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

November 14, 2013

EA-13-209

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: NOTICE OF VIOLATION AND BRAIDWOOD STATION, UNITS 1 AND 2, NRC BASELINE INSERVICE INSPECTION REPORT 05000456/2013008; 05000457/2013008

Dear Mr. Pacilio:

On October 29, 2013, the U.S. Nuclear Regulatory Commission (NRC) completed a baseline Inservice Inspection at your Braidwood Station, Units 1 and 2. The NRC inspectors discussed the results of this inspection with Mr. Bashor and other members of your staff. Inspectors documented the results of this inspection in the enclosed inspection report.

During this inspection, the NRC staff examined activities conducted under your license as they relate to public health and safety to confirm compliance with the Commission's rules and regulations, and with the conditions of your license. Within these areas, the inspection consisted of selected examination of procedures and representative records, observations of activities, and interviews with personnel.

Based on the results of this inspection, the NRC has determined that a Severity Level IV violation of NRC requirements occurred involving incomplete and inaccurate information in a license amendment. The violation was evaluated in accordance with the NRC Enforcement Policy. The current Enforcement Policy is included on the NRC's web site at http://www.nrc.gov/about-nrc/regulatory/enforcement/enforce-pol.html. The NRC takes the issue of complete and accurate license submittals very seriously. For this reason, the NRC considered citing this as a Severity Level III violation, as discussed in the Enforcement Policy, as the NRC had approved a licensing action based on the incorrect information. However, after consideration by NRC management, and with the approval of the Director of the Office of Enforcement, it was determined that a Severity Level IV Cited Violation was appropriate. This decision was based on the very low safety significance (Green) of the associated finding. Therefore, this violation of Title 10 of the *Code of Federal Regulations* (10 CFR) 50.9(a) "Completeness and Accuracy of Information" is cited in the enclosed Notice of Violation (Notice) and the circumstances surrounding this violation are described in detail in the enclosed report.

You are required to respond to this letter and should follow the instructions specified in the enclosed Notice when preparing your response. If you have additional information that you believe the NRC should consider, you may provide it in your response to the Notice. The NRC review of your response to the Notice will also determine whether further enforcement action is necessary to ensure compliance with regulatory requirements.

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter, its enclosure, and your response 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 Document Access and Management System (ADAMS). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room). To the extent possible, your response should not include any personal privacy or proprietary information so that it can be made available to the Public without redaction.

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

Enclosures:
1. Notice of Violation, EA-2013-209
2. Inspection Report 05000456/2013008 and 05000457/2013008
   Attachment: Supplemental Information

cc w/encl: Distribution via ListServ™

NOTICE OF VIOLATION

Exelon Generation Company, LLC
Braidwood Station, Unit 2

Docket No. 50-457
License No. NPF-77
EA-13-209

During a U.S. Nuclear Regulatory Commission (NRC) inspection conducted from September 9, 2013, to October 29, 2013, a violation of NRC requirements was identified. In accordance with the NRC Enforcement Policy, the violation is listed below:

> Title 10 of the *Code of Federal Regulations* (10 CFR) 50.9(a), “Completeness and Accuracy of Information,” requires that “Information provided to the Commission by an applicant for a license or by a licensee or information required by statute, or by the Commission's regulations, orders, or license conditions to be maintained by the applicant or the licensee shall be complete and accurate in all material respects.”
>
> In Letter No. RS-05-103, “License Amendment Request Regarding Reactor Coolant System Pressure, and Temperature Limits Report and Request for Exemption from 10 CFR 50.60, Acceptance Criteria for Fracture Prevention Measures for Lightwater Nuclear Power Reactors for Normal Operation,” Attachment 4, “Justification for Exemption from 10 CFR 50.60,” the licensee [Exelon Generation Company, LLC] stated “WCAP-16143 provides a valid basis for changing the RPV [Reactor Pressure Vessel] closure head flange limit and maintains the relative margin of safety commensurate with that which existed at the time the 10 CFR [Part] 50, Appendix G requirement was issued.”
>
> Contrary to the above, on October 3, 2005, in Letter No. RS-05-103, the licensee [Exelon Generation Company, LLC] failed to provide information to the Commission that was complete and accurate in all material respects, in that, WCAP-16143, “Reactor Vessel Closure Had/Vessel Flange Requirements Evaluation for Byron/Braidwood Units 1 and 2,” did not provide a valid basis for changing the RPV closure head flange limit for Braidwood Unit 2. Specifically, WCAP-16143, Section 4, “Flange Integrity,” demonstrated adequate vessel margins based upon the original closure head flange configuration and did not represent the modified closure head configuration (53 head studs applicable to the Unit 2 reactor vessel). Operation of the Braidwood Unit 2 vessel with 53 closure head studs was not within the bounds and limitations of what the NRC had reviewed in Letter No. RS-05-103 and found to be an acceptable basis to grant the exemption request. Therefore, this information was considered material to the NRC.

This is a Severity Level IV Violation (Section 6.9).

Pursuant to the provisions of 10 CFR 2.201, Exelon Generation Company, LLC is hereby required to submit a written statement or explanation to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555-0001 with a copy to the Regional Administrator, Region III, and a copy to the NRC Resident Inspector at the Braidwood facility, within 30 days of the date of the letter transmitting this Notice of Violation (Notice). This reply should be clearly marked as a “Reply to a Notice of Violation; EA-13-209” and should include for the violation: (1) the reason for the violation, or, if contested, the basis for disputing the violation, (2) the corrective steps that have been taken and the results achieved, (3) the corrective steps that will be taken, and (4) the date when full compliance will be achieved. Your response may reference or include previous docketed correspondence, if the correspondence adequately addresses the required response. If an adequate reply is not received within the time specified in this Notice, an Order or a Demand for Information may be issued as to why the license should not be modified, suspended, or revoked, or why such other action as may be proper should not be taken. Where good cause is shown, consideration will be given to extending the response time.

