UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

April 19, 2012

Mr. Michael J. Pacilio
President and Chief Nuclear Officer
Exelon Nuclear
4300 Winfield Road
Warrenville, IL 60555

SUBJECT: BRAIDWOOD STATION, UNITS 1 AND 2 - SAFETY EVALUATION IN SUPPORT OF THE THIRD 10-YEAR INSERVICE INSPECTION INTERVAL REQUEST FOR RELIEF I3R-08 (TAC NOS. ME6024 AND ME6025)

Dear Mr. Pacilio:

By letter to the U.S. Nuclear Regulatory Commission (NRC) dated April 11, 2011, and supplemented by letters dated June 6, 2011, November 2, 2011, and March 12, 2012 (Agencywide Documents Access and Management System (ADAMS) Accession Numbers ML111020263, ML111580106, ML113070349, and ML120730196 respectively), Exelon Generation Company, LLC (the licensee) requested relief from the requirements of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code for Braidwood Station, Units 1 and 2 (Braidwood).

Specifically, pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR) 50.55a(g)(5)(iii), the licensee requested relief from the depth-sizing uncertainty qualification requirement for ultrasonic examinations conducted from the inside diameter of pipes, i.e., root mean square error not greater than 0.125 inches, contained in ASME Code Cases N-695, "Qualification Requirements for Dissimilar Metal Piping Welds, Section XI, Division 1," and N-696, "Qualification Requirements for Appendix VIII Piping Examinations Conducted From the Inside Surface, Section XI, Division 1," due to impracticality.

The licensee initially requested that relief be granted for the remainder of the third 10-year inservice inspection interval of Braidwood, however, in its March 12, 2012 submittal, the licensee revised its request to include only examinations conducted at Braidwood, Unit 1 prior to the end of refueling outage A1R16 which is scheduled to begin on April 16, 2012 and examinations conducted at Braidwood, Unit 2 prior to the end of refueling outage A2R16 which is scheduled to begin in the fall of 2012.

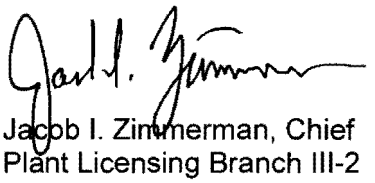
The NRC staff determined that granting relief pursuant to 10 CFR 50.55a(g)(6)(i) is authorized by law and will not endanger life or property or the common defense and security, and is otherwise in the public interest giving due consideration to the burden upon the licensee that could result if the requirements were imposed on the facility. Accordingly, the NRC staff concludes that the licensee has adequately addressed all of the regulatory requirements set forth in 10 CFR 50.55a.

Therefore, the NRC staff grants the use of the alternate depth-sizing qualification (0.189 inch for ASME Code Case N-695 welds and 0.245 inch for ASME Code Case N-696 welds), subject to the licensee providing, for NRC staff review and approval prior to the expiration of the relief, information identified in the safety evaluation (SE) should a crack requiring depth-sizing be identified. This relief is granted for Braidwood Units 1 and 2, until prior to startup following refueling outage A1R16 and A2R16, respectively.

The NRC staff's SE is enclosed.

Please contact the Braidwood Project Manager, Brenda Mozafari at (301) 415-2020 if you have any questions on this action.

Sincerely,



Jacob I. Zimmerman, Chief
Plant Licensing Branch III-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-456 and 50-457

Enclosure:
Safety Evaluation

cc w/encl: Distribution via Listserv

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

ON THE THIRD 10-YEAR INSERVICE INSPECTION INTERVAL

REQUEST FOR RELIEF I3R-08, REVISION 0

EXELON GENERATION COMPANY, LLC

BRAIDWOOD STATION, UNITS 1 AND 2

DOCKET NUMBERS 50-456 AND 50-457

1.0 INTRODUCTION

By letter to the U.S. Nuclear Regulatory Commission (NRC, the Commission) dated April 11, 2011, and supplemented by letters dated June 6, 2011, November 2, 2011, and March 12, 2012 (Agencywide Documents Access and Management System (ADAMS) Accession Numbers ML111020263, ML111580106, ML113070349, and ML120730196, respectively), Exelon Generation Company, LLC (the licensee) requested relief from the requirements of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code for Braidwood Station, Units 1 and 2 (Braidwood).

