April 2, 2008

Mr. Charles G. Pardee
Chief Nuclear Officer
and Senior Vice President
Exelon Generation Company, LLC
4300 Winfield Road
Warrenville, IL 60555

SUBJECT: BRAIDWOOD STATION, UNITS 1 AND 2 - ISSUANCE OF AMENDMENTS RE: REQUEST FOR AMENDMENT TO TECHNICAL SPECIFICATION 5.5.16, “CONTAINMENT LEAKAGE RATE TESTING PROGRAM” (TAC NOS. MD5149 AND MD5150)

Dear Mr. Pardee:

The U.S. Nuclear Regulatory Commission (the Commission) has issued the enclosed Amendment No. 149 to Facility Operating License No. NPF-72 and Amendment No.149 to Facility Operating License No. NPF-77 for the Braidwood Station (Braidwood), Units 1 and 2, respectively. The amendments are in response to your application dated April 4, 2007 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML070950418), as supplemented by letters dated October 10, 2007, January 31, and February 26, 2008, (ADAMS Accession Nos. ML072840458, ML080320555, and ML080570644, respectively).

The amendments revise TS 5.5.16, “Containment Leakage Rate Testing Program,” to reflect a one-time, 5-year extension of the current containment Type A test (containment integrated leakage rate test (IRLT)) interval requirement, under Title 10 of the *Code of Federal Regulations*, Part 50, Appendix J, Option B, from 10 years to 15 years. The amendments allow the next Type A ILRT to be performed within 15 years of the most recent Type A test at Braidwood, but no later than October 5, 2013, for Unit 1, and no later than May 4, 2014, for Unit 2.

A copy of the Safety Evaluation is also enclosed. The Notice of Issuance will be included in the Commission’s biweekly *Federal Register* notice.

Sincerely,

/RA/

Marshall J. David, Senior Project Manager
Plant Licensing Branch III-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. STN 50-456 and STN 50-457

Enclosures:
1. Amendment No.149 to NPF-72
2. Amendment No.149 to NPF-77
3. Safety Evaluation

cc w/encls: See next page

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ADAMS Accession Nos. Pkg ML080640312 **Ltr. Accession No. ML080640290** *SE dated NRR-058
TS Pages Accession No. ML081010177

OFFICE	LPL3-2/PM	LPL3-2/LA	DRA/APLA/BC	DSS/SCVB/BC	DE/EMCB/BC	OGC	LPL3-2/BC
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DATE	3/19/08	3/19/08	12/03/07	12/03/07	03/10/08	3/24/08	4/2/08

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Braidwood Station, Units 1 and 2

cc:

Corporate Distribution
Exelon Generation Company, LLC
Via e-mail

Braidwood Distribution
Exelon Generation Company, LLC
Via e-mail

Mr. Dwain W. Alexander, Project Manager
Westinghouse Electric Corporation
Via e-mail

Ms. Bridget Little Rorem
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Howard A. Learner
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Chairman, Ogle County Board
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EXELON GENERATION COMPANY, LLC

DOCKET NO. STN 50-456

BRAIDWOOD STATION, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 149
License No. NPF-72

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Exelon Generation Company, LLC (the licensee) dated April 4, 2007, as supplemented by letters dated October 10, 2007, January 31, and February 26, 2008, and complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. NPF-72 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendix A as revised through Amendment No. 149 and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, are hereby incorporated into this license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of the date of its issuance and shall be implemented within 30 days of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Russell Gibbs, Chief
Plant Licensing Branch III-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications and Facility Operating License

Date of Issuance:
April 2, 2008

EXELON GENERATION COMPANY, LLC

DOCKET NO. STN 50-457

BRAIDWOOD STATION, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 149
License No. NPF-77

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Exelon Generation Company, LLC (the licensee) dated April 4, 2007, as supplemented by letters dated October 10, 2007, January 31, and February 26, 2008, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. NPF-77 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendix A as revised through Amendment No. 149 and the Environmental Protection Plan contained in Appendix B, both of which were attached to License No. NPF-72, dated July 2, 1987, are hereby incorporated into this license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of the date of its issuance and shall be implemented within 30 days of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Russell Gibbs, Chief
Plant Licensing Branch III-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications and Facility Operating License

Date of Issuance: April 2, 2008

ATTACHMENT TO LICENSE AMENDMENT NOS. 149 AND 149

FACILITY OPERATING LICENSE NOS. NPF-72 AND NPF-77

DOCKET NOS. STN 50-456 AND STN 50-457

Replace the following pages of the Facility Operating Licenses and the Appendix A Technical Specifications (TSs) with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

Remove	Insert
License NPF-72	License NPF-72
License Page 3	License Page 3
License NPF-77	License NPF-77
License Page 3	License Page 3
TSs	TSs
5.5-20	5.5-20

(3) Exelon Generation Company, pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess, and use at any time any byproduct, source and special nuclear material as sealed neutron sources for reactor startup, sealed sources for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as required;

(4) Exelon Generation Company, pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess, and use in amounts as required any byproduct, source or special nuclear material without restriction to chemical or physical form, for sample analysis or instrument calibration or associated with radioactive apparatus or components; and

(5) Exelon Generation Company, pursuant to the Act and 10 CFR Parts 30, 40 and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility.

