February 23, 2007

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 EVALUATION OF CHANGES, TESTS, OR EXPERIMENTS, AND PERMANENT PLANT MODIFICATIONS BASELINE INSPECTION REPORT 05000456/2007002(DRS); 05000457/2007002(DRS)

Dear Mr. Crane:

On January 26, 2007, the U.S. Nuclear Regulatory Commission (NRC) completed baseline inspections of Evaluation of Changes, Tests, or Experiments and Permanent Plant Modifications at the Braidwood Station. The enclosed report documents the results of the inspection which was discussed with Mr. T. Couto and other members of your staff at the completion of the inspection on January 26, 2007.

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, two NRC-identified findings of very low safety significance were identified. 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 (NCVs) in accordance with Section VI.A.1 of the NRC's Enforcement Policy.

If you contest the subject or severity of a NCV, 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 Braidwood Station.

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter, its enclosure, and your response (if any), 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), 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. 50-456; 50-457
License Nos. NPF-72; NPF-77

Enclosure: Inspection Report 05000456/2007002(DRS);05000457/2007002(DRS)
w/Attachment: Supplemental Information

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

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Vice President - Operations Support
Director Licensing
Manager Licensing - Braidwood and Byron
Senior Counsel, Nuclear, Mid-West Regional Operating Group
Document Control Desk - Licensing
Assistant Attorney General
Illinois Emergency Management Agency
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:	05000456/2007002(DRS); 05000457/2007002(DRS)
Licensee:	Exelon Generation Company, LLC
Facility:	Braidwood Station, Units 1 and 2
Location:	Braceville, IL
Dates:	January 8 through January 26, 2007
Inspectors:	M. Holmberg, Reactor Inspector, Lead M. Munir, Reactor Inspector
Observer:	D. Szwarc, Reactor Engineer
Approved by:	D. Hills, Chief Engineering Branch 1 Division of Reactor Safety (DRS)

SUMMARY OF FINDINGS

IR 05000456/2007002(DRS); 05000457/2007002(DRS); 1/08/2007 - 1/26/2007; Braidwood Station, Units 1 and 2; Evaluation of Changes, Tests, or Experiments (10 CFR 50.59), and Permanent Plant Modifications.

The inspection covered a 2-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. Two Green Non-Cited Violations (NCVs) 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. Inspector-Identified and Self-Revealed Findings

Cornerstone: Barrier Integrity

Green. The inspectors identified a Severity Level IV NCV of 10 CFR 50.59 "Changes, Tests, and Experiments," having very low safety significance (Green) for the licensee's failure to perform an adequate safety evaluation review for changes made to the facility per modification EC 357102. In safety evaluation BWR-E-2006-196 (for EC 357102), the licensee failed to provide an adequate basis as to why the installation of two 15,000 gallon sulfuric acid tanks sharing a common drain system with a sodium hypochlorite tank did not create conditions for an accident of a different type than any previously evaluated in the Updated Final Safety Analysis Report. When these chemicals mix, they would produce an on-site release of chlorine gas, which could potentially overcome the control room operators. The licensee entered this issue into the corrective action program and considered the control room heating, ventilation and air conditioning system operable because of the time of year (mid-winter) such that weather conditions favoring formation of a tornado does not occur. Also, the licensee stated that the sulfuric acid tanks would be drained to approximately 2000 gallons each during the times of year when tornadoes are more likely to minimize the magnitude of any chlorine release.

Because the issue potentially impacted the NRC's ability to perform its regulatory function, this finding was evaluated using the traditional enforcement process. The finding was determined to be more than minor because the inspectors could not reasonably determine that the activity to install the new sulfuric acid tanks without detection and alarm circuits to notify the control room would not have ultimately required NRC prior approval. This finding has a cross-cutting aspect in the area of human performance because the licensee did not make appropriate or conservative decisions with respect to reviewing the plant design and license basis. Specifically, the licensee staff chose a narrow interpretation of Regulatory Guide 1.78 "Assumptions for Evaluating the Habitability of a Nuclear Power Plant Control Room During a Postulated

Hazardous Chemical Release," such that a control room habitability evaluation for an on-site chlorine gas release was not completed. The finding was not suitable for a significance determination process evaluation, but has been reviewed by NRC Management in accordance with qualitative criteria of Appendix M of IMC 0609 and is determined to be a finding of very low safety significance. (Section 1R02.1.b.1)

Cornerstone: Mitigating Systems

Green. The inspectors identified a Severity Level IV NCV of 10 CFR 50.59(d)(1) for the licensee's failure to document an evaluation which provides a basis for the determination that the change, test, or experiment did not require a license amendment. Specifically, for Revision 4 and Revision 101 of procedure 1/2BwOA ELEC-4 "Loss of Off-site Power" the licensee failed to provide an evaluation as to why replacing an automatic pump start function with a manual action (e.g., placing the A motor driven auxiliary feed pump in "pull out") did not present more than a minimal increase in the likelihood of occurrence of a malfunction of a structure system or component important to safety previously evaluated in the Updated Final Safety Analysis Report. The licensee entered this issue into the corrective action program, and initiated actions to complete a 10 CFR 50.59 evaluation to determine if these procedure changes were acceptable without a license amendment. Because the licensee's procedure changes implemented the appropriate Technical Specification requirements, this issue did not affect the operability of the auxiliary feedwater system.

