UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION III
2443 WARRENVILLE ROAD, SUITE 210
LISLE, IL 60532-4352

October 28, 2011

Mr. Michael J. Pacilio
Senior Vice President, Exelon Generation Company, LLC
President and Chief Nuclear Officer (CNO), Exelon Nuclear
4300 Winfield Road
Warrenville, IL 60555

SUBJECT: BRAIDWOOD STATION, UNITS 1 AND 2
EVALUATION OF CHANGES, TESTS, OR EXPERIMENTS AND PERMANENT PLANT MODIFICATIONS BASELINE INSPECTION REPORT 05000456/2011008; 05000457/2011008 (DRS)

Dear Mr. Pacilio:

On September 30, 2011, the U.S. Nuclear Regulatory Commission (NRC) completed an Evaluation of Changes, Tests, or Experiments and Permanent Plant Modifications inspection at your Braidwood Station, Units 1 and 2. The enclosed inspection report documents the inspection results which were discussed on September 30, 2011, with Ms. A. Ferko and other members of your staff.

The inspection 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 this inspection, two NRC-identified findings of very low safety significance were identified. The findings involved a violation of NRC requirements. However, because of their very low safety significance, and because the issues were entered into your corrective action program, the NRC is treating these issues as Non-Cited Violations (NCVs) in accordance with Section 2.3.2 of the NRC Enforcement Policy.

If you contest the subject or severity of any 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 addition, if you disagree with the cross-cutting aspect assigned to any finding in this report, you should provide a response within 30 days of the date of this inspection report, with the basis for your disagreement, to the Regional Administrator, Region III, and the NRC Resident Inspector 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 System (PARS) component of NRC's Agencywide Documents Access and Management System (ADAMS), accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

/RA/

Robert C. Daley, Chief
Engineering Branch 3
Division of Reactor Safety

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

Enclosure: Inspection Report 05000456/2011008; 05000457/2011008;
w/Attachment: Supplemental Information

cc w/encl: Distribution via ListServ

U. S. NUCLEAR REGULATORY COMMISSION

REGION III

Docket No:	50-456; 50-457
License No:	NPF-72; NPF-77
Report No:	05000456/2011008; 05000457/2011008
Licensee:	Exelon Generation Company, LLC
Facility:	Braidwood Station, Units 1 and 2
Location:	Braceville, IL
Dates:	September 12 – 30, 2011
Inspectors:	J. Bozga, Reactor Inspector (Lead) J. Gilliam, Reactor Inspector M. Jones, Reactor Inspector
Approved by:	R. Daley, Chief Engineering Branch 3 Division of Reactor Safety

Enclosure

SUMMARY OF FINDINGS

IR 05000456/2011008, 05000457/2011008; 09/12/2011 – 09/30/2011; Braidwood Station Units 1 and 2; Evaluation of Changes, Tests, or Experiments and Permanent Plant Modifications.

This report covers a two-week announced baseline inspection on evaluation of changes, tests, or experiments and permanent plant modifications. The inspection was conducted by Region III based engineering inspectors. Two NRC-identified Green findings were identified by the inspectors. Both findings were considered as Non-Cited Violation (NCV) of NRC regulations. The significance of most findings is indicated by their color (Green, White, Yellow, Red) using Inspection Manual Chapter (IMC) 0609, "Significance Determination Process" (SDP). The cross-cutting aspects were determined using IMC 0310, "Components Within the Cross-Cutting Areas." 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 4, dated December 2006.

A. NRC-Identified and Self-Revealed Findings

Cornerstone: Mitigating Systems

- Green: The inspectors identified a finding of very low safety significance (Green) and associated Non-Cited Violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," for the failure to properly evaluate the structural steel embedment plate which supports Safety Injection (SI) pipe supports 1SI06025V and 1SI06030S. Specifically, the licensee failed to demonstrate compliance with the American Institute of Steel Construction (AISC) and Seismic Category I linear elastic requirements. The licensee entered this issue into their corrective action program and planned calculation revisions and modifications as needed to restore design margins.

