was deleted:

> For each operating division of RHR, after adequate core cooling has been established and one of the running RHR pumps is manually secured, cleared interlocks will permit both RHR Service Water pumps in the associated division to be manually started.

While a 10 CFR 50.59 evaluation was performed regarding this design and USAR change, the 10 CFR 50.59 evaluation relied on the already established use of jumpers and contact boots as a justification for the modification to the control logic. The evaluation never discussed the appropriateness of bypassing the interlock and the acceptability of deletion of the USAR wording which described the interlock. Additionally, licensee personnel could provide no such evaluation for the original use of jumpers and contact boots to bypass the interlock. Without an adequate safety evaluation, the inspectors questioned whether or not the modification of the interlock circuit constituted an unreviewed safety question.

During discussions, licensee staff pointed out that on a design basis loss of coolant accident, water level could not reach 48 inches; it would stop at ⅔ core height. Consequently, the logic for the interlock could never be satisfied (cleared). Recognizing that establishing torus cooling was required for continued accident mitigation, the licensee elected to procedurally install a jumper to bypass the interlock, allowing restoration of RHRSW and initiation of torus cooling. The licensee staff considered the jumpering of the interlock as the equivalent of clearing it. The inspectors considered bypassing the logic as distinctly different from meeting the physical conditions necessary to satisfy it, were concerned by this philosophy, and referred this issue to station management for evaluation.

The failure to perform an adequate evaluation, in accordance with station procedures, to justify the acceptability of bypassing the RHRSW Pump ECCS Load Shed Bypass interlock, is a violation of 10 CFR 50, Appendix B, Criterion V. This violation is considered a Non-Cited Violation (50-263/00-17-02(DRS)), consistent with the General Statement of Policy and Procedure for NRC Enforcement Actions (NUREG 1600) (Enforcement Policy), Section VI.A.1. This violation was identified by the NRC and promptly entered by the licensee into the corrective action program as CR 2000-4494.

Attention to Detail in Engineering Activities

While reviewing samples of the various engineering activities encompassed by permanent plant modifications, the inspectors found recurrent examples of lack of attention to detail. These activities included design changes, jumper/bypasses, calculations, setpoint changes, procedure changes, and safety review items. Examples of these deficiencies included:

- Setpoint change request: 99-016 P111A-“ESW 480VAC supply, Overload trip setpoint change.” No formal calculation was done to show whether or not the fuse/breaker coordination study was adversely impacted nor was any analysis done to show that there was no adverse impact on other systems. An informal calculation and analysis were done and results were included in a related CR which was not referenced. Consequently there was no documented basis for the setpoint change.

- When installing new sensing lines for a turbine pressure signal to an interlock, no pressure rating could be found in the documentation for the original lines. Based on a normal operating pressure of 697 pounds as identified in the USAR, engineering selected a pressure rating of 750 pounds, giving an apparent margin of 53 pounds. When the inspectors pointed out that a combined intercept valve closure would cause a pressure spike, engineering calculated the pressure that would occur as 744 pounds, a margin of six pounds.
- When modifying the air operator for the torus vacuum breaker to double stroke speed, engineers chose to ignore increased inertial forces due to increased valve speed based on a phone call from a vendor representative.

- To resolve an issue (CR 20003071) concerning the ability of the HPCI pump to deliver required flow to the vessel with the minimum flow valve failed open, licensee engineers relied on phone call with a vendor representative to substantiate greater than rated capability of HPCI turbine.

  In dealing with the same HPCI minimum flow issue, engineering referenced Calculation 97-0232 which evaluated the required submergence of the HPCI suction line to avoid vortexing. The engineers did not recalculate the required submergence with the additional 600 gallons per minute attributed to a failed open minimum flow valve, but stated that the additional flow would not significantly impact the required submergence. The inspectors recalculated the submergence with the additional flow and found that for the limiting case, required submergence was increased by approximately two feet.

- For a battery ventilation fan (EF-40) motor change out, engineering did not assess motor starting current in the fuse/breaker coordination study, on the assumption that the motor was small. When starting currents were received from the vendor and actually finally plotted, it was revealed that current thermal overload settings did not completely protect the motor.

- A 10 CFR 50.59 evaluation was prepared because of revision to a drawing in USAR. After inspection, it was revealed that drawing was not in the USAR.