Specifically, pursuant to Title 10 of the *Code of Federal Regulations* (CFR), Section 50.55a(g)(5)(iii), the licensee requested relief from the depth-sizing uncertainty qualification requirement for ultrasonic (UT) examinations conducted from the inside diameter (ID) of pipes, i.e., root mean square (RMS) error not greater than 0.125 inches, contained in ASME Code Cases N-695, "Qualification Requirements for Dissimilar Metal Piping Welds, Section XI, Division 1," and N-696, "Qualification Requirements for Appendix VIII Piping Examinations Conducted From the Inside Surface, Section XI, Division 1," due to impracticality.

The licensee initially requested that relief be granted for the remainder of the third 10-year inservice inspection interval (ISI) for Braidwood, however, in its March 12, 2012 submittal, the licensee revised its request to include only examinations conducted at Braidwood, Unit 1 prior to the end of refueling outage A1R16 which is scheduled to begin on April 16, 2012 and examinations conducted at Braidwood, Unit 2 prior to the end of refueling outage A2R16 which is scheduled to begin in the fall of 2012.

2.0 REGULATORY REQUIREMENTS

In its letter dated June 6, 2011, the licensee requests relief from the depth-sizing acceptance criteria contained in ASME Code Cases N-695 and N-696 pursuant to 10 CFR 50.55a(g)(5)(iii).

ASME Code Cases N-695 and N-696 are accepted for use in NRC Regulatory Guide (RG) 1.147, Revision 16, "Inservice Inspection Code Case Acceptability ASME Section XI, Division 1," and incorporated by reference in 10 CFR 50.55a(b).

Section 50.55a(g)(4)(ii) of 10 CFR states, in part, that inservice examination of components must comply with the latest edition of the Code incorporated by reference in 10 CFR 50.55a(b) or the optional ASME Code cases listed in RG 1.147.

Section 50.55a(g)(5)(iii) of 10 CFR states, in part, that licensees may determine that conformance with certain code requirements is impractical and that the licensee shall notify the Commission and submit information in support of the determination.

Section 50.55a(g)(6)(i) of 10 CFR states, in part that the Commission will evaluate determinations under 10 CFR 50.55a(g)(5) that code requirements are impractical and that the Commission may grant such relief and may impose such alternative requirements as it determines is authorized by law and will not endanger life or property.

Based on the above, and subject to the following technical evaluation, the NRC staff finds that regulatory authority exists for the licensee to request and the Commission to grant the relief requested by the licensee.

3.0 TECHNICAL EVALUATION

3.1 Component Descriptions

The licensee stated that this relief request addresses nozzle-to-safe end dissimilar metal welds and adjacent safe end-to-pipe or safe end-to-elbow austenitic welds at the locations listed in Tables 1 and 2 below.

Table 1: Dissimilar Metal Category B-F Item Number B5.10/RI-ISI Category R-A Item Number R1.15 Welds

Component Number	Unit	Description
1 RV-01-022	1	Reactor Pressure Vessel (RPV) Nozzle-to-Safe End Hot leg, loop C
1 RV-01-023	1	RPV Nozzle-to-Safe End Cold leg, loop C
1 RV-01-024	1	RPV Nozzle-to-Safe End Cold leg, loop D
1 RV-01-025	1	RPV Nozzle-to-Safe End Hot leg, loop D
1 RV-01-026	1	RPV Nozzle-to-Safe End Hot leg, loop A
1 RV-01-027	1	RPV Nozzle-to-Safe End Cold leg, loop A
1 RV-01-028	1	RPV Nozzle-to-Safe End Cold leg, loop B
1 RV-OI-029	1	RPV Nozzle-to-Safe End Hot leg, loop B
2RV-01-022	2	RPV Nozzle-to-Safe End Hot leg, loop D
2RV-01-023	2	RPV Nozzle-to-Safe End Cold leg, loop D
2RV-01-024	2	RPV Nozzle-to-Safe End Cold leg, loop C
2RV-01-025	2	RPV Nozzle-to-Safe End Hot leg, loop C
2RV-01-026	2	RPV Nozzle-to-Safe End Hot leg, loop B
2RV-01-027	2	RPV Nozzle-to-Safe End Cold leg, loop B