  The finding was determined to be more than minor because the finding was associated with the Mitigating Systems cornerstone attribute of design control and affected the cornerstone objective of the availability, reliability, and capability of the SI piping and pipe supports. Specifically, the licensee used the actual material yield stress to ensure the structural steel embedment plate would maintain structural integrity when subjected to design loads. This is contrary to the AISC and Seismic Category I linear elastic requirements to use the specified minimum yield stress of the material. The inspectors determined that the finding was of very low safety significance because the finding did not result in loss of operability or functionality. The inspectors did not identify a cross-cutting aspect associated with this finding because it was associated with a calculation from the 1980s and was not reflective of current performance. (Section 1R17.2.b.(1))

- Green. The inspectors identified a finding of very low safety significance (Green) and associated Non-Cited Violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," for the failure to properly evaluate the Unit 1 SI subsystem 1SI06 and the Unit 1 Chemical Volume and Control System (CVCS) subsystem 1CV18 piping and pipe supports. Specifically, the licensee failed to demonstrate compliance with the AISC and the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code for the 1SI06 and 1CV18 piping and pipe supports. The licensee entered this issue into their corrective action program and planned calculation revisions and modifications as needed to restore design margins.

The finding was determined to be more than minor because the finding was associated with the Mitigating Systems cornerstone attribute of design control and affected the cornerstone objective of the availability, reliability, and capability of SI piping and pipe supports and CVCS piping and pipe supports. Specifically, the licensee did not perform an analysis to ensure compliance with AISC and ASME Section III requirements with the addition of permanent lead shielding to ensure the 1SI06 and 1CV18 piping and pipe supports would maintain structural integrity when subjected to design basis loads. The inspectors determined that the underlying finding was of very low safety significance because the finding did not result in loss of operability or functionality. The inspectors did not identify a cross-cutting aspect associated with this finding because this was a calculational deficiency that did not occur within the past three years and was not reflective of current performance. (Section 1R17.2.b.(2))

B. Licensee-Identified Violations

No violations of significance were identified.

REPORT DETAILS

1. REACTOR SAFETY

Cornerstone: Initiating Events, Mitigating Systems, and Barrier Integrity

1R17 Evaluation of Changes, Tests, or Experiments and Permanent Plant Modifications (71111.17)

.1 Evaluation of Changes, Tests, or Experiments

a. Inspection Scope

From September 12, 2011 through September 30, 2011, the inspectors reviewed six safety evaluations performed pursuant to 10 CFR 50.59 to determine if the evaluations were adequate and that prior NRC approval was obtained as appropriate. The inspectors also reviewed 15 screenings where licensee personnel had determined that a 10 CFR 50.59 evaluation was not necessary. The inspectors reviewed these documents to determine if:

- the changes, tests, or experiments performed were evaluated in accordance with 10 CFR 50.59 and that sufficient documentation existed to confirm that a license amendment was not required;
- the safety issue requiring the change, tests or experiment was resolved;
- the licensee conclusions for evaluations of changes, tests, or experiments were correct and consistent with 10 CFR 50.59; and
- the design and licensing basis documentation was updated to reflect the change.

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.”

This inspection constituted six samples of evaluations and 15 samples of changes as defined in IP 71111.17-04.

b. Findings

No findings of significance were identified

.2 Permanent Plant Modifications

a. Inspection Scope

From September 12, 2011 through September 30, 2011, the inspectors reviewed 11 permanent plant modifications that had been installed in the plant during the last three years. This review included in-plant walkdowns for portions of the following installed modifications: SI and CVCS piping systems; 2A Emergency Diesel Generator (EDG) Diagnostic/Performance Monitoring System; Unit 1 Service Water (SW) Strainer Backwash Cable Re-Route; EDG Air start system; EDG Pressure control valve setpoint modification; Unit 1 and Unit 2 SW strainers and associated Motor Operated Valve modifications. The modifications were selected based upon risk-significance, safety significance, and complexity. The inspectors reviewed the modifications selected to determine if:

- the supporting design and licensing basis documentation was updated;
- the changes were in accordance with the specified design requirements;
- the procedures and training plans affected by the modification have been adequately updated;
- the test documentation as required by the applicable test programs has been updated; and
- post-modification testing adequately verified system operability and/or functionality.

