April 28, 2006

Mr. T. Palmisano
Site Vice President
Prairie Island Nuclear Generating Plant
Nuclear Management Company, LLC
1717 Wakonade Drive East
Welch, MN 55089

SUBJECT: PRAIRIE ISLAND NUCLEAR GENERATING PLANT, UNITS 1 AND 2 NRC EVALUATION OF CHANGES, TESTS, OR EXPERIMENTS AND PERMANENT PLANT MODIFICATIONS BASELINE INSPECTION REPORT 05000282/2006006 (DRS); 05000306/2006006 (DRS)

Dear Mr. Palmisano:

On March 24, 2006, the U.S. Nuclear Regulatory Commission (NRC) completed a combined baseline inspection of the Evaluation of Changes, Tests, or Experiments and Permanent Plant Modifications at the Prairie Island Nuclear Generating Plant. The enclosed report documents the results of the inspection, which were discussed with you and others of your staff at the completion of the inspection on March 24, 2006.

The inspectors examined activities conducted under your license as they relate to safety and compliance with the Commission's rules and regulations and with the conditions of your license. The inspectors reviewed selected procedures and records, observed activities, and interviewed personnel.

Based on the results of the inspection, one NRC-identified finding of very low safety significance was identified which involved a violation of NRC requirements. However, because this violation was of very low safety significance, not willful, and because it was entered into your corrective action program, the NRC is treating the issue as a Non-Cited Violation in accordance with Section VI.A.1 of the NRC's Enforcement Policy.

If you contest the subject or severity of a 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 U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555-0001, with a copy to the Regional Administrator, U.S. Nuclear Regulatory Commission - Region III, 2443 Warrenville Road, Suite 210, Lisle, IL 60532-4352; the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001; and the Resident Inspector Office at the Prairie Island Nuclear Generating Plant facility.

In accordance with 10 CFR 2.390 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/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

/RA/

David E. Hills, Chief
Engineering Branch 1
Division of Reactor Safety

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

Enclosure: Inspection Report 05000282/2006006 (DRS); 05000306/2006006 (DRS)

cc w/encl: C. Anderson, Senior Vice President, Group Operations
M. Sellman, Chief Executive Officer and Chief Nuclear Officer
Regulatory Affairs Manager
J. Rogoff, Vice President, Counsel and Secretary
Nuclear Asset Manager
Tribal Council, Prairie Island Indian Community
Administrator, Goodhue County Courthouse
Commissioner, Minnesota Department
of Commerce
Manager, Environmental Protection Division
Office of the Attorney General of Minnesota

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Commissioner, Minnesota Department of Commerce
Manager, Environmental Protection Division Office of the Attorney General of Minnesota

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U.S. NUCLEAR REGULATORY COMMISSION

REGION III

Docket No:	50-282; 50-306
License No:	DPR-42; DPR-60
Report No:	05000282/2006006 (DRS); 05000306/2006006 (DRS)
Licensee:	Nuclear Management Company, LLC
Facility:	Prairie Island Nuclear Generating Plant
Location:	1717 Wakonade Drive East Welch, MN 55089
Dates:	March 6 through March 24, 2006
Inspectors:	J. Neurauter, Senior Reactor Inspector, Team Leader Alan Dahbur, Reactor Inspector
Approved by:	D. Hills, Chief Engineering Branch 1 Division of Reactor Safety (DRS)

SUMMARY OF FINDINGS

IR 05000282/2006006 (DRS); 05000306/2006006 (DRS); 03/06/2006 - 03/24/2006; Prairie Island Nuclear Generating Plant, Units 1 and 2; Evaluation of Changes, Tests, or Experiments (10 CFR 50.59) and Permanent Plant Modifications.

The inspection covered a two-week announced baseline inspection on evaluations of changes, tests, or experiments and permanent plant modifications. The inspection was conducted by two regional based engineering inspectors. One Green Non-Cited Violation (NCV) was identified. The significance of most 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, may be Green, or be assigned a severity level after NRC management review. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, “Reactor Oversight Process,” Revision 3, dated July 2000.

