October 28, 2005

Mr. Christopher M. Crane
President and Chief Nuclear Officer
Exelon Nuclear
Exelon Generation Company, LLC
4300 Winfield Road
Warrenville, IL 60555

SUBJECT: BRAIDWOOD STATION, UNITS 1 AND 2
NRC BIENNIAL SAFETY SYSTEM DESIGN AND PERFORMANCE CAPABILITY INSPECTION REPORT 05000456/2005007; 05000457/2005007(DRS)

Dear Mr. Crane:

On September 2, 2005, and subsequent telephone calls on September 6th and October 18, 2005, the U.S. Nuclear Regulatory Commission (NRC) completed a baseline inspection at your Braidwood Station, Units 1 and 2. The enclosed report documents the results of this inspection discussed on site September 2, 2005, with the Site Vice President, K. Polson, and other members of your staff and during a final exit meeting on October 18, 2005.

The inspection examined activities conducted under your license as they relate to safety and to compliance with the Commission's rules and regulations and with the conditions of your license. The inspectors reviewed selected procedures and records, performed walkdowns of equipment, observed activities, and interviewed personnel. This inspection specifically focused on the Service Water System and the Loss of AC Power Event.

Based on the results of this inspection, five NRC-identified findings of very low safety significance were identified, four of which involved violations of NRC requirements. However, because these violations were of very low safety significance and because they were entered into your corrective action program, the NRC is treating the issues as Non-Cited Violations 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 NRC Resident Inspectors' Office at the Braidwood Station.

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/

Ann Marie Stone, Chief
Engineering Branch 2
Division of Reactor Safety

Docket Nos. 50-456; 50-457
License Nos. NPF-72; NPF-77

Enclosure: Inspection Report 05000456/2005007; 05000457/2005007 (DRS)
w/Attachment 1: Supplemental Information

cc w/encl: Site Vice President - Braidwood Station
Braidwood Station Plant Manager
Regulatory Assurance Manager - Braidwood Station
Chief Operating Officer
Senior Vice President - Nuclear Services
Senior Vice President - Mid-West Regional Operating Group
Vice President - Mid-West Operations Support
Vice President - Licensing and Regulatory Affairs
Director Licensing - Mid-West Regional Operating Group
Manager Licensing - Byron and Braidwood
Senior Counsel, Nuclear, Mid-West Regional Operating Group
Document Control Desk - Licensing
Assistant Attorney General
Illinois Department of Nuclear Safety
State Liaison Officer
Chairman, Illinois Commerce Commission

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Docket Nos. 50-456; 50-457
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Director Licensing - Mid-West Regional Operating Group
Manager Licensing - Byron and Braidwood
Senior Counsel, Nuclear, Mid-West Regional Operating Group
Document Control Desk - Licensing
Assistant Attorney General
Illinois Department of Nuclear Safety
State Liaison Officer
Chairman, Illinois Commerce Commission

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U.S. NUCLEAR REGULATORY COMMISSION

REGION III

Docket Nos:	50-456; 50-457
License Nos:	NPF-72; NPF-77
Report No:	05000456/2005007; 05000457/2005007 (DRS)
Licensee:	Exelon Generation Company, LLC
Facility:	Braidwood Station, Units 1 and 2
Location:	35100 S. Route 53 Suite 79 Braceville, IL 60407-9617
Dates:	August 15, 2005, through September 2, 2005
Inspectors:	R. Daley, Senior Engineering Inspector, Lead M. Holmberg, Senior Engineering Inspector D. Schrum, Engineering Inspector M. Munir, Engineering Inspector M. Jordan, Contractor
Approved by:	Ann Marie Stone, Chief Engineering Branch 2 Division of Reactor Safety (DRS)

Enclosure

SUMMARY OF FINDINGS

IR 05000456/2005007; 05000457/2005007(DRS); 08/15/2005 - 09/02/2005; Braidwood Station; Safety System Design and Performance Capability

The inspection consisted of a review of the Safety System Design and Performance Capability (SSDPC) of the Essential Service Water System (SX) and the Loss of AC Power event. The inspection was conducted by regional engineering inspectors. Four Green Non-Cited Violations, one Green finding, and 1 Unresolved Item (URI) were 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. <u>Inspector-Identified and Self-Revealed Findings</u>

Cornerstone: Mitigating Systems

- Green. A finding of very low safety significance was identified by the inspectors associated with a violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," where the licensee had no design basis calculation supporting adequate voltage levels for safety related equipment during a safety injection (SI). Voltage drop during an SI transient can be large and could result in operation of required safety-related equipment outside its design basis. After identification by the team, the licensee was able to demonstrate adequate voltage to support the operation of safety related equipment during this bounding voltage transient scenario.

  This finding was more than minor because if left uncorrected, the finding would become more significant. Modifications to the electrical distribution system can adversely affect the voltage for safety related equipment. Without a bounding voltage drop analysis to support the reliable operation of safety related equipment during an SI, these effects would go unnoticed causing adverse conditions during an actual SI with off-site power available. This finding was of very low safety significance because it screened out using the Phase 1 worksheet. (Section 1R21.1.b.1)

- Green. A finding of very low safety significance associated with a 10 CFR Part 50, Appendix B, Criterion III, "Design Control" was identified by the inspectors. The finding involved the operation of the emergency diesel generator jacket water coolers in a cross-connected configuration that was not supported by the plant's license and design basis. The licensee is evaluating the procedure for possible revision.

  This finding was more than minor because the licensee's established design and license basis for these coolers required a higher level of flow than that actually observed in the coolers during this cross-connected operation. The licensee had inappropriately relied on a manual operator action to justify operation in this configuration. This condition, if left uncorrected, would become more significant. This finding was of very low safety significance because it screened out using the Phase 1 worksheet. (Section 1R21.2.b.1)

- Green. The inspectors identified a finding involving a Non-Cited Violation (NCV) violation of 10 CFR Part 50.55a(g)4 having very low safety significance for failure to perform periodic leakage testing required by the American Society of Mechanical Engineers Code on the buried portions of the essential service water (SX) system intake piping.

  This finding was more than minor because failure to perform periodic leakage testing could have allowed undetected through-wall flaws to remain inservice. These undetected flaws could grow in size until leakage from the buried SX intake pipe degrades system operation or if sufficient general corrosion occurs, a gross rupture or collapse of the SX piping sections could occur. The finding was of very low safety significance because the licensee concluded that the piping systems were currently operable based upon pump surveillance testing which measured adequate SX system flow. The licensee also documented that piping failure was not anticipated due to the external pipe coating, cathodic protection and low system operating pressure.
  (Section 1R21.2.b.2)

- Green. A finding of very low safety significance was identified by the inspectors associated with a violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control" where the licensee failed to maintain an accurate design basis for the condensate storage tank (CST) useable inventory. The team identified an additional depletion path of CST water, the makeup valve (1(2)CD0035) from the CST to the condenser hotwell, that was not accounted for in the plant's calculation for useable CST volume.

