Exelon Generation®

Michael P. Gallagher
Vice President, License Renewal
Exelon Nuclear

200 Exelon Way
Kennett Square, PA 19348

610 765 5958 Office
610 765 5956 Fax
www.exeloncorp.com

michaelp.gallagher@exeloncorp.com

10 CFR 50
10 CFR 51
10 CFR 54

RS-14-149

May 23, 2014

U. S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, DC 20555-0001

Braidwood Station, Units 1 and 2
Facility Operating License Nos. NPF-72 and NPF-77
NRC Docket Nos. STN 50-456 and STN 50-457

Byron Station, Units 1 and 2
Facility Operating License Nos. NPF-37 and NPF-66
NRC Docket Nos. STN 50-454 and STN 50-455

Subject: Responses to NRC Requests for Additional Information, Set 22, dated April 24, 2014, related to the Byron Station, Units 1 and 2, and Braidwood Station, Units 1 and 2, License Renewal Application

References:

1. Letter from Michael P. Gallagher, Exelon Generation Company LLC (Exelon) to NRC Document Control Desk, dated May 29, 2013, "Application for Renewed Operating Licenses."

2. Letter from Lindsay R. Robinson, US NRC to Michael P. Gallagher, Exelon, dated April 24, 2014, "Request for Additional Information for the Review of the Byron Station, Units 1 and 2, and Braidwood Station, Units 1 and 2, License Renewal Application, Set 22 (TAC NOS. MF1879, MF1880, MF1881, and MF1882)"

In the Reference 1 letter, Exelon Generation Company, LLC (Exelon) submitted the License Renewal Application (LRA) for the Byron Station, Units 1 and 2, and Braidwood Station, Units 1 and 2 (BBS). In the Reference 2 letter, the NRC requested additional information to support staff review of the LRA.

Enclosure A contains the responses to these requests for additional information.

Enclosure B contains updates to sections of the LRA (except for the License Renewal Commitment List) affected by the responses.

Enclosure C provides an update to the License Renewal Commitment List (LRA Appendix A, Section A.5). There are no other new or revised regulatory commitments contained in this letter.

If you have any questions, please contact Mr. Al Fulvio, Manager, Exelon License Renewal, at 610-765-5936.

I declare under penalty of perjury that the foregoing is true and correct.

Executed on 5-23-2014

Respectfully,

Michael P. Gallagher
Vice President - License Renewal Projects
Exelon Generation Company, LLC

Enclosures: A. Responses to Requests for Additional Information
B. Updates to affected LRA sections
C: License Renewal Commitment List Changes

cc: Regional Administrator – NRC Region III
NRC Project Manager (Safety Review), NRR-DLR
NRC Project Manager (Environmental Review), NRR-DLR
NRC Senior Resident Inspector, Braidwood Station
NRC Senior Resident Inspector, Byron Station
NRC Project Manager, NRR-DORL-Braidwood and Byron Stations
Illinois Emergency Management Agency - Division of Nuclear Safety

Enclosure A

Byron and Braidwood Stations (BBS), Units 1 and 2
License Renewal Application

Responses to Requests for Additional Information

RAI B.2.1.19-1a
RAI 4.7.1-1
RAI 4.7.1-2
RAI 4.7.1-3

RAI B.2.1.19-1a

Applicability:

Byron Station (Byron) and Braidwood (Station), all units

Background:

By letter dated December 12, 2013, the staff issued a request for additional information (RAI) titled, "RAI B.2.1.19-1," requesting an updated surveillance capsule withdrawal schedule for each unit "including, but not limited to: identification of the capsule and associated neutron fluence value which will provide test results consistent with the [Generic Aging Lessons Learned] GALL Report recommendation of a neutron fluence exposure of between one and two times the peak reactor vessel wall neutron fluence at the end of the period of extended operation, and identification of a date for the submittal of each summary technical report." In its response dated January 13, 2014, the applicant stated that each technical summary report for the next surveillance capsule testing "will be submitted to the NRC prior to entering the associated period of extended operation." Currently, each unit of Byron and Braidwood stores the untested surveillance capsules in the spent fuel pool for future use.

Per Appendix H of Title 10 of the Code of Federal Regulations (10 CFR) Part 50, "Each capsule withdrawal and the test results must be the subject of a summary technical report to be submitted…within one year of the date of capsule withdrawal, unless an extension is granted by the Director, Office of Nuclear Reactor Regulation." The Byron and Braidwood Pressure-Temperature Limits Reports (PTLRs) include Tables for surveillance capsule withdrawal schedules and state that "surveillance capsule testing has been completed for the original operating period. Other capsules will be removed to avoid excessive fluence accumulation should they be needed to support life extension." These surveillance capsule withdrawal schedules are no longer applicable beyond the original operating period.

Issue:

In its response, the applicant did not clearly address the withdrawal dates and summary technical report submittal dates. The surveillance capsules have already received neutron fluence exposures of 1-2 times the projected neutron fluence values at the end of the period of extended operation and have been withdrawn from the reactor vessel and moved to the spent fuel pool. Since the current surveillance capsule withdrawal schedule is valid for the current operating period, the staff considers the initiation of a new surveillance capsule withdrawal schedule to be necessary for the period of extended operation. Upon receiving a renewed operating license, the surveillance capsules, identified in Table 1 of the applicant's response dated January 13, 2014, would no longer be considered standby capsules; instead, they would be considered part of the program to meet the GALL Report and 10 CFR Part 50, Appendix H, requirements. Capsules should be tested and summary reports submitted within 1 year of receiving the renewed license, unless Byron and Braidwood submits a request for extension for approval by the Director, Office of Nuclear Reactor Regulation, within this period.