2RV-01-028	2	RPV Nozzle-to-Safe End Cold leg, loop A
2RV-01-029	2	RPV Nozzle-to-Safe End Hot leg, loop A

Table 2: Austenitic Category B-J Item Number B9.11 RI-ISI Category R-A Item Number R1.20 Welds

Component Number	Unit	Description
1 RV-01-030	1	RPV Safe End-to-Pipe Hot leg, loop C
1 RV-01-031	1	RPV Safe End-to-Elbow Cold leg, loop C
1 RV-01-032	1	RPV Safe End-to-Elbow Cold leg, loop D
1 RV-01-033	1	RPV Safe End-to-Pipe Hot leg, loop D
1 RV-01-034	1	RPV Safe End-to-Pipe Hot leg, loop A
1 RV-01-035	1	RPV Safe End-to-Elbow Cold leg, loop A
1 RV-01-036	1	RPV Safe End-to-Elbow Cold leg, loop B
1 RV-01-037	1	RPV Safe End-to-Pipe Hot leg, loop B
2RV-01-030	2	RPV Safe End-to-Pipe Hot leg, loop D
2RV-01-031	2	RPV Safe End-to-Elbow Cold leg, loop D
2RV-01-032	2	RPV Safe End-to-Elbow Cold leg, loop C
2RV-01-033	2	RPV Safe End-to-Pipe Hot leg, loop C
2RV-01-034	2	RPV Safe End-to-Pipe Hot leg, loop B
2RV-01-035	2	RPV Safe End-to-Elbow Cold leg, loop B
2RV-01-036	2	RPV Safe End-to-Elbow Cold leg, loop A
2RV-01-037	2	RPV Safe End-to-Pipe Hot leg, loop A

3.1 Applicable ASME Code and Addenda

The applicable ASME Code edition and addenda is ASME Code Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," 2001 edition through the 2003 addenda. In addition, as required by 10 CFR 50.55a, ASME Code Section XI, 2001 edition is used for Appendix VIII, "Performance Demonstration for Ultrasonic Examination Systems."

3.2 Applicable ASME Code Requirements

Volumetric examinations of the components contained in Tables 1 and 2 are mandated in ASME Code Section XI Table IWB-2500 Category B-F, Item BF5.10. Requirements for UT examinations used to conduct these inspections are contained in ASME Code Section XI Paragraph IWA-2232 and ASME Code Section XI Appendix I Paragraph I-2220. These paragraphs refer to ASME Code Section XI Appendix VIII including Supplements 2 and 10.

ASME Code Case N-695, "Qualification Requirements for Dissimilar Metal Piping Welds, Section XI, Division 1," provides alternative requirements to Appendix VIII, Supplement 10. It is unconditionally approved for use in NRC RG 1.147, Revision 16. Paragraph *3.3(c)* of ASME Code Case N-695 states "Examination procedures, equipment, and personnel are qualified for depth-sizing when the RMS error of the flaw depth measurements, as compared to the true flaw depths, do not exceed 0.125 in. (3 mm)."