  This finding was more than minor because it was associated with and affected the Mitigating Systems Cornerstone. Specifically, the capacity of the water source for the auxiliary feedwater (AFW) system was adversely affected by this additional depletion path. This finding was of very low safety significance because it screened out using the Phase 1 worksheet. (Section 1R21.3.b.1)

C Green. The inspectors identified a finding of very low risk significance for failure to provide operators with equipment, procedures and training to manually operate the essential service water (SX) strainers to recover the loss of automatic backwash capability. Specifically, the loss of automatic strainer backwash function following a seismic event would lead to SX strainer plugging and without adequate recovery procedures, the loss of SX system flow. This finding did not constitute a violation of NRC requirements because the strainers (aside from the pressure boundary) and associated backwash equipment were not considered safety-related.

  The inspectors determined that this finding was of more than minor significance because it would become a more significant safety concern if left uncorrected. Specifically, the failure to provide equipment, procedures and training for manually backwashing the SX strainers could result in loss of cooling to safety-related equipment cooled by SX following a seismic event. An NRC Regional III Senior Reactor Analyst (SRA) performed a qualitative Phase 3 risk evaluation and determined that the initiating event frequency of a seismic event was low. In performing this evaluation, the SRA considered the lack of data to support how long it would take to plug the strainers with sediment or debris and given that strainer plugging may take days, there was a high likelihood that recovery of the backwash function would occur. Although there were no plant procedures, the licensee had access to vendor documents which provided adequate instructions for the manual backwash operation, and the loss of off-site power operating procedure included actions to restore power to the 480 volt motor control center which supplied power to the SX strainer backwash motors and isolation valves. Based on these facts, the SRA determined that the finding was of very low safety significance. The licensee entered this deficiency into their corrective action program for resolution. (Section 1R21.3.b.2)

B. **Licensee-Identified Violations**

None.

REPORT DETAILS

1. REACTOR SAFETY

Cornerstone: Initiating Events, Mitigating Systems and Barrier Integrity

1R21 Safety System Design and Performance Capability (71111.21)

Introduction: Inspection of safety system design and performance capability verifies the initial design and subsequent modifications and provides monitoring of the capability of the selected systems to perform design bases functions. As plants age, the design bases may be lost and important design features may be altered or disabled. The plant risk assessment model is based on the capability of the as-built safety system to perform the intended safety functions successfully. This inspectable area verifies aspects of the Mitigating Systems and Barrier Integrity cornerstones for which there are no indicators to measure performance.

The objective of the SSDPC inspection is to assess the adequacy of calculations, analyses, other engineering documents, and operational and testing practices that were used to support the performance of the selected systems during normal, abnormal, and accident conditions. Specific documents reviewed during the inspection are listed in the attachment to the report.

The systems and events selected for inspection were the service water system (SX) and the loss of all AC power event (two samples). This system and event were selected for review based upon:

- having high probabilistic risk analysis rankings;
- design basis overlap; and
- not having received recent NRC review.

The criteria used to determine the acceptability of the system’s performance was found in documents such as:

- licensee technical specifications;
- applicable Updated Final Safety Analysis Report (UFSAR) sections; and
- the systems' design documents.

.1 System Requirements

a. Inspection Scope

The inspectors reviewed the UFSAR, technical specifications, system design basis documents, system descriptions, drawings, and other available design basis information, to determine the performance requirements of the service water system and support systems, as well as the successful implementation of the Loss of AC Power procedures for the Loss of AC Power event. The reviewed system and event attributes included process medium, energy sources, control systems, operator actions, and heat removal. The rationale for reviewing each of the attributes was:

Process Medium: This attribute required review to ensure that the SX system was capable of providing cooling during design basis events. For the Loss of AC event, this attribute required review of operating procedures to ensure that emergency power was available to required loads during the event.

Energy Sources: This attribute required review to ensure that the power source for major electrical equipment in the service water system was adequate for the proper functioning of the valves and other components. For the Loss of AC Power event, this attribute was reviewed by verifying power to required equipment during the event. This review concentrated on the emergency diesel generator (EDG) loading and reliability, offsite power availability, and battery sizing.

Controls: This attribute required review to ensure that required instrumentation calculations and surveillances for the service water Ultimate Heat Sink (UHS) were adequate. Additionally, for the Loss of AC Power event, this review consisted of controls necessary for implementation of the necessary operating procedures. It also included an indepth review of the swap over function of the condensate storage tank (CST) to the SX system during a Loss of Offsite Power Event.

Heat Removal: This attribute required a detailed review of the service water flow calculations. The heat removal function of auxiliary feedwater (AFW) as well as the reactor inventory control function were also reviewed for the Loss of AC Power event.

b. Findings

b.1 Lack of a Bounding Voltage Drop Calculation During a Safety Injection (SI) System Actuation

Introduction: The inspectors identified a Non-Cited Violation (NCV) having very low safety significance (Green) of 10 CFR 50, Appendix B Criterion III, “Design Control.” Specifically, the inspectors identified that the licensee had no design basis calculation determining transient voltage levels during a SI system initiation with offsite power available.

Description: The inspectors discovered that the licensee did not have a voltage drop calculation to support the various operational loading configurations during an SI with offsite power available. The inspectors were concerned, because this event normally is the bounding configuration for voltage drop. Because of the large influx of loads that occur during an SI, voltage has the potential to drop to a level which could affect the starting of safety related loads required to operate during the event. If voltage drops substantially, the consequences could range from delaying initiation of certain safety related loads to the inability to start safety related loads.

Because of the concerns raised by the team, the licensee was able to produce a pre-startup test that verified operation of safety related loads during an SI. While this test verified that loads would start properly during an SI, the test did not contain the incoming grid voltages at the time of the test. This is important, because if the test was not conducted at the lowest operational voltage, the test would not be bounding for all potential SI conditions. Additionally, while the SI showed that loads would operate as designed prior to plant operation, this test would not validate the distribution system for loading changes performed to the AC distribution system post-startup. Consequently, the inspectors determined that the licensee did not have an adequate design basis to support operation of safety related loads due to voltage drop during an SI condition.

Because of the inspectors' concerns, the licensee initiated an Issue Report (AR 00369124) and performed a formal SI system block start calculation in their electrical load management (ELMs) software which demonstrated the voltage drop during the unique conditions present during an SI with offsite power available. While several 480 VAC loads (ECCS pump cubicle cooler fans) were shown to drop below their required voltage, the licensee was able to demonstrate that this low voltage would be short in duration (4 or 5 seconds) and would not adversely affect accident mitigation capabilities.

Analysis: The inspectors determined that this issue was a performance deficiency since the licensee failed to meet the requirements of 10 CFR Part 50 Appendix B, Criterion III. Specifically, the licensee did not maintain a voltage drop transient analysis for the bounding conditions that would exist during an SI with offsite power available. The issue was more than minor because if left uncorrected, the finding would become more significant. Modifications to the electrical distribution system can adversely affect the voltage for safety related equipment. Without a bounding voltage drop analysis to support the reliable operation of safety related equipment during an SI these effects would go unnoticed causing adverse conditions during an actual SI with off-site power available. The finding screened as having 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.