Because the issue potentially impacted the NRC's ability to perform its regulatory function, this finding was evaluated using the traditional enforcement process. The finding was determined to be more than minor because the inspectors could not reasonably determine that the procedural change, that affected the Updated Final Safety Analysis Report described design function of equipment important to safety, would not have ultimately required NRC prior approval. Based upon this Phase 1 screening, the inspectors concluded that the issue was of very low safety significance. (Section 1R02.1.b.2)

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 January 8 through 26, 2007, the inspectors reviewed ten safety evaluations performed pursuant to 10 CFR 50.59 to determine if the evaluations were adequate and to determine if prior NRC approval was obtained if applicable. 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

b.1 Inadequate Basis in 10 CFR 50.59 Evaluation for Installation of Sulfuric Acid System

Introduction: The inspectors identified a Severity Level IV Non-Cited Violation (NCV) of 10 CFR 50.59 "Changes, Tests, and Experiments," having very low safety significance (Green) for the licensee's failure to perform an adequate safety evaluation review for changes made to the facility per modification EC 357102. In safety evaluation BWR-E-2006-196 (for EC 357102), the licensee failed to provide an adequate basis as to why the installation of two 15,000 gallon sulfuric acid tanks sharing a common drain system with a sodium hypochlorite tank did not create conditions for an accident of a different type than any previously evaluated in the Updated Final Safety Analysis Report (UFSAR). When these chemicals mix, they would produce an on-site release of chlorine gas, which could potentially overcome the control room operators. This accident scenario had not been previously considered in the UFSAR and was not bounded by the evaluation of chlorine gas released from off-site sources.

Description: The Braidwood Station Units 1 and 2 were originally equipped with chlorine gas detectors, which served to automatically isolate the control room heating, ventilation and air conditioning (HVAC) system from the outside air if chlorine gas was detected. The licensee had changed the plant Technical Specifications (TS) and UFSAR to remove the automatic chlorine gas detection system and associated TS surveillances based upon the capability to manually realign the control room HVAC and isolate the control room envelope from a potential off-site spill of chlorine gas (no on-site source of chlorine gas existed). In the NRC Safety Evaluation (SE) dated February 28, 1995, the NRC approved removal of the chlorine detection systems and associated TS surveillances for the control room HVAC system. In this SE, the NRC considered off-site transportation-related chlorine gas release sources and the basis for approval included "no potential for stationary chlorine release that could pose a threat to control room habitability." Mitigation of chlorine gas from off-site sources was discussed in Section 6.4.4.2 of the UFSAR and was dependent upon timely notification of chlorine gas spills from off-site authorities. This section stated: "The control room HVAC system is provided with control switches on the local control panels which can manually isolate the system upon notification of an accidental release of chlorine gas from sources external to the station."

The control of on-site storage facilities containing hazardous chemicals for the Braidwood Station is identified in Appendix A of the UFSAR which implemented Regulatory Guide (RG) 1.78 "Assumptions for Evaluating the Habitability of a Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release," Revision 0. This RG identified a detailed list and quantity of chemicals considered hazardous to control room habitability, which included both sulfuric acid and chlorine gas. This RG required a detailed analysis of the effects of a hazardous substances released (such as chlorine gas) on the habitability of the control room.

In December of 2006, the licensee installed a bulk sulfuric acid addition system at the lake screenhouse under EC 357102. This included installation of two 15,000 gallon capacity tanks of sulfuric acid in proximity to a sodium hypochlorite storage tank and the tanks share a common drainage system. These tanks were located at the lake screen house which the inspectors estimated was within approximately 0.5 miles of the control room air intake. In safety evaluation BWR-E-2006-196 "Sulfuric Acid System Addition at the Lake Screenhouse," the licensee recognized that a chlorine gas release would occur following a failure of these tanks due to tornado or tornado driven missiles. Specifically, these tanks were not designed to withstand the wind loads or postulated missiles described in Table 3.5-3 "Tornado Generated Missiles and Their Properties" of the UFSAR. In a tornado induced tank failure, the contents of the sulfuric acid and sodium hypochlorite tanks would mix in the common drainage piping system surrounding these tanks and generate chlorine gas. In BWR-E-2006-196, the licensee determined that release of chlorine gas was a low probability event, but did not evaluate how this event would be mitigated, nor how the storage system deviated from the design and licensing basis described in the NRC SE or UFSAR Sections discussed above. Specifically, Appendix B of RG 1.78 required a detailed diffusion analysis for release of a hazardous chemicals (e.g., chlorine) to assess the impact on control room habitability. Further, step C.3 of RG 1.78 stated any hazardous chemicals stored on-site should be accompanied by instrumentation that will detect its escape, set off an alarm, and provide

a readout in the control room. The licensee staff stated that they met RG 1.78, because the vapor pressure of sulfuric acid was below 10 torr as discussed in step C.5 of RG 1.78 and therefore, no evaluation of the toxic effects from a sulfuric acid spill on control room habitability was required. Also, with no direct storage of chlorine gas on-site, the licensee did not believe RG 1.78 controls applied to the release of chlorine gas generated by the mixing of sulfuric acid and sodium hypochlorite. The inspectors did not believe this narrow interpretation of RG 1.78 was appropriate, because Step C.12 of RG 1.78 required that concurrent chemical release of container contents during an earthquake, tornado or flood should be considered for chemical container facilities that are not designed to withstand these natural events.