Request:

For each surveillance capsule identified in Table 1 of the applicant's response dated January 13, 2014, provide the withdrawal date and expected date of submittal of the summary technical report. A request for extension must be submitted for approval by the Director, Office of Nuclear Reactor Regulation, if the expected date for the submittal of the summary technical report exceeds 1 year from the date of capsule withdrawal.

Exelon Response:

Exelon understands that upon receiving a renewed operating license, the surveillance capsules, identified in Table 1 of our response dated January 13, 2014, would no longer be considered standby capsules; instead, they would be considered part of the Reactor Vessel Surveillance program to meet the NUREG-1801, Revision 2, GALL Report guidelines and the 10 CFR Part 50, Appendix H requirements. Since the capsules were previously withdrawn, the date of the issuance of the renewed license establishes the date of capsule withdrawal. Exelon also acknowledges the requirement to comply with 10 CFR 50 Appendix H, section IV.A which states; "Each capsule withdrawal and the test results must be the subject of a summary technical report to be submitted, as specified in §50.4, within one year of the date of capsule withdrawal, unless an extension is granted by the Director, Office of Nuclear Reactor Regulation."

Since the testing of the capsules is being controlled by 10 CFR 50 Appendix H, the Reactor Vessel Surveillance aging management program enhancements and commitments specified in our response dated January 13, 2014 are no longer required and are being withdrawn.

LRA Appendix A, Section A.2.1.19, Appendix B, Section B.2.1.19, and Appendix A.5, commitment 19, are revised as shown in Enclosures B and C.

RAI 4.7.1-1

Applicability:

Byron and Braidwood

Background:

Per 10 CFR Part 50, Criterion 4 of Appendix A, "General Design Criteria for Nuclear Power Plants" (GDC-4), systems, structures, and components (SSCs) important to safety are required to be appropriately protected against dynamic effects associated with postulated pipe ruptures, unless analyses reviewed and approved by the Commission demonstrate that the probability of rupture is extremely low under conditions consistent with the design basis for the piping. An approved leak-before-break analysis permits a licensee to remove protective hardware such as pipe whip restraints and jet impingement barriers; redesign pipe connected components, their supports, and their internals; and other related changes. License renewal application (LRA) Section 4.7.1 describes the applicant's time limited aging analyses (TLAA) evaluation for the Byron and Braidwood leak-before-break analyses. The LRA states that the applicant updated the existing leak-before-break analysis for the reactor coolant primary loop piping and concludes that the updated analysis meets the requirements of 10 CFR 54.21(c)(1)(ii).

Issue:

To meet the requirements of 10 CFR 54.21(c)(1)(ii), the applicant must demonstrate that its updated leak-before-break analysis, which has been projected to the end of the period of extended operation, satisfies the requirements of GDC-4. The LRA provides a general description of how the applicant updated the leak-before-break analysis for the reactor coolant primary loop piping. However, the LRA does not clearly identify the methodology used for the updated analysis, nor does it contain a sufficient level of technical detail for the NRC staff to confirm that the updated analysis complies with GDC-4.

Request:

Provide for the NRC staff review and approval the full update to the leak-before-break analyses for the reactor coolant primary loop piping. The submitted analysis should contain a sufficient level of technical information to demonstrate compliance with the GDC-4 requirements for extremely low probability of rupture. A sufficient level of technical information would address items 1 through 11 from NUREG-0800, "Standard Review Plan," Section 3.6.3, "Leak-Before-Break Evaluation Procedures," Subsection III, dated March 2007. Otherwise, provide the rationale for not submitting a full update to the leak-before-break analysis.

Exelon Response:

The existing leak-before-break (LBB) analysis for the Byron and Braidwood Stations' reactor coolant primary loop piping was submitted to the NRC on April 30th, 1996. This document, WCAP-14559, "Technical Justification for Eliminating Large Primary Loop Pipe Rupture as the Structural Design Basis for the Byron and Braidwood Units 1 and 2", Revision 1, was approved by the NRC for Byron and Braidwood Stations, Units 1 and 2, in a safety evaluation letter dated

October 25,1996 (ADAMS Accession Number 9610290229). The analysis in WCAP-14559 used the NUREG-0800, "Standard Review Plan," Section 3.6.3, "Leak-Before-Break Evaluation Procedures," Subsection III [Review Procedures], dated 1987 to satisfy the GDC-4 requirements.

The license renewal LBB analysis used the newer NRC-approved criteria contained in NUREG-0800, "Standard Review Plan," Section 3.6.3, "Leak-Before-Break Evaluation Procedures," Subsection III [Review Procedures], dated March 2007. In addition, the analysis used the methods to determine loss of fracture toughness contained in NUREG/CR-4513, "Estimation of Fracture Toughness of Cast Stainless Steels During Thermal Aging in LWR Systems", Revision 1, dated May 1994. This license renewal LBB analysis was performed to reflect the differences in fracture toughness calculations to the saturated condition applicable for 60 years, and confirmation of the design transients and cycles for the fatigue crack growth analysis. The license renewal LBB analysis addressed the revisions to the Standard Review Plan for the leak-before-break evaluation. The margins and conclusions from the existing LBB analysis were maintained in the license renewal LBB analysis. The criteria in the current Standard Review Plan have been met, demonstrating compliance with GDC-4.

<u>Differences</u>

Standard Review Plan:
The major difference in the Standard Review Plans between 1987 and 2007 was the addition of the criterion for material susceptibility to corrosion. More specifically, the later version required the evaluation of the material susceptibility to corrosion, the potential for high residual stresses, and environmental conditions – all which could lead to degradation by primary water stress corrosion cracking. Primary water stress corrosion cracking (PWSCC) is considered to be a degradation mechanism in Alloy 82/182. The Byron and Braidwood Units have Alloy 82/182 welds in the reactor coolant loops. Therefore, Byron and Braidwood addressed the stress corrosion cracking of the Alloy 82/182 with the implementation of stress relief on the hot and cold leg reactor vessel connections. The Mechanical Stress Improvement Process (MSIP) has been completed for the Byron and Braidwood units, which has mitigated the potential for PWSCC of Alloy 82/182. Since the PWSCC degradation mechanism has been mitigated by the MSIP, the leak-before-break methodology was applied to the reactor vessel hot and cold leg connections. The results of this evaluation are discussed below.