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 Braidwood Station had no design basis calculation supporting adequate voltage levels for safety related equipment during an SI. Voltage drop during the SI transient can be large and could result in operation of required safety-related equipment outside its design basis.

Because the failure to have a bounding design basis voltage drop calculation during an SI block start was determined to be of very low safety significance and because it was entered in the licensee's corrective action program as AR 00369124, this violation is being treated as an NCV, consistent with Section VI.A of the NRC Enforcement Policy. (NCV 05000456/2005007-01; NCV 05000457/2005007-01)

.2 System Condition and Capability

a. Inspection Scope

The inspectors reviewed design basis documents and plant drawings, abnormal and emergency operating procedures, requirements, and commitments identified in the UFSAR and technical specifications. The inspectors compared the information in these documents to applicable electrical, instrumentation and control, mechanical calculations, and plant modifications. The inspectors used applicable industry standards, such as the American Society of Mechanical Engineers (ASME) Code and the Institute of Electrical and Electronics Engineers (IEEE), to evaluate acceptability of the systems' design. Select operating experience was reviewed to ensure the issue was adequately evaluated and corrective actions implemented, as necessary. The inspectors also reviewed operational procedures to verify that instructions to operators were consistent with design assumptions.

The inspectors reviewed information to verify that actual system condition and tested capability were consistent with the identified design bases. Specifically, the inspectors reviewed the installed configuration, the system operation, the detailed design, and the system testing, as described below.

Installed Configuration: This attribute required detailed system walkdowns of the installed configuration of the service water system and components necessary to perform the Loss of AC Power procedures. The walkdowns focused on the configuration of piping, components, and instruments as well as the environmental conditions in the areas and the potential vulnerabilities in regard to flooding and seismic events. The walkdowns also verified the installed configuration of components with design and licensing bases assumptions and design input values.

Operation: The inspectors verified that the service water system and systems required to operate during the Loss of AC Power event were operated in accordance with design basis documents and station procedures. The inspectors evaluated the effects on the systems of permanent changes, temporary changes, operator workarounds, or equipment being out of service and ensured that operations staff would have required access to equipment if needed during postulated scenarios.

Design: The inspectors reviewed the mechanical and electrical design of the service water system and systems necessary for the Loss of AC event to verify that the systems and subsystems would function as required. This included a review of the design basis, license basis, design assumptions, calculations, boundary conditions, and a review of selected modification packages. Additionally, the inspectors verified that these design and license basis attributes were translated properly into the plant's operating procedures, particularly for the Loss of AC event procedures.

Testing: The inspectors reviewed records of selected periodic testing and calibration procedures as well as surveillance procedures to verify that the design requirements of calculations, drawings, and procedures were incorporated in the system and were adequately demonstrated by test results. Test results were also reviewed to ensure that testing was consistent with design basis information.

b. Findings

b.1 EDG SX Cross-Connect not Supported by Design Basis

Introduction: The inspectors identified a Non-Cited Violation (NCV) having very low safety significance (Green) of 10 CFR Part 50, Appendix B Criterion III, “Design Control”. Specifically, Braidwood Station operated the service water to the EDG jacket water coolers in a cross-connected configuration that was not supported by the plant’s license and design basis.

Description: The inspectors identified that during performance of procedures for testing service water to the EDG the licensee was relying on manual valve manipulation to ensure the operability of the EDG. Specifically, procedures 1BwOS DG-M1 and BwOP SX-10 established cross-connected service water lineups to both EDGs that required operator action to restore the lineup to normal should a diesel auto-start occur. For BwOP SX-10, this manual operator action formed the underlying basis for ensuring the operability of one of the two EDGs, while in 1BwOS DG-M1, this action was the basis for both EDGs being operable during performance of the procedure. This manual action was outlined in the Limitations and Actions sections for each of these procedures. However, the EDGs were designed to automatically start and supply load independent of operator actions.

Additionally, the team identified that the licensee had established flow rates as low as 1600 gallons per minute (gpm) through the EDG jacket water coolers while in this cross-connected lineup. This flowrate represented a degraded condition from the design flow identified in the Braidwood UFSAR (Table 9.2-11) and as used in service water system design flow calculations. To address operability of this degraded flow condition, the licensee used results from a Byron calculation BYR04-055 which showed that as little as 1450 gpm would be acceptable to provide sufficient flow to the coolers during EDG operation. The Byron calculation was based upon the same EDG jacket water cooler design as Braidwood. Because the cooler performance and SX flowrates for the Byron EDG cooler bound the configuration for the Braidwood EDG jacket water coolers, the licensee concluded that the Braidwood EDGs had been operable during this cross-connected lineup.

This issue was entered into the licensee’s corrective action program as AR 00368849 and 00368877. Even though the licensee was able to show that the EDG had sufficient flow to the EDG jacket water coolers for proper operation, the design basis for this cross-connected service water lineup was not established until after the inspectors identified their concerns. In fact, the operation was justified by proceduralized manual actions to restore the service water configuration to normal should the EDG auto-start.

Analysis: The inspectors determined that this issue was a performance deficiency, since the licensee failed to meet the requirements of 10 CFR Part 50 Appendix B, Criterion III. Specifically, the licensee did not have a design basis established for operation of the service water to the EDG jacket water coolers cross-connected. In fact, both the license basis and the design basis calculation in place supported a higher flow than that actually established in this configuration.

The issue was more than minor because if left uncorrected, the finding would become more significant. If the inspectors had not questioned the proceduralized manual actions in place to ensure EDG Operability, a correct design basis flow for this configuration may not have been established. Without this design basis, future plugging of the coolers could have resulted in insufficient flowrates through the EDG jacket water coolers to support operation during a design basis event. The finding screened as having 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.

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.

Both the licensee's design basis calculation and Table 9.2-11 of the Braidwood UFSAR identified the SX design flowrate as 1650 gpm.

Contrary to the above, as of September 2, 2005, the licensee had operated the EDG with a cross-connected service water configuration to the jacket water coolers with 1600 gpm flow-rates. Because the licensee was able to confirm past operability of the EDG in this cross-connected service water configuration, this issue was determined to be of very low safety significance. Because it was entered in the licensee's corrective action program as AR 00368849 and 00368877, this violation is being treated as an NCV, consistent with Section VI.A of the NRC Enforcement Policy. (NCV 05000456/2005007-02; NCV 05000457/2005007-02)

b.2 Failure to Test Buried Service Water Intake Headers

Introduction: The inspectors identified a finding involving a Non-Cited Violation (NCV) of 10 CFR Part 50.55a(g)4 having very low safety significance (Green) for failure to perform periodic leakage testing required by the American Society of Mechanical Engineers (ASME) Code on the buried portions of the SX system intake piping.

