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10 CFR 50
10 CFR 51
10 CFR 54

RS-14-129

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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 19, dated April 8, 2014, related to the Braidwood Station, Units 1 and 2, and Byron 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 8, 2014, "Requests 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)"

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 affected by the responses.

There are no 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 5-6-2014

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

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 4.2.6-1
RAI 4.2.4-1/A.4.2.4-1
RAI 4.2.5-1/A.4.2.5-1
RAI 4.2.5-2
RAI A.4.2.4-2

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 bases (CLBs) already specify 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 Technical Specification 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.

Exelon Response:

The current licensing bases (CLBs) specify the processes that the Byron and Braidwood stations must use to update and report the Low Temperature Overpressure Protection (LTOP) system setpoints. Data is obtained from the reactor surveillance specimens in accordance with the Reactor Vessel Surveillance program. In particular, Technical Specification (TS) 5.6.6 identifies the analytical methods that both stations must use for establishing these LTOP system setpoints. The process also requires the stations 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.

For the period of extended operation, Byron and Braidwood Units 1 and 2 will submit the appropriate analysis for the LTOP system setpoints in accordance with the PTLR process. The LTOP setpoints are determined based on the accepted methodology in the NRC letter dated January 21st, 1998 (Accession Number 9802040389), which approved the use of WCAP-14040, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves." Byron and Braidwood have implemented changes to the LTOP setpoints throughout the current period of operation using the PTLR process, and will continue to establish, document, and report new LTOP setpoints using the PTLR process in accordance with the requirements of Technical Specification 5.6.6 for the period of extended operation.

LRA Section 4.2.6 and Appendix A, Section A.4.2.6, are revised as shown in Enclosure B.

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).

Exelon Response:

1. Exelon provided the projected 1/4T ART values for information only to show the effect of the 60-year fluence projections on the ART values. It was not the intent to provide the ART values in preparation for a formal review to support a P-T Limit change. The projected 1/4T ART values were provided to demonstrate reasonable assurance that the 1/4T ART values projected to the end of the period of extended operation are acceptable. The disposition of the TLAA on the ART is being changed from "in accordance with 10 CFR 54.21(c)(1)(ii)" to "in accordance with 10 CFR 54.21(c)(1)(iii)." The revised disposition also references LRA Section 4.2.5, "Pressure-Temperature Limits" which is also dispositioned in accordance with 10 CFR 54.21(c)(1)(iii). The limiting 1/4T and 3/4T ART values will continue to be provided with the PTLR report to maintain the P-T Limits in accordance with Technical Specification Requirements during the period of extended operation as presented in LRA Section 4.2.5, "Pressure-Temperature Limits." The revised LRA sections associated with this change are contained in Enclosure B to this submittal.

2. As stated in the response to Request 1 above, LRA Sections 4.1.4 (Table 4.1-2), 4.2.4, and A.4.2.4 are revised to disposition the TLAA on the ART in accordance with 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

[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

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.

Requests:

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

Exelon Response:

1. 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 will be factored into the update of the PTLRs that will be submitted to the NRC in accordance with plant Technical Specification (TS) 5.6.6, Part c. The revisions to the P-T limits beyond the current P-T limits will continue to consider the requirements established in GL 96-03, which would have this

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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.

type of assessment performed in accordance with 10 CFR Part 50, Appendix G requirements. The LRA is amended to provide this further clarification of the consideration of the 10 CFR 50, Appendix G requirements. 10 CFR 50, Appendix G recognizes the ASME Section XI, Appendix G limits as an acceptable approach for analysis and providing sufficient margins of safety. ASME Section XI, Appendix G is included within the scope of TS 5.6.6, Part b, since ASME Section XI, Appendix G methodologies are used, and have been approved for use in accordance with 10 CFR 50.55(a) . Since TS 5.6.6, Part b, implements ASME Section XI, Appendix G requirements, and ASME Section XI, Appendix G includes the consideration of the assessment of RPV non-beltline structural discontinuities, no enhancement was intended by the LRA statement.

