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

April 8, 2014

Mr. Michael P. Gallagher
Vice President, License Renewal Projects
Exelon Generation Company, LLC
200 Exelon Way
Kennett Square, PA 19348

SUBJECT: 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 19 (TAC NOS. MF1879, MF1880, MF1881, AND MF1882)

Dear Mr. Gallagher:

By letter dated May 29, 2013, Exelon Generation Company, LLC, submitted an application pursuant to Title 10 of the *Code of Federal Regulations* Part 54, to renew the operating licenses NPF-37, NPF-66, NPF-72, and NPF-77 for Byron Station, Units 1 and 2, and Braidwood Station, Units 1 and 2, respectively, for review by the U.S. Nuclear Regulatory Commission (NRC or the staff). The staff is reviewing the information contained in the license renewal application and has identified, in the enclosure, areas where additional information is needed to complete the review.

These requests for additional information were discussed with John Hufnagel, and a mutually agreeable date for the response is within 30 days from the date of this letter. If you have any questions, please contact me at 301-415-4115 or by e-mail at Lindsay.Robinson@nrc.gov.

Sincerely,

Lindsay R. Robinson, Project Manager
Projects Branch 1
Division of License Renewal
Office of Nuclear Reactor Regulation

Docket Nos. 50-454, 50-455, 50-456, and 50-457

Enclosure:
Request for Additional Information

cc w/encl: Listserv

April 8, 2014

Mr. Michael P. Gallagher
Vice President, License Renewal Projects
Exelon Generation Company, LLC
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Kennett Square, PA 19348

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Letter to M.P. Gallagher from Lindsay R. Robinson dated April 8, 2014

SUBJECT: 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 19 (TAC NOS. MF1879, MF1880, MF1881, AND MF1882)

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BYRON STATION, UNITS 1 AND 2,
AND BRAIDWOOD STATION, UNITS 1 AND 2,
LICENSE RENEWAL APPLICATION
REQUEST FOR ADDITIONAL INFORMATION, SET 19
(TAC NOS. MF1879, MF1880, MF1881, AND MF1882)

RAI 4.2.6-1

Applicability:

Byron Station (Byron) and Braidwood Station (Braidwood), Units 1 and 2

Background:

License renewal application (LRA) Section 4.2.6 describes the time-limited aging analysis (TLAA) for calculation of the low temperature overpressure protection (LTOP) system setpoints. The LRA states that, in accordance with 10 CFR 54.21(c)(1)(iii), the applicant will use its Reactor Vessel Surveillance Program to establish and report the LTOP system setpoints in order to manage the effects of aging for the period of extended operation (PEO). As described in LRA Section B.2.1.19, the Reactor Vessel Surveillance Program is a condition monitoring program that provides material and dosimetry data for monitoring irradiation embrittlement through the PEO.

Issue:

To satisfy the requirements of 10 CFR 54.21(c)(1)(iii), the applicant should describe the processes it will use to ensure that the LTOP system setpoints are updated and reported to the NRC prior to entering the PEO. LRA Section 4.2.6 states that the applicant will use its Reactor Vessel Surveillance Program for this purpose. However, the current licensing basis (CLB) already specifies certain processes that the applicant must use to update the LTOP system setpoints. In particular, Technical Specification (TS) 5.6.6 identifies the analytical methods that the applicant must use for establishing the setpoints, and it also requires the applicant to document the setpoints in a Pressure and Temperature Limits Report (PTLR) and provide the report to the NRC for each reactor vessel fluence period and for any revision or supplement thereto. Since the primary purpose of the Reactor Vessel Surveillance Program is for data collection only, the program does not include the specific analytical methods and processes that must be used to establish, document, and report the new LTOP system setpoints in accordance with TS 5.6.6. Because the Reactor Vessel Surveillance Program does not fully implement the requirements of TS 5.6.6, it is not clear to the staff why the applicant credits this program for the demonstration required by 10 CFR 54.21(c)(1)(iii).

Request:

Explain why the procedures that implement the requirements of TS 5.6.6 will not be used to establish, document, and report the new LTOP system setpoints prior to entering the PEO, in order to satisfy the requirements of 10 CFR 54.21(c)(1)(iii). If any analytical methods or processes outside the requirements of TS 5.6.6 will be used to establish, document, and report the new LTOP system setpoints for the PEO, identify and explain the TS changes or additions that are needed per the requirements of 10 CFR 54.22. Based on this response, revise LRA Sections 4.2.6 and A.4.2.6 accordingly.