Description: On August 17, 2005, the inspectors identified that the licensee had not performed the periodic pressure drop test or change in flow rate test to confirm the absence of leakage from the buried SX intake header piping as required by ASME Section XI. The licensee was committed to the 1989 Edition of Section XI for Code Interval No. 2, which required the licensee to complete a visual VT-2 inspection of isolable buried piping by performing a leakage test to measure the rate of pressure drop, or alternatively through a test to measure the change in flowrate between the ends of the buried pipe. The inspectors identified that the licensee had never performed either of these tests for the 36 and 48 inch diameter SX system (Code Class 3 system) intake headers (lines 2SX01BB, 1SX01BB, 1SX01AA, 2SX01BA, 1SX01BA) which could be isolated using the boundary valves located at the end of these buried pipe segments. The licensee had incorrectly classified this section of piping as nonisolable pipe in a nonredundant system which did not require leak testing.

The purpose of the ASME Code Section XI inservice test for buried piping is to detect leakage. Leaking SX pipe could be caused by the active corrosion processes present in the Braidwood SX system or by external pipe corrosion induced by contact with the soil/backfill and groundwater (the SX piping is buried below the groundwater table elevation). Further, the licensee had not performed internal visual examinations nor other types of nondestructive examinations of the buried portions of the SX intake piping headers to determine the extent of general or localized corrosion. Therefore, the inspectors were concerned that without completing the Code leakage tests, the licensee may operate the buried SX system intake piping with through-wall leaks, which could ultimately progress to a gross pipe failure/rupture. The licensee documented the failure to perform periodic leak testing in Assignment Report (AR) No. 00364793. The licensee did not identify the cause of the errors which led to the incorrect Code classification and testing for the affected SX piping.

Analysis: The licensee's performance deficiency associated with this finding, is the failure to perform the required periodic leakage testing of the buried SX intake piping. The inspectors noted that the licensee had numerous opportunities to identify this error. Specifically, each time the licensee staff performed the VT-2 surveillance pressure test, the procedure required licensee staff to evaluate and select the applicable test requirements based upon whether the buried piping was isolable. The inspectors concluded that the finding was greater than minor because, if left uncorrected, the failure to perform the required periodic tests could have allowed undetected through-wall flaws to remain inservice. These undetected flaws could grow in size until leakage from the buried SX intake pipe degrades system operation or if sufficient general corrosion occurs, a gross rupture or collapse of the SX piping sections could occur. This finding was assigned to the Mitigating System Cornerstone because the affected headers were in the SX system (mitigating system) and the finding affected the Mitigating System Cornerstone objective of equipment reliability. This finding was determined to be of very low safety significance, because the buried SX headers were operable and piping failure was not anticipated due to the external pipe coating, cathodic protection and low system operating pressure (25 pounds per square inch gage (psig)). Additionally, it screened as a finding 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.

Enforcement: ASME Code Section XI, Table IWD-2500-1, Item D.1.10 (Pressure Retaining Components) required a System Inservice Test (IWD-5221) be performed each Code inspection period.

ASME Code Section XI, IWD-5210 (a)(1) required a System Inservice Test in accordance with IWA-5211(c).

ASME Code Section XI, IWA-5211(c) required a visual examination VT-2 while the system is in service under operating pressure.

ASME Code Section XI, IWA-5244 "Buried Components" required "(a) In nonredundant systems where the buried components are isolable by means of valves, the visual examination VT-2 shall consist of a leakage test that determines the rate of pressure loss. Alternatively, the test may determine the change in flow between the ends of the buried components."

Contrary to these requirements, as of September 2, 2005, the licensee failed during the Code Periods completed for the Second Inservice Inspection Interval (Period 1 ending on July 28, 2001 and Period 2 ending on July 28, 2005 for Unit 1; Period 1 ending on October 16, 2001 for Unit 2), to perform the pressure drop or change in flow rate testing of the isolable buried portions of the 36 and 48 inch diameter SX system (Code Class 3 system, Item D.1.10) intake headers (lines 2SX01BB, 1SX01BB, 1SX01AA, 2SX01BA, 1SX01BA) which were bounded by SX isolation valves. Because of the very low safety significance of this finding and because the issue was entered into the licensee's corrective action program (AR 00364793 and AR 366352), it is being treated as a NCV, consistent with Section VI.A.1 of the Enforcement Policy (NCV 05000456/2005007-03; 05000457/2005007-03).

.3 Components

a. Inspection Scope

The inspectors examined the service water system and systems necessary for the Loss of AC power procedures to ensure that component level attributes were satisfied. The inspectors specifically focused on EDG loading and the CST for the Loss of AC Power event, and they additionally focused on the service water pumps and the service water strainers in the service water system. The following component level attributes of the service water system and the Loss of AC Power event were reviewed:

Component Degradation: This attribute was reviewed to ensure that components were being maintained consistent with the design basis. The inspectors reviewed service water surveillance tests to ensure that equipment degradation, if present, was within allowable limits. Additionally, the inspectors performed a selective review to determine if the licensee was performing inservice testing in accordance with applicable requirements. Selected testing of the EDGs was also reviewed to ensure that these components would operate as required for the Loss of AC Power event. Maintenance history was also reviewed for various components to ensure that there was not excessive degradation present.

Component Inputs/Outputs: The inspectors reviewed selected components in the service water system to ensure proper operation and input assumptions. Additionally, the inspectors verified selected component operation for the Loss of AC Power event to ensure that the expected output/operation was consistent with desired outcomes.

Equipment/Environmental Qualification: This attribute verifies that the equipment is qualified to operate under the environment in which it is expected to be subjected to under normal and accident conditions. The inspectors reviewed design information, specifications, and other documentation to ensure that the systems necessary for mitigation of the Loss of all AC Power event were qualified to operate within the temperatures specified in the station blackout documentation.

Equipment Protection: The inspectors reviewed design information, specifications, and documentation to ensure that the service water system and the systems necessary for the Loss of AC Power event were adequately protected from those hazards identified in the UFSAR which could impact their ability to perform their safety function. Specifically, the inspectors verified flood scenarios and design basis for the service water system. Additionally, the inspectors verified adequate heating ventilation and air conditioning and freeze protection for systems relied upon to cope with the station blackout scenario.

Operating Experience: This attribute ensures that applicable industry and site operating experience has been considered and applied to the components or systems. To verify this attribute, the inspectors reviewed licensee evaluations of operating experience including regulatory OE and site OE (Corrective action documents and maintenance history) to ensure that the licensee had appropriately applied applicable insights to the systems and components reviewed.

b. Findings

b.1 Non-Conservative CST Inventory Calculation

Introduction: The inspectors identified an NCV having very low safety significance (Green) of 10 CFR Part 50, Appendix B Criterion III, “Design Control”. Specifically, Braidwood Station failed to maintain an accurate design basis for the CST useable inventory. An additional depletion path of CST water during a Loss of Offsite Power, the makeup valve (1(2)CD0035) from the CST to the condenser hotwell, was not accounted for in the design basis calculation for useable CST volume.