C. The license shall be deemed to contain and is subject to the conditions specified in the Commission's regulations set forth in 10 CFR Chapter I and is subject to all applicable provisions of the Act and to the rules, regulations and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) Maximum Power Level

The licensee is authorized to operate the facility at reactor core power levels is not in excess of 3586.6 megawatts thermal (100 percent rated power) in accordance with the conditions specified herein and other items identified in Attachment 1 to this license. The items identified in Attachment 1 to this license shall be completed as specified. Attachment 1 is hereby incorporated into this license.

(2) Technical Specifications

The Technical Specifications contained in Appendix A as revised through Amendment No. 149, and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, are hereby incorporated into this license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

(3) Emergency Planning

In the event that the NRC finds that the lack of progress in completion of the procedures in the Federal Emergency Management Agency's final rule, 44 CFR Part 350, is an indication that a major substantive problem exists in achieving or maintaining an adequate state of emergency preparedness, the provisions of 10 CFR Section 50.54(s)(2) will apply.

material as sealed neutron sources for reactor startup, sealed sources for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as required;

(4) Exelon Generation Company, LLC pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess, and use in amounts are required any byproduct, source or special nuclear material without restriction to chemical or physical form, for sample analysis or instrument calibration or associated with radioactive apparatus or components; and

(5) Exelon Generation Company, LLC, pursuant to the Act and 10 CFR Parts 30, 40 and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility.

C. The license shall be deemed to contain and is subject to the conditions specified in the Commission's regulations set forth in 10 CFR Chapter I and is subject to all applicable provisions of the Act and to the rules, regulations and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) Maximum Power Level

The licensee is authorized to operate the facility at reactor core power levels is not in excess of 3586.6 megawatts thermal (100 percent rated power) in accordance with the conditions specified herein and other items identified in Attachment 1 to this license. The items identified in Attachment 1 to this license shall be completed as specified. Attachment 1 is hereby incorporated into this license.

(2) Technical Specifications

The Technical Specifications contained in Appendix A as revised through Amendment No. 149, and the Environmental Protection Plan contained in Appendix B, both of which are attached to License No. NPF-72, dated July 2, 1987, are hereby incorporated into this license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

(3) Emergency Planning

In the event that the NRC finds that the lack of progress in completion of the procedures in the Federal Emergency Management Agency's final rule, 44 CFR Part 350, is an indication that a major substantive problem exists in achieving or maintaining an adequate state of emergency preparedness, the provisions of 10 CFR Section 50.54(s)(2) will apply.

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 149 TO FACILITY OPERATING LICENSE NO. NPF-72

AND AMENDMENT NO. 149 TO FACILITY OPERATING LICENSE NO. NPF-77

EXELON GENERATION COMPANY, LLC

BRAIDWOOD STATION, UNITS 1 AND 2

DOCKET NOS. STN 50-456 AND STN 50-457

1.0 INTRODUCTION

By letter to the U.S. Nuclear Regulatory Commission (NRC, the Commission) dated April 4, 2007 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML070950418), as supplemented by letters dated October 10, 2007, January 31, and February 26, 2008, (ADAMS Accession Nos. ML072840458, ML080320555, and ML080570644, respectively), Exelon Generation Company, LLC (the licensee), requested an amendment to Appendix A, Technical Specifications (TSs) of Facility Operating License Nos. NPF-72 and NPF-77 for Braidwood Station (Braidwood), Units 1 and 2. The proposed amendment would revise TS 5.5.16, "Containment Leakage Rate Testing Program," to reflect a one-time, 5-year extension of the current containment Type A test (containment integrated leakage rate test (ILRT)) interval requirement from 10 to 15 years. The proposed change would allow the next Type A ILRT to be performed within 15 years of the most recent Type A test at Braidwood, but no later than October 5, 2013, for Unit 1, and no later than May 4, 2014, for Unit 2.

The supplements contained clarifying information and did not change the NRC staff's original proposed no significant hazards consideration.

2.0 REGULATORY EVALUATION

Paragraph 50.54(o) of Title 10 of the *Code of Federal Regulations* (10 CFR), and 10 CFR Part 50, Appendix J, Option B requires that a Type A test be conducted at a periodic interval based on historical performance of the overall containment system. The Type A test must be conducted (1) after a containment system has been completed and is ready for operation, and (2) at a periodic interval based on historical performance of the overall containment system. Section V.B.3 of 10 CFR 50, Appendix J, Option B, requires that the regulatory guide (RG) or other implementation document used by a licensee to develop a performance-based leakage-testing program must be included, by general reference, in the plant TSs. Furthermore, the submittal for TS revisions must contain justification, including supporting analyses, if the licensee chooses to deviate from methods approved by the Commission and endorsed in a RG.

Braidwood TS 5.5.16, “Containment Leakage Rate Testing Program,” requires that leakage rate testing be performed as required by 10 CFR 50.54(o) and 10 CFR Part 50, Appendix J, Option B, as modified by approved exemptions, and in accordance with the guidelines contained in RG 1.163, “Performance-Based Containment Leak-Test Program,” dated September 1995. This RG endorses, with certain exceptions, Nuclear Energy Institute (NEI) report NEI 94-01, Revision 0, “Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J,” dated July 26, 1995.