Leak-Before-Break Analysis:
The primary difference between the existing LBB analysis and the license renewal LBB analysis that was performed for the Byron and Braidwood stations was the calculation method for the fracture toughness. When the Westinghouse evaluation (WCAP-14559) was performed in April 1996 to support the Commonwealth Edison submittal for the leak-before-break evaluation, there were several acceptable methods of calculating the fracture toughness of materials. The existing LBB analysis covered the current period of operation up to 40 years of service. The method to perform these fracture toughness calculations in the existing Westinghouse analysis was delineated in WCAP-10931, "Toughness Criteria for Thermally Aged Cast Stainless Steel, dated July 1986." The license renewal LBB analysis used the methodology contained in NUREG/CR-4513 to compute fracture toughness properties. In NUREG/CR-4513, some analytical outputs (e.g., J_{IC}, J_{max}, T_{mat}) differ due to the change in fracture toughness calculation methods. The results of this analysis are discussed below.

Results

The existing LBB analysis performed for the leak-before-break evaluation did not account for PWSCC, since it was not recognized as a concern with the Alloy 82/182 weld material at the time. However, the license renewal LBB analysis considered PWSCC at the hot and cold leg weld locations with these materials. The input data for the affected locations (reactor vessel cold and hot leg safe end welds) was used to determine the critical and leakage flaw sizes with the Standard Review Plan (2007) LBB methodology. The analysis determined that the 10-gpm leakage flaw size was greater than ten (10) inches for all Alloy 82/182 weld material. The critical flaw sizes of all the analyzed welds were in excess of 38 inches and satisfied all the leak-before-break margin criteria.

The license renewal LBB analysis was developed using the most current NRC guidance including the guidance in NUREG/CR-4513. The critical locations for the reactor coolant system piping were determined using two different methods as specified in the current Standard Review Plan: load critical location and toughness critical location. These critical locations were on the hot leg elbow fitting into the steam generator and the cold leg discharge piping of the reactor coolant pump. Flaw sizes (10-gpm) were then determined for these critical locations. In both the 1996 and the 2013 LBB analyses, the critical flaw and leakage flaw sizes at both locations were the same. Therefore, the limiting flaw sizes and locations in the license renewal LBB analysis have not changed from the existing LBB analysis performed in 1996. Although the absolute values of the elastic-plastic J-integral changed due to the differences in the fracture toughness calculation methods, the fracture stability criteria were satisfied for the calculated fracture toughness and tearing modulus at the same toughness critical location.

The results of the existing LBB analysis and the license renewal LBB analysis are very similar and both met the acceptance criteria in the associated Standard Review Plan. The SRP-required factor-of-10 margin still exists between the calculated leak rate from the leakage flaw and the plant leakage detection capability of one (1) gpm. For the license renewal LBB analysis at the critical locations, the leakage flaw was shown to be stable using the faulted loads obtained by the absolute sum method, and a factor-of-one (1) margin is satisfied on loads using the absolute summation of faulted load combinations. Since the 40-year design transients and cycles bounded the 60-year projected cycles, the 40-year design transients and cycles were used for the fatigue crack growth evaluation. The results of the fatigue crack growth evaluation were found to be acceptable.

Conclusion

The original margins and conclusions were maintained in the license renewal LBB analysis as reflected in the license renewal application. These margins and conclusions had been previously approved in the existing LBB analysis by the NRC in 1996. The license renewal LBB analysis is consistent with the current methodology in the Standard Review Plan. The criteria in the current Standard Review Plan have been met, demonstrating compliance with GDC-4. This response summarizes the full update to the LBB analysis.

RAI 4.7.1-2

Applicability:

Byron and Braidwood

Background:

LRA Section 4.7.1 describes the applicant's TLAA evaluation for the Byron and Braidwood leak-before-break analyses. For the safety injection accumulator piping and the reactor coolant bypass piping, the LRA states that the existing loads from Sargent and Lundy Report SL-4518 will still govern in the period of extended operation. Therefore, the LRA concludes that the analyses from this report remain valid for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(i).

Issue:
LRA Section 4.7.1 does not explicitly identify the time-dependent loads or other parameters from Sargent and Lundy Report SL-4518 that are applicable to the safety injection accumulator piping and the reactor coolant bypass piping. In addition, the LRA does not demonstrate how these time-dependent parameters remain valid for the period of extended operation.

Request:

From Sargent and Lundy Report SL-4518, specifically identify all of the time-dependent parameters used in the leak-before-break analyses for the safety injection accumulator piping and the reactor coolant bypass piping. Justify why each of these time-dependent parameters remain valid for the period of extended operation.

Exelon Response:

The analysis method used in the Sargent and Lundy Report SL-4518 for the leak-before-break assessment is similar to the modified load limit analysis methodology described in NUREG-0800, SRP 3.6.3. The parameters used as inputs to the modified limit load analysis consist of material properties, dimensions of the piping, and the piping loads. The loads are inputs to the limit load analysis methodology for excluding the dynamic effects of postulated ruptures in system piping. The piping loads were used to determine the actual stress index at the bounding locations. Upon further review of the six (6) criteria which are used to determine if a TLAA exists, it has been determined that none of these parameters are time dependent. Since the modified limit load analysis does not rely on time dependent parameters, the leak-before-break analysis for the safety injection accumulator piping and the reactor coolant bypass piping is not considered a TLAA. Therefore, the LRA is revised as indicated in Enclosure B to remove consideration of this analysis as a TLAA.