While none of these deficiencies were significant enough to raise operability questions, they represent a pattern of lack of rigor and attention to detail. This was referred to station management for their evaluation.

4. OTHER ACTIVITIES

4OA5 Management Meetings

Exit Meeting Summary

The inspectors presented the inspection results to Mr. M. Hammer, Site General Manager, and other members of licensee management and staff at the conclusion of the inspection on November 17, 2000. The licensee acknowledged the findings presented.

PARTIAL LIST OF PERSONS CONTACTED

Licensee

P. Albares, Engineering
P. Burke, Engineering
B. Day, Plant Manager
S. Engelke, Engineering
R. Frederickson, Technical Services
J. Grubb, Engineering
M. Hammer, Site General Manager
S. Hammer, Engineering
S. Ludders, Engineering
J. Nelson, Site Supply
M. Petitclair, Engineering
J. Rootes, Quality Services
C. Schibonski, Safety Assessment
R. Seipel, Engineering
S. Shirey, Licensing
D. Wegner, Engineering
D. Zerchner, Engineering

NRC

S. Burton, Senior Resident Inspector
D. Kimble, Resident Inspector

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

(NCV) 50-263/2000017-01	Failure to perform a 10 CFR 50.59 screening or evaluation for USAR changes made in accordance with 10 CFR 50.68
(NCV) 50-263/2000017-02	Failure to peform an adequate 10 CFR 50.59 evaluation for modification of interlock circuits described in the USAR

Closed

(NCV) 50-263/2000017-01	Failure to perform a 10 CFR 50.59 screening or evaluation for USAR changes made in accordance with 10 CFR 50.68
(NCV) 50-263/2000017-02	Failure to peform an adequate 10 CFR 50.59 evaluation for modification of interlock circuits described in the USAR

Discussed

None

LIST OF ACRONYMS USED

ADAMS	Agency Wide Documents Access and Management System
CFR	Code of Federal Regulations
CR	Condition Report
DRS	Division of Reactor Safety
ECCS	Emergency Core Cooling System
LC	Load Center
NRC	Nuclear Regulatory Commission
PARS	Publicly Available Records
PERR	Public Electronic Reading Room
RHRSW	Residual Heat Removal Service Water
SGTS	Standby Gas Treatment System
SRI	Safety Review Item
TS	Technical Specifications
USAR	Updated Safety Analysis Report

LIST OF DOCUMENTS REVIEWED

The following is a list of licensee documents reviewed during the inspection, including documents prepared by others for the licensee. Inclusion on this list does not imply that NRC inspectors reviewed the documents in their entirety, but, rather that selected sections or portions of the documents were evaluated as part of the overall inspection effort.

Condition Reports (CR) Nos.

19991403, EDG Droop Setting and Frequency Considerations While in Standby Mode, May 24, 1999
19991446, Modification of Instrument Tubing, Revision 0, June 4, 1999
19991515, Non-safety related steel requisitioned for safety-related mod.
19991654, Outboard MSIV Limit Switch, Revision 0, June 28, 1999
19992771, Incorrect Seismic Criteria USAR Wording, Revision 0, June 6, 2000
19992779, Incorrect CST Inventory USAR Wording, Revision 0, September 28, 1999
19992822, Installation of Incorrect Pipe Schedule for CST-189, September 21, 2000
19992965, 10 CFR 50, Part 50.59, Changes, October 4, 1999
19993486, Numerous Defects in valves purchased for installation
20000194, Firestop Sealant is Bonding to Blowout Panels and may require excessive force for panels to relieve during HELB, January 14, 2000
20000208, HPCI CV-2065 air accumulator Check Valve failed Leak Rate Test 0255-06-ID-3, January 14, 2000
20000307, New 14 ESW Pump Vibration Levels Increased significantly between Preoperational Testing Runs, January 19, 2000
20000392, Clarification of EDG Loading Acceptance Criteria, January 22, 2000
20000705, Unauthorized Modification to Instrument Air Line without a Work Order - Air Line not in Service at the Time, February 11, 2000
20000727, Change 99Q020 Removed Needed Plant Consulting I&C, February 2, 2000
20000956, BWROG Raised Concerns about the Local Suppression Pool Temp Limits and the Supporting Analysis, February 2, 2000