A. Inspector-Identified and Self-Revealed Findings

Cornerstone: Mitigating Systems

Green. A Non-Cited violation of 10 CFR Part 50, Appendix B, Criterion III, “Design Control,” having very low safety significance was identified by the inspectors. Specifically, the licensee had not evaluated and updated the associated plant cable ampacity calculation to determine the potential consequences of adverse effects to cabling due to higher temperatures in the auxiliary feedwater (AFW) pump rooms and other auxiliary building areas. After identification by the inspectors, the licensee was able to demonstrate that even though the higher temperatures decreased the ampacity margins for the affected cabling, it did not decrease the margins to the limit where the cabling would fail if called upon to provide power to equipment important to safety.

The finding was more than minor because it affected the mitigating system cornerstone objective to ensure the availability, reliability, and capability of systems that mitigate transients and accidents, and if left uncorrected, the finding could become a more significant safety concern. Specifically, if left uncorrected, the licensee may not account for high temperature conditions in plant areas that could adversely affect the ampacity of cabling that supply power to equipment important to safety. This finding was of very low safety significance because, the licensee’s preliminary evaluation determined that the higher temperatures in the AFW pump rooms and other auxiliary building areas would not prevent equipment important to safety from functioning. (Section 1R17.1.b.1)

Cornerstone: Barrier Integrity

No findings of significance were identified.

B. Licensee-Identified Violations

None.

REPORT DETAILS

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, and Barrier Integrity

1R02 Evaluations of Changes, Tests, or Experiments (71111.02)

.1 Review of 10 CFR 50.59 Evaluations and Screenings

a. Inspection Scope

From March 6 through March 24, 2006, the inspectors reviewed eight evaluations performed pursuant to 10 CFR 50.59. The inspectors confirmed that the evaluations were thorough and that prior NRC approval was obtained as appropriate. The inspectors also reviewed seventeen screenings where licensee personnel had determined that a 10 CFR 50.59 evaluation was not necessary. In regard to the changes reviewed where no 10 CFR 50.59 evaluation was performed, the inspectors verified that the changes did not meet the threshold to require a 10 CFR 50.59 evaluation. The evaluations and screenings were chosen based on risk significance, safety significance, and complexity. The list of documents reviewed by the inspectors is included as an attachment to this report.

The inspectors used, in part, Nuclear Energy Institute (NEI) 96-07, “Guidelines for 10 CFR 50.59 Implementation,” Revision 1, to determine acceptability of the completed evaluations and screenings. The NEI document was endorsed by the NRC in Regulatory Guide 1.187, “Guidance for Implementation of 10 CFR 50.59, Changes, Tests, and Experiments,” dated November 2000. The inspectors also consulted Part 9900 of the NRC Inspection Manual, “10 CFR Guidance for 10 CFR 50.59, Changes, Tests, and Experiments.”

b. Findings

No findings of significance were identified.

1R17 Permanent Plant Modifications (71111.17B)

.1 Review of Permanent Plant Modifications

a. Inspection Scope

From March 6 through March 24, 2006, the inspectors reviewed twelve permanent plant modifications that had been installed in the plant during the last two years. The modifications were chosen based upon risk significance, safety significance, and complexity. The inspectors 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 adversely 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 inspectors also verified that the modifications performed did not place the plant in an increased risk configuration.

The inspectors also used applicable industry standards to evaluate acceptability of the modifications. The list of modifications and other documents reviewed by the inspectors is included as an attachment to this report.