Recent issuance of the draft RIS 2014-XX, "Information on Licensing Applications for Fracture Toughness Requirements for Ferritic Reactor Coolant Pressure Boundary Components" also addresses the need to consider the non-beltline structural discontinuities. With the issuance of the final RIS, further clarification will be provided on the expectations for future PTLR submittals on this subject. After issuance of the final RIS, and with PTLRs required for the PEO, Exelon will, in accordance with Technical Specifications, provide PTLRs which sufficiently address all ferritic materials of pressure-retaining components of the reactor coolant pressure boundary, including the impact of structural discontinuities, and address the impact of neutron fluence accumulation in accordance with the requirements of 10 CFR Part 50, Appendix G.

2. The current TS 5.6.6 required methodologies and the plant procedures for implementing the PTLR process will be valid for updating the P-T limit curves that will be generated for the period of extended operation. 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, as an interim measure, commitments have been made in Exelon's response to Notice of Violation dated December 13, 2013 to take corrective steps to revise WCAP-16143-P to reflect the Braidwood Unit 2 configuration of 53 RPV head bolts. In addition, the revision of WCAP-16143-P will include the 53 RPV head bolt configuration at Byron Unit 2. The revision of WCAP-16143-P will bring the methodology in agreement with the current configuration. With regard to the period of extended operation, a commitment was made to restore the configuration for Byron Unit 2 and Braidwood Unit 2 RPV closure flange assemblies to that analyzed in WCAP-16143-P (all 54 reactor head studs tensioned) prior to the period of extended operation. This commitment was made in Exelon's response to NRC RAI B.2.1.3-2 in letter RS-13-285, dated December 13, 2013. Implementing these commitments will maintain the current TS 5.6.6 methodologies and plant procedures for implementing the PTLR process valid for the current operating period and the period of extended operation.

3. Based on the responses to Requests (1) and (2) above, there are no changes to TS 5.6.6 or to the methodologies invoked by TS 5.6.6 for the LRA in accordance with the requirement in 10 CFR 54.22. LRA Sections 4.2.5 and A.4.2.5 are revised as presented in Enclosure B to clarify the need to meet 10 CFR 50, Appendix G 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.

Exelon Response:

Exelon agrees that the subject statement in UFSAR Section 5.3.2.1 needs to be corrected. The statement is inconsistent with UFSAR Section 5.3.1.6 which states, "For the schedule for removal of the capsules for postirradiation testing which follows that of 10 CFR 50 Appendix H, refer to Table 4.1 of the PTLR." The statement is also inconsistent with the NRC Safety Evaluation Report (SER)(Accession Number 9802040391) authorizing the relocation of the capsule withdrawal schedule from the plant Technical Specifications to the Pressure-Temperature Limits Report (PTLR). The NRC SER states, "The staff also concludes that it is acceptable to relocate the surveillance capsule withdrawal schedule to the licensee controlled

PTLR since changes to this schedule are controlled by the requirements of Appendix H to 10 CFR 50." This issue has been entered into the corrective action program to revise the UFSAR.

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 ºF) value in Section C.3 of Regulatory Guide (RG) 1.99, "Radiation Embrittlement of Reactor Vessel Materials," Revision 2, and implies that the 200 ºF 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 ºF 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 ºF 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.

Exelon Response:

LRA Sections 4.2.4 and A.4.2.4 are revised to remove the reference to the 200 degree Fahrenheit (200 ºF) value specified in Section C.3 of Regulatory Guide (RG) 1.99, Revision 2. The revisions to LRA Sections 4.2.4 and A.4.2.4 are contained in Enclosure B of this submittal.

Enclosure B

Byron and Braidwood Stations, Units 1 and 2
License Renewal Application (LRA) updates resulting
from the response to the following RAIs:

RAI 4.2.4-1/A.4.2.4-1
RAI 4.2.5-1/A.4.2.5-1
RAI 4.2.6-1

Note: To facilitate understanding, portions of the original LRA 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 the response to RAI 4.2.4-1/A.4.2.4-1, LRA Section 4.1.4, Table 4.1-2, page 4.1-6 is revised as shown on the following page. Revisions are indicated with ***bold italics*** for inserted text and ~~strikethroughs~~ for deleted text.