20001323, Inadequate procedure Revisions for #11 CW Pump Exciter Power Supply Modification, March 24, 2000
20002342, CS check valve bypass lines not analyzed for higher pressure rating under DCP.
20002614, Jumper Bypass 00-22 stated calculations were revised; however, the calculations were not approved prior to installation
20002683, Eng. Evaluation for Jumper Bypass 00-106 did not address impact of chiller weight on HPCI building roof
20003071, Allowable leak rate evaluation for HPCI min flow air accumulator check valve, AI-611, may not have been bounding
20004185, Revise Procedure 7130 to Incorporate Changes Resulting From SCR 00-024 for PT-23-100, October 31, 2000

Procedures

4AWI-02.07.01, Updated Safety Analysis Report (USAR) Control, Revision 3, June 30, 2000
4AWI-05.01.01, Introduction Design Change Process, Revision 7, November 21, 1996
4AWI-05.01.04, General Instructions for Design Changes, Revision 3, November 21, 1996
4AWI-05.01.05, Project Initiation, Revision 8, August 2, 2000
4AWI-05.01.06, Design Input, Revision 2, February 16, 1999
4AWI-05.01.07, Design Document Review, Revision 1, November 21, 1996
4AWI-05.01.08, Specifications and Drawings, Revision 2, February 11, 1999
4AWI-05.01.09, Design Checking and Verification, Revision 4, November 21, 1999
4AWI-05.01.10, Project Descriptions, Revision 5, July 17, 2000
4AWI-05.01.11, Design Change Package Content, Revision 2, June 1, 1999
4AWI-05.01.12, Design Change Installation Plan, Revision 1, November 21, 1996
4AWI-05.01.13, Design Change Package Review and Approval, Revision 8, June 4, 1999
4AWI-05.01.14, Installation and Test Procedures, Revision 3, October 17, 2000
4AWI-05.01.15, Engineering Change Request, Revision 1, November 21, 1996
4AWI-05.01.16, Turnover for Operation, Revision 5, May 19, 2000
4AWI-05.01.17, Design Change Closeout, Revision 3, October 9, 1997
4AWI-05.01.18, Design by Consultants, Revision 1, November 21, 1996
4AWI-05.01.19, Generic Design Change Process, Revision 2, August 26, 1997
4AWI-05.06.02, 10CFR 50.59 Applicability Screening, Revision 3, November 10, 2000
4AWI-05.06.03, 10 CFR 50.59 Evaluations, Revision 1
4 AWI-08.03.02, Programmable Digital Equipment Requirements, Rev 1, June 30, 1999
4 AWI-08.03.03, Software Quality Assurance Requirements, Rev 3, January 30, 1998
7130, HPCI System Instrument Maintenance Procedure, Rev 13*, November 29, 1999
8247, AGASTAT Relay Replacement, Rev 4, March 3, 2000
MWI-8-M-4.17, Motor Replacement and Termination, Revision 1, February 22, 1999
SGP-07.03, USAR Change Procedure, Revision 1, September 26, 2000
Licensing Group Procedure 12.1, Updated Safety Analysis Report, Revision 8, December 29, 1998

Design Changes

98Q125, Main Steam Line Drain replacement, Revision 1, October 20, 1999.
98Q130, CST check valves to ECCS pump replacement, Revision 4, August 5, 1999.
99Q005, CRD piping replacement, Revision 0, February 20, 2000.
99Q110, “B” RHR room block wall modification, Revision 0, May 27, 1999

96Q175, Digital Feedwater Replacement, Rev 0, October 26, 1999
98Q105, Generic Timing Relay Replacement, Rev 0, February 18, 1999
98Q140, RHRSW pump ECCS load-shed bypass
98Q175, Miscellaneous Fire Protection Modification
99Q010, Offgas Instrument Replacement, Revision 0, Addendum/Part 0/0, August 27, 1999
99Q055, Improvements for MO-2014, MO-2015, MO-2035, MO-2397 & MO-2398, Revision 0, August 30, 2000
99Q095, HP Turbine Sensing Line/Vacuum Breaker FME, Revision 1, November 15, 1999
99Q095, MO-2397 Control Circuit Change, Revision 0, May 14, 1999
99Q145, Replacement of CRD-113 valves
99Q215, Outboard MSIV Limit Switch Service Temperature Reduction, Rev 0, December 22, 1999
99Q220, Emergency diesel droop projects
00Q010, Torus Vacuum Breaker Tubing Replacement, Revision 0, February 3, 2000
00Q027, SBGT Makeup Air Improvements, Revision 0, February 24, 2000
00Q180, V-EF-40A & B Fan Motor Replacement, May 12, 2000