A Type A test is an overall ILRT of the containment structure. NEI 94-01 specifies an initial test interval of 48 months, but allows an extended interval of 10 years, based upon two consecutive successful tests. There is also a provision for extending the test interval an additional 15 months in certain circumstances. The most recent two Type A tests at Braidwood have been successful, so the current interval requirement would normally be 10 years.

However, by the current application dated April 4, 2007, the licensee is seeking a deviation from the NEI 94-01 guidelines by requesting a one-time extension of the Type A test interval from 10 to 15 years based on historical performance of its containment supported by a risk-informed analysis. Specifically, the licensee is requesting a change to TS 5.5.16 which would add exceptions from the guidelines of RG 1.163 and NEI 94-01, Revision 0, regarding the Type A test interval. The exceptions state that for Braidwood:

> The first Unit 1 Type A test performed after the October 5, 1998, Type A test shall be performed no later than October 5, 2013.
>
> The first Unit 2 Type A test performed after the May 4, 1999, Type A test shall be performed no later than May 4, 2014.

The local leakage rate tests (LLRTs) (Type B and Type C tests), including their schedules, are not affected by this request. The proposed TS change does not involve any other changes to licensing commitments or acceptance criteria.

3.0 TECHNICAL EVALUATION

Braidwood containments are post-tensioned reinforced concrete structures with a carbon steel liner on the inside surface. Each containment consist of a cylindrical wall, a flat foundation mat, a shallow dome roof, and penetrations through the structure. The post-tensioning system consists of vertical and horizontal tendons in the cylinder wall and three-way tendons in the dome. The steel liner and its penetrations establish the leakage-limiting boundary of the containment. The post-tensioned reinforced concrete structures provide containment structural integrity. The leak-tight integrity of the containment penetrations (equipment hatch, airlocks, flanges, and sealing mechanisms) and isolation valves are verified through Type B and Type C LLRTs. The overall leak-tight integrity and structural integrity of the containment is verified through a Type A ILRT as required by 10 CFR 50, Appendix J. These tests are performed at the calculated peak containment internal pressure related to the design-basis loss-of-coolant accident (LOCA). Under Option B, the two units at Braidwood currently have an ILRT interval of 10 years. By letter dated April 4, 2007, the licensee requested a one-time 5-year extension of the Type A test interval from 10 to 15 years. The licensee justified the proposed change based on historical plant-specific containment leakage testing program results and containment in-service inspection (ISI) program results, supported by a risk-informed analysis.

The leakage rate testing requirements of 10 CFR 50, Appendix J, Option B (Type A ILRT and Type B and Type C LLRTs) and the containment ISI requirements mandated by 10 CFR 50.55a, together, help ensure the continued leak-tight and structural integrity of the containment during its service life. Therefore, the NRC staff requested information regarding the licensee's program for LLRTs, containment ISI, and potential areas of weaknesses in the containment that may not be apparent in the risk assessment. The review of the Section 4.1 of Attachment 1 of the licensee's April 4, 2007, request also warranted certain additional information. The information presented in the licensee's request, the NRC staff's request for additional information (RAI) dated September 7, 2007 (ADAMS Accession No. ML072490143), and the licensee's supplemental responses in letters dated October 10, 2007, January 31, and February 26, 2008, are discussed and evaluated below.

3.1 Containment ISI Program and Structural/Leak-Tight Integrity Considerations

In its request, the licensee stated that the results of previous Type A ILRTs performed at Braidwood demonstrate that the containment structure of each unit remains essentially a leak-tight barrier. The licensee presented the plant-specific results from the recent two previous Type A ILRTs for Unit 1 (November 1995 and October 1998) and Unit 2 (November 1994 and May 1999), which were successful. The most recent of these tests indicated a leakage rate of 0.071 percent of containment air weight per day and 0.063 percent of containment air weight per day for Unit 1 and Unit 2, respectively, in comparison to the allowable containment leakage rate (0.75La) of 0.075 percent of containment air weight per day. The licensee stated that License Amendment 140 for Braidwood granted by the NRC staff on September 8, 2006, which fully implemented an alternative accident source term pursuant to 10 CFR 50.67, increased the acceptance criteria for allowable leakage rate (0.75La) for all future ILRT results from less than 0.075 percent of containment air weight per day to less than 0.15 percent of containment air weight per day. This increase in the acceptance criteria for future ILRTs provides additional margin of assurance that Braidwood containment structures will continue to perform their design function following the design-basis accident.

The licensee also stated that adoption of an Option B performance-based leakage testing program did not alter the basic test methods nor the acceptance criteria, but it did alter the test frequency of containment leakage testing in Type A, B and C tests based on an evaluation which utilizes the as-found leakage history. The licensee stated that continued satisfactory results from the Type B and Type C LLRTs and containment inspections support the proposed extension of the Type A test interval. The initial test interval for Type B and Type C tests is 30 months but may be extended to 120 months for Type B tests and 60 months for Type C tests based on acceptable performance, established by passing two as-found LLRTs. Type B components whose test intervals are extended to more than 60 months are tested on a staggered basis for early detection of common mode failures. In its RAI, the NRC staff requested the licensee to provide the current test intervals under Option B for the Type B and Type C LLRTs. The NRC staff also requested a schedule for the Type B and Type C tests on containment pressure retaining boundaries that are or will be scheduled to be performed prior to and during the requested 5-year extension period.