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

This inspection constituted 11 permanent plant modification samples as defined in IP 71111.17-04.

b. Findings

(1) Embedment Plate Design Deficiencies

Introduction: The inspectors identified a finding of very low safety significance (Green) and associated Non-Cited Violation of 10 CFR Part 50, Appendix B, Criterion III, “Design Control,” for the failure to properly evaluate the structural steel embedment plate which supports SI pipe supports 1SI06025V and 1SI06030S. Specifically, the licensee failed to demonstrate compliance with the AISC and Seismic Category I linear elastic requirements.

Description: The SI system is part of the Emergency Core Cooling System (ECCS). The Braidwood Updated Final Safety Analysis Report (UFSAR), Section 6.3.1, states the primary function of the ECCS is to remove the stored and fission product decay heat from the reactor during accident conditions. The ECCS also provides shutdown capability for design basis accidents by means of boron injection. The SI system is classified as a safety Category I system in UFSAR Section 3.2.

Piping Subsystem 1SI06 is part of the SI System and is a safety-related, ASME Class II, Seismic Category I subsystem located in the curved wall area of the Auxiliary Building. The structural steel embedment plate supports safety-related pipe supports 1SI06025V and 1SI06030S and is located in the Auxiliary Building, which is a Seismic Category I structure. The UFSAR Section 3.8.4.5.2 provides requirements for structural steel design inside the auxiliary building. Section 3.8.4.5.2 states, "The stresses and strains of structural steel are limited to those specified in the AISC Specification...." Also, this section requires that stresses are held within the elastic range and no plastic deformation is allowed.

The inspectors reviewed Calculation No. 13.2.29, "Structural Calculation for Mechanical Component Support 1SI06030S", Revision 4. The purpose of this calculation was to evaluate pipe support 1SI06025V and 1SI06030S structural elements for design and licensing basis requirements. The structural steel embedment plate evaluation was also contained in this calculation. The applied bending stress onto the embedment plate was greater than the allowable bending stress by 53 percent. The calculation used the following engineering judgment to justify compliance with their design and licensing basis requirements. The calculation used actual material yield stress of the embedment plate member and not specified material yield stress to calculate allowable bending stress. Also, the calculation used as an acceptance criteria, which allowed for the plastic or permanent deformation through yielding of the structural steel embedment plate and redistribution of stresses in the plate due to applied loads.

The inspectors determined that the engineering judgment used was not valid because the licensee used the actual material yield stress of material to determine the allowable bending stress as opposed to the requirement in the AISC for the allowable bending stress to use the specified minimum yield stress of the material. In addition, UFSAR Section 3.8.4.5.2 requires that no plastic or permanent deformation occur due to applied stresses. The inspectors also identified that structural steel embedment plate design loads were not correct and were non-conservative.

This issue was entered into the licensee's corrective action process as Action Request (AR) 1267356, "NRC Mod/50.59 Inspection-Pipe Support Calculations," dated September 23, 2011. The licensee performed an analysis that determined the embedment plate would not experience ultimate structure failure or collapse when subjected to the design loads and determined the plate was operable but nonconforming.

<u>Analysis</u>: The inspectors determined that the inadequately designed structural steel embedment plate was a performance deficiency because the structural steel embedment plate was not in conformance with AISC and Seismic Category I linear elastic requirements.

The finding was determined to be more than minor in accordance with IMC 0612 because the finding was associated with the Mitigating Systems cornerstone attribute of design control and affected the cornerstone objective of the availability, reliability, and capability of safety injection piping and pipe supports. Specifically, the licensee used the actual material yield stress to ensure the structural steel embedment plate would maintain structural integrity when subjected to design loads. This is contrary to the AISC and Seismic Category I linear elastic requirements to use the specified minimum yield stress of the material.

The inspectors determined the finding could be evaluated using the SDP in accordance with IMC 0609, "Significance Determination Process," Attachment 0609.04, "Phase I - Initial Screening and Characterization of findings," Table 4a for the Mitigating Systems cornerstone. The inspectors answered "yes" to Question 1 under the Mitigating Systems cornerstone column of IMC 0609, Attachment 4, Table 4a, Phase I worksheet. Specifically, the design deficiency was confirmed not to result in a loss of operability of the structural steel embedment plate. The inspectors agreed with the licensee's position that the structural steel embedment plate was operable because the licensee performed an analysis that determined the embedment plate would not experience ultimate structure failure or collapse when subjected to the design loads. Therefore, the inspectors concluded that the finding did not represent an actual loss of safety function, and the issue screened out as having very low safety significance (Green).

