November 30, 1999

Mr. M. Wadley
President, Nuclear Generation
Northern States Power Company
414 Nicollet Mall
Minneapolis, MN 55401

SUBJECT: NRC INSPECTION REPORT 50-282/99015(DRS); 50-306/99015(DRS)

Dear Mr. Wadley:

On October 22, 1999, the NRC completed the pilot baseline annual inspection of Changes to the Safety Analysis Report and the biennial Permanent Plant Modifications inspection at your Prairie Island Nuclear Generating Plant, Units 1 and 2. The results of this inspection were discussed on October 22, 1999, with Mr. T. Amundson and other members of your staff. A subsequent telephone re-exit was held on November 8, 1999, with Mr. G. Gore and other members of your staff to discuss changes to the original conclusions reached during the inspection. The enclosed report presents the results of this inspection.

The inspection was an examination of activities conducted under your license as they relate to changes to the Updated Safety Analysis Report under the provisions 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 procedures and representative records, observations of activities, and interviews with personnel.

Based on the results of this inspection, NRC identified one issue which was categorized as being of low risk significance. The issue concerned inadequate corrective actions to resolve the hot short issue with two motor-operated valves, which was evaluated under the risk significance determination process and determined to be of low risk significance (Green), although regulatory requirements were violated. This issue has been entered into your corrective action program. Therefore, one non-cited violation was identified. This issue is listed in the summary of findings and is discussed in the report.

If you contest the non-cited violation, 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, DC 20555-0001, with copies to the Regional Administrator, Region III and the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington, DC 20555-0001.

In accordance with 10 CFR 2.790 of the NRC=s ARules of Practice@, a copy of this letter and its enclosure will be placed in the NRC Public Document Room.

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

Sincerely,

/s/ J. M. Jacobson

John M. Jacobson, Chief
Mechanical Engineering Branch

Docket Nos. 50-282; 50-306
License Nos. DPR-42; DPR-60

Enclosure: Inspection Report 50-282/99015(DRS); 50-306/99015(DRS)

cc w/encl: Site General Manager, Prairie Island
Plant Manager, Prairie Island
S. Minn, Commissioner, Minnesota
Department of Public Service
State Liaison Officer, State of Wisconsin
Tribal Council, Prairie Island Dakota Community

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

Sincerely,

John M. Jacobson, Chief
Mechanical Engineering Branch

Docket Nos. 50-282; 50-306
License Nos. DPR-42; DPR-60

Enclosure: Inspection Report 50-282/99015(DRS); 50-306/99015(DRS)

cc w/encl: Site General Manager, Prairie Island
Plant Manager, Prairie Island
S. Minn, Commissioner, Minnesota
Department of Public Service
State Liaison Officer, State of Wisconsin
Tribal Council, Prairie Island Dakota Community

Distribution:
CAC (E-Mail)
RPC (E-Mail)
TJK3 (Project Mgr.) (E-Mail)
J. Dyer, RIII w/encl
J. Caldwell, RIII w/encl
B. Clayton, RIII w/encl
SRI Prairie Island w/encl
DRP w/encl
DRS w/encl
RIII PRR w/encl
PUBLIC IE-01 w/encl
Docket File w/encl
GREENS
IEO (E-Mail)
DOCDESK (E-Mail)
M. Branch, NRR (E-Mail)
T. Frye, NRR (E-Mail)
A. Madison, NRR (E-Mail)
S. Stein, NRR (E-Mail)

DOCUMENT NAME: G:DRS\PRA99015.WPD
To receive a copy of this document, indicate in the box: "**C**" = Copy without attachment/enclosure "**E**" = Copy with attachment/enclosure "**N**" = No copy

OFFICE	RIII		RIII		RIII			
NAME	ADunlop:jp		RLanksbury		JJacobson			

DATE	11/ /99	11/ /99	11/ /99	

OFFICIAL RECORD COPY

U.S. NUCLEAR REGULATORY COMMISSION

REGION III

Docket Nos:	50-282; 50-306
License Nos:	DPR-42; DPR-60
Report No:	50-282/99015(DRS); 50-306/99015(DRS)
Licensee:	Northern States Power Company
Facility:	Prairie Island Nuclear Generating Plant Units 1 & 2
Location:	1717 Wakonade Dr. East Welch, MN 55089
Dates:	October 18-22, 1999
Re-exit Date:	November 8, 1999
Inspectors:	A. Dunlop, Reactor Engineer, Team Leader G. O=Dwyer, Reactor Engineer D. Schrum, Reactor Engineer R. Winter, Reactor Engineer
Approved by:	John M. Jacobson, Chief, Mechanical Engineering Branch Division of Reactor Safety