Description: The inspectors identified that, during a Loss of Offsite Power (LOOP) event, depending upon plant conditions, the licensee could potentially leave a condensate makeup valve open which could unnecessarily and unknowingly drain water from the CST. Specifically, valve 1(2)CD0035, which the licensee used for makeup from the CST to the condenser hotwell, could be left open during a LOOP event, since there was no guidance for the operators to shut this manual valve. Guidance for ensuring that this valve was shut was contained in procedural guidance for a station blackout (SBO) event; however, even in this case the procedural guidance would occur well into the event, leaving a period of time when the 1(2)CD0035 would be draining the CST into the condenser hotwell.

The inspectors discovered that this CST depletion path was not accounted for in the licensee’s design basis calculation for CST useable volume. This useable volume was based upon the required amount of CST water needed to cooldown the plant to 350 degrees Fahrenheit using the AFW pumps, at which point, the RHR pump could be used to provide a forced cooldown. This extra depletion path (1(2)CD0035) could cause the plant to swap water sources - from the CST to the service water system - to the AFW pump prior to reaching 350 degrees. This would cause an unnecessary reliance on the safety related service water system. While the service water system served as the safety related source for the AFW pumps (The CST is neither safety related or seismically qualified.), a premature swap during a LOOP was not within the plant’s licensing basis and could potentially cause the CST to be inoperable based upon Technical Specification

LCO 3.7.6. Additionally, since both the CST and the service water system were considered by the plant's PRA analysis as fully capable water sources for bringing the plant to cooldown conditions during a LOOP event, the risk consequences of having the CST unavailable for this function was clearly adverse.

Based upon the inspector's concerns, the licensee issued AR 00367805. As part of the corrective actions for this issue, the licensee placed administrative controls on these valves to ensure quick closure should a LOOP occur. Additionally, the licensee performed a past operability evaluation using conservative calculation methodologies to show that in the past, even though this extra depletion path was substantial, the CST still contained enough excess volume to be capable of providing a cooldown to 350 degrees Fahrenheit as designed. Although the licensee was able to conclude that the CST was operable in the past, the licensee's operating procedures and design basis calculation for supporting operability did not account for this leakage path from the CST.

Analysis: The inspectors determined that this issue was a performance deficiency since the licensee failed to meet the requirements of 10 CFR Part 50 Appendix B, Criterion III. Specifically, the licensee did not maintain an accurate design basis to show that the CST had sufficient volume to support a plant cooldown during a LOOP event. Additionally, operating procedures did not account for this potential depletion path.

The issue was more than minor because it was associated with and affected the Mitigating Systems Cornerstone. Specifically, the capability of the water source for the AFW system during cooldown was adversely affected by the additional depletion path through the manual makeup valves. The finding screened as having 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.

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, at the Braidwood Station, neither the licensee's procedures nor the design basis calculations reflected the potential additional CST depletion path of the 1(2)CD0035 manual valve. Leaving this valve open would have had adverse consequences on the CST water source for the safety related AFW system.

Because the failure to account for this unidentified depletion path in the plant's operations procedures and design basis was determined to be of very low safety significance and because it was entered in the licensee's corrective action program as AR 00367805, this violation is being treated as an NCV, consistent with Section VI.A of the NRC Enforcement Policy. (NCV 05000456/2005007-04; 05000457/2005007-04)

b.2 Lack of Procedures, Equipment and Training for Recovery of Plugged SX Strainers

Introduction: The inspectors identified a finding of very low risk significance (Green) for failure to provide operators with equipment, procedures and training to manually operate the SX system strainers to recover the loss of automatic backwash capability.

Specifically, the loss of automatic strainer backwash function following a seismic event would lead to SX strainer plugging and without adequate recovery procedures, the loss of SX system flow. This finding did not constitute a violation of NRC requirements because the strainers (aside from the pressure boundary) and associated backwash equipment were not considered safety-related.

Discussion: The inspectors identified a concern for recovery from SX strainer plugging due to the loss of automatic backwash capability following a seismic event. The Braidwood SX system consists of two trains on each Unit and each SX system train contains a pump discharge strainer in the 36 inch pipe through which all SX flow must pass (e.g. cannot be bypassed). These strainers contain vertically oriented stacked metal disc type Poro-Edge™ filter elements with 0.0625 inch slots that prevented larger debris from entering the SX system and serve to minimize fouling of the SX cooled heat exchangers and components. The Poro-Edge™ filter elements in the strainer are routinely cleaned by a motor rotated internal backwash arm that directs the blow down flow through the Poro-Edge™ filter elements to the circulating water system through motor operated isolation valves 1(2)SX150A(B). The SX strainer backwash function normally occurs automatically every 8 hours or on high strainer differential pressure. The strainer backwash motor and isolation drain valve power supplies and control circuits are not safety-related or seismically qualified and following a seismic event these components could lose power. Specifically, the SX strainer motor is powered through an electrical panel with an automatic/manual control switch mounted within the SX pump room and if damage to internal electrical control panel circuits occurred during a seismic event, the SX strainer motor may not function. Under this scenario, corrective actions (repair electrical damage or manually operate the strainer) to restore the SX strainer backwash function would be required to prevent the accumulation of sediment/debris on the stainers which would cause a loss of SX flow to safety-related equipment.

On August 1, 2003, the 2A SX strainer high differential pressure alarm was received followed by a rapid drop in SX system pressure to 80 psig as documented in condition report No.169943. The licensee attributed the cause of this event to the buildup of lake material on the strainer without the automatic backwash feature which was unavailable due to a problem with the relay in the backwash control circuit. The operators took manual control of the backwash motor (power was available to the motor) to perform the backwash and clean the strainer for this event. Therefore, if a seismic event had occurred causing loss of power to the strainer motor, the licensee would not have been able to rapidly recover the loss of SX system flow caused by SX strainer plugging. To operate the backwash motor and strainer without power required an operator to first disconnect the motor and gear unit from the shaft, remove the shear key and reassemble the shaft assembly. The inspectors identified that the licensee had not staged equipment (ladder, flashlight, screwdriver, wrenches, needle nose pliers, flashlights), nor provided procedures (incorporating the vendor technical manual instructions and drawings) and training to operations personnel to perform the strainer backwash function without power to the backwash motor. The licensee estimated that the time to gather tools, and have a mechanic put strainer in a condition for manual backwash would be less than 4 hours. The licensee did not identify the cause of the errors which led to the failure to provide operators with equipment, procedures and training to manually operate the SX strainers to recover the loss of automatic backwash capability.

Analysis: The inspectors determined that the licensee's failure to provide equipment, procedures and training for manually backwashing the SX strainers was a performance deficiency that warranted a significance evaluation. The licensee had opportunities to identify the need for a procedure to manually backwash the SX strainers during prior SX strainer plugging events such as that documented in condition report No. 169943. The inspectors concluded that the finding was greater than minor, because the failure to provide equipment, procedures and training for manually backwashing the SX strainers (without power available to the backwash motor) could result in loss of cooling to safety-related equipment cooled by the SX system. Because the service water system provided cooling to systems primarily associated with decay heat removal following certain design basis accidents, the inspectors concluded that this issue was associated with the Mitigating Systems cornerstone. The inspectors performed a Phase 1 SDP review of this finding using the guidance provided in IMC 0609, Appendix A, "Significance Determination of Reactor Inspection Findings for At-Power Situations," and determined that this finding was a licensee performance deficiency of potential risk significance due to a seismic event that would degrade two or more trains of a multi-train system, which required a Phase 3 risk evaluation.