The inspectors did not identify a cross-cutting aspect associated with this finding because the calculation was from the 1980s and was not representative of current performance.

Enforcement: Title 10 CFR Part 50, Appendix B, Criterion III, "Design Control," requires, in part, that design control measures shall provide for verifying or checking the adequacy of design, such as by the performance of design reviews, by the use of alternate or simplified calculational methods, or by the performance of a suitable testing program.

Contrary to the above, on September 23, 2011, the licensee failed to demonstrate the design adequacy of the embedment plate. Specifically, the performance of design reviews for the structural steel embedment plate were inadequate, in that Calculation No. 13.1.29 did not demonstrate that the embedment plate would meet AISC and Seismic Category I linear elastic requirements as required by the Braidwood UFSAR Section 3.8.4.5.2.

Because this violation was of very low safety significance and it was entered into the licensee's corrective action program as AR 1267356, this violation is being treated as an NCV, consistent with Section 2.3.2 of the NRC Enforcement Policy. (NCV 05000456/2011008-01: Embedment Plate Design Deficiencies).

(2) Permanent Lead Shielding added to Safety Injection System and Chemical Volume and Control System Piping

Introduction: The inspectors identified a finding of very low safety significance (Green) and associated Non-Cited Violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," for the failure to properly evaluate the Unit 1 SI subsystem 1SI06 and the Unit 1 CVCS subsystem 1CV18 piping and pipe supports. Specifically, the licensee failed to demonstrate compliance with the AISC and the ASME Boiler and Pressure Vessel Code for the 1SI06 and 1CV18 piping and pipe supports.

Description: Braidwood UFSAR, Section 9.3.4, states the CVCS system provides safety-related seal water injection to the reactor coolant pumps and maintains required water inventory in the reactor coolant system. The CVCS system also provides control of reactor coolant water chemistry conditions, activity level, and chemical neutron absorber concentration and makeup. The CVCS system is classified as a safety Category I system in UFSAR Section 3.2.

The SI system is part of the ECCS. The Braidwood UFSAR, Section 6.3.1, states the primary function of the ECCS is to remove the stored and fission product decay heat from the reactor during accident conditions. The ECCS also provides shutdown capability for design basis accidents by means of boron injection. The ECCS is classified as a safety category I system in UFSAR Section 3.2.

The SI and CVCS piping were designed to the ASME Boiler and Pressure Vessel Code Section III and the SI and CV pipe supports were designed to the AISC code as required in UFSAR Section 3.9.3.

The inspectors reviewed Analysis No. 065613, “Stress Report for Chemical Volume and Control Piping Subsystem 1CV18,” Minor Revision 006M and Analysis No. 065643, “Piping Stress Report for Safety Injection/Residual Heat Removal Subsystem 1SI06/1RH06,” Minor Revision 004F. The calculation used NCIG 05, “Guidelines for Piping System Reconciliation,” Revision 1 to evaluate and accept a permanent addition of lead shielding to the 1SI06 and 1CV18 piping system.

The inspectors determined that use of NCIG-05 was not valid because the calculation did not demonstrate compliance with the AISC and ASME Section III requirements for piping and pipe supports with the addition of permanent lead shielding. Specifically, the licensee did not perform an analysis to demonstrate compliance with the AISC and ASME Section III requirements for piping and pipe supports with the addition of permanent lead shielding as required by Braidwood UFSAR Section 3.9.3.

This issue was entered into the licensee's corrective action process as AR 1269227, “NRC Mod/50.59 Inspection-Use of NCIG-05 for Lead Shielding,” dated September 28, 2011. The licensee performed an evaluation to demonstrate compliance with ASME Section III Appendix F operability criteria for piping and pipe supports and determined the 1SI06 and 1CV18 piping and pipe supports were operable but nonconforming.