In its response dated October 10, 2007, the licensee provided, for each unit, a comprehensive table that identified all the penetrations subjected to Type B and C testing and their current test frequencies established under Option B based on performance. The licensee identified only one penetration (i.e., the fuel transfer tube) in each unit with bellows. The licensee indicated that the test frequencies are re-evaluated after each refueling outage for potential changes. The licensee also provided the date of the last test and the due date for the next scheduled test for between now and the next ILRT. The tabular information indicates that each unit has approximately 66 penetrations that are subject to local leak rate tests. The current performance-based test frequencies for these penetrations are approximately as follows: 3 percent every 3 months; 3 percent tested every 6 months; 6 percent every 18 months; 18 percent every 24 months; 8 to 15 percent every 30 months; 42 to 48 percent every 48 months; and 12 percent every 96 months. Based on the information in the tables, approximately 40 to 45 percent of the penetrations in each unit will be tested two or more times between now and the next proposed ILRT and the remaining will be tested at least one time. Thus, the response indicates that the performance of each of the containment pressure boundary penetrations will be monitored by a Type B or Type C test at least once and about half of them two to three times during the requested extension period for the ILRT interval. The NRC staff finds that the licensee is effectively implementing its Type B and Type C LLRT program, under Option B, in a rational and systematic manner that is consistent with industry standards and regulatory guidance, and will continue to do so during the requested ILRT interval extension period.

The licensee stated that the containments at Braidwood are examined by the Appendix J visual inspections program, containment ISI program, and coatings inspections program and discussed these programs in Section 4.1.3 of its request.

The licensee stated that, as part of the 10 CFR 50 Appendix J program, Braidwood performs visual inspections of accessible interior and exterior containment surfaces to uncover any evidence of structural deterioration that may affect containment structural and leak-tight integrity. These examinations are conducted consistent with the requirements of 10 CFR 50 Appendix J, Option B, RG 1.163, and NEI 94-01, prior to any Type A test, and during two refueling outages prior to the next Type A test, based on a 10-year frequency. The licensee stated that additional visual inspections are conducted in accordance with the requirements of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code) Section XI, Subsections IWE and IWL. The most recent visual inspections of accessible interior and exterior containment surfaces at Braidwood, completed in 2006 for Units 1 and 2, indicated that there were no structural problems that could affect containment structural or leakage integrity.

In its RAI, the NRC staff requested the licensee to provide information with regard to how the licensee was implementing the general visual inspection requirements of 10 CFR 50 Appendix J, Option B. The NRC staff requested the licensee to discuss its program for Braidwood for visual inspections (with schedules and methods) that meet the RG 1.163 requirement. The NRC staff also requested the licensee to indicate, with a schedule, how it would supplement this 10-year interval-based visual inspection requirement for the requested 15-year ILRT interval to ensure a continuing means of uncovering, early, evidence of containment structural deterioration.

In its response dated October 10, 2007, the licensee stated that it is using containment ISI program visual examinations pursuant to ASME Code, Section XI, Subsections IWE and IWL to satisfy the visual examination requirement in Regulatory Position C.3 of RG 1.163 for the performance-based Option B Containment Leakage Testing Program. Subsection IWE requires the licensee to perform general visual examinations of the liner and penetrations three times in a 10-year containment ISI interval. Subsection IWL requires the licensee to perform general visual examinations of the accessible concrete surfaces two times in a 10-year containment ISI interval. During the 15-year ILRT interval, this will result in three IWL visual examinations of the concrete surfaces and more than three IWE visual examinations of accessible metallic containment surfaces for Braidwood. Prior to performing an ILRT, the licensee will schedule its IWE and IWL examinations in a way that it is considered as a pre-ILRT examination. This process satisfies the intent and frequency of visual examinations required by Regulatory Position C.3 of RG 1.163 even for a 15-year interval. The NRC staff finds that the licensee's implementation of the visual examination program provides an acceptable level of quality and safety even for the 15-year ILRT interval.

As mentioned previously, the licensee stated that Braidwood has implemented a comprehensive containment ISI program in accordance with the frequency and requirements of ASME Code, Section XI, Subsections IWE and IWL. The first containment ISI interval examinations at Braidwood were performed in accordance with the 1992 Edition and 1992 Addenda of the ASME Code, Section XI, Subsections IWE and IWL, as modified by 10 CFR 50.55a. In its RAI response, the licensee clarified that the current (first) containment ISI intervals end on July 28, 2008 and October 16, 2008, for Braidwood, Unit 1 and Unit 2, respectively. For the next (second) containment ISI interval, which includes the requested 5-year extension period, Braidwood is committed to the 2001 Edition through 2003 Addenda of the ASME Code, Section XI. The licensee stated that during these IWE/IWL inspections, various indications were identified that were either repaired or documented and evaluated as acceptable by the Responsible Individual/Responsible Engineer, without affecting structural integrity. In its RAI, the NRC staff requested the licensee to substantiate this statement by providing the historic highlights of plant-specific results of inspections/examinations performed on structures and components of the containment pressure boundary at Braidwood with significant findings and actions taken that demonstrate effective implementation of the containment ISI program in managing containment degradation and ensuring its structural and leak-tight integrity. The NRC staff also requested the schedule of IWE/IWL examinations that will be performed during the requested extension period.