If you contest this enforcement action, you should also provide a copy of your response, with the basis for your denial, to the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington, DC 20555-0001.

Because your response will be made available electronically for public inspection in the NRC Public Document Room or from the NRC's Agencywide Documents Access and Management System (ADAMS), accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html, to the extent possible, it should not include any personal privacy, or proprietary information so that it can be made available to the public without redaction. If personal privacy or proprietary information is necessary to provide an acceptable response, then please provide a bracketed copy of your response that identifies the information that should be protected and a redacted copy of your response that deletes such information. If you request withholding of such material, you must specifically identify the portions of your response that you seek to have withheld and provide in detail the bases for your claim of withholding (e.g., explain why the disclosure of information will create an unwarranted invasion of personal privacy or provide the information required by 10 CFR 2.390(b) to support a request for withholding confidential commercial or financial information).

In accordance with 10 CFR 19.11, you may be required to post this Notice within 2 working days of receipt.

Dated this 14 day of November 2013.

U.S. NUCLEAR REGULATORY COMMISSION

REGION III

Docket Nos:	50-456, 50-457
License Nos:	NPF-72, NPF-77
Report No:	05000456/2013008; 05000457/2013008
Licensee:	Exelon Generation Company, LLC
Facility:	Braidwood Station, Units 1 and 2
Location:	Braceville, IL
Dates:	September 9, 2013 through October 29, 2013
Inspectors:	M. Holmberg, Reactor Inspector V. Meghani, Reactor Inspector A. Garmoe, Acting Senior Resident Inspector B. Metrow, Illinois Emergency Management Agency
Approved by:	D. E. Hills, Chief Engineering Branch 1 Division of Reactor Safety

SUMMARY OF FINDINGS

IR 05000456/2013008; 050000457/2013008; 09/09/2013 – 10/29/2013; Braidwood Station, Inservice Inspection.

This report covers an announced baseline inspection by regional inspectors. A finding of very low safety significance and associated Severity Level IV Violation of an NRC requirement was identified. The significance of inspection findings is indicated by their color (i.e., greater than Green, or Green, White, Yellow, Red) and determined using Inspection Manual Chapter (IMC) 0609, "Significance Determination Process," dated June 2, 2011. The cross-cutting aspect is determined using IMC 0310, "Components Within the Cross-Cutting Areas," dated October 28, 2011. All violations of NRC requirements are dispositioned in accordance with the NRC's Enforcement Policy dated January 28, 2013. 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.

SUMMARY OF FINDINGS

A. NRC-Identified and Self-Revealed Findings

Cornerstone: Barrier Integrity

- Severity Level IV Violation. The inspectors identified a finding of very low safety significance (Green) and an associated Severity Level IV Violation of 10 CFR 50.9 "Completeness and Accuracy of Information," for the licensee's failure to provide information to the NRC that was complete and accurate in all material respects. Specifically, in Letter RS-05-103 "License Amendment Request Regarding Reactor Coolant System Pressure and Temperature Limits Report and Request for Exemption from 10 CFR 50.60," the licensee stated that WCAP-16143 provides a valid basis for changing the reactor pressure vessel (RPV) closure head flange limit and maintains the relative margin of safety commensurate with that which existed at the time the 10 CFR Part 50, Appendix G requirement was issued. However, the analysis documented in WCAP-16143 demonstrated adequate vessel margins based upon the original closure head flange configuration and did not represent the modified closure head configuration (53 head studs) applicable to the Unit 2 reactor vessel. Therefore, this analysis did not provide a valid basis for changing the Unit 2 RPV closure head flange limits in 10 CFR Part 50, Appendix G. The licensee entered this issue into the Corrective Action Program (AR 01558067), performed an operability evaluation, and was evaluating several options for corrective measures. The corrective actions under consideration by the licensee included: completing a calculation to validate the Westinghouse Electric vendor letter, revision to WCAP-16143, installation of a 54th head stud, submittal of a license amendment request with a revised WCAP-16143, or negate the existing exemption methodology and return to the pressure temperature limit curves based upon 10 CFR Part 50, Appendix G requirements.