Analysis: The inspectors determined that the inadequately designed piping and pipe supports was a performance deficiency because the piping and pipe supports were not in conformance with AISC and ASME Boiler and Pressure Vessel Code Section III requirements with the addition of permanent lead shielding.

The finding was determined to be more than minor in accordance with IMC 0612 because the finding was associated with the Mitigating Systems cornerstone attribute of design control and affected the cornerstone objective of the availability, reliability, and capability of safety injection piping and pipe support and chemical volume and control piping and pipe supports. Specifically, the licensee did not ensure compliance with AISC and ASME Boiler and Pressure Vessel Code Section III requirements to ensure the 1SI06 and 1CV18 piping and pipe supports would maintain structural integrity when subjected to design basis loads.

The inspectors determined the finding could be evaluated using the SDP in accordance with IMC 0609, “Significance Determination Process,” Attachment 0609.04, “Phase I -

Initial Screening and Characterization of findings,” Table 4a for the Mitigating Systems cornerstone. The inspectors answered “yes” to Question 1 under the Mitigating Systems cornerstone column of IMC 0609, Attachment 4, Table 4a, Phase I worksheet. Specifically, the design deficiency was confirmed not to result in a loss of operability of the 1SI06 and 1CV18 piping and pipe supports. The inspectors agreed with the licensee’s position that the 1SI06 and 1CV18 piping and pipe supports were operable because the licensee demonstrated compliance with ASME Section III Appendix F operability criteria for piping and pipe supports when subjected to the design loads. Therefore, the inspectors concluded that the finding did not represent an actual loss of safety function, and the issue screened out as having very low safety significance (Green).

The inspectors did not identify a cross-cutting aspect associated with this finding because the calculational deficiency did not occur with the last three years and was not representative of current performance.

Enforcement: Title 10 CFR Part 50, Appendix B, Criterion III, “Design Control,” requires, in part, that design control measures shall provide for verifying or checking the adequacy of design, such as by the performance of design reviews, by the use of alternate or simplified calculational methods, or by the performance of a suitable testing program.

Contrary to the above, on September 28, 2011, the licensee failed to demonstrate the design adequacy of 1SI06 and 1CV18 piping and pipe supports. Specifically, the performance of design reviews for 1SI06 and 1CV18 piping and pipe supports were inadequate, in that Analysis No. 065613, Minor Revision 006M and Analysis No. 065643, Minor Revision 004F did not demonstrate that the piping and pipe supports would meet ASME Section III and AISC requirements as required by the Braidwood UFSAR Section 3.9.3.

Because this violation was of very low safety significance and it was entered into the licensee’s corrective action program as AR 1269227, this violation is being treated as an NCV, consistent with Section 2.3.2 of the NRC Enforcement Policy. (NRC 05000456/2011008-02: Permanent Lead Shielding added to Safety Injection and Chemical Volume and Control System Piping).

4. OTHER ACTIVITIES (OA)

4OA2 Identification and Resolution of Problems

.1 Routine Review of Condition Reports

a. Inspection Scope

From September 12 through September 30, 2011, the inspectors reviewed corrective action process documents that identified or were related to 10 CFR 50.59 evaluations and permanent plant modifications. The inspectors reviewed these documents to evaluate the effectiveness of corrective actions related to permanent pant 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 Summary

On September 30, 2011, the inspectors presented the inspection results to Ms. A. Ferko and other members of the licensee staff. The licensee personnel acknowledged the inspection results presented and did not identify any proprietary content. The inspectors confirmed that all proprietary material reviewed during the inspection was returned to the licensee staff.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee

A. Ferko, Engineering Director
G. Dudek, Training Director
R. Radulovich, Nuclear Oversight Manager
P. Raush, Senior Design Engineering Manager
C. VanDenburg, Regulatory Assurance Manager
C. Mokijewski, Design Engineering
M. Grachowski, Regulatory Assurance

Nuclear Regulatory Commission

R. Daley, Chief, Engineering Branch 3, Division of Reactor Safety
J. Benjamin, Senior Resident Inspector
A. Garmoe, Reactor Inspector

LIST OF ITEMS OPENED, CLOSED AND DISCUSSED

Opened

05000456/2011008-01;	NCV	Embedment Plate Design Deficiencies (Section 1R17.2.b.(1))
05000456/2011008-02;	NCV	Permanent Lead Shielding added to Safety Injection and Chemical Volume and Control System Piping. (Section 1R17.2.b.(2))