The Prairie Island Unit 1 reactor vessel head replacement modification, which affects the barrier integrity cornerstone, was not selected as part of this inspection. This modification will be inspected at a later date in accordance with inspection procedure 71007, "Reactor Vessel Head Replacement Inspection."

b. Findings

b.1 Failure to Consider Adverse Ampacity Effects of High Ambient Temperature Conditions in the Auxiliary Feedwater Pump Rooms

Introduction: On March 15, 2006, the inspectors identified a Non-Cited Violation of 10 CFR Part 50, Appendix B Criterion III, "Design Control," of very low safety significance (Green). Specifically, the licensee had not evaluated and updated the associated plant cable ampacity calculation to determine the potential consequences of adverse effects to cabling due to higher temperatures in the AFW pump rooms and other auxiliary building areas

Discussion: Revised licensee calculation ENG-ME-021, "Auxiliary Feedwater Pump Room Heat-up," indicated that the potential maximum ambient temperature in the AFW pump rooms could reach up to 127EF. The potential high ambient temperature could occur during post accident mitigation (an extended loss of off-site power) and when the initial ambient temperature in the rooms was at 104EF.

The licensee evaluated the effects of the high ambient temperature on the safety-related equipment (i.e., motor driven AFW pump motors, motor operated valves, motor control centers, transformers and hot shutdown panel) located in the AFW pump rooms. The evaluation was documented in calculation ENG-ME-021, Revision 2 and concluded that the operability of the safety-related equipment located in the rooms was acceptable for an ambient room temperature of 127EF. This conclusion was also documented in the licensee's 10 CFR 50.59 screening number 2469, Revision 0. However, the licensee failed to address the effects of these heightened temperatures on the ampacity of electrical cables in the rooms. The inspectors also reviewed the licensee's 10 CFR 50.59 Safety Evaluation Number 1037 "Affect of Revised Unit 1 Main Steam Line Break on Auxiliary Building Environment," which identified that the ambient temperature could also reach up to 122EF in several areas in the auxiliary building.

Prairie Island Engineering Manual for Electrical Cables Design, Fabrication and Installation Summary was based on an ambient temperature of 104EF. Other plant specific evaluations (i.e. Calculation ENG-EE-019 and Safety Evaluation 369) which have previously evaluated potential cable ampacity issues were also based on an ambient temperature of 104EF. The licensee failed to evaluate and update the cable ampacity calculation to evaluate the effects of potential high ambient temperatures on the ampacity of electrical cables located in these rooms. Since higher temperatures adversely affect the ampacity of electrical cables, the higher temperatures in the AFW pump rooms and other plant areas had the potential to adversely affect the functionality and/or operability of equipment important to safety fed by cabling in these rooms. The inspectors were concerned that the possibility existed that some of the equipment fed by cables located in these areas may not function due to possible faulting of the supply cables. As a result of the inspectors' concerns, the licensee issued Action Request CAP 01018612.

After performing a preliminary evaluation that assessed cabling in AFW pump rooms and the auxiliary building areas, the licensee determined that there was no evidence that safety related structures, systems, and components would not function as required. While the higher temperatures decreased the ampacity margins for the affected cabling, the licensee preliminarily determined that the margins it did not decrease to the limit where the cabling would fail if called upon to provide power to equipment important to safety.

Analysis: The inspectors determined that this issue was a performance deficiency warranting a significance evaluation, since the licensee failed to account for high temperature conditions in the AFW pump rooms and other several rooms located in the auxiliary building that adversely affected cables supplying power to equipment important to safety.

The finding was greater than minor in accordance with IMC 0612, "Power Reactor Inspection Reports," Appendix B, "Issue Screening," because it affected the mitigating system cornerstone objective to ensure the availability, reliability, and capability of systems that mitigate transients and accidents, and if left uncorrected, the finding could become a more significant safety concern. Specifically, if left uncorrected, the licensee may not account for high temperature conditions in plant areas that could adversely affect the ampacity of cabling that supply power to equipment important to safety.