An NRC Regional III Senior Reactor Analyst (SRA) performed a qualitative Phase 3 risk evaluation and determined that the initiating event frequency of a seismic event was low. In performing this evaluation, the SRA considered the lack of data to support how long it would take to plug the strainers with sediment or debris and given that strainer plugging may take days, there was a high likelihood that recovery of the backwash function would occur. Although there were no plant procedures, the licensee had access to vendor documents which provided adequate instructions for the manual backwash operation, and the loss of off-site power operating procedure included actions to restore power to the 480 volt motor control center which supplied power to the SX strainer backwash motors and isolation valves. Based on these facts, the SRA determined that the finding was of very low safety significance.

Enforcement: No violation of regulatory requirements was identified because the strainers (aside from the pressure boundary) and associated backwash equipment were not considered safety-related. This finding was determined to be of very low risk significance based upon a Phase 3 risk analysis (FIN 05000456/2005007-05; 05000457/2005007-05). The licensee entered this finding into its corrective action program as AR 00367473.

b.3 Review of Seismic/Safety Classification for the SX Strainer Backwash System

In UFSAR (Updated Final Safety Analysis Report) Table 3.2-1 the SX strainer backwash motor was identified as Safety Category II and with Non-1E power. Section 3.2.1.2 of the UFSAR described Safety Category II structures, systems, and components as not specifically designed to remain functional in the event of the safe shutdown earthquake or other design basis events. As described in Section b.2 above, the SX strainer backwash motor and isolation drain valve power supplies and control circuits are not safety-related or seismically qualified and following a seismic event these components could lose power. Without corrective actions to restore the SX strainer backwash function, the accumulation of sediment/debris present in the SX system would build up on the strainers and cause a loss of SX flow to safety-related equipment. Because the

SX strainer backwash system was required to maintain the safety-related SX function following a seismic event, the inspectors were concerned that it should have been provided with seismically qualified control circuits and power supplies in accordance with 10 CFR 50 Appendix A, General Design Criteria (GDC) No. 2 and No. 17. Additionally, Section 3.2.1.2 of the UFSAR stated that Category II systems have no public health or safety implications, which appeared to be inconsistent with the importance of the SX strainer backwash system in maintaining the SX system function.

The licensee provided the inspectors with four points considered by the original Sargent & Lundy Engineers involved in Classification of the SX backwash system;

- Power can be restored to strainer backwash in the event of a loss of offsite power.
- Strainer backwash can be operated manually.
- There will be sufficient time to accomplish strainer backwash before adverse effects on the SX system.
- Strainer backwash capability is expected to function after a seismic event.

The inspectors were concerned that these points may not have been discussed with the NRC during the review of the plants original Safety Analysis Report. Further, the inspectors identified data which appeared to conflict with the presumption that sufficient time exists to accomplish strainer backwash before adverse affects on the SX system occur. From data recorded in operator logs for the Unit 2A SX train transient which occurred on August 1, 2003, (described in condition report No.169943), it took less than four minutes after receiving the low SX header pressure alarm condition (90 psig) to a condition that affected SX header pressure/flow (e.g., 80 psig discharge pressure). This condition was caused by debris accumulation fouling the 2A strainer due to the unavailability of the automatic backwash system. The licensee staff did not believe that this short time to foul the strainers was typical and may have been due to a debris intrusion. However, during a seismic event, the licensee staff acknowledged that additional small debris (above that found during normal lake conditions) may enter the SX system and foul strainers.

Based upon the importance of the automatic strainer backwash function, the inspectors were concerned that a formal seismic qualification of backwash control systems and circuits was required under 10 CFR 50 Appendix A, GDC 2. Additionally, if the SX strainer backwash control system should be considered safety-related, the SX strainer backwash system (strainer motors and isolation valves) would need to have a safety-related on-site power supply in accordance with GDC-17. This issue is considered an unresolved item pending NRC review of the plant licensing basis for the SX system backwash strainer function (URI 05000456/2005007-06; URI 05000455/2005007-06).

4. OTHER ACTIVITIES (OA)

4OA2 Problem Identification and Resolution

Review of Condition Reports

a. Inspection Scope

The inspectors reviewed a sample of service water system problems and issues related to the Loss of AC Power event that were identified by the licensee and entered into the corrective action program. The inspectors reviewed these issues to verify an appropriate threshold for identifying issues and to evaluate the effectiveness of corrective actions related to design issues. In addition, issue reports written on issues identified during the inspection were reviewed to verify adequate problem identification and incorporation of the problem into the corrective action program. The specific corrective action documents that were sampled and reviewed by the inspectors are listed in the attachment to this report.

b. Findings

No findings of significance were identified.

4OA6 Meetings, Including Exits

.1 Exit Meeting

The inspectors presented the inspection results to Mr. K. Polson and other members of licensee management at the conclusion of the inspection on September 2, 2005 and in telephone conversations with Mr. F. Lentine on September 6, 2005 and October 18, 2005. No proprietary information was identified.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee

K. Polson, Site Vice President
G. Boerschig, Plant Manager
M. Smith, Site Engineering Director
F. Lentine, Design Engineering Manager
D. Reidinger, Electrical/I & C Design Manager
J. Panfill, Design Engineering Supervisor
C. Mokijewski, Design Engineering
D. Ibrahim, Design Engineering
B. Koenig, Design Engineering
S. Mullins, Design Engineering
C. Furlow, Design Engineering
V. Gloria, System Engineering
D. Baran, Operations
D. Ambler, Regulatory Assurance Manager
S. Butler, Regulatory Assurance

Nuclear Regulatory Commission
C. Pederson, Director, Division of Reactor Safety
A. M. Stone, Chief, Engineering Branch 2, Division of Reactor Safety
N. Shah, Senior Resident Inspector
G. Roach, Resident Inspector
S. Burgess, Senior Reactor Analyst
L. Kozak, Senior Reactor Analyst
R. Moore, Senior Reactor Inspector - Region II

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

05000454/2005007-06; 05000455/2005007-06	URI	Review of Seismic/Safety Classification for the SX Strainer Backwash System

Opened and Closed

05000454/2005007-01; 05000455/2005007-01	NCV	Lack of a Bounding Voltage Drop Calculation During an SI
05000456/2005007-02; 05000457/2005007-02	NCV	EDG SX Cross-Connect not Supported by Design Basis
05000456/2005007-03; 05000457/2005007-03	NCV	Failure to Leak Test Buried SX Intake Header Piping

05000456/2005007-04; 05000457/2005007-04	NCV	Non-Conservative CST Inventory Calculation
05000454/2005007-05; 05000455/2005007-05	FIN	Failure to provide Procedure For Recovery of Plugged SX Strainer

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.