Closed

05000456/2011008-01;	NCV	Embedment Plate Design Deficiencies. (Section 1R17.2.b.(1))
05000456/2011008-02;	NCV	Permanent Lead Shielding added to Safety Injection and Chemical Volume and Control System Piping. (Section 1R17.2.b.(2))

Discussed

None

LIST OF DOCUMENTS REVIEWED

The following is a list of documents reviewed during the inspection. Inclusion on this list does not imply that the NRC inspectors reviewed the documents in their entirety, but rather, that selected sections of portions of the documents were evaluated as part of the overall inspection effort. Inclusion of a document on this list does not imply NRC acceptance of the document or any part of it, unless this is stated in the body of the inspection report.

CALCULATIONS

Number	Description or Title	Date or Revision
48630	Piping Stress Report For Subsystem 1SI16	4
49258	Piping Calculation for Subsystem 1CV21, Chemical and Volume Control	001E
53790	Addendum to Piping Stress Report Chemical & Volume Control System 2CV53	000E
65613	Stress Report for Chemical and Volume Control Piping System 1CV1	006M
65643	Piping Stress Report for Safety Injection/Residual Heat Removal Subsystem 1SI06/1RH06	004F
13.2.2-BRW-08-0049-S	Qualification of Existing Pipe Support M-2AF08005G for Revised Loads	0
13.2.2-BRW-08-0051-S	Qualification of Existing Pipe Support M-2AF08014R for Revised Loads	0
13.2.2-BRW-08-0052-S	Qualification of Existing Modified Pipe Support M-2AF08023G for Revised Loads	0
13.1.2-BRW-08-0053-S	Qualification of Existing Pipe Support M-1AF08024G for Revised Loads	0
13.2.18BR-BRW-03-0015-S	Mechanical Component Supports 0IA74A2S184T and 0IA74A2S185T	0
13.2.29BR	Mechanical Component Support 1SI06030S	4
BRW-00-0010-M	Byron/Braidwood Uprate Project – Spent Fuel Pool Temperature Analysis	0
BRW-08-0074	Auxiliary Feedwater Cross Tie Hydraulic Analysis	0
BRW-98-0724-E	Motor Operated Valve (MOV) Actuator Terminal Voltage and Thermal Overload Sizing Calculation-SX System	0
BRW-96-089-M	Verification of Braidwood 125 VDC Battery 111, 112, 211 and 212 Ventilation Requirements and Hydrogen Concentration Evaluation Following a Loss of Battery Room	1
SITH-1	Refueling Water Storage Tank (RWST) Level Setpoints	7

CORRECTIVE ACTION PROGRAM DOCUMENTS REVIEWED

Number	Description or Title	Date or Revision
0562372	Byron CDBI Issue – CALC BRW-04-0005-M/BYR04-016	November 27, 2006
0813142	Abnormal System Response During 1SX01PA ASME	September 2, 2008
0882419	1B DG 30# Regulator Needs Adjusted – 1DG5228B	February 18, 2009
0908920	2A DG Recorder Reading High, Verify Calibration	April 18, 2009
0914792	Determination of Sx Strainer Manipulation on the Sx System	May 1, 2009
0960770	Overhaul for MOV 1SX150B and Diagnostically Test	Sept 2, 2009
0960766	Overhaul Actuator for MOV 1SX150A and Diagnostically Test	October 2, 2009
1022480	2SX150A SX Late Changes In the On-Line Work Scheduling – 2SX150A	February 27, 2010
1098702	EC 374543 Late Issuance Challenges Station Resources	August 6, 2010
1117907	U-2 RCS Arrow Range RTD Drifting	Sept 26, 2010
1128271	Sx MOD Cost Impacted by Missed Engineering Milestones	October 19, 2010
1132815	Wiring in MCC 133X1-A-D1 and D4 cannot be Restored	October 29, 2010
1171787	1A DG 40 No. Regulator Requires Replacement – 1DG5228A	February 7, 2011