SUMMARY OF FINDINGS

Prairie Island Nuclear Generating Plant, Units 1 & 2
NRC Inspection Report 50-282/99015(DRS); 50-306/99015(DRS)

This report covers the pilot baseline inspections for the annual review of changes to the safety analysis report and the biennial review of permanent plant modifications.

Inspection findings were assessed according to potential risk significance, and were assigned colors of GREEN, WHITE, YELLOW, or RED. GREEN findings are indicative of issues that, while not necessarily desirable, represent little risk to safety. WHITE findings would indicate issues with some increased risk to safety, and which may require additional NRC inspections. YELLOW findings would be indicative of more serious issues with higher potential risk to safe performance and would require the NRC to take additional actions. RED findings represent an unacceptable loss of margin to safety and would result in the NRC taking significant actions that could include ordering the plant shut down. No individual finding by itself would be indicative of either acceptable or unacceptable performance. The findings, considered in total with other inspection findings and performance indicators, will be used to determine overall plant performance.

Cornerstone: Mitigating Systems

Green: The inspectors identified that the corrective actions to address the hot short issue for the residual heat removal vessel injection valves were inadequate, which resulted in a non-cited violation. The modification did not address the potential for both a hot short and a ground that could allow a fire-induced spurious energization of the valves. This condition could have prevented the valves from performing their safety function. This issue was determined to be of low risk significance because the valves were still considered operable based on the compensatory measures in place to address a potential hot short event.

Cornerstone: Barrier Integrity

No findings were identified in this area.

Report Details

1. **REACTOR SAFETY**
Cornerstones: Mitigating Systems and Barrier Integrity

1R02 Changes to License Conditions and Safety Analysis Report (IP 71111, Attachment 2)

.1 Review of 50.59 Evaluations for Changes to the Safety Analysis Report

a. Inspection Scope

The inspectors reviewed seven evaluations performed pursuant to 10 CFR 50.59, one of which pertained to the barrier integrity cornerstone. All seven evaluations related to changes to the updated final safety analysis report. The inspectors also reviewed a number of changes to the updated final safety analysis report where the licensee had determined that a 50.59 evaluation was not necessary. In regard to the changes reviewed where no 50.59 evaluation was performed, the inspectors verified that the changes were minor editorial clarifications that did not meet the threshold of a "change to the facility as described in the safety analysis report." For the 50.59 evaluations, the inspectors confirmed that prior NRC approval was not required for any of the changes.

b. Observations and Findings

There were no findings identified and documented during this inspection.

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

.1 Review of Recent Plant Modifications

a. Inspection Scope

The inspectors reviewed five plant modifications that were recently installed. The packages were chosen based upon their affecting systems that had high Maintenance Rule safety significance or high risk significance in the licensee's Individual Plant Evaluation. Four of the modifications involved changes to mitigating systems, while the last one affected both mitigating systems and barrier integrity. The inspectors reviewed the modifications to confirm that the changes did not affect any systems' safety function. Design and testing aspects were verified to ensure the functionality of the modification, it's associated system, and any support systems. Walkdowns were conducted to verify proper installation of the modifications.

b. Observations and Findings

Brief Overview

The inspectors reviewed Design Change # 98FP01, which was implemented to address potential hot shorts for the residual heat removal (RHR) vessel injection valves (MV-32064 and MV-32065) and identified that the design had not addressed all potential failure mechanisms. The motor-operated valve (MOV) modification failed to address a hot short and a ground in the control circuit as required by 10 CFR Part 50, Appendix R, based on the guidance in Generic Letter 86-10, "Implementation of Fire Protection Requirements".

Background

On February 28, 1992, the NRC issued Information Notice 92-18, "Potential for Loss of Remote Shutdown Capability During a Control Room Fire". This information notice identified a potential common mode failure mechanism of MOVs in which a postulated fire could cause damage to the valve's control circuit in such a way as to bypass the valve's protective features (i.e., valve limit switches and torque switches). The resulting fire-induced spurious operation of a valve could result in physical damage to the valve operator or the valve itself, or prevent the valve from performing it's intended function.