RAI 4.2.4-1/A.4.2.4-1

Applicability:

Byron and Braidwood

Background:

LRA Section 4.2.4 states that the TLAA on the adjusted reference temperature (ART) calculations is acceptable in accordance with the 10 CFR 54.21(c)(1)(ii). LRA Section 4.2.5 states that "the P-T [pressure-temperature] limits for the period of extended operation will be updated prior to expiration of the P-T limits for the current period of operation" and concludes that the TLAA on P-T limits satisfies the requirements of 10 CFR 54.21(c)(1)(iii).

Issue:

The methods of analysis in ASME Section XI, Appendix G, as referenced in 10 CFR Part 50, Appendix G, require an analysis of neutron fluence values at the crack tips of flaws that are postulated to initiate at both the inside (i.e., clad-to-base metal) and outside surfaces of the reactor pressure vessel (RPV) and projected to extend from the postulated crack initiation site to a depth one-quarter of the wall thickness. To be consistent with these regulatory requirements, the methodology in WCAP-14040, Revision 4 (Reference 4.8.2 in the LRA), as mandated by TS 5.6.6, requires the ART calculations (i.e., nil-ductility reference temperature (RT_{NDT}) calculations) to be performed based on an assessment of both the 1/4T and 3/4T neutron fluence values for the RPV beltline and extended beltline components. LRA Section 4.2.4 does not include any ART values for RPV beltline and extended beltline components that are based on the 3/4T fluence values for the components at 57 effective full-power years (EFPY).

LRA Section A.4.2.4 states that "57 EFPY 1/4T fluence values were used to compute ART values for Byron and Braidwood beltline and extended beltline materials in accordance with Regulatory Guide 1.99, Revision 2 requirements." This is not consistent with WCAP-14040-NP-A as described above.

Request:

1. Amend LRA Section 4.2.4 to provide the ART tables and values that are based on an assessment of the 3/4T neutron fluences for the RPV beltline and extended beltline components at 57 EFPY. Amend LRA Section A.4.2.4 to state that both the 57 EFPY 1/4T and 3/4T fluence values were used to compute ART values for Byron and Braidwood beltline and extended beltline materials in accordance with WCAP-14040-NP requirements, as mandated by TS 5.6.6.

2. Provide a basis for dispositioning the TLAA on the ART in terms of 10 CFR 54.21(c)(1)(ii), given that these values will be factored into the P-T limits for the period of extended operation, which are being dispositioned as 10 CFR 54.21(c)(1)(iii). Otherwise, revise the LRA to disposition the TLAA for projected ART values in terms of 10 CFR 54.21(c)(1)(iii).

RAI 4.2.5-1/A.4.2.5-1

Applicability:

Byron and Braidwood

Background:

LRA Section 4.2.5 provides the TLAA on the P-T Limits. LRA Section 4.2.5 states that the TLAA is acceptable in accordance with the requirements of 10 CFR 54.21(c)(1)(iii) because the applicant's PTLR process will be used to generate the P-T limit curves for the PEO.

Generation of the P-T limit curves using the applicant's PTLR process is currently governed by the TS 5.6.6 and the associated plant implementing procedures. The provisions in TS 5.6.6 require the P-T limits to be generated in accordance with the following NRC-approved methodologies:

- Those methodologies referenced in the NRC letter of January 21, 1998, "Byron Station, Units 1 and 2, and Braidwood Station, Units 1 and 2, Acceptance of Referencing Pressure Temperature Limits Report," which include WCAP-14040-NP-A (current NRC approved version, which is Revision 4 of the report; LRA Reference 4.8.2).

- Those methodologies referenced in the NRC letter of August 8, 2001, "Issuance of Exemption from the requirements of 10 CFR 50.60 and Appendix G for Byron Station, Units 1 and 2, and Braidwood Station, Units 1 and 2."

- WCAP-16143-P, "Reactor Vessel Closure Head/Vessel Flange Requirements Evaluation for Byron/Braidwood Units 1 and 2."

The fracture toughness of reactor vessel materials may decrease with time in the presence of sufficient neutron irradiation. Therefore, NRC regulations require monitoring of reactor vessel material fracture toughness during plant operation. P-T limits define the pressure and temperature operating conditions that must be maintained to ensure adequate margins of safety exist on material fracture toughness.

10 CFR Part 50 Appendix G, "Fracture Toughness Requirements," Section I, "Introduction and Scope," states the following:

> This appendix specifies fracture toughness requirements for ferritic materials of pressure-retaining components of the reactor coolant pressure boundary (RCPB) of light water nuclear power reactors to provide adequate margins of safety during any condition of normal operation, including anticipated operational occurrences and system hydrostatic tests, to which the pressure boundary may be subjected over its service lifetime.