The inspectors concluded that the licensee had not provided an adequate basis to answer Question 5 of safety evaluation BWR-E-2006-196 which asked "Does the proposed activity create a possibility for an accident of a different type than any previously evaluated in the UFSAR?" The licensee incorrectly answered "no" to this question based upon a determination that the on-site release of chlorine gas was unlikely and that the control room response to an on-site release would be the same as that discussed in the UFSAR for an off-site chlorine gas release. These statements are true, but the licensee failed to identify, and evaluate the lack of any notification systems such as the automatic detection and alarm systems that read out in the control room as described in RG 1.78 for storage of on-site chemicals which pose a hazard to control room habitability. Without automatic alarm systems, the inspectors concluded, that in the event of an on-site generation of chlorine gas, the control room operators may not have sufficient notification to preclude chlorine intrusion into the control room. Because the licensee had not completed a detailed diffusion analysis for the potential chlorine gas generated as discussed in Appendix B of RG 1.78, the inspectors could not determine the consequences of this postulated event. Therefore, the inspectors concluded that the storage of bulk sulfuric acid in proximity to the sodium hypochlorite tank created conditions for a different type of accident than any previously considered in the UFSAR.

The licensee entered this issue into the corrective action program (AR 00583639) and considered the control room HVAC system operable because of the time of year (mid-winter) such that weather conditions favoring formation of a tornado does not occur. Also, the licensee stated that the sulfuric acid tanks would be drained to approximately 2000 gallons each during the times of year when tornadoes are more likely to minimize the magnitude of any chlorine release.

Analysis: The inspectors determined that this issue was a performance deficiency warranting a significance evaluation, because in December of 2006, the licensee failed to provide an adequate basis for changes made to the facility in accordance with 10 CFR 50.59. Specifically, the licensee failed to provide an adequate basis as to why changes made per modification EC 357102 did not create conditions for a different type of accident than any previously considered in the UFSAR. The finding was determined to be more than minor because the inspectors could not reasonably determine that the activity to install the new sulfuric acid tanks without detection and alarm circuits to notify the control room would not have ultimately required NRC prior approval. This finding has a cross-cutting aspect in the area of human performance because the licensee did not make appropriate or conservative decisions with respect to reviewing the plant design and license basis. Specifically, the licensee staff chose a narrow interpretation of

RG 1.78, such that a control room habitability evaluation for an on-site chlorine gas release was not completed. Further, the licensee staff had not reviewed the basis for the NRC SE related to control room habitability from an off-site chlorine gas release to determine if it would bound an on-site release of chlorine gas. Because violations of 10 CFR 50.59 are considered to be violations that potentially impede or impact the regulatory process, they are dispositioned using the traditional enforcement process instead of the significance determination process (SDP). However, if possible, the underlying technical issue is evaluated under the SDP to determine the severity of the violation. In this case, the underlying technical issue affected the Barrier Integrity Cornerstone. The finding was evaluated under the SDP using NRC's inspection manual chapter (IMC) 0609, Appendix A, "Significance Determination of Reactor Inspection Findings for At-Power Situations," and the inspectors answered "yes" to the question "Does the finding represent a degradation of the barrier function of the control room against smoke or a toxic atmosphere"? The SDP required a phase 3 analysis to resolve this type of finding. However, after consultation with a Region 3 Senior Reactor Analyst it became apparent that no SDP methods or tools exist to determine the significance of the finding. Therefore, the finding was not suitable for evaluation using the SDP, so the risk significance was established in accordance with the qualitative criteria of Appendix M of IMC 0609. Specifically, the qualitative decision-making attribute from Table 4.1 of Appendix M "Period of time (exposure) effect on the performance deficiency" was applicable to this finding. Because this modification had only been installed since December of 2006, the exposure time for this accident scenario was low. Further, the probability of tornado or tornado generated missiles hitting and rupturing both the sulfuric acid tank and the sodium hypochlorite tank concurrently causing the contents of the tanks to mix in the common drainage to produce chlorine gas was also very low. Therefore, based upon a qualitative measure of risk determined in accordance with Appendix M, NRC Management concluded that the issue was of very low safety significance (Green).

Enforcement: Title 10 CFR 50.59(d)(1) states, in part, that the licensee shall maintain records of changes in the facility, of changes in procedures, and of tests and experiments as described in the UFSAR. These records must include a written evaluation which provides a basis for the determination that the change, test, or experiment does not require a license amendment.