ASME Code Case N-696, "Qualification Requirements for Appendix VIII Piping Examinations Conducted From the Inside Surface, Section XI, Division 1," provides alternative requirements to Appendix VIII Supplements 2, 3, and 10 examinations conducted from the inside surface. ASME Code Case N-696 is unconditionally approved for use in NRC RG 1.147, Revision 16. Supplement 3 is not applicable to the welds referenced under this relief request. Paragraph *3.3(d)* of ASME Code Case N-696 states, "Supplement 2 or Supplement 3 examination procedures, equipment, and personnel are qualified for depth-sizing when the flaw depths estimated by UT examinations, as compared to true depths, do not exceed 0.125 in. (3 mm) RMS, when they are combined with a successful Supplement 10 qualification."

3.4 Impracticality of Compliance

The licensee stated that although examination vendors have qualified for detection and length sizing of flaws in accordance with the ASME Code requirements for examinations from the ID surface, none have met the established RMS error of 0.125-inch for flaw indication depth-sizing. The ISI vendor that will be used in the upcoming inspections at Braidwood has demonstrated the ability to depth-size flaw indications in dissimilar metal welds with a RMS error of 0.189 inch instead of the 0.125-inch RMS error required by Appendix VIII Supplement 10 (ASME Code Case N-695) and an RMS error of 0.245-inch instead of the 0.125 inch RMS error required by the combined Appendix VIII Supplement 2 and 10 qualification (ASME Code Case N-696).

The NRC staff also notes that inspection of these welds from the outer diameter (OD) of the pipe is often precluded by the geometry of the weld or the presence of obstructions. The NRC staff further notes that, due to the location of these welds, OD inspections are normally associated with significant dose to the inspector.

3.5 Licensee's Proposed Alternative

For welds governed by ASME Code Case N-695, the licensee proposed using an alternative depth-sizing RMS error qualification criterion of 0.189 inches. To compensate for the greater uncertainty in the depth of a measured flaw which will result from this acceptance criterion, the licensee proposed to add to the measured depth of any cracks detected a value equal to the difference between the proposed acceptance criterion (0.189 inches) and the acceptance criterion contained in the code case (0.125 inches), in this case 0.064 inches.

For welds governed by ASME Code Case N-696, the licensee proposed using an alternative depth-sizing RMS error qualification criterion of 0.245 inches. To compensate for the greater uncertainty in the depth of a measured flaw which will result from this acceptance criterion, the licensee proposed to add to the measured depth of any cracks detected a value equal to the difference between the proposed acceptance criterion (0.245 inches) and the acceptance criterion contained in the code case (0.125 inches), in this case 0.120 inches.

3.6 Duration of Relief

In its original request the licensee requested relief for the remainder of the third 10-Year ISI Interval for Braidwood, Units 1 and 2, which is currently scheduled to end July 28, 2018, for Unit 1, and on October 16, 2018, for Unit 2. In its letter dated March 12, 2012, the licensee revised its request to include only examinations conducted at Braidwood, Unit 1 prior to the end of refueling outage A1R16 which is scheduled to begin in April 2012 and examinations conducted at Braidwood, Unit 2 prior to the end of refueling outage A2R16 which is scheduled to begin in the fall of 2012.

3.7 NRC Staff Evaluation

As described above, the licensee has requested relief from the requirements of ASME Code Case N-695, "Qualification Requirements for Dissimilar Metal Piping Welds" and ASME Code Case N-696, "Qualification Requirements for Appendix VIII Piping Examinations Conducted From the Inside Surface, Section XI, Division 1." These ASME Code Cases require that procedures used to inspect welds from the ID be qualified by performance demonstration. The acceptance criterion established by the code case is an RMS error of not greater than 0.125 inches.

The licensee has reported and the NRC staff has confirmed that attempts have been made to qualify ID UT inspection procedures since 2002 and that, to date, no inspection vendor has been able to meet the acceptance criteria established by ASME Code cases despite the fact that numerous individuals from several companies have attempted to do so. The NRC staff finds that this repeated inability to qualify ID UT inspection techniques in accordance with ASME Code Cases N-695 and N-696 constitutes an impracticality as described in 10 CFR 50.55a(g)(5)(iii). Use of an alternate approach, such as an OD examination represents a burden to the licensee as the welds under consideration may be inaccessible from the OD or inspectors may be subject to significant radiation dose in performing the inspections.