In its responses dated October 10, 2007, and February 26, 2008, the licensee stated that the containment ISI examinations will proceed in accordance with schedules and requirements in Subsections IWE and IWL of the ASME Code Section XI, 2001 Edition through 2003 Addenda during the requested extension period. The licensee also provided two tables summarizing the results of the most recent (2001 and 2006) containment ISI program inspection findings of indications and their disposition of the containment concrete and post-tensioning system performed in accordance with ASME Code, Section XI, Subsection IWL for Braidwood. The licensee stated that none of the conditions found from the IWL inspections were severe enough to compromise the structural integrity of the containment structures. However, indications were found that required evaluations and, in some instances, repair and/or augmented examinations. An important finding was that water was discovered in several tendon ducts. For these tendons, sheathing filler samples obtained and analyzed per ASME Code, Section XI, Subsection IWL revealed no evidence of active corrosion. The sheathing filler grease was replaced in the tendon grease caps for most affected tendons and in the entire tendon duct for the other affected tendons. Also, a tendon wire was removed and subjected to examination and physical testing. Furthermore, a continuity test was performed on all wires of one tendon. The licensee identified and summarized the augmented examination of selected concrete locations, where grease leakage through the containment wall exterior was noted, and post-tensioning system grease caps on an annual basis. The annual examinations of all tendon grease caps, at below-grade locations and on the dome, are performed at Braidwood due to a history of free water at specific locations. During the examinations, the grease caps were inspected for leakage, deformation, and evidence of moisture. Also, during every post-tensioning surveillance at a 5-year frequency, specific tendons that have exhibited free water are selected for examination in addition to the samples required by Subsection IWL. Tendon and grease cap inspections are discussed further, below.

Since degradation of bellows is a source for potential leakage, in its RAI, the NRC staff requested the licensee if bellows were used on penetrations through containment pressure-retaining boundaries at Braidwood. If bellows were used, the RAI requested the licensee to provide information on their location, inspection, testing and operating experience.

In its response dated October 10, 2007, the licensee stated that there is only one pressure-retaining boundary penetration in each unit at Braidwood that employs bellows. These bellows are around the fuel transfer tube and are in the fuel transfer canal between the containment and the fuel handling building, which houses the spent fuel pool. These bellows are not regularly inspected. However, they are local leak rate tested in accordance with 10 CFR 50, Appendix J. The licensee provided dates of the last completed test, next planned test, and the current test frequency in its RAI response. The licensee stated that no leakage issues with these bellows have been identified at Braidwood. The licensee's response stated that the bellows are local leak-rate tested every 8 years and that no leakage issues have been identified to date with these bellows. The NRC staff finds that the information provided was responsive to the NRC staff's concern with regard to leakage through bellows and is, therefore, acceptable.

Since management of degradation in inaccessible and uninspectable areas of the containment is an area of concern, the NRC staff requested the licensee to provide information of instances, if any, during implementation of the IWE/IWL containment ISI program at Braidwood where existence of or potential for degradation conditions in inaccessible areas of the containment structure and metallic liner were identified and evaluated based on conditions found in accessible areas as required by 10 CFR 50.55a(b)(2)(viii)(E) and 10 CFR 50.55a(b)(2)(ix)(A). If there were any instances of such conditions, the licensee was requested to discuss the findings and actions taken.

In its response dated October 10, 2007, the licensee stated that there have been no conditions at Braidwood where existence of or potential for degradation conditions in inaccessible areas of the containment structure and metallic liner were identified and evaluated based on conditions found in accessible areas as required by 10 CFR 50.55a(b)(2)(viii)(E) and 10 CFR 50.55a(b)(2)(ix)(A). The licensee, however, stated that for Braidwood, Unit 1, there are approximately 20 localized liner areas that are inaccessible below the containment moisture barrier (MB) that were identified, in the March 2000 refueling outage, as areas for augmented examination per Table IWE-2500-1, Examination Category E-C. For Braidwood, Unit 2, there are approximately 80 localized liner areas that are inaccessible below the containment MB that were identified, during the May 1999 and October 2000 refueling outages, as areas for augmented examination. These areas have some metal loss due to pitting and corrosion. The maximum metal reduction identified was 5/64" and 6/64" deep for Braidwood, Unit 1 and Unit 2, respectively, and the nominal thickness of the liner is 1/4". The licensee stated that the degradation of the liner is suspected to be the result of MB aging and mechanical damage to the MB from impacts during maintenance activities. This combined with impingement of water, possibly the result of fan cooler condensate leakage, established the conditions for liner degradation. The water had a flow path between the degraded MB and the metal liner and became entrapped below the MB Tremco epoxy sealant into the Cerablanket material below it and above the compressible material in the 2" annular space between the basemat and the containment liner.