Contrary to the above, as of January 26, 2007, the licensee failed to provide an adequate basis for the determination that the change to the facility per EC 357102 in December of 2006, was acceptable without a licensee amendment. Specifically, in BWR-E-2006-196, the licensee failed to provide an adequate basis as to why the installation of two 15,000 gallon sulfuric acid tanks in proximity to a sodium hypochlorite tank did not create conditions for an accident of a different type than any previously evaluated in the UFSAR. In this case, the accident scenario involving an on-site release of chlorine gas overcoming the control room operators had not been previously evaluated in the UFSAR and was not bounded by the UFSAR evaluation of chlorine releases from off-site sources. In accordance with the Enforcement Policy, this violation of the requirements of 10 CFR 50.59 was classified as a Severity Level IV Violation because the underlying technical issue was of very low safety significance. Because this non-willful violation was non-repetitive, and was captured in the licensee's corrective action program (AR 00583639), it is considered a NCV consistent with VI.A.1 of the NRC Enforcement Policy. (NCV 05000456/2007002-01; 05000457/2007002-01 (DRS))

b.2 Lack of 50.59 Evaluation for Substitution of Manual Actions for Automatic Start Signals on the Motor Driven AFW Pump

Introduction: The inspectors identified a Severity Level IV NCV of 10 CFR 50.59 "Changes, Tests, and Experiments," having very low safety significance (Green) for the licensee's failure to document an evaluation which provides a basis for the determination that the change, test, or experiment did not require a license amendment. Specifically, for Revision 4 and Revision 101 of procedure 1/2BwOA ELEC-4 "Loss of Off-site Power" the licensee failed to provide an evaluation as to why replacing an automatic pump start function with a manual action (e.g., placing the A motor driven auxiliary feed pump in "pull out") did not present more than a minimal increase in the likelihood of occurrence of a malfunction of a structure system or component (SSC) important to safety previously evaluated in the UFSAR.

Description: UFSAR Sections 10.4.9.3.1, 15.2.6.1 described that the AFW pumps are started automatically on either a low-low level in any steam generator, a safety injection signal, or a loss of off-site power. One AFW pump utilizes a direct diesel-engine so that auxiliary feedwater can be supplied in the event that both on-site and off-site sources of AC power are lost. In the event of a loss of off-site power, the motor driven AFW pump will auto-start and continue to run until the off-site power condition can be cleared (AC power provided by the on-site emergency diesel generators).

The motor driven AFW pump delivers flow to the steam generators from the condensate storage tank. This water is colder than the normal feedwater and is not accounted for in the calorimetric calculation making the value of reactor power non conservative with auxiliary feedwater flow. Therefore, in the event of a loss of off-site power where the AFW pump auto-starts and continues to run, the licensee changed 1/2BwOA ELEC-4 "Loss of Off-site Power" to place the A motor driven AFW pump in "pull out" to prevent continued flow of cold feedwater entering the steam generators causing an undesirable positive reactivity addition (e.g., power increase).

In screening BWR-S-2006-207, the licensee evaluated the effect of adding steps in Revision 101 to abnormal operating procedure 1/2BwOA ELEC-4 "Loss of Off-site Power" that included placing the motor driven train A auxiliary feedwater (AFW) pump in "pull out" to prevent automatic restart and concluded that no safety evaluation was required. In Attachment B of this procedure similar steps already existed to place the A AFW pump in "pull out" and as such the licensee intended this screening to support the incorporation of these steps at an earlier point in the procedure. The licensee first added the instructions in Attachment B to place the pump in "pull out" in 1996 during Revision 4 of 1/2BwOA ELEC-4 and had performed a screening dated September 9, 1996, to accept this change. With the pump in "pull out," the auto start signals for the A AFW pump are bypassed and the pump will not start without manual operator action. Replacing an automatic function with a manual action is considered to be an adverse change to the reliability of the AFW pump function (reference Section 4.2.1 of NEI 96-07, "Guidelines for 10 CFR 50.59 Implementation") and required a 10 CFR 50.59 evaluation to determine why this adverse action did not present more than a minimal increase in the likelihood of occurrence of a malfunction of a SSC important to safety previously evaluated in the UFSAR. Further, based upon examples in Section 4.3.2 of NEI 96-07, a permanent substitution of manual for automatic actions requires NRC prior approval because it constitutes more than minimal increase in likelihood of malfunction of a SSC.

Both screenings accepted the “pull out” pump condition, and the licensee failed to identify that the change adversely affected UFSAR described design basis functions of the AFW pump. Instead, the licensee concluded this action did not affect UFSAR described design basis functions because the procedure 1/2BwOA ELEC-4 steps verified that there were no accidents requiring the AFW pump at the point that the control switch was placed in “pull out” and that the TS LCO 3.7.5 actions would be implemented due to the inoperable A AFW pump. For this case, compliance with the plant TS did not ensure the plant design and licensing basis was met, because the resulting equipment configuration changed the plant response to design basis accidents. By procedurally directing operators to take the A AFW pump to “pull out,” the pump could no longer automatically respond to accidents as described in the UFSAR Chapter 15. Therefore, the licensee failed to recognize these changes adversely affected on the plant design basis function for AFW pump to automatically respond to UFSAR Chapter 15 accidents (loss of main feedwater, secondary pipe breaks, loss of coolant accident and cooldown transients).