The inspectors determined the finding was of very low significance (Green) using IMC 0609, Appendix A, "Significance Determination of Reactor Inspection Findings for the At-Power Situations," because the inspectors answered "no" to all five questions under the Mitigating Systems Cornerstone column of the Phase 1 worksheet. In particular, the licensee's preliminary evaluation determined that the higher temperatures in the AFW pump rooms and other auxiliary building areas would not prevent equipment important to safety from functioning.

Enforcement: 10 CFR Part 50, Appendix B, Criterion III, "Design Control" states, in part, that measures shall be established to assure that applicable design basis are correctly translated into specifications, drawings, procedures and instructions. Contrary to the above, the licensee did not have a design basis calculation for cable ampacity that supported the high temperatures that the AFW pump rooms and other plant areas could experience. The Prairie Island calculation and engineering manual that did address cable ampacity were significantly less conservative, since temperatures of 104EF were assumed where temperatures in these areas could exceed 122EF.

Because this issue was of very low safety significance, not willful, and because it was entered in the licensee's corrective action program as CAP 01018612, this violation is being treated as an NCV, consistent with Section VI.A of the NRC Enforcement Policy. (NCV 05000282/2006006-01; 05000306/2006006-01)

4. OTHER ACTIVITIES (OA)

4OA2 Identification and Resolution of Problems

.1 Routine Review of Condition Reports

a. Inspection Scope

From March 6 through March 24, 2006, the inspectors reviewed thirteen Corrective Action Process documents that identified or were related to 10 CFR 50.59 evaluations and permanent plant modifications. The inspectors reviewed these documents to evaluate the effectiveness of corrective actions related to permanent plant modifications and evaluations for changes, tests, or experiments issues. In addition, corrective action documents written on issues identified during the inspection were reviewed to verify adequate problem identification and incorporation of the problems into the corrective action system. The specific corrective action documents that were sampled and reviewed by the team are listed in the attachment to this report.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA6 Meetings

.1 Exit Meeting

The inspectors presented the inspection results to Mr. T. Palmisano and others of the licensee's staff, on March 24, 2006. Licensee personnel acknowledged the inspection results presented. Licensee personnel were asked to identify any documents, materials, or information provided during the inspection that were considered proprietary other than those returned. No additional proprietary information was identified.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee

T. Palmisano, Site Vice President
C. Mundt, Design Engineering Manager
J. Kivi, Senior Regulatory Compliance Engineer
S. Thomas, Design Engineering Supervisor
L. Gunderson, Mechanical Design Engineer
C. Sansome, Mechanical Design Engineer

Nuclear Regulatory Commission

J. Adams, Senior Resident Inspector

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

None.

Opened and Closed

05000282/2006006-01; 05000306/2006006-01	NCV	Failure to Consider Adverse Ampacity Effects of High Temperature Conditions in the Auxiliary feedwater Pump Rooms

Discussed

None.

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. Inclusion of a document in this list does not imply NRC acceptance of the document, unless specifically stated in the inspection report.

IR02 Evaluation of Changes, Tests, or Experiments (71111.02)

10 CFR 50.59 Screenings

No. 1691; ENG-ME-538, Structural Evaluation of Bolts on 11 and 21 Fan Coil Unit Motors; Revision 0

No. 2056; Modification 04CT02, Revise Cooling Tower Undervoltage Relaying; Revision 0

No. 2272; Permanent Plant Modification 04RC03 - Pressurizer PORV Block Valve Replacement, Addendum 2 to Westinghouse Stress Analysis 0951s, Revision 0

No. 2301; ENG-ME-576, AFW Pump Minimum Acceptance Criteria, Revision 0

No. 2303; USAR Input Item #05001; Revision 0

No. 2307; Calculation ENG-ME-443, Revision 3, PCRs 20050890, 20050891, 20050892, 20050893; Revision 0

No. 2350; Calculation ENG-CS-278, Seismic Qualification of Components in Component Cooling System Pressure Boundary; Revision 0

No. 2365; Calculation ENG-ME-615, Tube Plugging Limits for 21 Containment Fan Coil Unit; Revision 0