The licensee stated that the surface of the liner remained wetted adjacent to the Cerablanket material; with the presence of oxygen, the liner coating and liner started to degrade. The licensee performed engineering evaluations and took corrective actions, such as, surface preparation of the degraded liner areas, application of new coatings, installation of new MB material, and categorization of the degraded liner for augmented examination to prevent further propagation of the liner degradation. For Braidwood, Unit 1, these actions were completed during the March 2000 refueling outage, and subsequent refueling outages as needed. For Braidwood, Unit 2, these actions were completed during the October 2000 refueling outage, and subsequent refueling outages as needed. The licensee stated that these corrective actions have been determined to be effective, because follow up VT-1 and ultrasonic examinations of the augmented examination areas in May 2006 and October 2006 for Braidwood, Unit 1 and Unit 2, respectively, have confirmed that the MB was intact, prior to removal, and the Cerablanket material below the MB was dry. In addition, negligible changes from the previous augmented examinations in 2003 were found, and no active corrosion was identified. The licensee stated that these augmented examinations will be performed again on each unit during the next containment ISI period.

The licensee's RAI response identified MB degradation and related pitting corrosion degradation of the liner plate below the MB in both units. The licensee evaluated these degradations and took appropriate corrective actions, such as, surface preparation of the degraded liner areas, application of new coatings, installation of new MB material, and categorization of the degraded liner for augmented examination (IWE Table-2500-1, Examination Category E-C) to prevent further liner degradation. The licensee's follow-up examination of the augmented category areas in 2006 confirmed that the material below the MB was dry, with no active corrosion and no significant change of conditions from the previous augmented examinations. The augmented examination of these areas will continue over the next containment ISI period. The NRC staff finds that the licensee has appropriately managed the MB and liner plate degradations. (The licensee also provided a risk analysis to estimate the likelihood and risk implications of corrosion-induced leakage of the steel liners going undetected in inaccessible areas during the extended test interval. The analysis and the NRC staff's associated findings are discussed in the next section of this safety evaluation.)

The licensee stated that, as part of the IWL-2520 examination of the post-tensioning systems, it verified that the prestressing forces for the tendons selected for examination met the acceptance criteria for predicted forces. The licensee stated that a regression analysis, using individual tendon lift-off forces measured during tendon surveillances and performed as recommended in NRC Information Notice 99-10, indicated acceptable margin for all tendons for the design life of

Braidwood. The licensee concluded that the IWL and IWE program examinations have demonstrated that the structural integrity and leak-tightness of the Braidwood containments have not been compromised. The license stated that there will be no change to the containment ISI program schedule as a result of the proposed changes.

With regard to the regression analyses of the tendon prestressing forces, the NRC staff requested the licensee to confirm whether the analyses were based on individual tendon lift-off forces measured during tendon surveillances and to indicate the number of surveillances from which data was used. The NRC staff further requested the licensee to confirm if accessible grease caps are visually examined as part of the containment ISI programs at Braidwood as required by 10 CFR 50.55a(b)(2)(viii)(A).

In its response dated October 10, 2007, the licensee stated that the regression analysis was performed primarily from data obtained from the control tendon for each group (i.e., vertical, horizontal, and dome groups). Since the non-control tendons have not undergone more than two lift-off inspections, sufficient data is not available for non-control tendons to perform a regression analysis to predict future lift-off forces (i.e., more than two data points are required to identify a trend). Non-control tendon data were, however, considered in the analysis. A set of regression analyses was performed for the control tendons. The results of the regression analyses were then extended to the non-control tendons. Data from surveillance activities extending back to 1987 (i.e., the first surveillance) were used in the regression analysis.

The licensee stated that Braidwood conducts an annual examination of grease cans in areas susceptible to moisture intrusion because there has been a history of existence of free water at specific tendon anchorage locations. These are primarily located below grade level and also at a few dome tendon anchorage locations. Due to this known condition, grease cans are examined for evidence of water leakage, grease leakage, and conditions that could indicate degradation of the anchorage components. These examinations are performed as an annual surveillance in the summer months when grease leakage would be most likely detected. The licensee further stated that, as required by 10 CFR 50.55a(b)(2)(viii)(A), grease caps installed at Braidwood are visually inspected during every surveillance with a 5-year frequency. Each of the grease caps in each unit is inspected for evidence of grease leakage, grease cap deformation, evidence of free water, and evidence of corrosion that challenges the capability of the grease cap to contain the grease.

The NRC staff finds that the licensee performed the regression analysis of the tendon lift-off forces in a rational manner within the limitations of available data. The NRC staff also finds that the licensee has an adequate program for examination of grease caps for evidence of water leakage, grease leakage, and degradation of tendon anchorage components. The licensee is also performing visual inspections of the grease caps as required by the modification in 10 CFR 50.55a(b)(2)(viii)(A).

The licensee stated that it conducts periodic inspections of Service Level 1 coatings inside containment that meet RG 1.54, “Service Level I, II, and III Protective Coatings Applied to Nuclear Power Plants,” during each refueling outage at Braidwood, as required by the plant licensing basis and station procedures. Any localized areas of degraded coatings identified by these inspections are evaluated and repaired or replaced, as necessary. The licensee stated that recent inspections of containment coatings at Braidwood indicate that coatings were generally in a good condition. In some areas on the liner plate and other surfaces where indications of degraded coatings were observed, the licensee implemented measures to evaluate and repair the coating deterioration to an acceptable condition. The inspections requirements of the containment coatings program will not be changed as a result of the requested ILRT interval extension.