The licensee entered this issue into the corrective action program (AR 00583152), and initiated actions to complete a 10 CFR 50.59 evaluation to determine if these procedure changes to 1/2BwOA ELEC-4 were acceptable without a license amendment. Because the licensee’s procedure changes implemented the appropriate TS requirements, this issue did not affect the operability of the AFW system.

Analysis: The inspectors determined that the failure to perform an evaluation of these adverse changes to the design basis of the plant as described in the UFSAR was a performance deficiency warranting a significance evaluation. Specifically, the licensee failed to provide a basis as to why changes made to 1/2BwOA ELEC-4 in Revision 101 and Revision 4 did not present more than a minimal increase in the likelihood of occurrence of a malfunction of an SSC important to safety (e.g., Train A motor driven AFW pump). The finding was determined to be more than minor because the inspectors could not reasonably determine that the activity to defeat the automatic start function of the motor driven AFW pump by going to “pull out” would not have ultimately required NRC prior approval.

Because violations of 10 CFR 50.59 are considered to be violations that potentially impede or impact the regulatory process, they are dispositioned using the traditional enforcement process instead of the SDP. However, if possible, the underlying technical issue is evaluated under the SDP to determine the severity of the violation. In this case, the underlying technical issue affected the Mitigating Events Cornerstone because it affected the reliability of systems required to respond to events to prevent undesirable consequences. The inspectors determined that 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 “yes” to question 1 of the Mitigating Systems Cornerstone column of the Phase 1 worksheet. Specifically, the inspectors concluded that this was a design basis deficiency confirmed not to result in loss of operability. Based upon this Phase 1 screening, the inspectors concluded that the issue was of very low safety significance (Green).

Enforcement: 10 CFR 50.59(d)(1) states, in part, that the licensee shall maintain records of changes in the facility, of changes in procedures, and of tests and experiments as

described in the UFSAR. These records must include a written evaluation which provides a basis for the determination that the change, test, or experiment does not require a license amendment.

UFSAR Section 15.2.6 “Loss of Nonemergency AC Power to the Plant Auxiliaries” states “The auxiliary feedwater system is started automatically as follows: One motor driven and one diesel-driven auxiliary feedwater pump are started on any of the following: a.) Low-low level in any steam generator; b.) Any safety injection signal; c.) Loss of off-site power; and d.) Manual actuation.”

Contrary to the above, as of January 26, 2007, the licensee failed to provide an adequate basis for the determination that the change to the facility per procedure 1/2BwOA ELEC-4 “Loss of Off-site Power” Revision 4 and Revision 101 was acceptable without a license amendment. Specifically, the licensee failed to provide a basis as to why placing the motor driven A AFW pump in “pull out,” which defeated automatic pump start signals for low-low level in steam generators, safety injection signal and loss of off site power, did not present more than a minimal increase in the likelihood of occurrence of a malfunction of SSC important to safety (e.g., substitution of manual actions for automatic functions). In accordance with the Enforcement Policy, this violation of the requirements of 10 CFR 50.59 was classified as a Severity Level IV Violation because the underlying technical issue was of very low safety significance. Because this non-willful violation was non-repetitive, and was captured in licensee’s corrective action program (AR 00583152), it is considered a NCV consistent with VI.A.1 of the NRC Enforcement Policy. (NCV 05000456/2007002-02; 05000457/2007002-02 (DRS))

1R17 Permanent Plant Modifications (71111.17B)

.1 Review of Permanent Plant Modifications

a. Inspection Scope

From January 8 through 26, 2007, the inspectors reviewed eight permanent plant modifications that had been installed in the plant during the last two years. The modifications were selected based upon risk significance, safety significance, complexity, and one modification was chosen that affected the Barrier Integrity Cornerstone. 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 are included as an attachment to this report.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES (OA)

4OA2 Identification and Resolution of Problems

.1 Routine Review of Condition Reports

a. Inspection Scope

From January 8 through 26, 2007, the inspectors reviewed ten corrective action documents that identified or were related to 10 CFR 50.59 evaluations or 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 inspectors are listed in the attachment to this report.

b. Findings

No findings of significance were identified.

4OA6 Meetings

.1 Exit Meeting

The inspectors presented the inspection results to Mr. T. Couto and others of the licensee's staff, on January 26, 2007. 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. No proprietary information was identified.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee

T. Couto, Site Vice President
G. Boerschig, Plant Manager
D. Ambler, Regulatory Assurance Manager
M. Smith, Engineering Director
J. Panfil, Engineering
R. Belair, Engineering
D. Ibrahim, Engineering
D. Reddinger, Engineering
B. Wunder, Engineering
R. Wolen, Engineering

Nuclear Regulatory Commission

S. Ray, Senior Resident Inspector
D. Hills, EB1 Branch Chief

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened and Closed

05000456/2007002-01; 05000457/2007002-01	NCV	Inadequate Basis in 10 CFR 50.59 Evaluation for Installation of Sulfuric Acid System
05000456/2007002-02; 05000457/2007002-02	NCV	Lack of 50.59 Evaluation for Substitution of Manual Actions for Automatic Start Signals on the A Motor Driven AFW Pump