Ferritic materials of pressure-retaining components of the RCPB include the following:

(1) Reactor pressure vessel (RPV) forgings (e.g., RPV nozzles and flanges) and their associated structural welds

(2) Plates or forgings from which the RPV shells and heads were manufactured and their associated structural welds
(3) Ferritic materials in other portions of the RCPB, including those used to fabricate ferritic piping, pumps, valves, and other pressure vessels in the RCPB

For LRAs, the regulation in 10 CFR 54.22 requires the LRA to include any TS additions or changes that are necessary to manage the effects of aging during the PEO and the justification for such TS changes or additions to be included in the application.

Issues:

1. Licensees must be able to demonstrate that the P-T limits developed for the plant are bounding for all ferritic components in the RCPB, as required by Section I of 10 CFR Part 50, Appendix G. To demonstrate compliance with 10 CFR Part 50, Appendix G, the evaluation of P-T limits considers several factors, including the initial properties and composition of the ferritic materials used to fabricate the RPV components, the accumulated neutron fluence for each component (and hence the neutron embrittlement of the material), and the stress levels applied to the components resulting from operating loads and structural discontinuities. The evaluation of P-T limits that are based solely on an evaluation of ferritic RPV components in the beltline region of the vessel may be insufficient to demonstrate compliance with 10 CFR Part 50, Appendix G. This is because the effects of structural discontinuities for an RPV component with a lower reference temperature (such as a nozzle with a lower neutron fluence) may result in more conservative P-T limits than those that are based on an RPV shell component with a higher reference temperature. Thus, the development of P-T limits for the RPCB must consider not only the RPV beltline shell components with the highest reference temperature but also other RPV components with structural discontinuities, including those that are located outside of the beltline region of the RPV.

   The applicant has proposed to address the RCPB and RPV discontinuity issue through an enhancement in LRA Sections 4.2.5 and A.4.2.5 that states the following:

   > The analysis for the P-T curves will consider locations outside of the beltline such as nozzles, penetrations and other discontinuities to determine if more restrictive P-T limits are required than would be determined by considering only the reactor vessel beltline materials.

   It is not evident to the staff why this issue can be resolved through an enhancement that is defined in LRA Section A.4.2.5. The calculation of the Byron and Braidwood P-T limits is driven by a PTLR process that is mandated by TS 5.6.6 and the PTLR criteria in Generic Letter (GL) 96-03[1], "Relocation of the Pressure Temperature Limit Curves and Low

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[1] Generic Letter (GL) 96-03, establishes the NRC policy for processing license amendments to move P-T limit and low temperature overpressure protection system setpoint requirements into an owner's report (the PTLR) controlled by the Administrative Controls Section of the TS. The GL also establishes the TS criteria that need to be proposed for the processing of these license amendments, the minimum criteria that should be included in the methodologies for generating the P-T limits and LTOP system setpoint values for the facilities, and the information that should be included in the PTLRs. GL 96-03 may be accessed at: http://www.nrc.gov/reading-rm/doc-collections/gen-comm/gen-letters/1996/gl96003.html.

Temperature Overpressure Protection System Limits," dated January 31, 1996, which dictates that this should be part of the approved methodologies that are referenced in TS 5.6.6.

2. The applicant modified its RPV closure flange configuration in 1995 (Braidwood, Unit 2) and in 2010 (Byron, Unit 2), such that one stud cannot be tensioned. However, the methods of analysis in WCAP-16143-P are based on the original plant design configuration, with all original reactor vessel closure studs fully tensioned.

3. Based on the issues raised in Parts (1) and (2) above, the staff seeks clarification why a change to TS 5.6.6, Part b., or to the methodologies invoked by TSS 5.6.6, Part b., would not need to be processed as part of the LRA, as mandated by 10 CFR 54.22.

<u>Requests</u>:

1. Clarify how the assessment of RPV non-beltline structural discontinuities for its impact on future P-T limits will be performed in accordance with 10 CFR 54.21(c)(1)(iii) and how this will be factored into the update of the PTLRs that will be submitted to the NRC in accordance TS 5.6.6, Part c. Explain why the assessment of RPV non-beltline structural discontinuities is proposed as part of an enhancement that is defined in LRA Section A.4.2.5 rather than the NRC policy established in GL 96-03, which would have this type of assessment performed in accordance with 10 CFR Part 50, Appendix G, requirements and included within the scope of at least one of the P-T limit methodologies that are invoked by TS 5.6.6, Part b.