The licensee had identified that 32 Appendix R related MOVs were susceptible to physical damage based on this scenario. These problems were documented in Licensee Event Report 98010 and NRC Inspection Report 50-282/98016; 50-306-98016. Escalated Enforcement action was taken for the plant being outside of the Appendix R post-fire safe shutdown design basis. The licensee established plans to address the hot short issue for the 32 valves and implement corrective actions by the end of the Cycle 21 Unit 1 refueling outage. The licensee implemented compensatory actions to address operability issues until the corrective actions for these 32 MOVs were complete. Some of these MOVs, such as MV-32064 and MV-32065, required a modification to correct the hot short problem.

Appendix R, Section III.G.2 specified separation requirements for cables and equipment, including associated circuits, as a means of ensuring that one redundant train of safe shutdown equipment remained free of fire damage. Generic Letter 86-10, Section 5.3.1, stated, "Sections III.G.2 and III.L.7 of Appendix R define the circuit failure modes as hot shorts, open circuits, and shorts to ground. For consideration of spurious actuations, all possible functional failure states must be evaluated, that is, the component could be energized or de-energized by one or more of the above failure modes."

Assessment

The design change implemented a wiring change to MV-32064 and MV-32065 circuitry to resolve the hot short issue. However, the modification did not address a hot short coincident with a ground that could allow a fire-induced spurious energization of the MOVs to a stall condition, thus potentially damaging the MOVs and preventing subsequent manual operation from outside the control room to achieve and maintain safe shutdown during a fire. Additionally, these RHR vessel injection valves were normally open and were required to be closed for safe shutdown. A spurious failure that opened these valves would provide a parallel path via the reactor vessel injection lines in addition to the loop B cold leg. This would allow a potential reduction in the cooldown rate because some injection flow would not be forced through the core. The core, however, would remain covered using safety injection during the event. The licensee stated there was no analyzed alternate method to accomplish the valves' function.

Requirements

10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Action", required, in part, that for conditions adverse to quality, measures shall assure that the cause of the condition was determined and corrective action taken to preclude repetition. Contrary to the above, the licensee did not take appropriate corrective actions to ensure that a potential fire-induced hot short condition would prevent valves MV-32064 and MV-32065 from meeting their function to safely achieve and maintain hot shutdown conditions. This Severity Level IV violation is being treated as a Non-Cited Violation (NCV), consistent with Interim Enforcement Policy for Use During the NRC Power Reactor Oversight Process Pilot Plant Study. This violation is in the licensee=s corrective action program as Condition Report 19993152 (50-282/99015-01(DRS); 50-306/99015-01(DRS)).

Significance Determination

The inspectors performed a Phase 1 screening of this issue under the significance determination process. This condition was determined to be of low risk significance because the MOVs were still considered operable based on the compensatory measures the licensee put in place and the issue screened out as AGreen.@

.2 Review of Seismic Controls for Scaffolding

a. Inspection Scope

The inspectors reviewed Maintenance Procedure D80, "Scaffolding, Ladders, and Cable Tray Platforms", on controls for erecting scaffolding in safety-related areas of the plant. The review included discussions with cognizant licensee personnel.

b. Observations and Findings

The inspectors identified that the controls for seismic scaffolding in procedure D80, were weak. The procedure only required a seismic evaluation for scaffolding if it was located within 10 feet of safety-related equipment and was more than 20 feet high or expected to be in service more than 21 days. As a result, redundant (both) trains would be susceptible to common mode damage during a seismic event.

The inspectors determined that the following weaknesses existed in procedure D80 and the scaffolding process: (1) the procedure did not require a seismic evaluation for scaffolding installed adjacent to safety-related equipment until after 21 days; (2) controls were not in place requiring that Operation personnel be informed of the status of previously installed scaffolding or when installed scaffolding was removed from the plant; (3) the procedure did not prohibit the installation of scaffolding adjacent to redundant safety-related equipment or adjacent to one safety train when the other train was out-of-service without a seismic evaluation; and (4) the licensee did not have any bounding criteria to demonstrate that the scaffolding would remain standing during a design basis seismic event. In response to the NRC finding, the licensee initiated Condition Report 19993148 to address this issue.