CORRECTIVE ACTION PROGRAM DOCUMENTS GENERATED

Number	Description or Title	Date or Revision
1267356	NRC Mod/50.59 Inspection – Pipe Support Calculations	Sept 23, 2011
1268631	TRM Implementation not Timely	Sept 27, 2011
1269263	NRC Mod/50.59 Inspection – Concrete Compressive Strength	Sept 27, 2011
1269227	NRC Mod 50.59 Inspection – Use of NCIG-05 for Lead Shielding	Sept 28, 2011

DRAWINGS

Number	Description or Title	Date or Revision
1A-AF-34	Auxiliary Feedwater Large Bore Isometric	B
2A-AF-29	Auxiliary Feedwater Auxiliary Building	E
20E-1-4030DG32	Schematic Diagram Diesel Generator 1A Starting Sequence Control 1 DG01KA Part 2	AI
M-1SI06025V	Support No. M-1SI06025V	H
M-1SI06030S	Support No. M-1SI06030S	G
M-37	Diagram of Auxiliary Feedwater Unit 1	BG
M-122	Diagram of Auxiliary Feedwater Unit 2	BA
M-152	Manufacturer's Supplemental Diagram of Diesel Generator Control Diagram Shutdown System Units 1 and 2	L
S-497	Turbine Room Concrete Column Schedule	AC
S-1404	Refueling Water Storage Tank Section and Details	W

10 CFR 50.59 EVALUATIONS

Number	Description or Title	Date or Revision
BRW-E-2008-154	Zinc Injection into the CV Letdown Flow Path	Sept, 25, 2008
BRW-E-2008-168	Modify U2 RWST Level Transmitter Drain Line 2SI99F-1" to Eliminate Single Point of Vulnerability	October 8, 2010
BRW-E-2010-10	Zinc Injection into the Unit 1 CV Letdown Flow Path	May 14, 2010
BRW-E-2010-57	Rewire/Bypass Sensors 1 and 4 in Order to Restore Remaining 2B RVLIS Probe Sensors	June 28, 2010
BRW-E-2010-126	Change In-Core Decay Time for A1R15	October 2, 2010
BRW-E-2010-163	2C DT/TA Loop-Two RTD Operation for Cycle BR2C15	October 30, 2010

10 CFR 50.59 SCREENINGS

Number	Description or Title	Date or Revision
BRW-S-2008-150	ECs to Provide Temporary Reliable Power Feed to SX Strainers	Sept 4, 2008
BRW-S-2008-151	Revised Procedure for Installation of Safety-Related Power Supplies to Sx Strainers. ECs 372033, 372036, 372037, 372044	October 9, 2008
BRW-S-2009-4	Installation of Polymer Dispersant Injection System to Unit 2 Feedwater System	January 13, 2009
BRW-S-2009-15	Revised NPSH Values for the CV and SI	April 3, 2009

10 CFR 50.59 SCREENINGS

Number	Description or Title	Date or Revision
	Pumps	
BRW-S-2009-23	Replace Diesel Generator Starting Air Dryer 2DG01SB-D with New Model	February 13, 2009
BRW-S-2009-92	U-1 (U-2) Rod Drive Power Supply-Addition of Load Resisters	Sept 28, 2009
BRW-S-2009-124	2SI06-Modify Supports due to Re-Analysis per NRC finding	October 7, 2009
BRW-S-2009-135	Emergency Diesel Generator Governor Booster Modification	December 4, 2009
BRW-S-2010-164	Develop Calculation Supporting AF Diesel Fuel Storage Tech Spec Requirement and revise Tech Spec Basis	December 21, 2010
BRW-S-2010-172	Temporary Scaffold for WO No. 00612535 per AR 01130014-03	Nov 12, 2010
BRW-S-2010-177	Temporary Scaffold for WO #1236986	Nov 23, 2010
BRW-S-2011-35	Disable Alarms for Fire Detection Zone 2D-31 (2W MPT)	Sept 7, 2011
BRW-S-2011-54	Containment Fuel Transfer Area Permanent Supports and Removable Screen Doors	April 7, 2011
BRW-S-2011-75	Disable Iso Phase Bus Duct Temp Switch ITS-MP031	May 11, 2011
BRW-S-2011-84	Temporary Scaffold for WO No. 01275408 and WO No. 01274736	June 9, 2011