In summary, the NRC staff finds that the licensee has effectively implemented adequate LLRT, containment ISI and safety-related coatings inspection programs to periodically examine, monitor, and manage age-related and environmental degradations of the Braidwood containment structures. The results of the past ILRTs and the containment ISI programs demonstrate that the structural and leak-tight integrity of the containment structures is sound and adequately managed. The containment structures will continue to be periodically monitored by these programs during the requested 5-year extension period for the ILRT interval. The NRC staff finds that there is reasonable assurance that the containment structural and leak-tight integrity will continue to be maintained without undue risk to safety during the requested 5-year extension period for the ILRT interval. Therefore, the NRC staff finds it acceptable to grant the requested one-time extension of the ILRT interval to 15 years for Braidwood. However, the NRC staff notes that the dates for the next ILRTs being approved by this TS amendment may not coincide with future refueling outage dates. Therefore, the NRC staff recommends that the licensee plan well ahead to conduct the next ILRT for Braidwood, within the 15-year interval being approved without seeking a further extension.

3.2 Risk Impact Assessment Considerations

The licensee performed a risk impact assessment of extending the Type A test interval to 15 years. The risk assessment was provided in the April 4, 2007, application for license amendment. In performing the risk assessment, the licensee considered the guidelines of NEI 94-01, the methodology used in Electric Power Research Institute (EPRI) Topical Report (TR)-104285, “Risk Impact Assessment of Revised Containment Leak Rate Testing,” the NEI Interim Guidance for Performing Risk Impact Assessments in Support of One-Time Extensions for Containment Integrated Leakage Rate Surveillance Intervals, and RG 1.174, “An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis.”

The basis for the current 10-year test interval is provided in Section 11.0 of NEI 94-01 and was established in 1995 during the development of the performance-based Option B to Appendix J. Section 11.0 of NEI 94-01 states that NUREG-1493, “Performance-Based Containment Leak-Test Program,” provides the technical basis to revise leakage rate testing requirements contained in Option B to Appendix J. The basis consistsd of qualitative and quantitative assessments of the risk impact (in terms of increased public dose) associated with a range of extended leakage rate test intervals. To supplement this basis, industry undertook a similar study. The results of that study are documented in EPRI Research Project Report TR-104285.

The EPRI study used an analytical approach similar to that presented in NUREG-1493 for evaluating the incremental risk associated with increasing the interval for Type A tests. The Appendix J, Option A, requirements that were in effect for Braidwood early in the plant’s life required a Type A test frequency of three tests in 10 years. The EPRI study estimated that relaxing the test frequency from three tests in 10 years to one test in 10 years would increase the average time that a leak, which was detectable only by a Type A test, goes undetected from 18 to 60 months. Since Type A tests only detect about 3 percent of leaks (the rest are identified during local leak-rate tests based on industry leakage rate data gathered from 1987 to 1993), this results in a 10 percent increase in the overall probability of leakage. The risk contribution of pre-existing leakage for the pressurized-water reactor and boiling-water reactor representative plants in the EPRI study confirmed the NUREG-1493 conclusion that a reduction in the frequency of Type A tests from three tests in 10 years to one test in 20 years leads to an “imperceptible” increase in risk that is on the order of 0.2 percent and a fraction of one-person-rem (roentgen equivalent man) per year in increased public doses.

Building upon the methodology of the EPRI study and the NEI Interim Guidance, the licensee assessed the change in the predicted person-rem per year frequency. The licensee quantified the risk from sequences that have the potential to result in large releases if a pre-existing leak were present. Since the Option B rulemaking was completed in 1995, the NRC staff has issued RG 1.174 on the use of probabilistic risk assessment (PRA) in evaluating risk-informed changes to a plant’s licensing basis. The licensee proposed using RG 1.174 to assess the acceptability of extending the Type A test interval beyond that established during the Option B rulemaking.

RG 1.174 provides risk-acceptance guidelines for assessing the increases in core damage frequency (CDF) and large early-release frequency (LERF) for risk-informed license amendment requests. Since the Type A test does not impact CDF, the relevant criterion is the change in LERF. The licensee estimated the change in LERF for the proposed change based on the cumulative change from the original frequency of three tests in a 10-year interval. RG 1.174 also discusses defense-in-depth and encourages the use of risk analysis techniques to help ensure and show that key principles, such as the defense-in-depth philosophy, are met. The licensee estimated the change in the conditional containment failure probability for the proposed change to demonstrate that the defense-in-depth philosophy is met.