  The inspectors determined that this issue was more than minor because it adversely affected the Barrier Integrity Cornerstone attribute of Design Control. The inspectors also answered "yes" to the More-than-Minor screening question, "If left uncorrected, would the performance deficiency have the potential to lead to a more significant safety concern"? Specifically, the inspectors determined that this issue was more than minor because, if left uncorrected, the failure to provide complete and accurate information for the Unit 2 vessel head stud configuration could have resulted in non-conservative pressure temperature limit curves that allowed operation in an unacceptable region that would increase the possibility of vessel failure during a pressurized thermal shock event. The inspectors performed a Phase I SDP screening using IMC 0609, Attachment 0609 Appendix A, Exhibit 3-Barrier Integrity Screening Questions, and selected the box under the Reactor Coolant System Boundary (e.g., pressurized thermal shock issues), which required a detailed risk-evaluation. A Region III Senior Reactor Analyst performed a detailed risk-evaluation of this finding. A potential increase in the probability for vessel failure would exist if the plant was operated in the unacceptable pressure temperature regions and a pressurized thermal shock event occurred. Based on the licensee and supporting vendor assessments which concluded that no substantial increase in vessel stresses will occur due to operation with 53 head studs, the driving force for crack propagation (e.g., K1) will remain essentially unchanged. However, to bound the delta risk-evaluation, it was assumed that the initiating event frequency for a reactor vessel failure increased by 10 percent. From the Braidwood Standardized Plant Analysis Risk Model Version 8.21, the initiating event frequency for reactor vessel failure from any cause was 1E-7/yr. Core damage is expected to occur if reactor vessel failure occurs. The exposure time for the finding was the maximum of one year. Thus, a bounding risk-assessment yields a delta risk of 1E-8/yr. Therefore, based on the detailed risk-evaluation, this finding is of very low risk significance (Green). Because the failure to provide complete and accurate information to the NRC had the potential to impede or impact the regulatory process, the finding was also evaluated in accordance with NRC Enforcement Policy for traditional enforcement. This violation was similar to an example of a Severity Level III violation identified in Section 6.9.c.1 of the NRC Enforcement Policy. However, after consideration by NRC management, and with the approval of the Director of the Office of Enforcement, it was determined that a Severity Level IV Cited Violation was appropriate. This decision was based upon the very low safety significance (Green) of the associated finding. The inspectors concluded that no cross-cutting aspect was applicable as the performance deficiency was not reflective of current performance because the issue was in excess of three years old. (Section 1R08.5)

B. Licensee-Identified Violation

None.

REPORT DETAILS

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, and Barrier Integrity

1R08 Inservice Inspection (ISI) Activities (71111.08P)

From September 9 through October 29, 2013, the inspectors conducted a review of the implementation of the licensee's Inservice Inspection (ISI) Program for monitoring degradation of the Unit 1 reactor coolant system (RCS), steam generator tubes, emergency feedwater systems, risk significant piping, and components and containment systems.

The inspections described in Sections 1R08.1, 1R08.2, R08.3, IR08.4, and 1R08.5 below constituted one inservice inspection sample as defined in Inspection Procedure (IP) 71111.08-05.

.1 Piping Systems Inservice Inspection

a. Inspection Scope

The inspectors observed and/or reviewed records of the following non-destructive examinations (NDE) mandated by the American Society of Mechanical Engineers (ASME) Section XI Code to evaluate compliance with the ASME Code Section XI and Section V requirements, and if any indications and defects were detected, to determine whether these were dispositioned in accordance with the ASME Code or an NRC-approved alternative requirement:

- Ultrasonic examination (UT) of reactor vessel head studs;
- UT of a risk-informed (R-A , R01.11 and R01.18), Pipe-to-Elbow, Weld, 1FW-01-18;
- UT of a risk-informed (R-A , R01.11 and R01.18), Pipe-to-Elbow, Weld, 1FW-01-17;
- UT of a risk-informed (R-A , R01.11 and R01.18), Pipe-to-Penetration, Weld, 1FW-01-05;
- UT of a risk-informed (R-A , R01.11 and R01.18), Pipe-to-Valve, Weld, 1FW-01-04;
- Liquid dye penetrant (PT) examination of Pipe Lug Welds, 1SI-24-SW12, 18, 19 and 20; and
- PT of vessel head Penetration Nozzle 69 J-groove weld overlay repair.

The inspectors reviewed the following examination records with recordable indications accepted for continued service to determine whether acceptance was in accordance with the ASME Code Section XI or an NRC-approved alternative.

- Unit 1 - Subsurface indications identified in weld ISI-39-04 (line 1SI08B-4”).

The inspectors observed and/or reviewed records for the following pressure boundary welds completed for risk significant systems during the outage to determine if the licensee applied the preservice non-destructive examinations and acceptance criteria required by the construction Code, and/or the NRC approved Code relief request. Additionally, the inspectors reviewed the welding procedure specification and supporting weld procedure qualification records to determine whether the weld procedures were qualified in accordance with the requirements of the Construction Code and the ASME Code Section IX.

- Weld overlay piping weld No. 1 on line 1SX27DA-10-Inch (EC-394727).
- Elbow-to-pipe weld FW-7C, on replacement segment in line 1SX27DA-10-inch fabricated under work order (WO) No.01661641-01.

b. Findings

No findings were identified.

.2 Reactor Pressure Vessel Upper Head (RPVUH) Penetration Inspection Activities

a. Inspection Scope

For the Unit 1 RPVUH, a bare metal visual (BMV) examination and a non-visual examination were required this outage pursuant to 10 CFR 50.55a(g)(6)(ii)(D).

The inspectors observed the BMV examination conducted on the Unit 1 RPVUH at penetration Nozzles 25, 26, 44, 56 and 61 to determine if the activities were conducted in accordance with the requirements of ASME Code Case (CC) N-729-1 and 10 CFR 50.55a(g)(6)(ii)(D). Specifically, to determine:

- If the required visual examination scope/coverage was achieved and limitations (if applicable) were recorded, in accordance with the licensee procedures;
- If the licensee criteria for visual examination quality and instructions for resolving interference and masking issues were adequate; and
- For indications of potential through-wall leakage, that the licensee entered the condition into the Corrective Action Program and implemented appropriate corrective actions.