No. 2370; SP 1450, 31 Battery Refueling Outage Discharge Test; Revision 0

No. 2371; T-Mod 05T186; Revision 0

No. 2443; Calculation ENG-ME-621, CV-31998 and CV-31999 Air Receiver Capacity; Revision 0

No. 2452; Containment Spray Pump Discharge Check Valve Closure Acceptance Criteria; Revision 0

No. 2469; Calculation ENG-ME-021, Auxiliary Feedwater Pump Room Heat-up; Revision 0

No. 2486; Modification No. 05SA02, Structural Calculations S-11164-039-01 and S-11164-039-02; Revision 1

No. 2509; Design Change 05CL03: Cooling Water Pump Bearing Water, Part 1: Enhance Well Water Normal Supply to Safeguards Cooling Water Pump Bearings; Revision 0

No. 2513; ENG-ME-576, AFW Pump Minimum Acceptance Criteria; ENG-ME-454, Pressure Drop between Steam Generator and Safety Valve; TCNs and PCRs for SP 1102, 1103, 2102 and 2103; Revision 0

No. 2523; ENG-ME-646 Revision 0 Addendum 1, Reinforcing of Component Cooling Heat Exchanger Divider Plate; Revision 0

<u>10 CFR 50.59 Evaluations</u>

No. 1025; Zebra Mussel Treatment; Revision 1 dated April 22, 2005

No. 1032; Revised Containment Integrity Analysis with New Mass and Energy Methods; Revision 0; dated November 4, 2004

No. 1035; Compensated Hi-Tavg Parameter Changes (TM-0401H); Revision 0; dated January 20, 2005

No. 1037; Affected Revised Unit 1 Main Steam Line Break on Auxiliary Building Environment; Revision 0; dated October 24, 2004

No. 1038; Use of Ultimate Strength Design Methodology to Evaluate Vertical Seismic Loads on Floors; Revision 0 dated November 4, 2005

No. 1046; Unit 2 Cycle 23 Core Reload; Revision 1 dated May 20, 2005

No. 1047; Changes to Primary Chemistry Program Lithium and Hydrogen Limits; Revision 0 dated May 18, 2005

No. 1050; Revised Small Break LOCA Analysis Using the NORTUMP Code, SI into the Broken Loop and COSI Condensation Model (WCAP-10054-P-A Add. 2 Revision. 1); Revision 0 dated January 13, 2006

<u>IR17</u> <u>Permanent Plant Modifications</u> (71111.17B)

<u>Modifications</u>

EEC No. 1378; Generic Change from Carbon Steel Globe Hancock Valves to Carbon Steel Globe Vogt Valves; Revision 0

EEC No. 1503; Replace SV-33535; Revision 0

EEC No. 1576; Upgrade CC HX TCV Positioners and F/Rs; Revision 0

EEC No. 1616; Replace Breaker 121B-31 THEF MCCB with a THED; Revision 0

EEC No. 1618; Replace 600 lb Valves with 800 lb Vogt Valves; Revision 0

EEC No. 1636; Longer Bolts for SI Accumulator Hangers and As Found Condition; Revision 0

04RC04; PRT Level Transmitter Replacement; Revision 0

05CL03; Enhance Well Water Normal Supply to Safeguards Cooling Water Pump Bearings; Revision 0

05SA02; Replacement of No. 121 and No. 122 Instrument Air Dryers; Revision 0

05ST01; CT Underground Cable Replacement; dated August 31, 2005

1TM-401H; Change Operating Parameters on 1TM-401H; dated February 04, 2005

2TM-401H; Change Operating Parameters on 2TM-401H; dated February 07, 2005

<u>Other Documents Reviewed During Inspection</u>

<u>Corrective Action Program Documents Generated As a Result of Inspection</u>

CAP 01018063; Acceptance Criteria in SP 1450 and 2450 Need Review; dated March 9, 2006