The licensee provided analyses, as discussed below. The following comparisons of risk are based on a change in test frequency from three tests in 10 years (the test frequency under Appendix J, Option A) to one test in 15 years. This bounds the impact of extending the test frequency from one test in 10 years to one test in 15 years. The following conclusions can be drawn from the analysis associated with extending the Type A test frequency:

1. Given the change from a three in 10-year test frequency to a one in 15-year test frequency, the increase in the total integrated plant risk is estimated to be about 0.1 person-rem per year or less for both units. This increase is comparable to that estimated in NUREG-1493, where it was concluded that a reduction in the frequency of tests from three in 10 years to one in 20 years leads to an “imperceptible” increase in risk. Therefore, the increase in the total integrated plant risk for the proposed change is considered by the NRC staff as small and supportive of the proposed change.

2. The increase in LERF resulting from a change in the Type A test frequency from the original three in 10 years to one in 15 years is estimated to be about 5.4×10^{-7} per year for Braidwood based on the unit-specific internal events PRA. This estimate assumes all non-LERF end states are assigned to EPRI Class 1 (intact containment) and, as such, are included as potential LERF contributors in

the event of a large undetected leak pursuant to the NEI methodology. In fact, many of these Class 1 sequences would involve successful operation of containment sprays, in which case the potential for a pre-existing leak to result in a large release would be greatly reduced. Based on a separate assessment, the licensee estimates that 50 percent or more of the Class 1 frequency would involve operation of containment sprays, even after accounting for dependent operator action failures and hardware dependencies. Thus, the estimated increase in LERF reported above is conservative. There is also some likelihood that the flaws in the containment estimated as part of the Class 3b frequency would be detected as part of the visual examination of the containment surfaces, performed in accordance with ASME Code, Section XI, Subsections IWE/IWL. Visual inspections are expected to be effective in detecting large flaws in the visible regions of containment, and this would reduce the impact of the extended test interval on LERF. The licensee's risk analysis considered the potential impact of age-related corrosion/degradation in inaccessible areas of the containment shell on the proposed change. The increase in LERF associated with corrosion events is estimated to be less than 1×10^{-8} per year for both units.

When the calculated increase in LERF is in the range of 10^{-7} per year to 10^{-6} per year, applications are considered by the NRC staff if the total LERF is less than 10^{-5} per year. The licensee estimated that the total LERF for internal and external events, including the requested change, was about 7.8×10^{-6} and 9.4×10^{-6} per year for Braidwood, Unit 1 and Unit 2, respectively. Thus, the total LERF including the requested change would remain below 10^{-5} per year. The LERF contribution from external events was determined based on a simplifying assumption that the LERF (i.e., the CDF and the fraction of core damage events contributing to LERF) for external events is comparable to that for internal events less the LERF contribution from interfacing-system LOCAs and steam generator tube rupture events (since these types of events would not typically occur from an external event initiator). This assumption is reasonable and somewhat conservative since the results of the Individual Plant Examination of External Events (IPEEE) for Braidwood indicate that fire events are the dominant external-event risk contributor, and that the fire CDF as reported in the IPEEE is approximately one decade lower than the internal events CDF for all units. The LERF impact of the ILRT extension for external events was also assumed to be the same as that for internal events. This is also conservative given that the estimated LERF increase for internal events conservatively neglects the impact of containment spray operation on LERF, as discussed above. The NRC staff concludes that increasing the Type A interval to 15 years results in only a small change in LERF and is consistent with the acceptance guidelines of RG 1.174.

3. RG 1.174 also encourages the use of risk analysis techniques to help ensure and show that the proposed change is consistent with the defense-in-depth philosophy. Consistency with the defense-in-depth philosophy is maintained if a reasonable balance is preserved between prevention of core damage, prevention of containment failure, and consequence mitigation. The licensee estimated the change in the conditional containment failure probability to be an increase of approximately one percentage point for all units for the cumulative change of

going from a test frequency of three in 10 years to one in 15 years. The NRC staff finds that the defense-in-depth philosophy is maintained based on the small magnitude of the change in the conditional containment failure probability for the proposed amendment.

Based on these conclusions, the NRC staff finds that the increase in predicted risk due to the proposed change is within the acceptance guidelines, while maintaining the defense-in-depth philosophy, of RG 1.174 and, therefore, is acceptable.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Illinois State official was notified of the proposed issuance of the amendment. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendments change requirements with respect to installation or use of a facility's component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration, and there has been no public comment on such finding (72 FR 31100; June 5, 2007). Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

6.0 CONCLUSION

The NRC staff concludes that the structural and leak-tight integrity of the Braidwood, containment structures is sound and adequately managed. Further, since the licensee has adequate LLRT and containment ISI programs in place that will continue to examine, monitor and manage potential degradations of the pressure-retaining components of the containments, there is reasonable assurance that the containment structural and leak-tight integrity will continue to be maintained if the ILRT interval is extended, as proposed, to 15 years. Additionally, the NRC staff concludes that the increase in predicted risk due to the proposed change is within the acceptance guidelines, while maintaining the defense-in-depth philosophy, of RG 1.174. Therefore, the one-time extension of the ILRT interval to 15 years until the next Type A test and the proposed change to TS 5.5.16 is acceptable at Braidwood.

The NRC staff notes that the dates for the next ILRTs being approved by this TS amendment may not coincide with future refueling outage dates. Therefore, the NRC staff recommends that the licensee plan well ahead to conduct the next ILRT for Braidwood, within the 15-year interval being approved, without seeking a further extension.

Thus, the Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public.

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Date: April 2, 2008