The inspectors observed and reviewed data for non-visual examinations conducted on the reactor vessel head penetrations No’s 51, 66, and 69 to determine whether the activities were conducted in accordance with the requirements of ASME CC N-729-1 and 10 CFR 50.55a(g)(6)(ii)(D). Specifically, to determine:

- If the required examination scope (volumetric and surface coverage) was achieved and limitations (if applicable) were recorded, in accordance with the licensee procedures;
- If the UT examination equipment and procedures used were demonstrated by blind demonstration testing;

- For indications or defects identified, that the licensee documented the conditions in examination reports and/or entered this condition into the Corrective Action Program and implemented appropriate corrective actions; and
- For the head penetration Nozzle No. 69 and the associated repair weld overlay indications accepted for continued service, that the licensee evaluation and acceptance criteria were in accordance with the ASME Section III Code, 10 CFR 50.55a(g)(6)(ii)(D) and/or the NRC-approved alternative (relief request 13R-09).

The inspectors reviewed records of the welded repairs to head penetration No. 69 J-groove weld overlay completed during the current outage to determine whether the licensee applied the preservice examinations and acceptance criteria required by the construction Code and NRC approved relief request 13R-09. Additionally, the inspectors reviewed the welding procedure specification and supporting procedure qualification records to determine whether the weld procedure used was qualified in accordance with the Construction Code and the ASME Code Section IX requirements.

b. Findings

No findings were identified.

.3 Boric Acid Corrosion Control (BACC)

a. Inspection Scope

On September 9, 2013, the inspectors observed the licensee staff performing visual examinations of the reactor coolant system within containment to determine whether these examinations focused on locations where boric acid (BA) leaks can cause degradation of safety significant components and to determine whether components with boric acid leaks were properly identified in the Corrective Action Program.

The inspectors reviewed the following licensee evaluations of RCS and connected system components with BA leaks/deposits to determine whether the licensee properly applied corrosion rates and assessed the effects of corrosion on structural or pressure boundary integrity.

- Boric acid evaluation for AR 01361504, Reactor Vessel Head Penetration No. 75; and
- Boric acid evaluation for AR 01536176, RH Letdown Booster Pump Upstream Isolation Valve.

The inspectors reviewed the following corrective actions related to evidence of BA leakage to determine if the corrective actions completed were consistent with the requirements of the ASME Code Section XI and 10 CFR Part 50, Appendix B, Criterion XVI:

- AR 01354783; 1SI 121 Relief Valve (1SI8811A) Leak At Threaded Connection;
- AR 01354192; Dry Boric Acid at Body to Bonnet Bolt on 1RC8037A Reactor Coolant Loop 1A Drain Valve; and

- AR 01354181; Dry Boric Acid at Body to Bonnet Bolt on 1RC8037B – Reactor Coolant Loop 1B Drain Valve.

b. Findings

No findings were identified.

.4 Steam Generator Tube Inspection Activities

a. Inspection Scope

No steam generator examinations were required this refueling outage pursuant to Technical Specification requirements in Section 3.4.19 “Steam Generator Tube Integrity,” and Section 5.5.9 “Steam Generator (SG) Program.” Because the licensee did not conduct SG tube examinations, no NRC review was completed for this inspection attribute.

b. Findings

No findings were identified.

.5 Identification and Resolution of Problems

a. Inspection Scope

The inspectors performed a review of ISI-related problems entered into the licensee’s Corrective Action Program and conducted interviews with licensee staff to determine if:

- the licensee had established an appropriate threshold for identifying ISI-related problems;
- the licensee had performed a root cause (if applicable) and taken appropriate corrective actions; and
- the licensee had evaluated operating experience and industry generic issues related to ISI and pressure boundary integrity.

The inspectors performed these reviews to evaluate compliance with 10 CFR Part 50, Appendix B, Criterion XVI, “Corrective Action,” requirements. The corrective action documents reviewed by the inspectors are listed in the Attachment to this report and this review included AR 1549725 “Pressure Temperature Limit Report (PTLR) Analysis not Revised for Reactor Head Stud Configuration.”

In August of 2013 the licensee documented in AR 1549725 a discrepancy between the Braidwood Unit 2 reactor vessel head stud configuration (operating with 53 head studs) and the stress analysis used to develop the vessel pressure-temperature limit (PTL) curves established in the PTLR. This discrepancy was first identified by the NRC during a license renewal audit, conducted at the Byron Station and as being applicable to both Byron and Braidwood. The inspectors reviewed the Braidwood licensee’s resolution for this issue, which was only applicable to the Unit 2 reactor vessel.

b. Findings

Inaccurate/Incomplete Information for Exemption Request from 10 CFR 50.60, Acceptance Criteria for Fracture Prevention Measures for Lightwater Nuclear Power Reactors for Normal Operation

Introduction: A finding of very low safety significance (Green) and associated Severity Level (SL) IV Violation of 10 CFR 50.9 "Completeness and Accuracy of Information," were identified by the inspectors for the licensee's failure to provide information to the NRC that was complete and accurate in all material respects. Specifically, in Letter RS-05-103 "License Amendment Request Regarding Reactor Coolant System Pressure and Temperature Limits Report and Request for Exemption from 10 CFR 50.60," the licensee stated that WCAP-16143 provides a valid basis for changing the reactor pressure vessel (RPV) closure head flange limit and maintains the relative margin of safety commensurate with that which existed at the time the 10 CFR Part 50, Appendix G requirement was issued. However, the analysis documented in WCAP-16143 demonstrated adequate vessel margins based upon the original closure head flange configuration and did not represent the modified closure head configuration (53 head studs) applicable to the Unit 2 reactor vessel. Therefore, the analysis did not provide a valid basis for changing the Unit 2 RPV closure head flange limits in 10 CFR Part 50, Appendix G.