MODIFICATIONS

Number	Description or Title	Date or Revision
41716	Add Supports for Air Line to 2SX114B, Containment Chiller	2
360532	Change Setpoint from 30 PSIG to 40 PSIG for the 1DG5228A/B and 1DG5231A/B Pressure Control Valves	0
364360	U2 Permanent Lead Shielding – Exelon TSP's 01-043 to 045, 049-051, 053, 03-033, -035, 036, and 052. Systems CV, RH, SI	0
366445	U0 and U1 Permanent Lead Shielding – S and L TSP's 01-023, 040, 052, 02-037, 039, 03-009, 010, 057, 058, 06-026, 041, 060	0
369972	AFW Cross Tie	3
370521	2A Diesel Generator Diagnostic Performance Monitoring System	1

MODIFICATIONS

Number	Description or Title	Date or Revision
372033	Provide Temporary Reliable Power Feed to 1A SX Strainer	0
373770	Unit 1 SX Strainer Upgrade	3
374543	Unit 1 SX Strainer Backwash Cable Re-Route	4
376814	Revise Calc BRW-96-089 to Evaluate Battery Room Ventilation Requirements for the Existing 125 VDC Batteries	0
381707	Re-Analysis of Piping Subsystem 1SI16	0

OTHER DOCUMENTS

Number	Description or Title	Date or Revision
1Bw0A	Essential Service Water Malfunction Unit 1	Revision 102
ATI 813142	Bryzoa Deposition And Growth in the CW Forebays Resulted in Rapid Fouling Of SX Strainers, and Inoperability of the 1A and 2A SX Trains	December 10, 2008
BwMP	SX Strainer Manual Backwash Operation of Loss of Power	Revision 1
BwOP AF-3	Filling and Venting the Auxiliary Feedwater System	Revision 26
BwOP SX-6	Essential Service Water Strainer Manual Operation	Revision 8
CC-AA-112	Temporary Configuration Changes	Revision 17
ER-AA-300	Motor Operated Valve Program Administrative Procedure	Revision 6
FDRP 24-004	Revised Fire Loading in the Unit 1 Turbine Building Mezzanine Floor	October 12, 2008
LA-AA-107-1001	UFSAR Update T&RM	Revision 1
LS-AA-104	Exelon 50.59 Review Process	Revision 6
TRM 11-003	Revise Technical Requirements Manual (TRM) Table T3.8.b-1, “Thermal Overload Protection Device-Unit 1”, Adding Valves 1SX150A/B	June 9, 2011
TRP 1PI-DG8040A	Calibration of DG 1A Control Air Valve 1 DG 5231A Pressure Indicator	November 18, 2010
WO 01266623	Overhaul Actuator for MOV 2SX150A and Diagnostically Test	February 17, 2011

LIST OF ACRONYMS USED

ADAMS	Agencywide Documents Access and Management System
AISC	American Institute of Steel Construction
AR	Action Request
ASME	American Society of Mechanical Engineers
CFR	Code of Federal Regulations
CNO	Chief Nuclear Officer
CVCS	Chemical Volume and Control System
DRS	Division of Reactor Safety
EC	Engineering Change
ECCS	Emergency Core Cooling System
EDG	Emergency Diesel Generator
IMC	Inspection Manual Chapter
NCV	Non-Cited Violation
NRC	U.S. Nuclear Regulatory Commission
PARS	Public Available Records System
SDP	Significance Determination Process
SI	Safety Injection
SW	Service Water
UFSAR	Updated Final Safety Analysis Report

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Sincerely,

/RA/

Robert C. Daley, Chief
Engineering Branch 3
Division of Reactor Safety

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

Enclosure: Inspection Report 05000456/2011008; 05000457/2011008
w/Attachment: Supplemental Information

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Letter to Mr. Michael J. Pacilio from Mr. Robert C. Daley dated October 28, 2011.

SUBJECT: BRAIDWOOD STATION, UNITS 1 AND 2
EVALUATION OF CHANGES, TESTS, OR EXPERIMENTS AND PERMANENT PLANT MODIFICATIONS BASELINE INSPECTION REPORT
05000456/2011008; 05000457/2011008

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