CAP 01018337; Clarity Regarding Application of 50.59 to IST Acceptance Criteria; dated March 13, 2006

CAP 01018612; Cable Ampacities Have Not Considered Increased Ambient Temperatures; dated March 15, 2006

CAP 01019410; EEC 1378 Is Not Clear Regarding Not Using for Throttle Valve Replacement; dated March 20, 2006

CAP 01019730; USAR Section 8.5.2 Needs to Be Clarified; dated March 22, 2006

CAP 01019811; EEC 1636 Does Not Address Potential Reduction in Hanger Capacity; dated March 22, 2006

CAP 01019822; Control of U-Bolt Configurations During Modifications; dated March 22, 2006

CAP 01019883; DC Battery SP 2314 Contains Incorrect Data; dated March 22, 2006

CAP 01020014; 50.59 Evaluation 1037 Does Not Clearly Address Input Parameters; dated March 23, 2006

CAP 01020123; Errors on Logic Diagram; dated March 23, 2006

Corrective Action Program Documents Reviewed During the Inspection

CAP 0039552; Auxiliary Building HELB Analysis Temperature Assumptions, dated October 28, 2004

CAP 0078984; RC System Head Vent System - Design Configuration; dated December 20, 2004

CAP 00841009; 50.59 Evaluation Bypassed Two Reviews Required by the Controlling AWI; dated March 10, 2005

CAP 00843566; Change in Primary Lithium/PH Control; dated May 10, 2005

CAP 00846108; Unable to Complete Repairs on CL Valves Due to Valve Configuration Issues; dated May 17, 2005

CAP 00851046; The Design Basis for the Air Receiver for CV-31998 and CV-31999 is Unclear; dated May 28, 2005

CAP 00854365; New Security Fencing Installed Without Apparent Design Change Controls; dated June 7, 2005

CAP 00884715; Temporary Cooling Added to TP 1636 and 1637 Needs a 50.59 Screening; dated September 8, 2005

CAP 00888817; 50.59 Process Self-Assessment for Screening No. 2042; dated September 21, 2005

CAP 00889055; 50.59 Process Self-Assessment Finding for Screening No. 2350; dated September 21, 2005

CAP 01017232; Configuration Management Self Assessment Finding - Modification 05CL03; March 3, 2006

CAP 01017244; EEC 1559- Replace the Pressurizer Porv Accumulator Air Inlet Check Valve; dated March 03, 2006

CAP 01017281; Setpoint Calculations have not been Screened by 50.59 Process; dated March 03, 2006

Calculations

ENG-CS-278; Seismic Qualification of Components in Component Cooling System Pressure Boundary; Revision 1

ENG-EE-019; Evaluation to Resolve Overfilled Cable Trays Identified in Follow on Item A0457; Revision 0

ENG-ME-021; Auxiliary Feedwater Pump Room Heat-Up; Revision 2

ENG-ME-454; Pressure Drop between SG and Safety Valve; Revision 0, Addendum 1

ENG-ME-538; Structural Evaluation of Bolts on 11 and 21 FCU Motors; Revision 0

ENG-ME-576; AFW Pump Minimum Acceptance Criteria; Revision 1

ENG-ME-577; Prairie Island Characterization of Zebra Mussel Transport in Pump Intake Structures; Revision 0

ENG-ME-605; Prairie Island Zebra Mussel Transport Shell Deposition as a Function of Mussel Concentration; Revision 0

ENG-ME-615; Tube Plugging Limits for 21 Containment Fan Coil Unit; Revision 0

ENG-ME-621; CV-31998 and CV-31999 Air Receiver Capacity; Revision 0

ENG-ME-646; Reinforcing of CC Hx Divider Plate - Vendor Calculation PI-S-021; Revision 0, Addendum 1

PI-605044-P01; Evaluation of Well / Filtered Water Piping Below Elevation 695 Screenhouse; Revision 0