Section 50.55a(g)(6)(i) to 10 CFR permits the Commission to grant relief in cases of impracticality when it determines (in part) that such relief will not endanger life or property. Given the fact that there is always some uncertainty in measuring cracks, and given that relaxation of the ASME Code Cases N-695 and N-696 acceptance criteria may increase the uncertainty of crack measurements, there is a possibility that, for a given crack, the depth measured using a technique which fails to meet the ASME Code Case acceptance criteria will underestimate the depth of the crack to a greater extent than a similar measurement using a procedure which meets the acceptance criteria. This situation could be viewed as a decreasing assurance that structural integrity or leak tightness of the subject welds will be maintained and, therefore, a failure to meet the criteria contained in 10 CFR 50.55a(g)(6)(i).

To address the issue of increased potential for undersizing of flaws by inspection procedures which do not meet the ASME Code Cases N-695 and N-696 acceptance criterion, the licensee has proposed to increase the measured depth of the crack by the difference between the actual RMS error obtained for the procedure employed and the code case acceptance criterion. In this case, the RMS error for the procedure employed is 0.189 inches for ASME Code Case N-695 welds and 0.245 inches for ASME Code Case N-696 welds.

For example, a crack in an ASME Code Case N-696 weld which measured 0.2 inches deep would be treated as though it was 0.32 inches deep:

$$0.2" + (0.245" - 0.125") = 0.32"$$

While this approach provides some additional margin of safety when compared to not adjusting the crack length, the NRC staff did not have sufficient evidence to confirm that the adjustment to the crack depth was sufficient to provide reasonable assurance that cracks measured using procedures with larger RMS error values would not be undersized when compared to those meeting the RMS error criteria. This lack of assurance was primarily based on the difficulty of sizing cracks when the ID surfaces of the weld are rough and the fact that the testing program was not designed to address this issue.

In November of 2011, the NRC staff reviewed the Performance Demonstration Initiative program (administered by Electric Power Research Institute) data used to qualify welds for ASME Code Cases. Based on this review, the NRC staff was able to determine:

1. When using inspection procedures which do not meet the ASME Code Case N-695 or N-696 acceptance criterion, a significant number of data points are undersized by greater than 0.064 inches (0.189" – 0.125") for ASME Code Case N-695 welds and 0.12 inches for ASME Code Case N-696 welds (0.245" – 0.125") indicating that reasonable assurance that cracks will not be significantly undersized does not exist when using this approach.

2. When using inspection procedures which do not meet the ASME Code Case N-695 or N-696 acceptance criterion, only a very few data points are undersized by greater than twice the RMS error of the procedure employed (in this case 2 x 0.189" = 0.378" for ASME Code Case N-695 welds and 2 x 0.245" = 0.490" for ASME Code Case N-696 Welds). The addition of twice the RMS error to the measured depths of any flaws found during the ID inspections provides reasonable assurance that cracks will not be significantly undersized.

3. Due to the qualification procedures used in ASME Code Case N-696, for some service providers, the reported RMS error value may understate the actual RMS error for austenitic stainless steel welds (Supplement 2 welds).

Based on these findings the NRC staff issued a request for additional information (RAI) on March 9, 2012 (ADAMS Accession No. ML120600603). In this RAI the NRC staff proposed two alternative approaches to the issue of crack depth-sizing that could potentially be found to be acceptable. These were:

1. Adjust the measured depth of the crack upward by 0.378 inches (2 x 0.189) for ASME Code Case N-695 welds and 0.490 inches (2 x 0.245 inch) for ASME Code Case N-696 welds; or

2. Provide a commitment such that if cracks are identified for which sizing is required during the scheduled examinations, a flaw evaluation will be submitted to the NRC for review and approval.