RAI 4.7.1-3

Applicability:

Byron and Braidwood

Background:

GDC-4 requires SSCs important to safety are required to be appropriately protected against dynamic effects associated with postulated pipe ruptures, unless analyses reviewed and approved by the Commission demonstrate that the probability of rupture is extremely low under conditions consistent with the design basis for the piping. An approved leak-before-break analysis permits a licensee to remove protective hardware such as pipe whip restraints and jet impingement barriers, redesign pipe connected components, their supports, and their internals, and other related changes. LRA Section 4.7.1 describes the applicant's TLAA evaluation for the Byron and Braidwood leak-before-break analysis for the safety injection accumulator piping cold leg nozzles, which are made of cast austenitic stainless steel (CASS). Because this material is susceptible to the effects of thermal aging, the LRA states that the applicant determined the fracture toughness properties for the materials at the fully aged condition applicable to the period of extended operation, and it used these properties to update the existing leak-before-break analysis. The LRA concludes that the updated analysis meets the requirements of 10 CFR 54.21(c)(1)(ii).

Issue:

To meet the requirements of 10 CFR 54.21(c)(1)(ii), the applicant must demonstrate that its updated leak-before-break analysis, which has been projected to the end of the period of extended operation, satisfies the requirements of GDC-4. The LRA does not clearly identify the methodology used for the leak-before-break analysis for the safety injection accumulator piping cold leg nozzles, nor does it contain a sufficient level of technical detail for the NRC staff to confirm that the updated analysis complies with GDC-4.

Request:

1. Provide for the NRC staff review and approval the full update to the leak-before-break analysis for the safety injection accumulator piping cold leg nozzles. The submitted analysis should contain a sufficient level of technical information to demonstrate compliance with the GDC-4 requirements for extremely low probability of rupture. A sufficient level of technical information would address items 1 through 11 from NUREG-0800, "Standard Review Plan," Section 3.6.3, "Leak-Before-Break Evaluation Procedures," Subsection III, dated March 2007. Otherwise, provide the rationale for not submitting a full update to the leak-before-break analysis.

2. Identify and provide justification for the methodology that was used to determine the CASS fracture toughness at the end of the period of extended operation.

Exelon Response:

1. The existing leak-before-break (LBB) analysis for Byron and Braidwood Stations' safety injection accumulator piping cold leg nozzles and reactor coolant bypass piping was performed by Sargent and Lundy on May 12th, 1989. This analysis (SL-4518, "Leak-Before-Break Evaluation for Stainless Steel Piping for Byron and Braidwood Nuclear Power Stations Units 1 and 2") was approved by the NRC for Byron and Braidwood Stations, Units 1 and 2, in a safety evaluation letter dated April 19th, 1991 (ADAMS Accession Number 9104290051). This analysis used the NUREG-0800, "Standard Review Plan," Section 3.6.3, "Leak-Before-Break Evaluation Procedures," Subsection III [Review Procedures], dated 1987 to satisfy the GDC-4 requirements. The discussion below focuses on the safety injection accumulator piping cold leg nozzles portion of the analysis.

   The license renewal LBB analysis used the newer NRC-approved criteria contained in NUREG-0800, "Standard Review Plan," Section 3.6.3, "Leak-Before-Break Evaluation Procedures," Subsection III [Review Procedures], dated March 2007. In addition, the license renewal LBB analysis used the methods to determine loss of fracture toughness contained in NUREG/CR-4513, "Estimation of Fracture Toughness of Cast Stainless Steels During Thermal Aging in LWR Systems", Revision 1, dated May 1994. This license renewal LBB analysis evaluated the differences in fracture toughness calculations to the saturated condition applicable to 60 years. The changes between the two versions of the Standard Review Plans and fracture toughness calculation methods did not affect the conclusion that the license renewal LBB analysis is consistent with the current Standard Review Plan methodology. The criteria in the current Standard Review Plan have been met, demonstrating compliance with GDC-4.

   The major difference in the Standard Review Plans between 1987 and 2007 was the addition of the criterion for material susceptibility to corrosion. More specifically, the later version required the evaluation of the material susceptibility to corrosion, the potential for high residual stresses, and the impact of environmental conditions – all which could lead to degradation by stress corrosion cracking. Primary water stress corrosion cracking (PWSCC) is considered a degradation mechanism in Alloy 82/182, which is applicable to the Byron and Braidwood Units for the hot and cold leg reactor vessel nozzles. These results were discussed in RAI 4.7.1-1 as part of the reactor coolant system piping, and the discussion of the Alloy 82/182 welds is not applicable to the safety injection accumulator piping cold leg nozzle.

   The primary difference between the two analyses for the safety injection accumulator piping cold leg nozzle is the calculation of the fracture toughness for the safety injection system cast austenitic stainless steel (CASS) nozzles. The existing LBB analysis from 1989 used the NRC-approved database (PIFRAC) to determine the fracture toughness properties (J-T curves) for the safety injection CASS nozzles. The values in this analysis were based on material properties from generic material heats for cast austenitic stainless steel. The license renewal LBB analysis used the plant-specific certified material test reports for the safety injection accumulator cold leg nozzles. The updated fracture toughness properties were determined in accordance with the methodology in NUREG/CR-4513. Based on the plant-specific LBB evaluation, the critical flaw size to the postulated 10-gpm circumferential leakage flaw size ratio is 3.30 based on limit load criterion of greater than 2.00. Additionally,

the plant-specific J-Integral values in the license renewal LBB analysis (e.g., J_{IC}, J_{max}, T_{mat}) maintained the fracture criteria for stability with significant margin.