Table 4.1-2 SUMMARY OF RESULTS – BBS TIME-LIMITED AGING ANALYSES		
TLAA DESCRIPTION	**DISPOSITION**	**LRA SECTION**
REACTOR VESSEL NEUTRON EMBRITTLEMENT ANALYSIS		4.2
Neutron Fluence Projections	§54.21(c)(1)(ii)	4.2.1
Upper-Shelf Energy	§54.21(c)(1)(ii)	4.2.2
Pressurized Thermal Shock	§54.21(c)(1)(ii)	4.2.3
Adjusted Reference Temperature	§54.21(c)(1)~~(ii)~~***(iii)***	4.2.4
Pressure-Temperature Limits	§54.21(c)(1)(iii)	4.2.5
Low Temperature Overpressure Projection (LTOP) Analyses	§54.21(c)(1)(iii)	4.2.6
METAL FATIGUE		4.3
Transient Inputs to Fatigue Analyses	§54.21(c)(1)(ii)	4.3.1
ASME Section III, Class 1, Class 2, and Class 3 Fatigue Analyses	§54.21(c)(1)(iii)	4.3.2
ASME Section III, Class 2 and 3 and ANSI B31.1 Allowable Stress Analyses	§54.21(c)(1)(iii) and §54.21(c)(1)(i)	4.3.3
Class 1 Component Fatigue Analyses Supporting GSI-190 Closure	§54.21(c)(1)(iii)	4.3.4
Reactor Vessel Internals Fatigue Analyses	§54.21(c)(1)(iii)	4.3.5
High-Energy Line Break (HELB) Analyses Based on Fatigue	§54.21(c)(1)(iii)	4.3.6
NRC Bulletin 88-11 Revised Fatigue Analysis of the Pressurizer Surge Line for Thermal Cycling and Stratification	§54.21(c)(1)(iii)	4.3.7
ASME Section III, Subsection NF, Class 1 Component Supports Allowable Stress Analyses	§54.21(c)(1)(iii)	4.3.8
Fatigue Design of Spent Fuel Pool Liner and Spent Fuel Storage Racks for Seismic Events	§54.21(c)(1)(iii)	4.3.9
Pressurizer Heater Sleeve Structural Assessment	§54.21(c)(1)(iii)	4.3.10

As a result of the response to RAI 4.2.4-1/A.4.2.4-1 and RAI A.4.2.4-2, LRA Section 4.2.4, pages 4.2-25 and 4.2-26 are revised as shown below. Revisions are indicated with ***bold italics*** for inserted text and ~~strikethroughs~~ for deleted text.

4.2.4 ADJUSTED REFERENCE TEMPERATURE

TLAA Description:

The adjusted reference temperature (ART) of the limiting beltline material is used to adjust the beltline P-T limit curves to account for irradiation effects. Regulatory Guide 1.99, Revision 2, provides the methodology for determining the ART of the limiting material. The initial nil-ductility reference temperature, RT_{NDT}, is the temperature at which a non-irradiated metal (ferritic steel) changes in fracture characteristics from ductile to brittle behavior. RT_{NDT} is evaluated according to the procedures in the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section III, Paragraph NB-2331. Neutron embrittlement increases the RT_{NDT} beyond its initial value.

10 CFR 50, Appendix G, defines the fracture toughness requirements for the life of the vessel. The shift in the initial RT_{NDT} (ΔRT_{NDT}) is evaluated as the difference in the 30 ft-lb index temperatures from the average Charpy curves measured before and after irradiation. This increase (ΔRT_{NDT}) means that higher temperatures are required for the material to continue to act in a ductile manner. The ART is defined as: Initial RT_{NDT} + (ΔRT_{NDT}) + Margin. Since the ΔRT_{NDT} value is a function of 32 EFPY fluence, associated with the 40-year licensed operating period, these ART calculations meet the criteria of 10 CFR 54.3(a) and have been identified as TLAAs requiring evaluation for 60 years.