Specific information to be included in the flaw evaluation, as requested in the RAI, includes:

1. Provide the measured flaw size.

2. Provide the personal qualification RMS error for the analyst who measured the flaw size and justification for adjusting the depth of the measured crack by less than twice the RMS error.

3. Calculate the RMS error for Supplement 2 and 10 examinations separately.

4. Conduct eddy current testing to determine if the flaw is surface breaking.

5. Provide the ID surface profile of the weld, pipe, nozzle, and safe-end (as applicable) in the region at and surrounding the transducer locations used to depth-size the detected flaw.

6. Describe the suspected flaw-degradation mechanism and the process used to determine the degradation mechanism.

In its response, dated March 12, 2012, the licensee agreed that as part of the relief request to provide the information requested in items 1, 4, 5, and 6 above. The licensee did not agree to provide the information requested in item 2 because the licensee believes that disclosure of individual qualification results could compromise the integrity of the qualification. The NRC staff finds that the risks to the integrity of the qualification program outweigh the benefit of obtaining this information. Item 3 is not necessary as the Supplement 2 RMS error was found in a relief request response using the same vendor dated July 7, 2005 (ADAMS Accession No. ML051960005). The NRC staff withdraws the requests for items 2 and 3 of the March 9, 2012 RAI.

Based on the concerted efforts by the industry to meet the acceptance criteria contained in ASME Code Case N-695 and the difficulties associated with other inspection methods, the NRC staff finds that meeting the 0.125 inch acceptance criterion in ASME Code Cases N-695 and N-696 is impractical and represents a burden to the licensee. This relief is subject to the licensee providing the information listed above in items 1, 3, 4, and 5, for the subject indicated flaws. On that basis, and subsequent to NRC staff review and approval of information submitted, the NRC staff finds that this alternative provides reasonable assurance of structural integrity or leak tightness of the subject component and therefore "will not endanger life or property" as required by 10 CFR 50.55a(g)(6)(i).

4.0 CONCLUSION

As set forth above, the NRC staff determines that granting relief pursuant to 10 CFR 50.55a(g)(6)(i) is authorized by law and will not endanger life or property or the common defense and security, and is otherwise in the public interest giving due consideration to the burden upon the licensee that could result if the requirements were imposed on the facility. Accordingly, the NRC staff concludes that the licensee has adequately addressed all of the regulatory requirements set forth in 10 CFR 50.55a.

Therefore, the NRC staff grants relief as specified above, i.e., the use of the alternate depth-sizing qualification (0.189 inch for ASME Code Case N-695 welds and 0.245 inch for ASME Code Case N-696 welds), subject to the licensee providing, for NRC staff review and approval prior to the expiration of the relief, the information identified above as items 1, 4, 5, and 6 should a crack requiring depth-sizing be identified. This relief is granted for Braidwood, Units 1 and 2, until prior to startup following refueling Cycle A1R16 and A2R16, respectively.

All other ASME Code, Section XI requirements for which relief was not specifically requested and approved in the subject request for relief remain applicable, including third-party review by the Authorized Nuclear Inservice Inspector.

Principal Contributor: S. Cumblidge, NRR

Date: April 19, 2012

Therefore, the NRC staff grants the use of the alternate depth-sizing qualification (0.189 inch for ASME Code Case N-695 welds and 0.245 inch for ASME Code Case N-696 welds), subject to the licensee providing, for NRC staff review and approval prior to the expiration of the relief, information identified in the safety evaluation (SE) should a crack requiring depth-sizing be identified. This relief is granted for Braidwood Units 1 and 2, until prior to startup following refueling outage A1R16 and A2R16, respectively.

The NRC staff's SE is enclosed.

Please contact the Braidwood Project Manager, Brenda Mozafari at (301) 415-2020 if you have any questions on this action.

Sincerely,

/RA/

Jacob I. Zimmerman, Chief
Plant Licensing Branch III-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-456 and 50-457

Enclosure:
Safety Evaluation

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