There are some differences in the calculation methods for fracture toughness due to the more recent NRC NUREG/CR-4513. The absolute values of the fracture toughness properties changed in the license renewal LBB analysis, but the fracture mechanics evaluations are consistent with the original margins and conclusions as reflected in the license renewal application. The license renewal LBB analysis satisfies items 1 through 11 of the current Standard Review Plan 3.6.3 (2007), demonstrating compliance with GDC-4. This response summarizes the full update to the LBB analysis.

2. The license renewal LBB analysis methodology for Byron and Braidwood to determine the fracture toughness values of the safety injection CASS nozzles used plant-specific certified material test reports and the NRC guidance in NUREG/CR-4513. The original margins and conclusions for the fracture mechanic evaluations that were determined in the existing LBB analysis remain valid in the license renewal LBB analysis for the period of extended operation.

Enclosure B

Byron and Braidwood Stations, Units 1 and 2

License Renewal Application Updates resulting from the responses to the following RAIs:

RAI B.2.1.19-1a
RAI 4.7.1-2

Note: To facilitate understanding, portions of the original LRA, as modified by previous RAI responses, have been repeated in this Enclosure, with revisions indicated. Existing LRA text is shown in normal font. Changes are highlighted with ***bolded italics*** for inserted text and ~~strikethroughs~~ for deleted text.

As a result of changes to the Reactor Vessel Surveillance aging management program identified in the response to RAI B.2.1.19-1a, LRA Appendix A, Section A.2.1.19, page A-25 is revised as shown below. Revisions are indicated with ***bolded italics*** for inserted text and ~~strikethroughs~~ for deleted text:

A.2.1.19 Reactor Vessel Surveillance

The Reactor Vessel Surveillance aging management program is an existing condition monitoring program that extends the scope of 10 CFR Part 50, Appendix H, "Reactor Vessel Material Surveillance Program Requirements." The program provides sufficient material and dosimetry data to monitor loss of fracture toughness due to neutron irradiation embrittlement until the end of the period of extended operation, and determine the need for operating restrictions on the irradiation temperature (i.e., cold leg operating temperature), neutron spectrum, and neutron fluence. There were six (6) specimen capsules installed in each Byron and Braidwood Station (BBS) reactor pressure vessel (RPV) prior to plant start-up. The capsules contain representative RPV material specimens, neutron dosimeters, and thermal monitors (eutectic alloy). All six (6) specimen capsules have been withdrawn from each of the BBS RPVs. Three (3) specimen capsules from each RPV were tested and the remaining three (3) untested specimen capsules from each RPV are currently stored in the spent fuel pool. Of the three (3) untested specimen capsules from each RPV at least one (1) untested specimen capsule has been irradiated in excess of the projected peak neutron fluence of the associated RPV at the end of the period of extended operation. Capsules that have been withdrawn will be tested as necessary to fulfill the surveillance capsule recommendations contained in ASTM 185-82 as required by 10 CFR Part 50, Appendix H. Operating restrictions will be established to ensure that the plant is operated under the conditions to which the surveillance capsules were exposed. All capsules tested for the period of extended operation will meet the test procedures and reporting requirements of ASTM E 185-82, "Standard Practice for Conducting Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessels" to the extent practicable for the configuration of the specimens in the capsule. Any changes to the capsule withdrawal schedule, including spare capsules, must be approved by the NRC prior to implementation. Untested capsules placed in storage must be maintained for possible future insertion.

The program also monitors plant operating conditions to ensure appropriate steps are taken if reactor vessel exposure conditions are altered, such as the review and updating of 60-year fluence projections to support upper shelf energy calculations and pressure-temperature limit curves. The program also includes condition monitoring by removal and analysis of ex-core neutron dosimetry sensor sets to validate neutron exposure projection calculations through the period of extended operation in accordance with Regulatory Guide 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence." These measures are effective in monitoring the extent of neutron irradiation embrittlement to prevent significant degradation of the reactor pressure vessel during the period of extended operation.

The Reactor Vessel Surveillance aging management program will be enhanced to:

1. Establish operating restrictions to ensure that the plant is operated under the conditions to which the surveillance capsules were exposed. The operating restrictions are as follows:

Byron Station, Unit 1:

- Cold leg operating temperature limitation: 525 degrees Fahrenheit (minimum) to 590 degrees Fahrenheit (maximum)
- RPV beltline material fluence: 3.21E+19 n/cm2 (E >1.0 MeV) (maximum)

Byron Station, Unit 2; Braidwood Station Unit 1:

- Cold leg operating temperature limitation: 525 degrees Fahrenheit (minimum) to 590 degrees Fahrenheit (maximum)
- RPV beltline material fluence: 3.19E+19 n/cm2 (E >1.0 MeV) (maximum)

Braidwood Station Unit 2:

- Cold leg operating temperature limitation: 525 degrees Fahrenheit (minimum) to 590 degrees Fahrenheit (maximum)
- RPV beltline material fluence: 3.16E+19 n/cm2 (E >1.0 MeV) (maximum)

If the reactor pressure vessel exposure conditions (neutron fluence, neutron spectrum) or irradiation temperature (cold leg inlet temperature) are altered, then the basis for the projection to the end of the period of extended operation needs to be reviewed and, if deemed appropriate, updates are made to the Reactor Vessel Surveillance program. Any changes to the Reactor Vessel Surveillance program must be submitted for NRC review and approval in accordance with 10 CFR Part 50, Appendix H.