4 OTHER ACTIVITIES

4OA5 Management Meetings

.1 Exit Meeting Summary

The inspector presented the inspection results to members of licensee management in an exit meeting on October 22, 1999. A subsequent telephone re-exit was held on November 8, 1999, to discuss changes to the original conclusions reached during the inspection. The licensee acknowledged the information and findings presented. No proprietary information was identified.

PARTIAL LIST OF PERSONS CONTACTED

Licensee

T. Amundson, General Superintendent, Engineering
D. Anderson, Updated Final Safety Analysis Project Manager
M. Brown, Station Air System Engineer
J. Daley, Nuclear Generating Services
J. Goldsmith, General Superintendent, Design Engineering
G. Gore, Superintendent, Civil/Mechanical Engineering
J. Hill, Quality Manager
B. Johnson, Senior Mechanical Design Engineer
M. Meyer, Civil Engineer
R. Parazin, Nuclear Generating Services
J. Schaefer, Probabilistic Risk Assessment Engineer
Y. Shen, Probabilistic Risk Assessment Project Manager
J. Sorensen, Site General Manager
R. Sitek, Nuclear Generating Services
G. Sundberg, Superintendent, Electrical Engineering

NRC

J. Jacobson, Chief, Mechanical Engineering Branch, Division of Reactor Safety
S. Ray, Senior Resident Inspector
S. Thomas, Resident Inspector

INSPECTION PROCEDURES (IPs) USED

IP 71111.02 (draft)	Changes to License Conditions and Safety Analysis Report
IP 71111.17 (draft)	Permanent Plant Modifications
IP 71150 (draft)	Plant Status

ITEMS OPENED, CLOSED AND DISCUSSED

Opened

50-282/306-99015-01	NCV	Failure to implement adequate corrective action to resolve the hot short issue with valves MV-32064 and MV-32065.

Closed

50-282/306-99015-01	NCV	Failure to implement adequate corrective action to resolve the hot short issue with valves MV-32064 and MV-32065.

Discussed

None

LIST OF ACRONYMS

CFR	Code of Federal Regulations
DRS	Division of Reactor Safety
IP	Inspection Procedure
MOV	Motor-Operated Valve
NCV	Non-cited Violation
NRC	Nuclear Regulatory Commission
RHR	Residual Heat Removal
SE	Safety Evaluation
SI	Safety Injection
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.

Calculations

ENG-CS-108	Evaluation of Selected Cabinets for Auxiliary Feedwater Relays, Rev. 0, Addenda 1
ENG-EE-121	Cable Sizing Calculation for MV-32159 for Project 98EB02, Rev. 0
12911.6254-E-001	MCC Voltage Drop Evaluation for Hot Shorts MOV Modification, Rev. 2

Condition Reports

19993148	Potential weakness in scaffolding procedure
19993152	NRC concern on hot short methodology

Drawings

B34-7	121/122 Instrument Air Dryer Flow Diagram, Rev. 1
FD-96SA01-1	121 Instrument Air Dryer Isolation Valve Piping Modification,@ Rev. 0
FD-96SA02-2	122 Instrument Air Dryer Isolation Valve Piping Modification, Rev. 0
FD-96SA01-3	121 Dryer Pressure Switch & Control Panels, Rev. 0
FD-96SA01-5	Pressure Switch Mounting Plate Fabrication & Installation Detail, Rev. 0
NX-19833-3	Terminal Block Conn Diagram for Rack EM-A1 Unit 1 AA@ Train, Rev K1-A
NX-19833-12	Conn Dia RCS Press (WR) SG Level (WR) Rack EM-A1 Unit 1 AA@ Train, Rev. H1-A
NF-39244	Flow Diagram Instrument Air Piping, Rev. AT
NF-39752-2	Main Control Board Panel E-1, Rev. R1-A
NL-39776-537-1	Primary Piping Remote Mounted Level XMTR & Pressure XMTR, Rev. AD
NE-40004	Sheet 28 for Autostop trip and reset, Rev. AL1-A
NE-40004	Sheet 29 for Overspeed trip, Rev. AL1I-A
NE-40006	Sheet 59 for 12 Aux. Feedwater Pump Bus 16 Cubicle 3, Rev. AF1-A
NE-40005	Sheet for 12 Reactor Coolant Pump Bus 12 Cubicle 2, Rev. CJ1A
NE-40008	Sheet 65 for Reactor Safety Inj. Cold Leg Isol. Va. , Rev. BS1-A
NE-40008	Sheet 128 for Reactor Safety Inj. Cold Leg Isol. Va. B, Rev. BX1-A
NF-40008	Sheet 43 for Loop A/B Cool Water Return Hdr XOVR Valve A, Rev. BT1-A
NF-40008	Sheet 45 for 11 Aux. Building Cool Water Return Hdr Isol., Rev. BT1-B
NF-40008	Sheet 85.2 for 121 Spent Fuel Pit Pump, Rev. F1-A
NF-40008	Sheet 102 for Loop A/B Cool Water Hdr Cross-over VA B, Rev. BT1-B
NF-40008	Sheet 149.1 for 121 Spent Fuel Pit Pump, Rev. F1-A
NF-40009	Sheet 97.2 for 11 TD Aux. FW PMP MN STM Supply VA CV-31998, Rev. DT1-A
NF-40009	Sheet 111.2 for B Train Safety Injection Aux. Relay Circuit, Rev. CE1-A
NF-40011	Sheet 29 for BOP Annunciator Point Schematic, Rev. R1-A