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

BWR- S-2004-14; Replace Diesel Generator Air Dryer 2DG01SA-C with New Model; Revision 0.
BWR-S-2004-17; EC 348144, 34870-74; Revision 0; DC Ground Detector Replacement; Revision 0
BWR-S-2004-22; EC 34634 (U1) and 346435 (U2); Revision 0; Revise CST Level Low Alarm Setpoint to Reflect Technical Specification Allowances; Revision 0
BWR-S-2004-145; EC 348271; Revision 0; Modify UCSR Area “D” and “J” VC Flow Paths and Install Balance Dampers; Cap Area Floor Drains (Control Room Habitability Project); Revision 0
BWR-S-2005-97; 0BwOA ELEC-1; Revision 1; Abnormal Grid Conditions; Revision 0
BWR-S-2005-101; BwOP SD-1; Revision 21; Start-Up of The Steam Generator Blowdown System; Revision 0
BWR-S-2005-162; 1/2BwOA ELEC-3; Revision 100/100; Loss of 4 KV ESF Bus - Unit 1 and 2; Revision 0
BWR-S-2006-175; TRM Change Package 06-016 for the Diesel Fuel Oil Testing Program; Revision 0.
BWR-S-2006-45; Draining the Spent Fuel Cask Chamber; Refueling Canal or Refuel Cavity Lower Areas below the 401 Elevation; Revision 0.
BWR-S-2006-68; BwOP DG-9; Revision 11; Chemical Addition to The Diesel Generator Jacket Water System; Revision 1
BWR-S-2006-89; Rod Bow Penalty UFSAR Update (DRP 11-066); Revision 0.
BWR-S-2006-93; UFSAR Change to Clarify SGTR Response Time Requirement; Revision 0.
BWR-S-2006-114; SG Foreign Object Evaluation; Revision 0.
BWR-S-2006-133; 1/2BwOA ELEC-7; Revision 0/0; Loss of Annunciators; Revision 0
BWR-S-2006-189; Fuel Repair for Braidwood Units 1 and 2; Revision 0.
BWR-S-2006-207; 1/2BwOA ELEC-1, Revision 101; Loss of Off-site Power; Revision 0
BWR-S-2006-231; Essential Service Water System Pump Discharge Flow Rate Adjustment; Revision 0.

10 CFR 50.59 Evaluations

BRW-SE-1977-583; EC 40940; Revision 8; Replace 2301 Woodward Governor System with new 2301A Woodward Governor System; dated August 1, 1997.
BWR-SE-1999-1196; 2DG01SA-C; Replace Diesel Generator Start Air Dryer w/new Model; Revision 0.

BWR-E-2005-70; Change In Core Decay Time for A2R11; Revision 0.
BWR-E-2005-81; Reactor Vessel Head Vents; Revision 0.
BWR-E-2005-165; Braidwood NAC-LWT Cask Operating Procedure; Revision 0.
BRW-E-2006-61; Letdown Booster Pump 1CV03P Post Maintenance Test; Revision 0.
BWR-E-2006-74; Change In Core Decay Time for A1R12; Revision 0.
BWR-E-2006-161; EC 349301, Replace Digital Electro-Hydraulic (DEH) Control System with Distributed Control System (DCS); Revision 0.
BWR-E-2006-192; Change in Core Decay Time for A2R12; Revision 0.
BWR-E-2006-196; EC 357102 and DRP 11-092; Revision 001; 0; Sulphuric Acid System Addition at The Lake Screen House; Revision 0.

Calculations

BYR200-007/BRW-00-0010-M; Spent Fuel Pool Temperature Analysis; Revision 0.
BWR-00-10-M/BYR2000-007; Byron/Braidwood Uprate Project - Spent Fuel Pool Temperature Analysis; Revision 1.

Commercial Grade Dedications (Applicability Determinations)

49996; Refurbish ESW Pump; dated April 18, 2006.
53306; 1" Swing Check Valve; dated October 22, 2006.

Procedures

BwOR NAC-414; Braidwood NAC-LWT Cask Operating Procedure; Revision 1.
SPP-05-008; Movement of Heavy Loads in the Fuel Building; Revision 0.
BwOR WEST-459; Fuel Repair for Braidwood Units 1 and 2; Revision 0.

Miscellaneous Documents

CN-COMED-195; Additional Evaluation of Rod Bow Penalty for Exelon Plant 3DFAC Analysis; Revision 0.

IR17 Permanent Plant Modifications (71111.17B)

Corrective Action Program Documents Generated As a Result of NRC Inspection

AR 00577792; WO Task 7, Testing Procedure Data Cannot be Found; dated January 11, 2007.
AR 00578124; Alarm Response Procedure did not Fully Incorporate Changes; dated January 10, 2007.
AR 00580952; Pzr Safety Valves- Discrepancy with Design Drawing; dated January 19, 2007.
AR 00581649; NRC- Mod/50.59 Inspection Screening Cover Sheet Error; dated January 22, 2007.
AR 00581884; BWR-S-2006-207 Listed Wrong Document Number on Coversheet; dated January 22, 2007.
AR 00581894; Screening BWR-S-2006-133 Identified the Wrong UFSAR Section; dated January 22, 2007.