~~2. One (1) specimen capsule per reactor vessel, as designated below, irradiated to a neutron fluence of one (1) to two (2) times the projected peak neutron fluence at the end of the period of extended operation will be withdrawn from the spent fuel pool, tested, and the summary technical report submitted to the NRC prior to entering the associated period of extended operation.~~

~~Reactor Vessel (Station , Unit)~~	~~Capsule ID~~	~~Capsule Fluence (n/cm^2)(E>1.0 MeV)~~
~~Byron, Unit 1~~	~~Y~~	~~3.97E+19~~
~~Byron, Unit 2~~	~~Y~~	~~4.19E+19~~
~~Braidwood, Unit 1~~	~~V~~	~~3.71E+19~~
~~Braidwood, Unit 2~~	~~V~~	~~3.73E+19~~

Th***is***~~ese~~ enhancement~~s~~ will be implemented prior to the period of extended operation.

As a result of changes to the Reactor Vessel Surveillance aging management program identified in the response to RAI B.2.1.19-1a, LRA Appendix B, Section B.2.1.19, pages B-127 through B-128, the Enhancements subsection is revised as shown below. Revisions are indicated with ~~strikethroughs~~ for deleted text:

Enhancements

Prior to the period of extended operation, the following enhancements will be implemented in the following program elements:

1. Establish operating restrictions to ensure that the plant is operated under the conditions to which the surveillance capsules were exposed. The operating restrictions are as follows:

   Byron Station, Unit 1:

   – Cold leg operating temperature limitation: 525 degrees Fahrenheit (minimum) to 590 degrees Fahrenheit (maximum)

   – RPV beltline material fluence: 3.21E+19 n/cm2 (E >1.0 MeV) (maximum)

   Byron Station, Unit 2; Braidwood Station, Unit 1:

   – Cold leg operating temperature limitation: 525 degrees Fahrenheit (minimum) to 590 degrees Fahrenheit (maximum)

   – RPV beltline material fluence: 3.19E+19 n/cm2 (E >1.0 MeV) (maximum)

   Braidwood Station, Unit 2:

   – Cold leg operating temperature limitation: 525 degrees Fahrenheit (minimum) to 590 degrees Fahrenheit (maximum)

   – RPV beltline material fluence: 3.16E+19 n/cm2 (E >1.0 MeV) (maximum)

   If the reactor pressure vessel exposure conditions (neutron fluence, neutron spectrum) or irradiation temperature (cold leg inlet temperature) are altered, then the basis for the projection to the end of the period of extended operation needs to be reviewed and, if deemed appropriate, updates are made to the Reactor Vessel Surveillance program. Any changes to the Reactor Vessel Surveillance program must be submitted for NRC review and approval in accordance with 10 CFR Part 50, Appendix H. **Program Elements Affected: Parameters Monitored/Inspected (Element 3), Detection of Aging Effects (Element 4), Monitoring and Trending (Element 5), Acceptance Criteria (Element 6)**

2. ~~One (1) specimen capsule per reactor vessel, as designated below, irradiated to a neutron fluence of one (1) to two (2) times the projected peak neutron fluence at the end of the period of extended operation will be withdrawn from the spent fuel pool, tested, and the summary technical report submitted to the NRC prior to entering the associated period of extended operation.~~

~~Reactor Vessel~~	~~Capsule ID~~	~~Capsule Fluence (n/cm^2)(E>1.0 MeV)~~
~~Byron, Unit 1~~	~~Y~~	~~3.97E+19~~
~~Byron, Unit 2~~	~~Y~~	~~4.19E+19~~
~~Braidwood, Unit 1~~	~~V~~	~~3.71E+19~~
~~Braidwood, Unit 2~~	~~V~~	~~3.73E+19~~

~~Program Elements Affected: Parameters Monitored/Inspected (Element 3), Detection of Aging Effects (Element 4), Monitoring and Trending (Element 5), Acceptance Criteria (Element 6)~~

As a result of the response to RAI 4.7.1-2, LRA Section 4.1.4, Table 4.1-2, on page 4.1-7, is revised as shown below. Revisions are indicated with ~~strikethroughs~~ for deleted text:

Table 4.1-2 **SUMMARY OF RESULTS – BBS TIME-LIMITED AGING ANALYSES**		
TLAA DESCRIPTION	**DISPOSITION**	**LRA SECTION**
ENVIRONMENTAL QUALIFICATION (EQ) OF ELECTRIC COMPONENTS	§54.21(c)(1)(iii)	4.4
CONCRETE CONTAINMENT TENDON PRESTRESS ANALYSIS	§54.21(c)(1)(iii)	4.5
CONTAINMENT LINER PLATE, METAL CONTAINMENTS, AND PENETRATIONS FATIGUE ANALYSES		4.6
Containment Liner Plates Fatigue	§54.21(c)(1)(iii)	4.6.1
Containment Airlocks and Hatches Fatigue	§54.21(c)(1)(iii)	4.6.2
Containment Electrical Penetrations Fatigue	§54.21(c)(1)(iii)	4.6.3
Containment Piping Penetrations Fatigue	§54.21(c)(1)(iii)	4.6.4
Fuel Transfer Tube Bellows Fatigue	§54.21(c)(1)(iii)	4.6.5
Recirculation Sump Guard Piping Bellows Fatigue	§54.21(c)(1)(iii)	4.6.6
OTHER PLANT-SPECIFIC TIME-LIMITED AGING ANALYSES		4.7
Leak-Before-Break	~~§54.21(c)(1)(i) and~~ §54.21(c)(1)(ii)	4.7.1
Crane Load Cycle Limits	§54.21(c)(1)(i)	4.7.2
Mechanical Environmental Qualification	§54.21(c)(1)(iii)	4.7.3
Residual Heat Removal Heat Exchangers Tube Side Inlet and Outlet Nozzles Fracture Mechanics Analysis	§54.21(c)(1)(iii)	4.7.4
Reactor Coolant Pump Flywheel Crack Growth Analysis	§54.21(c)(1)(i)	4.7.5
Byron Unit 2 Pressurizer Seismic Restraint Lug Flaw Evaluation	§54.21(c)(1)(iii)	4.7.6
Braidwood Unit 2 Feedwater Pipe Elbow Crack Growth Evaluation	§54.21(c)(1)(iii)	4.7.7
Analyses Supporting Flaw Evaluations of Primary	§54.21(c)(1)(i)	4.7.8