Description: On September 11, 2013, the inspectors identified that the licensee had provided incomplete/inaccurate information in Letter RS-05-103 because the vendor analysis (WCAP-16143 "Reactor Vessel Closure Head/Vessel Flange Requirements Evaluation for Byron/Braidwood Units 1 and 2") that supported this exemption request letter was based upon the original vessel closure head configuration instead of the modified closure head configuration (53 head studs) applicable to Unit 2. The inspectors were concerned that without an accurate stress evaluation applicable to the Unit 2 vessel, non-conservative PTL curves for heatup, and cooldown may have been approved by the NRC. Operation of the Unit 2 vessel in an unacceptable region (e.g., above the correct PTL curves for heatup and cooldown) could increase the possibility of vessel failure during a pressurized thermal shock event.

In 1991 the reactor vessel head stud No. 35 became stuck in the Unit 2 vessel flange. In 1994 the licensee cut-off the portion of this stud sticking out of the flange leaving a section of stud inside the stud hole. In 2002, the licensee removed the remnant portion of the stuck stud, and machined the inside surface of the flange to remove the existing damaged flange threads. The licensee originally intended to install a helical-coil threaded insert and reinstall a vessel head stud. However, the vendor equipment to machine the flange in support of the helical-coil insert installation failed and the licensee abandoned the repair effort. The licensee then permanently accepted the 53 flange bolt configuration of the vessel head (reference EC 334720) based upon analysis that demonstrated the vessel was acceptable for operation. The analysis credited by the licensee to operate with 53 head studs was not submitted to the NRC for review and approval.

On October 3, 2005, the licensee submitted Letter RS-05-103 "License Amendment Request Regarding Reactor Coolant System Pressure and Temperature Limits Report and Request for Exemption from 10 CFR 50.60 Acceptance Criteria for Fracture Prevention Measures for Lightwater Nuclear Power Reactors for Normal Operation." In this letter, the licensee requested, in accordance with 10 CFR 50.12, NRC approval for an exemption to the requirements in 10 CFR 50.60(a) and 10 CFR Part 50, Appendix G "Fracture Toughness Requirements." Specifically, the licensee requested to apply WCAP-16143, "Reactor Vessel Closure Head/Vessel Flange Requirements Evaluation for Byron/Braidwood Units 1 and 2," to calculate the RPV PTL for Braidwood Station, Units 1 and 2, in lieu of the 10 CFR Part 50, Appendix G, Paragraph IV.A.2.c requirements mandated by 10 CFR 50.60(a). In Attachment 4 "Justification for Exemption from 10 CFR 50.60" of RS-05-103, the licensee stated "WCAP-16143 provides a valid basis for changing the RPV closure head flange limit and maintains the relative margin of safety commensurate with that which existed at the time the 10 CFR Part 50, Appendix G requirement was issued." On November 22, 2006, the NRC issued a safety evaluation for this topic and on November 27, 2006, the NRC issued licensee Amendment No. 142 to the Braidwood Unit 2 operating license authorizing this exemption request and associated changes to the Braidwood operating license. The NRC decision to approve license Amendment No.142 was based in part upon the stress analysis identified in Section 4 "Flange Integrity" and in Appendix C of WCAP-16143, which demonstrated adequate vessel margins based upon the original closure head configuration for the Byron and Braidwood Units. Operation of Braidwood Unit 2 with a modified closure head configuration (53 head studs), would not be within the bounds and limitations of what the NRC had reviewed and found acceptable. Therefore, operation of Braidwood Unit 2 with 53 head studs was considered "material" to the NRC's review because the WCAP-16143 analysis did not provide a valid basis for exempting the Unit 2 RPV closure head flange from the PTL identified in 10 CFR Part 50, Appendix G.

The licensee performed operability evaluation No. 13-005 and concluded that the vessel was operable because the stress components at the governing locations were expected to remain essentially unchanged as a result of the bolt-up and reactor operation of Unit 2 with 53 head studs. The licensee based their operability evaluation upon a Westinghouse Electric (WE) letter dated August 27, 2013, in which the WE vendor concluded the vessel stress components evaluated near the head flange in WCAP-16143 would remain essentially unchanged and that the PTL for the vessel established in the current PTLR remain valid. The licensee entered this issue into the Corrective Action Program as AR 01558067 and was evaluating several options for corrective measures. The corrective actions under consideration by the licensee included: completion of a calculation to validate the WE vendor letter, revision to WCAP-16143, installation of a 54th head stud, submittal of a license amendment request with a revised WCAP-16143, or negate the existing exemption methodology and return to the PTL curves based upon 10 CFR Part 50, Appendix G requirements.

Analysis: The inspectors concluded that the licensee had reasonable opportunity to foresee and correct the inaccurate/incomplete information provided in Letter RS-05-103 prior to the information being submitted and approved by the NRC. Therefore, the failure to provide information to the NRC in Letter RS-05-103 that was complete and accurate in all material respects was considered a performance deficiency.

The inspectors reviewed this issue in accordance with IMC 0612, Appendix B, "Issue Screening," dated September 7, 2012. Because the failure to provide complete and accurate information to the NRC had the potential to impede or impact the regulatory process, the finding was evaluated in accordance with the NRC Enforcement Policy for traditional enforcement items and the underlying technical issue was evaluated using the SDP to determine the risk significance of this issue. Specifically, this violation is associated with a finding that has been evaluated by the SDP and communicated with an SDP color reflective of the safety impact of the deficient licensee performance. The SDP, however, does not specifically consider the regulatory process impact, or actual consequences. Thus, although related to a common regulatory concern, it is necessary to address the violation and finding using different processes to correctly reflect both the regulatory importance of the violation and the safety significance of the associated finding.