Safety Review Items

98-004, Operability of Div. II 125 VDC loads are cross-tied to Div. II 125 VDC battery
98-019, HPCI Discharge Line Temporary Pressurization, Revision 0, January 15, 1999
99-001, 1997 & 1998 ISI Outage and Pre-outage Discrepancies As-built Conditions, Revision 0, January 10, 2000
99-003, Compliance with 10 CFR 50.68 (criticality accident requirements)
99-006, Clarification of Plant Shielding Review Discrepancies in USAR, Revision 0, October 26, 1999
99-014, USAR Appendix I Changes, Revision 0, March 14, 2000
00-006, CR 19982250/Action 19982820Single Failure of RHR Heat Exchanger Bypass Valve
00-012, Elimination of Local Suppression Pool Temperature Limits, April 17, 2000
00-024, Operating with One PRessure Regulator in Service, Revision 0, November 9, 2000

Setpoint Changes

99-004, ECCS Valve Permissive Switch, February 17, 1999
99-007, HPCI Steam Line Hi Area Temp Isolation, July 14, 1999
99-009, RCIC Steam Line Hi Area Temp Isolation, April 15, 1999
99-016,P-111A #11 ESW Pump 480 Supply, May 26, 1999
99-018, Main Steam Line low pressure isolation switches
99-019, Reactor high pressure scram switches
99-021, HPCI high steam flow isolation
99-028, OGHU TR A Inlet Relief Control Gain and Reset, September 16, 1999
00-009, Main Steam Tunnel High Temperature Isolation, May 2, 2000
00-010, ATWS RCP Trip Reactor Pressure, May 16, 2000
00-011, RCIC Turbine steam supply low pressure isolation
00-012, HPCI Turbine steam supply low pressure isolation
00-024, HPCI Pump Suction Pressure, October 25, 2000

Jumper/Bypass

99-005, HPCI Discharge Line Temporary Pressurization, Revision 0, January 18, 1999
99-014, SW Pump Motor Restraint, Revision 0, March 16, 1999
99-028, Bypass HPCI Low Steam Line Pressure Auto Close Signal for MO-2035, April 23, 1999
99-041, Loop ‘B’ RHR - Shutdown Cooling #4179-02 OCD #12 & #14 RHR Pump Min Flow Valves Blocked Closed, May 20, 1999
99-089, Non Documented Filtering Capacitor Installed in HPCI Square Root Extractor, December 13, 1999
00-009, Block RHR minimum flow valves for Maintenance, Revision 0, January 16, 2000
00-014, Bypass Group I Isolation for MO-2373, November 27, 1999
00-072, Allow the Recirc Pumps to Operate above minimum speed with feedwater flow < 20%, February 16, 2000
00-080, Remove Latch from Door &^ (lower 4kV to Condensate Pump Area), February 21, 2000
00-102, Single Cell Battery Charger on Cell #82 of Battery #16, April 14, 2000
00-121, Jumper/Bypass is required to document acceptability of temporary equipment label installed on 13 RHRSW pump motor per WO 0003411, September 13, 2000
00-128, Temporary Beams in RHR rooms, Revision 0, September 28, 2000
00-129, Heating Boiler Valve Replacement, Revision 0, October 7, 2000.

Procedure Changes

93-0600, Integrated Primary Containment Leak Rate Test, April 16, 1993
95-2180, Integrated Primary Containment Leak Rate Test, October 16, 1995
98-0539, Integrated Primary Containment Leak Rate Test, February 27, 1998
98-2644, Integrated Primary Containment Leak Rate Test, July 3, 2000
99-0296, Integrated Primary Containment Leak Rate Test, February 2, 1999
99-0630, Reactor Protection System Channel Time Response Test Procedure, March 4, 1999
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20004494, 10 CFR 50.59 evaluation for modification 98Q140 did not adequately justify the USAR changes for the modification, November 17, 2000
2000-4536, “10 CFR 50.68 criticality requirement,” does not have a 10 CFR 50.59 screening or evaluation for USAR changes, November 17, 2000