1R21 Safety System Design and Performance Capability

Calculations

19-AN-1; Relay Settings for U-1 Generator, Main, Unit Auxiliary & System Auxiliary Transformers; Revision 05A

19-AN-3; Protective Relay Settings for 4.16 kV ESF Switchgear; Revision 16

19-AN-5; Diesel Generator Protective Relay Settings; Revision 3

19-AN-29; Second Level Undervoltage Relay Setpoint; Revision 2

19-AQ-66; Instrument Bus Inverter Electrical Loading - 1(2)IP05E, 6E, 7E, 8E; Revision 1A

AQ-68; Division Specific Degraded Voltage Analysis; Revision 6

ATD-0021; Heat Load TO The Ultimate Heat Sink During Station Blackout; Revision 0.

ATD-0051; Performance Analysis For SX System Cooled Lubricating Oil Heat Exchanger; Revision 0.

ATD-0063; Heat Load To The Ultimate Heat Sink During A Loss Of Cooling Accident; Revision 4.

ATD-0109; Thermal Performance Of UHS During Postulated Loss Of Coolant Accident; Revision 3.

BRW-01-0117-M; Flaw Evaluation Of Line 1WER7AB; April 24, 2001.

BRW-00-0017-M; Byron/Braidwood Uprate Project- Post LOCA Component Cooling water System Temperature Analysis; Revision 1.

WCAP-10541; Reactor Coolant Pump Seal Performance Following A Loss Of All AC Power; Revision 2.

BRW-99-0461-M; Essential SW Flow Model Updated; May 19, 2003.
90-0094; Station Blackout Analysis; Revision 0.

BRY97-224/BRW-97-0472-E; 125 VDC Voltage Drop Calculation; Revision 2

BRY97-227/BRW-97-0475-E, Revision 0; 125 VDC Fuse Sizing and Coordination; Revision 0

BRW-00-0018-M; Ultimate Heat Sink Evaluation for Power Uprate Heat Load Condition; dated May 18, 2000;

BRW-00-0237-E; Voltage Drop Calculation for 4160 V Switchgear Breaker Control Circuits; Revision 0

BRW-97-0384-E; 125 VDC Battery Sizing Calculation; Revision 3

BRW-97-0724-E; Motor Operated Valve Actuator Motor Terminal Voltage and Thermal Overload Sizing Calculation; Revision 0

BRW-03-0122-M; Evaluation of CST Technical Specification at Braidwood Station; Revision 0

CN-FSE-00-2; Byron/Braidwood Unit 1 Auxiliary Feedwater Storage Volume for Uprating to 3600.6 Mwt NSSS; Revision 1

MAD 90-0127; Reactor Cooldown; Revision 0

PC-01; Review of Containment Isolation Valves for Station Blackout; Revision 1

SX 1-85; Essential Service Water Pumps Net Positive Suction Head Available; Revision 1

SX2-85; SX Pump Head Check; Revision 0

3C8-0685-002; Auxiliary Building Flood Level Calculations (SX Area); Revision

VA-102; Auxiliary Building Energy Load Calculations for Elevation 330'; 346'; 364'; 383'; 401' & 426' in Abnormal Condition; Revision 3

L-VA-430; Auxiliary Building HVAC System Minimum Air Flow; Revision 2

T-3; Station Blackout - Diesel Generator Loading; Revision 1

<u>Drawings</u>

20E-1-4001A; Station One Line Diagram; Revision O

20E-2-4001A; Station One Line Diagram; Revision L

20E-2-4002C; Single Line Diagram 4.16 kV Switchgear Bus 241, 243; Revision R

20E-1-4006A; Key Diagram 4160 V ESF Switchgear Bus 141 (1AP05E); Revision H

20E-1-4006B; Key Diagram 4160 V ESF Switchgear Bus 142 (1AP06E); Revision K

20E-2-4018A; Relay & Metering Diagram 4160 V ESF Switchgear Bus 241; Revision N

20E-2-4018C; Relay & Metering Diagram 4160 V ESF Switchgear Bus 243; Revision F

20E-1-4008E; Key Diagram 480 V Auxiliary Building ESF MCC 131X2 (1AP25E) & 131X2A (1AP25E-A); Revision AC

20E-1-4008AW; Key Diagram 480 V Auxiliary Building MCC 133X1B (1AP36E); Revision AC

20E-1-4008BJ; Key Diagram 480 V Auxiliary Building MCC 134V1 (1AP39E); Revision U

20E-1-4030SX01; Schematic Diagram Essential Service Water Pump 1A, 1SX01PA; Revision U
20E-1-4030SX02; Schematic Diagram Essential Service Water Pump 1B, 1SX01PB; Revision V

M-900; Outdoor Piping Arrangement; Revision 0.

M-906; Outdoor Non-Essential And Essential-Supply Piping; Revision B.

<u>Engineering Changes</u>

DCP/EC 335668; Change Instantaneous Setting of the Circuit Breaker for 2SX016B; Revision 0

DCP 9900414; Extend SX Discharge Pipes Above Lake Level; August 25, 2000

DCR 350143; Addendum To Piping Stress Report Subsystem 2SX16 Revision 2 Essential Service Water System; July 13, 2004.

DCR 990725; Determine The Flood Level In The SX Pump Room; June 27, 2000.

EC E20-1-96-263; Increase SX Pump Room Water Removal Capacity System; January 22, 1997

EC 336758; Changed Valve 2SX033 Orientation Tolerance; April 24, 2002

EC 351271; Fail Open 1Auxiliary Feedwater Pump Oil Cooler Outlet Isolation Valve 1SX101A, October 8, 2004

EC 353912; Permanently Remove Valve 2SX101A; Revision 0

EC 41081; Reroute SX and CC Power Cables; October 22, 1997

EC 42321; Replace the Carbon Steel Sensing lines of 1FE-SX112 with Stainless Steel lines

EC 42687; Extend SX Discharge Pipe Above Lake Level; Revision 0

EC 42942; Supply FP Water to 2B CV Pump Oil Cooler; dated December 19, 2001

EC 357385; SI Block Start Evaluation; Revision 0

MPC P20-1-92-601; Replace SX Flow to CC Heat Exchanger Indicators OFI-SX044 and 1FI-SX031 with more accurate Indicators. Provide ESF Power Supply to Panel 1PA20JB;Revision 0

Issue Reports Generated Due to the Inspection

AR 00363142; Drawing 20E-1-4001A Identifies ACB 1411 as ACB 141; August 15, 2005

AR 00363277; NRC SSDPC - Drawing Discrepancy on 20E-2-4001A; August 15, 2005

AR 00363806; Calculation Shown Active in Passport when the Calc. Is Void; August 17, 2005

AR 00363869; Drawing Does Not Reflect Correct HP for Motors; August 17, 2005

AR 00364793; SSDPC Inspection - Pressure Test of SX Piping; August 19, 2005

AR 00364858; SSDPC Inspection - Enhancement to BWVSR 3.7.9.3; August 19, 2005

AR 00364915; Typographical Error in Calculation BRW-97-0472-E; August 19, 2005

AR 00366352; SSDPC Inspection Pressure Test SX Pipe Follow-Up to IR 364793; August 24, 2005