The inspectors determined that this issue was more than minor in accordance with IMC 0612, Appendix B "Issue Screening," dated September 7, 2012, because it adversely affected the Barrier Integrity Cornerstone attribute of Design Control. The inspectors also answered "yes" to the More-than-Minor screening question, "If left uncorrected, would the performance deficiency have the potential to lead to a more significant safety concern"? Specifically, the inspectors determined that this issue was more than minor because, if left uncorrected, the failure to provide complete and accurate information for the Unit 2 vessel head stud configuration could have resulted in non-conservative PTL curves that allowed operation in an unacceptable region that would increase the possibility of vessel failure during a pressurized thermal shock event. The inspectors performed a Phase I SDP screening using IMC 0609, Attachment 0609 Appendix A, Exhibit 3-Barrier Integrity Screening Questions, dated June 19, 2012, and selected the box under the RCS Boundary (e.g., pressurized thermal shock issues) which required a detailed risk-evaluation.

A Region III Senior Reactor Analyst performed a detailed risk-evaluation of this finding. A potential increase in the probability for vessel failure would exist if the plant was operated in the unacceptable pressure temperature region and a pressurized thermal shock event occurred. Based on the licensee and supporting vendor assessments which concluded that no substantial increase in vessel stresses will occur due to operation with 53 head studs, the driving force for crack propagation (e.g., K1) will remain essentially unchanged. However, to bound the delta risk evaluation, it was assumed that the initiating event frequency for a reactor vessel failure increased by 10 percent. From the Braidwood Standardized Plant Analysis Risk Model Version 8.21, the initiating event frequency for reactor vessel failure from any cause was 1E-7/yr. Core damage is expected to occur if reactor vessel failure occurs. The exposure time for the finding was the maximum of one year. Thus, a bounding risk assessment yields a delta risk of 1E-8/yr. Therefore, based on the detailed risk-evaluation, this finding is of very low risk significance (Green).

This violation was similar to an example of a SL III violation identified in Section 6.9.c.1 of the NRC Enforcement Policy, which stated "Incomplete or inaccurate information is provided or maintained. If this information had been completely and accurately provided or maintained, it would likely have caused the NRC to reconsider a regulatory position or undertake a substantial further inquiry." If the NRC had been provided with the information that Unit 2 was operating with only 53 head studs, the NRC would have likely reconsidered approval of the exemption, or requested a new analysis, or revision to the existing analysis (WCAP-16143) to support the licensee's exemption request for Unit 2.

No cross-cutting aspect was assigned to this SL IV violation as the performance deficiency was not reflective of current performance. Specifically, the issue was in excess of three years old, and therefore did not meet the definition of current performance as identified in Section 03.15 of IMC 0612.

Enforcement:

Title 10 of the *Code of Federal Regulations* (10 CFR) 50.9(a), "Completeness and Accuracy of Information," requires that "Information provided to the Commission by an applicant for a license or by a licensee or information required by statute or by the Commission's regulations, orders, or license conditions to be maintained by the applicant or the licensee shall be complete and accurate in all material respects."

On October 3, 2005, in Letter No. RS-05-103, the licensee provided information to the Commission that was not complete and accurate in all material respects in that, WCAP-16143 "Reactor Vessel Closure Had/Vessel Flange Requirements Evaluation for Byron/Braidwood Units 1 and 2" did not provide a valid basis for changing the RPV closure head flange limit for Braidwood Unit 2. Specifically, WCAP-16143 Section 4 "Flange Integrity" demonstrated adequate vessel margins based upon the original closure head flange configuration and did not represent the modified closure head configuration (53 head studs) applicable to the Unit 2 reactor vessel.

This violation was similar to an example of a SL III violation identified in Section 6.9.c.1 of the NRC Enforcement Policy. However, after consideration by NRC management, and with the approval of the Director of the Office of Enforcement, it was determined that a Severity Level IV Cited Violation was appropriate. This is a violation of 10 CFR 50.9(a) and a Notice of Violation is attached.

The licensee corrective actions for this issue included performing an operability evaluation and entering this violation into the CAP as AR 0155806. Because the licensee's corrective actions were under development at the time of the exit meeting, the inspectors assessed the basis for concluding that this non-compliance was not an immediate safety concern. Specifically, this violation was not an immediate safety concern because it was not expected to change the operating limits established in the current PTLR. (VIO 05000457/2013008-01 Inaccurate and Incomplete Information for Exemption Request from 10 CFR 50.60).

Because the finding discussed above was evaluated separately using the SDP, it is required to be tracked separately and will be given a separate tracking number (FIN 05000457/2013008-02 Inaccurate/Incomplete Information for Exemption Request from 10 CFR 50.60).

4OA6 Management Meetings

.1 Exit Meeting Summary

On October 29, 2013, the inspection results were presented to Mr. J. Bashor. The licensee acknowledged the issues presented. None of the potential report input discussed was considered proprietary.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee

M. Kanavos, Site Vice President
J. Bashor, Engineering Director
B. Casey, Exelon ISI
M. Abbas, Regulatory Assurance

Nuclear Regulatory Commission

E. Duncan, Branch Chief

LIST OF ITEMS OPENED, CLOSED AND DISCUSSED

Opened

05000457/2013008-01	VIO	Inaccurate/Incomplete Information For Exemption Request From 10 CFR 50.60. (Section 1R08.5)
05000457/2013008-02	FIN	Inaccurate/Incomplete Information For Exemption Request From 10 CFR 50.60. (Section 1R08.5)

LIST OF DOCUMENTS REVIEWED

The following is a partial list of documents reviewed during the inspection. Inclusion on this list does not imply that the NRC inspector 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 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.