NF-40036	480V Circuit Diagram Motor Control Center 1K, 1KA, Rev. S1-A

Drawings (continued)

NF-40151-2	Wiring Diagram 4.16KV SWGR BUS 11 Cub 2, Rev. J1-A
NF-40165	External Conn 4.16KV Switchgear 12 Aux. Feedwater Pump, Rev. N
NF-40174-4	Penetration Cabinets 1134 & 1135 Wiring Diagrams (Pen. C10), Rev. U1-A
NF-40208-2	Wiring Diagram Bus-1 Motor Control Center 1K, 1KA, Rev. AD1-A
NF-40224-3	External Connections Motor Control Center 1A, 1LA & IT, Rev. AB
NL-40259-1	Control Panel Switch Modules Internal Wiring Diagrams, Rev. BG
NL-40259-18.1	W2 Switch Module Style 505A715G05-3 Position Maintained, Rev. BG
NL-40259-145	Internal Wiring Diagram MO0 Control Switch Module, Rev. 0
NF-40260-112 OT2	Switch Module External Wiring Diagram Control Panel A, Rev. CD1-A
NF-40261-110	Monitor Light Module External Wiring Dia Control Panel B-1, Rev. BN1-A
NL-40264-1	Control Panel E-1 Switch Modules External Conn Diagrams, Rev. BB
NL-40264-148 W2	Switch Module External Wiring Diagram Control Panel E-1, Rev. BB
NL-40264-175	Internal Wiring Diagram Electroswitch Series 20M Switch Module, Rev. 0
NF-40273-9	External Wiring Diagram Computer Cabinet 1I/00, Zone 6, Rev. R1-A
NF-40275-1	Wiring Diagram Aux. Relay Cabinet 1215, Rev. AP1-A
NF-40275-2	Wiring Diagram Aux. Relay Cabinet 1216, Rev. AV
NF-40275-3	Wiring Diagram Terminal Cabinet 1217, Rev. AH
NF-40276-3	Wiring Diagram Aux. Relay Cabinet 1209, Rev. AG
NF-40277-23	Wiring Diagram AMSAC Cabinet Unit 1, sheet 1, Rev. C1-B
NF-40277-24	Wiring Diagram AMSAC Cabinet Unit 1, sheet 2, Rev. A
NF-40287-5	External Wiring Diagram Process Control System, Rev. R1-A
NF-40288-4	External Wiring Diagram Rod Drive Control System, 11AC Power, Rev. E1-A
NF-40291-2	External Wiring Diagram Reactor Safeguard System Train A, Rev. Y
NF-40291-8	External Wiring Diagram Reactor Safeguard System Train B, Rev. AB1-A
NF-40312-1	Interlock Logic Diagram Aux. Feedwater System - Unit 1, Rev. AB
NF-40313-4	121 Air Dryer Control Logic Diagram, Rev. A
NF-40315-1	Interlock Logic Diagram Cooling Water System Units 1 & 2, Rev. T1-A
NE-40404	Sheet 30 for Autotrip and Reset, Rev. Z1-A
NE-40404	Sheet 32 for Emergency trip and Overspeed trip, Rev. VX1-A
NF-40406	Sheet 78 for Loop A/B Cool Water Return Hdr XOVR Valve B, Rev. AF1-A
NF-40409	Sheet 81.1 for 22 TD Aux. Feedwater Pump MN STM Supply, Rev. EM
NF-40409	Sheet 97 for B Train Safety Injection Aux. Relay Circuit, Rev. X1-A
NF-40415	Sheet 23 for 21 AX BD CL Dump Open, Rev. AG1-A
NF-40415	Sheet 29 for CL X Over VLV A & B Closed, Rev. R1-A
NF-40426	480V Circuit Diagram Motor Control Center 2K, 2KA, Rev. S1-A
NF-40529-110	Monitor Light Module External Wiring Dia. Control Panel B-2, Rev. BU1-A
NL-40532-1	Control Panel E-2 Switch Modules External Conn Diagrams, Rev. AZ
NL-40532-85	W2 Switch Module External Wiring Diagram Control Panel E-2, Rev. AZ
NF-40544-1	Wiring Diagram Terminal Cabinet #2207, Rev. AQ1-A
NF-40544-3	Wiring Diagram Aux. Relay Cabinet 2209, Rev. X
NF-40564	External Conns 4.16KV SWGR, 21 Aux. Feedwater Pump, Rev. K
NF-40583-2	Motor Control Center 2LA Wiring Diagrams Bus-2, Rev. G1-A
NF-40592-3	External Connections Motor Control Center 2A, 2LA & IT, Rev. AF
NF-40626-8	External Wiring Diagram Reactor Safeguard System Train B, Rev. Q1-A