AR 00366863; Calculation Deficiencies - ATD-0021 SBO UHS Heat Load; August 25 2005

AR 00366895; Some References of Calc 90-0094 Are not Presently Retrievable; August 25, 2005

AR 00367473; Potential Enhancement to Strainer Backwash Response; August 27, 2005

AR 00367708; 1SXC5A-24" Wall Thickness Below Manufacturing Tolerance; August 29, 2005

AR 00367805; NRC SSDPC - Potential procedure Enhancement for CD System; August 29, 2005

AR 00367913; Calculation BRW-97-0472-E, Calculation Summary Typo; August 29, 2005

AR 00367993; Station Blackout Recovery Procedure Enhancements; August 29, 2005

AR 00367997; Procedure Enhancement to Restore Non-ESF Loads after LOOP; August 29, 2005

AR 00368201; Procedure ER-AA-340 (GL 89-13) Enhancements; August 30, 2005

AR 00368613; Catch Basin Controls Outside of RCA; August 31, 2005

AR 00368849; NRC SSDPC - Outage SX Cross-Tie Procedure Concerns; August 31, 2005

AR 00368877; NRC SSDPC - Monthly DG SX Cross-Tie Flush Procedure Concerns; August 31, 2005

AR 00368906; NRC SSDPC - Conflicting Statement in TS Bases 3.7.6; August 31, 2005

AR 00369088; SSDPC Inspection - Enhancement to Performance Monitoring; September 1, 2005

AR 00369124; Improvement to AP System Support Calculations; September 1, 2005

AR 00369134; ASME Code Case –513 for Class 3 Piping Leaks; September 1, 2005

AR 00369267; SX System CF Treatment Requirement with Respect to GL89-13; September 01, 2005

AR 00369569; SSDPC ID’ed Enhancement Opportunity to Performance Monitoring; September 2, 2005

AR 00369591; NRC SSDPC - Retrievability of Station Blackout Documentation; September 2, 2005

Issue Reports Reviewed During the Inspection

AR 00124315; Silt Build Up in 1A Forebay; September 24, 2001

AR 00132273; Silt Found in SX Line During pre-NDE Inspection of Freeze; November 19, 2002

AR 00140853; BwEPs Are Not Consistent with CST TS Bases 3.7.6; January 28, 2003

AR 00154319; Check Valve Seating Surfaces Found in Degraded Condition; April 17, 2001.

AR 00169834; 2A SX Strainer Alarm Concern For Lake Chemistry; July 31, 2003.

AR 00169943; Low SX System Pressure Due to High 2A Strainer D/P; August 1, 2003.

AR 00170524; A SX Strainer Has Biological Material in It; August 6, 2003.

AR 00189342; Improper Breaker Coordination on MPT Cooling System; December 5, 2003

AR 00199100;1B SX Pump Room Cubicle Cooler Flow Lower Than Required; February 3, 2004.

AR 00199206; Lake Chemistry Trend - Calcium Carbonate Issue; March 3, 2004.

AR 00200148;1B SX Strainer DP Slow to Decrease During Backwash; February 6, 2004.

AR 00200601; Flow to 1B Train RCFC's Coils Below Minimum - LCOAR ; March 9, 2004.

AR 00204227;1B CV Pump Lube Oil Cooler (1CV03SB) As-Found Data; February 26, 2004.

AR 00204229;1B CV Pump Gear Lube Oil Cooler (1CV02SB) As-Found Data; February 26, 2004.

AR 00206831; Review Condition of 1A CV Pump Gear Oil Cooler 1CV02SA; March 8, 2004.

AR 00207848; Unexpected CM - 1A SI PP Lube Oil Cooler Inspection Results; March 11, 2004.

AR 00208578; Unexpected CM - 2B CV Pump Gear Oil Cooler 2CV02SB; March 16, 2004.

AR 00208786; Unexpected CM - 2B CV PumP Lube Oil Cooler 2CV03SB; March 16, 2004.

AR 00215229; Results of Review of Historic Switchyard Voltages; April 15, 2004

AR 00220821; Minimum Wall Violation On 2SX26AB-10" Found By Pre-Freeze NDE; May 13, 2004.

AR 00221040; Submittal of Incorrect Data for AC Power System Availability; May 14, 2004

AR 00223964; Enter LCO 3.7.8 For Low 1B SX Pump Cubicle Cooler Flow; May 26, 2004.

AR 00224989; 1FI-SX118 Reading is Suspect (Unplanned LCO Entry); June 01, 2004.

AR 00226067; 0SX246B Check VLV Internal Leak By; June 4, 2004.

AR 00229568; Concerns Noted During Engineering Review of 0BWOA ELECT 1; June 18, 2005

AR 00232779; 1VP01AC Did Not Meet Thermal Performance Acceptance Criteria; June 30, 2004.

AR 00232869; SX Flow To RCFCS (Per 2BWOSR 3.6.6.3-1); June 30, 2004.

AR 00244690; 2FI-SX123 RCFC Flow - Needs Cleaning; August 16, 2004.

AR 00244706; WR Needed To Clean 2FE-SX122 RCFC 2A SX Outlet Flow; August 16, 2004.

AR 00244708; WR Needed To Clean 2FE-SX112 RCFC 2A SX Outlet; August 16, 2004.

AR 00244712; WR Needed To Clean 2FE-SX124 RCFC 2C SX Outlet; August 16, 2004.

AR 00249459; Clean 1FE-SX122 Annubar Per RCR 224989, September 2, 2004.

AR 00249460; Clean 1FE-SX123 Per RCR 224989; September 2, 2004.

AR 00251040; Minor Degradation (Corrosion) Of Bolting; September 8, 2004.

AR 00259371; Lower Than Expected Flow On 2FI-SX121 - 2D RCFC Flow; October 2, 2004.
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LIST OF ACRONYMS USED

AC	Alternating Current
ADAMS	Agencywide Documents Access and Management System
ADS	Automatic Depressurization System
ASME	American Society of Mechanical Engineers
CFR	Code of Federal Regulations
CST	Condensate Storage Tank
DC	Direct Current
DRS	Division of Reactor Safety
ECR	Engineering Change Request
EDG	Emergency Diesel Generator
FIN	Finding
GDC	General Design Criteria
HVAC	Heating, Ventilation and Air Conditioning
IMC	Inspection Manual Chapter
IEEE	Institute of Electrical and Electronics Engineers
IR	Issue Report, as used by licensee (and NRC Inspection Report)
LOCA	Loss of Coolant Accident
LOOP	Loss of Offsite Power
NCV	Non-Cited Violation
NEI	Nuclear Energy Institute
NPSH	Net Positive Suction Head
NRC	Nuclear Regulatory Commission
PARS	Publicly Available Records
OE	Operability Evaluation
SBO	Station Blackout
SDP	Significance Determination Process
SSDPC	Safety System Design and Performance Capability
SX	Essential Service Water
UFSAR	Updated Final Safety Analysis Report
URI	Unresolved Item