TLAA Evaluation:

As described in Section 4.2.1, 57 EFPY fluence values were determined for BBS for the RPV beltline and extended beltline components. These 57 EFPY 1/4T fluence values were used to compute the ART values of BBS, Units 1 and 2, in accordance with Regulatory Guide 1.99, Revision 2 (Reference 4.8.4). Table 4.2.4-1 and Table 4.2.4-2 present the ART results for Byron Units 1 and 2, respectively. The ART results for Braidwood Units 1 and 2 are presented in Table 4.2.4-3 and Table 4.2.4-4.

The Nozzle Shell Forging-to-Intermediate Shell Forging Circumferential Weld seam in both Byron Units 1 and 2 were fabricated using weld wire heat # 442011, which is the same weld material that is contained in the Braidwood Units 1 and 2 surveillance programs. The Braidwood surveillance weld material is deemed credible and also applicable to the Byron Units 1 and 2 Nozzle Shell Forging-to-Intermediate Shell Forging Circumferential Weld seams.

~~The ART values of the limiting beltline materials at 57 EFPY for each unit remain below 200 degrees F, which is the Nil-Ductility Transition (RT_{NDT}) limit specified in NRC Regulatory Guide 1.99, Revision 2, Section 3.~~ The limiting locations, taking credit for credible data, are listed below:

The limiting 57 EFPY ART values for Byron Unit 1 correspond to the Intermediate Shell Forging using non-credible surveillance data (Position 2.1) for axial flaw (forging) materials and the Intermediate Shell Forging-to-Lower Shell Forging Circumferential Weld Seam (Heat # 442002) using credible surveillance data (Position 2.1) for circumferential flaw (weld) materials.

The limiting 57 EFPY ART values for Byron Unit 2 correspond to the Nozzle Shell Forging for axial flaw (forging) materials and the Intermediate Shell Forging-to-Lower Shell Circumferential Weld Seam (Heat # 442002) using credible surveillance data (Position 2.1) for circumferential flaw (weld) materials.

The limiting 57 EFPY ART values for Braidwood Unit 1 correspond to the Nozzle Shell Forging for axial flaw (forging) materials and the Intermediate Shell Forging-to-Lower Shell Circumferential Weld Seam (Heat # 442011) using credible surveillance data (Position 2.1) for circumferential flaw (weld) materials.

The limiting 57 EFPY ART values for Braidwood Unit 2 correspond to the Nozzle Shell Forging for axial flaw (forging) materials and the Intermediate Shell Forging-to-Lower Shell Forging Circumferential Weld Seam (Heat # 442011) using credible surveillance data (Position 2.1) for circumferential flaw (weld) materials.

TLAA Disposition: 10 CFR 54.21 (c)(1)~~(ii)~~*(iii)* — ~~The ART analyses have been projected to the end of the period of extended operation. They may be used as inputs to 57 EFPY P-T limits for the period of extended operation.~~ ***The limiting 1/4T and 3/4T ART values will continue to be provided with the PTLR report to maintain the P-T Limits in accordance with Technical Specification requirements during the period of extended operation as presented in LRA Section 4.2.5, "Pressure-Temperature Limits."***

As a result of the response to RAI 4.2.4-1/A.4.2.4-1 and RAI A.4.2.4-2, LRA, Appendix A, Section A.4.2.4, page A-50 is revised as shown below. Revisions are indicated with ***bold italics*** for inserted text and ~~strikethroughs~~ for deleted text.