System Components	and §54.21(c)(1)(iii)	

As a result of the response to RAI 4.7.1-2, the Safety Injection Accumulator Piping and Reactor Coolant Bypass Piping subsection of LRA Section 4.7.1, page 4.7-2, is revised as shown below. Revisions are indicated with ~~strikethroughs~~ for deleted text:

~~Safety Injection Accumulator Piping and Reactor Coolant Bypass Piping~~

~~**TLAA Evaluation:**~~
~~Leak-Before-Break (LBB) analyses of Byron and Braidwood Units, 1 and 2 safety injection (SI) accumulator piping and reactor coolant bypass piping were performed in 1989. The results of these analyses were documented and accepted by the NRC (Reference 4.8.16).~~

~~A review of the current calculation packages of record for Byron and Braidwood, Units 1 and 2 was performed to obtain the latest piping loads on these piping systems. Based on the reviews, it was determined that the loads used in the original analysis are still governing. Therefore, LBB analysis results for the Byron and Braidwood, Units 1 and 2 safety injection accumulator piping and reactor coolant bypass piping remain valid for the period of extended operation.~~

~~**TLAA Disposition: 10 CFR 54.21(c)(1)(i)** – The Safety Injection Accumulator Piping and Reactor Coolant Bypass Piping LBB analyses for BBS Units 1 and 2 remain valid for the period of extended operation.~~

As a result of the response to RAI 4.7.1-2, LRA Section 4.8, page 4.8-2 is revised as shown below. Revisions are indicated with ~~strikethroughs~~ for deleted text and ***bolded italics*** for added text:

4.8.14. Westinghouse Electric Company Document WCAP-14559, Revision 1, "Technical Justification for Eliminating Large Primary Loop Piping Rupture as the Structural Design Basis for the Byron and Braidwood Units 1 and 2 Nuclear Power Plants."

4.8.15. NRC Letter to Commonwealth Edison, dated October 25, 1996, "Safety Evaluation (SE) Regarding Leak-Before-Break Analysis – Byron Stations, Units 1 and 2, and Braidwood Station, Units 1 and 2 (TAC Nos. M95342, M95343, M95344, and M95345)."

4.8.16. ~~NRC Letter to Commonwealth Edison dated April 19, 1991, "Safety Evaluation of Leak-Before-Break Methodology Applicable to Accumulator Piping and Reactor Coolant Bypass Piping (TAC Nos. 73306, 73307, 73308, and 73309)."~~ ***Not used***

4.8.17. Westinghouse Electric Company Document WCAP-14422, Revision 2A, "License Renewal Evaluation: Aging Management for Reactor Coolant system Supports," December 2000.

4.8.18. NUREG-0588, "Interim Staff Position on Environmental qualification of Safety-Related Electrical Equipment," July 1981.

4.8.19. Regulatory Guide 1.89, Revision 1, "Environmental Qualification of Certain electric Equipment Important to Safety for Nuclear Power Plants," U.S. Nuclear Regulatory Commission, Office of Nuclear Regulatory Research, June 1984.

4.8.20. Exelon, Byron and Braidwood Units 1 and 2 Updated Final Safety Analysis Report (UFSAR), Revision 14.

4.8.21. Regulatory Guide 1.35.1, "Determining Prestressing Forces for Inspection of Prestress Concrete Containments," July 1990.

4.8.22. NRC Information Notice 99-10, "Degradation of Prestressing Tendon Systems in Prestressed Concrete Containments;" Attachment 3, "Comparison and Trending of Prestressing Forces," April 13, 1999.

4.8.23. NUREG/CR-4513, Revision 1, "Estimation of Fracture Toughness of Cast Stainless Steels during Thermal Aging in LWR Systems," U.S. Nuclear Regulatory Commission, August 1994.

4.8.24. IEEE 323-1974, "Qualifying Class 1E Equipment for Nuclear Power Generating Stations," The Institute of Electrical and Electronics Engineers, Inc., 1974.

4.8.25. Commonwealth Edison Letter to Office of Nuclear Reactor Regulation dated November 13, 1991, "Braidwood Unit 2 Flaw Evaluation Report for RHR Heat Exchanger Nozzle to Shell Welds."

4.8.26. NRC Letter to Commonwealth Edison, dated November 21, 1991, “Residual Heat Removal Heat Exchanger Nozzle to Shell Welds (TAC NO. M82087).”

4.8.27. Westinghouse Electric Company Document WCAP-11063, "Handbook on Flaw Evaluations For Byron Unit 1 and 2 Steam Generators and Pressurizers,” March 1986.

As a result of the response to RAI 4.7.1-2, LRA Section A.4.7.1, page A-64, is revised as shown below. Revisions are indicated with ~~strikethroughs~~ for deleted text:

A.4.7 Other Plant-Specific Time-Limited Aging Analyses

A.4.7.1 Leak-Before-Break

Appendix A, Criterion 4, of 10 CFR 50 allows for the use of leak-before-break (LBB) methodology for excluding the dynamic effects of postulated ruptures in reactor coolant system piping. The fundamental premise of the LBB methodology is that the materials used in nuclear power plant piping are sufficiently tough that even a large through-wall crack would remain stable and would not result in a double-ended pipe rupture. Application of the LBB methodology is limited to those high-energy fluid systems not considered to be overly susceptible to failure from such mechanisms as corrosion, water hammer, fatigue, thermal aging or indirectly from such causes as missile damage or the failure of nearby components. The analyses associated with LBB have been identified as TLAAs.