2. Explain why the current TS 5.6.6 required methodologies and the plant procedures for implementing the PTLR process are valid for updating the P-T limit curves that will be generated for the PEO, given that the P-T limits minimum temperature requirement methodology in WCAP-16143-P is not based on the configurations of current RPV closure flange assemblies at Byron Unit 2 and Braidwood Unit 2.

3. Based on your responses to Requests (1) and (2) above, explain whether applicable changes to TS 5.6.6 or to the methodologies invoked by TS 5.6.6 need to be proposed for the LRA in accordance with the requirement in 10 CFR 54.22. Amend LRA Sections 4.2.5 and A.4.2.5, accordingly, if it is determined that either TS 5.6.6 or the methodologies invoked by TS 5.6.6 need to be amended in accordance with the 10 CFR 54.22 requirements.

RAI 4.2.5-2

Applicability:

Byron, Unit 2 and Braidwood, Unit 2

Background:

LRA Section 4.2.5 provides the applicant TLAA for accepting the Byron and Braidwood P-T limits in accordance with 10 CFR 54.21(c)(1)(iii) and the applicant's PTLR process, which is mandated and controlled by requirements in TS 5.6.6. Updated Final Safety Analysis Report (UFSAR) Section 5.3.2.1 states that the "surveillance program withdrawal schedule is contained in Table 4.1 of the PTLR document for each unit, respectively." UFSAR Section 5.3.2.1 also states that "[c]hanges to the withdrawal schedule may be made as part of an update to the PTLR under the provisions of 10 CFR 50.59."

Pursuant to the requirements in 10 CFR Part 50, Appendix H, "Reactor Vessel Surveillance Program Requirements," changes to a RPV surveillance withdrawal schedule must be submitted to the NRC for review and approval.

Issue:

The NRC's policy for approving license amendments for PTLR processes is given in GL 96-03. In this GL, the staff only indicated that P-T limit changes and the LTOP system setpoint changes could be processed through a licensee's 10 CFR 50.59 and PTLR processes, so long as the PTLR methodologies approved in the Administrative Controls Section of the TS would be used to make the changes to the P-T limits and the LTOP setpoint values. In contrast, proposed changes to the RPV surveillance program withdrawal schedules for the units are required by 10 CFR Part 50, Appendix H, to be submitted to the NRC for review and approval. The provisions of 10 CFR 50.59 do not apply to the processing of proposed changes to the RPV surveillance program withdrawal schedules.

Request:

Explain the basis for stating that future "[c]hanges to the withdrawal schedule may be made as part of an update to the PTLR under the provisions of 10 CFR 50.59," when the regulation in 10 CFR Part 50, Appendix H, requires proposed changes to RPV surveillance program withdrawal schedules to submitted to the NRC for review and approval.

RAI A.4.2.4-2

Applicability:

Byron and Braidwood

Background:

LRA Section A.4.2.4 provides the UFSAR Supplement summary description for the TLAA on the ART calculations, which were provided in LRA Section 4.2.4.

Issue:

UFSAR Supplement A.4.2.4 references a 200 degree Fahrenheit (200 °F) value in Section C.3 of Regulatory Guide (RG) 1.99, "Radiation Embrittlement of Reactor Vessel Materials," Revision 2, and implies that the 200 °F value was included in the RG section to place a limit on the calculation of 1/4T ART values (i.e., 1/4T RT_{NDT} values). Section C.3 of RG 1.99, Revision 2, relates to the bases for RPV material selection when choosing the ferritic steel materials that would be used to fabricate the RPVs of newly constructed plants. In this case, the 200 °F value referenced in Section C.3 of the RG serves only as a recommended ART basis for establishing and limiting the copper (Cu) alloying contents of ferritic steel materials that are procured and used for fabrication of the RPVs in new plants; it does not establish an upper-bound limit on the calculation of those 1/4T ART values after the RPVs are fabricated and the plants are operated.

The reference sentence from LRA Section A.4.2.4 states:

> The projections demonstrate that the ART values in the limiting material for each unit will remain below the NRC Regulatory Guide 1.99, Revision 2, Section 3 acceptance criteria of 200 degrees F through the period of extended operation.

Request:

Amend LRA Section A.4.2.4 to be consistent with the 200 °F value basis that is referenced in Section C.3 of RG 1.99, Revision 2, or provide a technical basis for the statement as written. Otherwise, amend LRA Section A.4.2.4 to delete that statement from UFSAR Supplement Section A.4.2.4.