S-11164-039-01; Design of foundations for Instrument Air Dryers Nos. 121 and 122; Revision 0

S-11164-039-02; Design of New Supports - Modification No. 05SA02; Revision 0

S-B01-VS-001; Structural Floor Analysis for Vertical Seismic; Revision 0

<u>Drawings</u>

FC-64-247A; Instrument Connection at LT 24060, Accumulator No. 22; Revision 4

FC-64-250; Instrument Piping Connection at LT 24058, Accumulator No. 21; Revision 4

ND-211543; 12,121 and 122 Cooling Water Pumps, Bearing Water Supply Isometric; Revision 0

ND-211544; 12,121 and 122 Cooling Water Pumps, Bearing Water Supply Piping Supports; Revision 0

NE-40009; Sheer 97.2; 11 TD Aux. Feedwater Pump Main Steam Supply Valve CV-31998; Revision DT

NF-40312-1; Interlock Logic Diagram, Aux. Feedwater System - Unit 1; Revision AC
X-HIAW-1106-6320; Pipe Support - Safety Injection; Revision A

X-HIAW-1106-6324; Pipe Support - Safety Injection; Revision A

Procedures

FP-E-EQV-01; Fleet Procedure; Equivalency Evaluations and Changes; Revision 0

FP-SC-GEN-02; Fleet Procedure; Requesting Materials; Revision 6

FP-SC-GEN-03; Fleet Procedure; Catalog Item Creation and Change; Revision 4

FP-WM-PLA-01; Fleet Procedure: Work Order Planning Process; Revision 0

SP 1353A: Surveillance Procedure; Quarterly Testing of CS-16 and CS-18, 11 CSP Suction and Discharge Check Valves; Revision 10

SP 2103: Surveillance Procedure; 22 Turbine-Driven Auxiliary Feedwater Pump Once Every Refueling Shutdown Flow Test; Revision 42

Miscellaneous Documents

Action Request No. 01015987; USAR Change for Modification 05SA02 - Instrument Air Dryer; dated February 23, 2006

PCR 2005-1849B; Procedure Change Request, SP 1353A Revision 8; Quarterly Testing of CS-16 and CS-18, 11 CSP Suction and Discharge Check Valves; dated August 12, 2005

PCR 2005-2881A; Procedure Change Request, SP 2103 Revision 41;22 Turbine-Driven Auxiliary Feedwater Pump Once Every Refueling Shutdown Flow Test; dated October 27, 2005

Safety Evaluation No. 342; Place CT BT112 in Manual for all Operating Condition; dated March 12, 1993

Safety Evaluation No. 369; Cable Tray Fill and Spacing Concerns; Revision 0; dated August 25, 1995

Safety Evaluation No. 478-A1-04; USAR Up-date Appendix I.11 (Compartment Pressure and Temperatures); Revision 0; dated February 24, 2000

SP-2314; 22 Battery Refueling Outage Discharge Test; Performed on May 19, 2005

TCN 2005-1104; Temporary Change Notice, SP 1353A Revision 8: Quarterly Testing of CS-16 and CS-18, 11 CSP Suction and Discharge Check Valves; dated July 18, 2005

TCN 2005-1117; Temporary Change Notice, SP 1353A Revision 8: Quarterly Testing of CS-16 and CS-18, 11 CSP Suction and Discharge Check Valves; dated July 22, 2005

LIST OF ACRONYMS USED

ADAMS	Agency-Wide Document Access and Management System
AFW	Auxiliary Feedwater
CFR	Code of Federal Regulations
DRS	Division of Reactor Safety
EEC	Equivalent Engineering Change
NCV	Non-Cited Violation
NEI	Nuclear Energy Institute
NRC	Nuclear Regulatory Commission
PCR	Procedure Change Request
SDP	Significance Determination Process
TCN	Temporary Change Notice
USAR	Updated Safety Analysis Report