AR 00583152; 50.59 Evaluation Should Have Been Performed; dated January 25, 2007.
AR 00583187; 50.59 Review for EC 359951; dated January 25, 2007.
AR 00583639; NRC Mod/50.59 Identified an Inadequate 50.59 Evaluation; dated January 25, 2007.

Corrective Action Program Documents Reviewed

AR 00344461; Need Evaluation of Tube Plug on 1DG01KB-X2 JW Cooler; dated June 15, 2006.
AR 00352672; UT Readings Reveal Pipe Thickness Below Screening Criteria; dated July 13, 2005.
AR 00372883; Configuration Issue-Design Change Installed but not Issued; dated September 13, 2005.
AR 00504208; CS Pumps - Design Calculation Discrepancy; dated June 27, 2006.
AR 00558142; Evaluation Needed for Bypassing EDG Air Dryers; dated November 15, 2006.
AR 513599; Error in Div 212 Battery Charger Sizing Calculation; dated July 26, 2006.
AR 362240; Auxiliary Building SBO Heat-Up Calculation Concern; dated August 11, 2005.
AR 445252; Turbine Overspeed Probability Calculation Discrepancy; dated January 24, 2006.
AR 366863; Calculation Deficiences - ATD 0021 SBO UHS Heat Load; dated August 25, 2005.
AR 508975; BwOA ELEC-4 Enhancement Needed to Minimize Reactor Overpwr; dated July 13, 2006

Calculations

050110; Piping Stress Analysis 1RH02; Revision 003N.
PSA-B-00-04; Byron/Braidwood Steam Generator Tube Rupture Analysis for Power Uprating; Revision 3.
BWR 96-015/ BYR96-238; Verification of Opening Capability for Braidwood and Byron (1)2 SI8802A&B Valves Susceptible to Pressure Locking; Revision 2.
Analysis No. 3C8-0685-002; Auxiliary Building Flood Level Calculations; Revision 13F.
BRW-97-0625-1; Condensate Storage Tank Level Error Analysis; Revision 0.

Drawings

M62; Diagram of Residual Heat Removal, Revision BO.
DS-C-56964; Nozzle Type Safety Valve 6-RV88-MSB; Revision G.
20E-0-4030VC63; Upper Cable Spreading Room CO2 Fire Protection Fire Damper Control; Revision J.
20E-2-4030DC05; 125V DC ESF Dist. Center Bus 211; Revision R.
M-61, Sheet 4; Diagram of Safety Injection Unit 1; Revision BB.
M-61, Sheet 7; Diagram of Chemical and Volume Control and Boron Thermal and Regeneration; Revision AV.
300-B50090; Outline of Motor Driven Auxiliary Feedwater Pump; Revision 4
300-B50090; Outline of Motor Driven Auxiliary Feedwater Pump; Revision 5
B-27347X; Diagram of Essential Service Water Pump; Revision A

Procedures

1BwOSR 5.5.8RH-4; Inservice Testing Requirements of RH Pressure Relief Check Valves 1RH8705A; 1RH8706A; and 1RH8705B; Revision 2.
1BwVSR 3.4.14.1; Reactor Coolant System Pressure Isolation Valve Leakage Surveillance; Revision 12.
BwAR 2-12-B2; Alarm 2-12-B2; Revision 10.
BwAR 2-12-D1; Alarm 2-12-D1; Revision 52.
BwAR 2-12-E1; Alarm 2-12-E1; Revision 52.
BwOS XCB-R1; U2 MCR Meter Color Banding; Revision 10.
BwVSR 3.4.10.1; Testing of Pressurizer Safety Valves; Revision 4.
BwOP DG-1; Diesel Generator Alignment to Standby Condition; Revision 26.
NWS-T-49; NWS Safety Valve Test Procedure for Exelon- Braidwood Nuclear Station Crosby Pressurizer Safety Valves; Revision 2.
BwVSR 3.7.10.4; Control Room Ventilation Pressurization Test; Revision 1E1.
1BwOA ELECT-4; Loss of Off-site Power Unit 1; Revision 101.
1BwOA ELEC-4; Loss of Off-site Power Unit 1; Revision 100.
BwOP AF-6; Motor Driven Auxiliary Feedwater Pump A Shutdown; Revision 13.
0BwOA ENV-6; Operation During Chlorine/Toxic Chemical Incident Unit 0; Revision 1.
0BwOA ENV-1; Adverse Weather Conditions Unit 0; Revision 104.