NF-40626-6	External Wiring Diagram Reactor Safeguard System Train B, Rev. N

Drawings (continued)

NF-40767-1	Interlock Logic Diagram Aux. Feedwater System - Unit 2, Rev. V
NF-40781-1	Interlock Logic Diagram Reactor Coolant System Unit 1, Rev. Q1-A
NF-40795	Interlock Logic Diagram AMSAC/DSS System Unit 1, Rev. BI-B
NF-40910	Sheet 24 for CLX Over Ret VLV A Open , Rev. GK1-A
NF-40910	Sheet 30 for 11 CA System Out Isol Open, Rev. V1-A
NF-74596-9	Remote Multiplexing Unit RMU-133 Cabinet 3 Conn Diagram, Rev. G1-A
NF-92982	480V Circuit Diagram Motor Control Center 1T, Rev. D1-B
NF-92983-1	Wiring Diagram Bus 1Motor Control Center 1T, Rev. E1-C
NF-93033	Wiring Diagram 11 TD Aux. Feedwater Pump Relay Cabinet A1848, Rev. J
NF-98213	Main Control Board, Panel E-1 Subpanel with Cutout, Rev. D1-A
NE-116785	Sheet 23 for 21 Aux. Feedwater Pump Bus 25, Cubicle 10, Rev. B
NF-120705-20	Internal Wiring Diagram 4KV Switchgear Bus 25 CUB 10 sheet 2, Rev. B
NF-120705-41	External Wiring Diagram 4KV Switchgear Bus 25, Rev. C
NF-122315-9	Internal Wiring Diagram ESF Rack M002 Train A Rear Compartment 6, Rev. D
X-HIAW-1-236	Logic Diagram Reactor Trip Signals Unit 1 & 2, Rev. D
X-HIAW-1-995	Feedwater Isolation Safeguards System Unit 1 & 2, Rev. E
X-HIAW-1001-885-10	Feedwater Isolation Safeguards System Unit 1 & 2, Rev. E
5D63658	NSP Rod Control System Diverse Scram System Circuit Board, Rev. 1

Engineering Transmittal Letter

Letter dated June 1, 1998, from EQ Coordinator to Modification Engineer, Subject: EQ Review for Design Change 96SA01

Miscellaneous

DBD SYS-34, "Design Basis Document for Station and Instrument Air System," Rev. 3
License Amendment Request, ATWS Mitigating System Actuating Circuitry/Diverse Scram System, 2/27/98
Licensee Response to July 2, 1998 Request for Additional Information, ATWS Mitigating System Actuating Circuitry/Diverse Scram System, 7.&/14/98
Al Kuroyama Memo, Non-impact on SQUG by Design Change 98EB02 90X002-0022, 3/16/99
Marc Meyer Memo, Seismic Evaluation of Containment Scaffold C-12, 5/8/99
Purchase Order PK2490MS for 2 Exhaust Purge Isolation Valves and 2 Control Panels