Original LBB analyses performed for Byron and Braidwood Stations, Units 1 and 2, demonstrated that postulated breaks can be eliminated from the structural design basis in the reactor coolant primary loop piping, safety injection accumulator piping and cold leg nozzles, and reactor coolant bypass piping. The reactor coolant primary loop piping includes cast austenitic stainless steel (CASS) elbows, and the safety injection accumulator piping cold leg nozzles are also fabricated from CASS material. The LBB analyses for these systems were updated for license renewal to consider the effects of additional thermal aging on the fracture toughness of the CASS materials through the period of extended operation. The fracture toughness properties used were based on the fully-aged condition (that has the lowest possible fracture toughness), which is applicable for the period of extended operation. The updated LBB analyses demonstrate that the dynamic effects of the pipe rupture resulting from postulated breaks in the reactor coolant primary loop piping and safety injection accumulator piping cold leg nozzles need not be considered in the structural design basis for BBS Units 1 and 2 for the license renewal period. The analyses have been projected to the end of the period of extended operation in accordance with 10 CFR 54.21(c)(1)(ii).

~~The safety injection accumulator piping (excluding the cold leg nozzle) and the reactor coolant bypass piping also have LBB analyses that were identified as TLAAs. However, these piping systems do not include CASS materials. A review was performed of the current calculation packages of record for Byron and Braidwood, Units 1 and 2, to obtain the latest piping loads on these systems. It was determined that the loads used in the original analyses are still governing. Therefore, the LBB analyses for the safety injection accumulator piping (excluding the cold leg nozzle) and reactor coolant bypass piping, remain valid for the period of extended operation, in accordance with 10 CFR 54.21(c)(1)(i).~~

Enclosure C

Byron and Braidwood Stations, Units 1 and 2
License Renewal Commitment List Changes

This Enclosure includes an update to the Byron and Braidwood Stations, Units 1 and 2, License Renewal Application (LRA) Appendix A, Section A.5 License Renewal Commitment List, as a result of the Exelon response to the following RAI:

RAI B.2.1.19-1a

Note: For clarity, portions of the original LRA License Renewal Commitment List, as modified by previous RAI responses, are repeated in this enclosure. Changes are highlighted with ~~strikethroughs~~ for deleted text and ***bolded italics*** for added text.

As a result of the response to RAI B.2.1.19-1a provided in Enclosure A of this letter, LRA Appendix A, Table A.5 License Renewal Commitment List, line item 19 on pages A-78 and A-79, is revised as shown below. The RAI that led to this commitment modification is listed in the "SOURCE" column. Any other actions described in this submittal represent intended or planned actions. They are described for the NRC's information and are not regulatory commitments.

A.5 License Renewal Commitment List

NO.	PROGRAM OR TOPIC	COMMITMENT	IMPLEMENTATION SCHEDULE	SOURCE
19	Reactor Vessel Surveillance	Reactor Vessel Surveillance is an existing program that will be enhanced to: 1. Establish operating restrictions to ensure that the plant is operated under the conditions to which the surveillance capsules were exposed. The operating restrictions are as follows: Byron Station, Unit 1: - Cold leg operating temperature limitation: 525 degrees Fahrenheit (minimum) to 590 degrees Fahrenheit (maximum). - RPV beltline material fluence: 3.21E+19 n/cm2 (E >1.0 MeV) (maximum). Byron Station, Unit 2; Braidwood Station Unit 1: - Cold leg operating temperature limitation: 525 degrees Fahrenheit (minimum) to 590 degrees Fahrenheit (maximum). - RPV beltline material fluence: 3.19E+19 n/cm2 (E >1.0 MeV) (maximum). Braidwood Station Unit 2: - Cold leg operating temperature limitation: 525 degrees Fahrenheit (minimum) to 590 degrees Fahrenheit (maximum). - RPV beltline material fluence: 3.16E+19 n/cm2 (E >1.0 MeV) (maximum). If the reactor pressure vessel exposure conditions (neutron	Program to be enhanced prior to the period of extended operation.	Section A.2.1.19 Exelon Letter RS-14-002 RAI B.2.1.19-1 01/13/2014

NO.	PROGRAM OR TOPIC	COMMITMENT	IMPLEMENTATION SCHEDULE	SOURCE
		fluence, neutron spectrum) or irradiation temperature (cold leg inlet temperature) are altered, then the basis for the projection to the end of the period of extended operation needs to be reviewed and, if deemed appropriate, updates are made to the Reactor Vessel Surveillance program. Any changes to the Reactor Vessel Surveillance program must be submitted for NRC review and approval in accordance with 10 CFR Part 50, Appendix H. ~~2. One (1) specimen capsule per reactor vessel, as designated below, irradiated to a neutron fluence of one (1) to two (2) times the projected peak neutron fluence at the end of the period of extended operation will be withdrawn from the spent fuel pool, tested, and the summary technical report submitted to the NRC prior to entering the associated period of extended operation.~~		***Exelon Letter RS-14-149 RAI B.2.1.19-1a 05/23/2014***

~~Reactor Vessel (Station, Unit)~~	~~Capsule ID~~	~~Capsule Fluence (n/cm^2)(E>1.0 MeV)~~
~~Byron, Unit 1~~	~~Y~~	~~3.97E+19~~
~~Byron, Unit 2~~	~~Y~~	~~4.19E+19~~
~~Braidwood, Unit 1~~	~~V~~	~~3.71E+19~~
~~Braidwood, Unit 2~~	~~V~~	~~3.73E+19~~