Miscellaneous Documents

NIS-2; Owners Report for Repairs or Replacements (2RY010A); dated June 21, 2005.
NIS-2; Owners Report for Repairs or Replacements (2RY010B); dated June 21, 2005.
NIS-2; Owners Report for Repairs or Replacements (2RY010C); dated June 21, 2005.
L-088; Installation Operation and Maintenance Crosby Style HB and HB-BP Self Actuated Nozzle-Type Safety Relief Valves; dated May 27, 1992.
WO 00712679; Bench Testing of Pressurizer Safety Valves; Completed August 5, 2005.
Data Acquisition System Calibration Sheet; CH 2 Pressure Indication; dated July 18, 2006.
Safety Valve Test Data; Valve S?N N56964-00-0032; dated November 23, 2004.
2702-220 Book 14, Westinghouse - Instruction Book Motor Operated Gate Valves/ Manually Operated/ Swing Check; Approved May 7, 1986.
WO 990090761; SI to Hot Leg 2B/2C Isol Vlv Assembly; dated October 24, 2000.
2BwOSR 5.5.8.SI-3B; Train B Unit 2 Safety Injection System SVAG Vlv Stroke Quarterly Surveillance; dated October 24, 2000.
2BwOSR 5.5.8.SI-3B; Train B Unit 2 Safety Injection System SVAG Vlv Stroke Quarterly Surveillance; dated October 29, 2006.
1BwOSR 5.5.8.SI-3B; Train B Unit 1 Safety Injection System SVAG Vlv Stroke Quarterly Surveillance; dated May 3, 2006.
WO 00720545-09; Operability Testing CRV Pressurization Test - EC 348271; dated January 5, 2007.
WO 00720545-08; Operability Testing (Temp Trending) for EC 438271; dated January 5, 2007.
2Bw0A PRI-1; Excessive Primary Plant Leakage Unit 2; Revision 102

2BwVSR 5.5.8.AF.1; Unit Two Motor Driven Auxiliary Feedwater Pump ASME Quarterly Surveillance; Revision 7; completed on September 29, 2006

2BwVSR 5.5.8.AF.1; Unit Two Motor Driven Auxiliary Feedwater Pump ASME Quarterly Surveillance; Revision 7; completed on July 6, 2006
2BwVSR 5.5.8.AF.1; Unit Two Motor Driven Auxiliary Feedwater Pump ASME Quarterly Surveillance; Revision 7; completed on April 3, 2006
2BwVSR 5.5.8.AF.1; Unit Two Motor Driven Auxiliary Feedwater Pump ASME Quarterly Surveillance; Revision 7; completed on January 5, 2006
WO 00720545-07; Perform Air Flow Testing - Modified Dampers - EC 348271; dated January 5, 2007.
WO 00662379-01; U1 Condensate Storage Tank Level Low Alarm Setpoint Change; dated April 24, 2005.
BRW-96-233; Nuclear Design Information Transmittal; Results of the 2A Auxiliary Feedwater Pump Auxiliary Lube Oil Pressure Interlock Time Test Under Cold Oil Conditions; dated December 5, 1996
CHRON No. 303095; SX Flow to Lube Oil Coolers at 32°F; dated October 10, 1994
DIT-BB-EXT-0484-02; External Design Information Transmittal; Evaluation of Pumps and Lubricating Systems for Auxiliary Feedwater, Safety Injection and Chemical & Volume Control services; dated June 9, 1994
Sargent & Lundy Project No. 9026-47; Evaluation of Pumps and Lubricating Systems for Auxiliary Feedwater, Safety Injection, and Charging Services at Braidwood Station – Units 1 & 2; dated November 24, 1992

Modifications

EC 0359951; Add Check Valve 1RH8706A to Line 1RH26AA; Revision 0.
EC 0347029; Revise the Setpoint for the Pressurizer Safety Valves From 2485 psig +/- 1 percent to 2460 +/- 2 percent; Revision 0.
EC 0042468; Upgrade of Motor Operators for Valves 1(2) SI 8802A/B; Revision 0.
EC 0042099; 2DG01SA-C, Replace Diesel Generator Start Air Dryer w/new Model; Revision 0.
EC 40940; Emergency Diesel Generator Governor Upgrade Modification; Revision 8.
EC 346434; Set Point Change to Unit 1 Condensate Storage Tank Level Low Alarm; Revision 0.
EC 348271; Modify Five Dampers OVC083Y, OVC084Y, OVC087Y, OVC088Y (Control Room Habitability Project); Revision 1.
EC 348470; Division 211 DC Bus Ground Detector Replacement; Revision 0.
2758-C; Pump Operation and Maintenance Manual; Pacific Pumps-Dresser; dated September 4, 1979

LIST OF ACRONYMS USED

AC	Alternating Current
ADAMS	Agency-Wide Document Access and Management System
AFW	Auxiliary Feedwater
AR	Assignment Report
CAP	Corrective Action Program
CFR	Code of Federal Regulations
HVAC	Heating Ventilation and Air Conditioning
IMC	Inspection Manual Chapter
NEI	Nuclear Energy Institute
NRC	Nuclear Regulatory Commission
PARS	Publicly Available Records
RG	Regulatory Guide
SE	Safety Evaluation
SDP	Significance Determination Process
SSC	Structure System or Component
TS	Technical Specifications
UFSAR	Updated Final Safety Analysis Report