Work Requests

Work Order 9702733	Perform Pre-Op Test 11 AFWP Low Press Protection, 4/26/97
Work Order 9702647	Pull New Replacement Cable 1CB-16, 4/15/97
Work Order 9811257	Perform Pre-Op Test on AMSAC/DSS - Block Mode, 12/08/98
Work Order 9811259	Perform Pre-Op Test on AMSAC/DSS - Auto Mode, 12/11/98
Work Order 9811257	Perform Pre-Op Test on AMSAC/DSS - Rod Drop Test, 12/22/98
Work Order 9811495	PE-112L-26 Breaker Elect 5 Year PM MV-32064

Work Order 9815243	Preform Pre-Op Test for 121 Instrument Air Dryer Purge Valve
Work Order 9815244	Preform Pre-Op Test for 122 Instrument Air Dryer Purge Valve
Work Order 9900590	P32064 Low Head Reactor Vessel Inj D70 Insp

Work Requests (continued)

Work Order 9900853	P32159 LP A/B HDR Xover Valve D70 Insp
Work Order 9901011	MV-32159 - Transfer Source to MCC IT2-A3
Work Order 9901005	B Train - Install Conduit and Pull Cable
Work Order 9901417	Prepare D-O-U IT2-A3 for MV-32159
Work Order 9901521	Remove and Reinstall Kaowool Tray Wrap

Procedures

5AWI 1.10.1	Condition Reporting Process, Rev. 1
5AWI 7.1.3	Commercial Grade Procurement, Rev. 3
1C20.6	Unit 1 - 480V System, Rev. 12
C34-4	Station Air System 121 Instrument Air Dryer, Rev. 4
D80	Scaffolding, Ladders, and Cable Tray Platforms, Rev. 11
PM 3510-1-121	121 Instrument Air Dryer Annual Inspection, Rev. 6
PM 3510-1-122	122 Instrument Air Dryer Annual Inspection, Rev. 6
SP1102	11 Turbine Driven Auxiliary Feedwater Pump Test, Rev. 58
SP1103	11 T D Aux FW Pump Once Every Refueling Shutdown Flow Test, Rev. 31
SP1376	AFW Flow Path Verification Test after Each Cold Shutdown,@ Rev. 4

Modifications

Design Change #96AF01	AFW Pump Runout Protection, Rev. 0
Design Change #96SA01	121 and 122 Instrument Air Dryer Isolation Valves, Rev. 0
Design Change #97AF02	AMSAC/Diverse Scram System, Rev. 0
Design Change #98FP01	MOV Hot Short, Rev. 0
Design Change #98EB02	Repower CL System Common Unit Motor Valves, Rev. 0

10 CFR 50.59 Evaluations and Screenings

SE No. 478-05-01	USAR Update - Sections 5.1 - 5.2, 4/8/99
SE No. 478-10-04	USAR Update - Table 10.2-1, 1/20/99
SE No. 527-05-01	USAR - Containment Isolation, 8/20/99
SE No. 527-11-02	Changes to USAR Section 11.9.4.2, 7/14/99
SE No. 531-0	Changes to USAR Section 12.2.7.1 Turbine Missiles, 7/14/99
SE No. 533-0	USAR Updates - Section 12.2 and Table 12.2-40, 7/20/99
SE No. 543-0	USAR - Natural Circulation Cooldown, 8/20/99
SE 96SA01	Modification 96SA01, 5/18/98

Updated Final Safety Analysis Report Change Packages

Submittal of Revision No. 14 to the USAR, 9/30/97
Submittal of Revision No. 18 to the USAR, 7/22/99
Submittal of Revision No. 19 to the USAR, 8/31/99
USAR Change Input Item #99081 Form

Updated Final Safety Analysis Report Sections

USAR Section I.12-2 to I.12-3
USAR Sections 5.1 - 5.2
USAR Section 6
USAR Table 10.2-1
USAR Section 10.3.10 and Figure 10.3-16
USAR Section 11.9.4.2
USAR Section 12.2 and Table 12.2-40
USAR Section 14.3.3