1R08 Inservice Inspection Activities (71111.08P)

- AR 1562139; Enhancement of RPV Calibration Stud Drawing M98003; dated September 22, 2013.
- AR 1561231; UT Data for Penetration 69 Missing UT Data 7.5 degrees; dated September 19, 2013.
- AR 1560253; Dray Boric Acid Residue at the CRDM Housing; dated September 18, 2013.
- AR 1559124; FAC Component UT Exam Failure; dated September 16, 2013.
- AR 1558897; Rejectable NDE of Penetration 69 After Buffing; dated September 15, 2013.
- AR 1558555; Results of Liquid Penetrant Examination of Penetration 69; dated September 14, 2013.
- AR 1558493; 1RC013B Support Ring Damage; dated September 13, 2013.
- AR 1558067; Potential NRC Violation of 10 CFR 50.9 for PTLR; dated September 13, 2013.
- AR 1549725; PTLR Analysis not Revised for Reactor Head Stud Configuration; dated August 22, 2013.
- AR 1544673; Scope Expansion Inspection to 2SX27DA-10"; Dated August 8, 2013.
- AR 1544658; A1R17 Scope Add: Scope Expansion Inspection 1SX27DB; 2nd LOC; Dated August 8, 2013.
- AR 1544651; A1R17 Scope Add: Scope Expansion Inspection 1SX27DB-10"; Dated August 8, 2013.
- AR 1542372; SX Piping Leak – 1SX27DA; dated August 1, 2013.
- AR 1537783; 1SX51AB Pre-Freeze UT Reading Below 87.5%; Dated July 19, 2013.
- AR 1536176: RH Letdown Booster Pump Upstream Suction Isolation Valve; dated July 15, 2013.
- AR 1515257; 1SX42AB Pre-Freeze UT Reading Below 87.5%; dated May17, 2013.
- AR 1506866; Eductor 1A Spray Addition Suction Check Valve; dated April 26, 2013.
- AR 1479975; A1R16 CRDM Penetration #4 Inspection Volume Coverage not Met; Dated February 26, 2013.
- AR 1433688; NDE Rejectable Indications on FW-1 for Valve 2RF031; Dated October 31, 2012.
- AR 1429452; NDE Results for Valve 2CV216; Dated October 21, 2012.
- AR 1426428; 2CV8321A Potential Through-Wall Leak at Leak-Off Line; Dated October 15, 2012.
- AR 1414934; Line 1SX93AB UT Reading of .067 inches; Dated September 18, 2012.
- AR 1373398; A1R17: Perform NDE and Bare Metal Visual Exams of RPV Head; Dated June 1, 2012.
- AR 1361504; A1R16 RPV Head BMV Inspection Results; dated May 2, 2012.
- AR 1361344; 1B CC Pump Casing Drain Through-Wall Leak; Dated May 2, 2012.

- AR 1360631; Close Out of Legacy Foreign Material in Unit 1 Rx Vessel; dated April 30, 2012
- AR 1360219; Legacy FM Discovered in Reactor Vessel – 1RC01R; dated April 30, 2012.
- AR 1360099; Craft Identified and QV Rejected a Questionable Root Pass; dated April 29, 2012
- AR 1356505; Acceptance Criteria for Reactor Head Examinations; Dated April 20, 2012
- AR 1354783; IST 121A Relief Valve (1SI8811A) Leak at Threaded Connection; dated April 16, 2012
- AR 1354192; Dry Boric Acid at Body to Bonnet Bolt on 1RC8037A; Dated April 16, 2012
- AR 1354181; Dry Boric Acid at Body to Bonnet Bolt on 1RC8037B; Dated April 16, 2012
- AR 1353906; 1RH03AA-8": Dry Boric Acid Residue in ASME Bolted Connection; dated April 14, 2012
- Letter- Wesdyne- Range Expansion of PDI Generic Procedure PDI-UT-5; dated March 11, 2005
- Letter- Exelon- Review and Approval PCI Welders dated September 17, 2013.
- NDE Report; Visual Examination - Unit 1 RPV Closure Head; dated September 23, 2013
- NDE Report 906281-001, PT Examination -Penetration 69- Repair Areas; dated September 20, 2013
- NDE Report A1R17-21, PT Examination -Penetration 69; dated September 13, 2013
- NDE Report A1R17-30, PT Examination -Penetration 69; dated September 15, 2013
- NDE Report A1R17-PT001; 1SI-24- SW12, 18, 19 & 20; dated September 12, 2013
- NDE Report; UT Report Data Sheet Penetration 69; dated September 20, 2013
- NDE Report; UT Report Data Sheet Penetration 69; dated September 19, 2013
- NDE Report; UT Calibration Data Sheet; Reactor Vessel Head Studs; dated September 13, 2013
- NDE Report A1R17- UT-007; UT Calibration Data Sheet; 1FW-01-17; dated September 12, 2013
- NDE Report A1R17-UT-008; UT Calibration Data Sheet; 1FW-01-18; dated September 12, 2013
- NDE Report A1R17-UT-027; UT Calibration Data Sheet; 1FW-01-05; dated September 16, 2013
- NDE Report A1R17-UT-026; UT Calibration Data Sheet; 1FW-01-04; dated September 16, 2013
- NDE Reports 2013-202 & 203; MT Data Sheet; 1SX27DA-10"; dated August 3, 2013.
- NDE Report 2013-206; UT Calibration Data Sheet;1SX27DA-10" Weld 1; dated August 3, 2013
- NDE Report; UT Calibration Data Sheet; ISI Weld ISI-39-25B; dated April 30, 2012.
- NDE Report; UT Calibration Data Sheet; ISI Weld ISI-39-4; dated April 17, 2012.
- NDE Report; UT Calibration Data Sheet; ISI Weld ISI-39-4; dated April 29, 2012.
- Procedure GQP-9.7; PCI- Solvent Removable Liquid Penetrant Examination and Acceptance Standards for Welds, Base Materials, and Cladding; Revision 14.
- Procedure ER-AA-335-002; Liquid Penetrant Examination; Revision 5.
- Procedure EXE-ISI-11; Liquid Penetrant Examination; Revision 2.
- Procedure EXE-PDI-UT-1; Ultrasonic Examination of Ferritic Piping Welds in accordance with PDI-UT-1; Revision 6
- Procedure EXE-PDI-UT-2; Ultrasonic Examination of Austenitic Piping Welds in accordance with PDI-UT-2; Revision 7
- Procedure ER-AP-335-001; Bare Metal Visual Examination for Alloy 600/82/182 Materials; Revision 3
- Procedure ER-AP-331; Boric Acid Corrosion Control Program; Revision 7

- Procedure ER-AP-331-1000; Boric Acid Corrosion Inspection Locations Implementation and Inspection Guidelines; Revision 7
- Procedure ER-AP-331-1002; Boric Acid Corrosion Control Program Identification Screening and Evaluation; Revision 8
- Procedure EXE-PDI-UT-5; Straight Beam Ultrasonic Examination of Bolts and Studs in Accordance with PDI-UT-05; Revision 1
- Procedure WDI-ET-004; IntraSpect Eddy Current Analysis Guidelines; Revision 17
- Procedure WDI-STD-114; Reactor Vessel Head Vent Tube Inside Diameter and Carbon Steel Wastage ET Examination; Revision 13
- Procedure WDI-STD-1040 - Procedure for Ultrasonic Examination of Reactor Vessel Head Penetrations; Revision 10
- Procedure WDI-STD-1041; Reactor Vessel Head Penetration Ultrasonic Examination Analysis; Revision 9
- Procedure WDI-STD-1042; Procedure for Eddy Current Examination of Reactor Vessel Head Penetrations; Revision 3
- PDQS; WDI-STD-1040; dated March 4, 2010
- PDQS; WDI-STD-1041; dated March 2, 2010
- PQR A-001; dated October 19, 1998
- PQR A-002; dated March 9, 1989
- PQR 1-50C; dated January 3, 1984
- PQR 644; dated April 29, 1999
- PQR 467; dated September 12, 1994
- PQR 307; dated June 29, 1992
- Rod Ticket; Penetration 69 Weld Filler Material; dated September 19, 2013
- Weld Process Traveler 906281; Penetration 69 Weld Overlay; dated September 21, 2013
- Weld Data Card 906281; Penetration 69 Weld Overlay; dated September 21, 2013
- Weld Data Sheet 906281; Penetration 69 Weld Overlay; dated September 20, 2013
- Welder BW41; Qualification dated September 12, 2013
- Welder L0031; Qualification dated September 12, 2013
- Welder BW-40; Qualification dated September 12, 2013
- Welder D246; Qualification dated September 12, 2013
- WO 01661641; Overlay Piping EC-394727; dated August 2, 2013
- WPS 1-1-GTSM-PWHT; Revision 2
- WPS 43 MN-GTAW/SMAW; Revision 8

LIST OF ACRONYMS USED

ADAMS	Agencywide Document Access Management System
AR	Action Request
ASME	American Society of Mechanical Engineers
AV	Apparent Violation
BA	Boric Acid
BACC	Boric Acid Corrosion Control
BMV	Bare Metal Visual
CAP	Corrective Action Program
CC	Code Case
CFR	Code of Federal Regulations
DRS	Division of Reactor Safety
EPRI	Electric Power Research Institute
ET	Eddy Current Testing
IMC	Inspection Manual Chapter
IP	Inspection Procedure
ISI	Inservice Inspection
MT	Magnetic Particle
NCV	Non-Cited Violation
NDE	Non-destructive Examination
NRC	U.S. Nuclear Regulatory Commission
PT	Dye Penetrant
PTL	Pressure Temperature Limit
PTLR	Pressure Temperature Limit Report
RCS	Reactor Coolant System
RPV	Reactor Pressure Vessel
RPVUH	Reactor Pressure Vessel Upper Head
SG	Steam Generator
SL	Severity Level
TS	Technical Specification
UT	Ultrasonic Examination
WE	Westinghouse Electric Company
WO	Work Order
TS	Technical Specification

You are required to respond to this letter and should follow the instructions specified in the enclosed Notice when preparing your response. If you have additional information that you believe the NRC should consider, you may provide it in your response to the Notice. The NRC review of your response to the Notice will also determine whether further enforcement action is necessary to ensure compliance with regulatory requirements.

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter, its enclosure, and your response 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 Document Access and Management System (ADAMS). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room). To the extent possible, your response should not include any personal privacy or proprietary, information so that it can be made available to the Public without redaction.

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

Enclosures:
1. Notice of Violation, EA-2013-209
2. Inspection Report 05000456/2013008 and 05000457/2013008
   Attachment: Supplemental Information

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[1]Office of Enforcement (OE) concurrence received on 11/12; Nuclear Reactor Regulation (NRR) concurrence received on 11/14