A.4.2.4 Adjusted Reference Temperature

The adjusted reference temperature (ART) of the limiting beltline material is used to adjust the beltline P-T limit curves to account for irradiation effects. Regulatory Guide 1.99, Revision 2, provides the methodology for determining the ART of the limiting material. The initial nil-ductility reference temperature, RT_{NDT}, is the temperature at which a non-irradiated metal (ferritic steel) changes in fracture characteristics from ductile to brittle behavior. RT_{NDT} is evaluated according to the procedures in the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section III, Paragraph NB-2331. Neutron embrittlement increases the RT_{NDT} beyond its initial value.

10 CFR 50, Appendix G, defines the fracture toughness requirements for the life of the vessel. The shift in the initial RT_{NDT} (ΔRT_{NDT}) is evaluated as the difference in the 30 ft-lb index temperatures from the average Charpy curves measured before and after irradiation. This increase (ΔRT_{NDT}) means that higher temperatures are required for the material to continue to act in a ductile manner. The ART is defined as: Initial RT_{NDT} + (ΔRT_{NDT}) + Margin. Since the ΔRT_{NDT} value is a function of 32 EFPY fluence, associated with the 40-year licensed operating period, these ART calculations meet the criteria of 10 CFR 54.3(a) and have been identified as TLAAs requiring evaluation for 60 years.

~~57 EFPY 1/4T fluence values were used to compute ART values for BBS beltline and extended beltline materials in accordance with Regulatory Guide 1.99, Revision 2 requirements.~~ ***The limiting 1/4T and 3/4T ART values will continue to be provided with the PTLR report to maintain the P-T Limits in accordance with Technical Specification requirements during the period of extended operation as presented in LRA Section 4.2.5, "Pressure-Temperature Limits."*** ~~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.~~ Therefore, these TLAAs are dispositioned in accordance with 10 CFR 54.21(c)(1)~~(ii)~~***(iii).***

As a result of the response to 4.2.5-1/A.4.2.5-1, LRA Section 4.2.5, page 4.2-37 is revised as shown below. Revisions are indicated with ***bold italics*** for inserted text and ~~strikethroughs~~ for deleted text.

- Inlet Nozzles
- Outlet Nozzles
- Inlet Nozzle-to-Nozzle Shell Forging Weld Seams
- Outlet Nozzle-to-Nozzle Shell Forging Weld Seams

The current P-T limits for Byron and Braidwood Units 1 and 2 are based on the latest fluence data and are located in the unit specific PTLRs. PTLRs containing updated P-T limits for the period of extended operation will be ~~submitted~~ ***provided*** to the NRC ~~for approval~~ prior to ***exceeding the current terms of applicability which is projected to be before entering the period of extended operation*** ~~exceeding the current terms of applicability~~. ~~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.~~ Maintenance of the P-T limits during the period of extended operation will be managed using the applicable process as described above to comply with 10 CFR 50, Appendix G.

TLAA Disposition: 10 CFR 54.21 (c)(1)(iii) – The effects of aging on the intended function(s) of the reactor vessels will be adequately managed for the period of extended operation. ~~The Reactor Vessel Surveillance (B.2.1.19) program~~ ***Plant Technical Specification 5.6.6*** will ensure that updated P-T limits based upon updated ART values will be submitted to the NRC for approval prior to exceeding the current terms of applicability for Byron and Braidwood, Units 1 and 2. ***The PTLR revision necessary to extend the P-T limits into the period of extended operation will consider all ferritic materials of pressure-retaining components of the reactor coolant pressure boundary including the impact of structural discontinuities, and address the impact of neutron fluence accumulation in accordance with the requirements of 10 CFR 50, Appendix G.*** The P-T limits will be maintained during the period of extended operation using the Administrative Section of Technical Specifications to amend the P-T limits through the PTLR process.

As a result of the response to 4.2.5-1/A.4.2.5-1, Appendix A, Section A.4.2.5, page A-51 is revised as shown below. Revisions are indicated with ***bold italics*** for inserted text and ~~strikethroughs~~ for deleted text.

A.4.2.5 Pressure–Temperature Limits

Appendix G of 10 CFR 50 requires that the reactor vessel be maintained within established pressure–temperature (P-T) limits. These limits specify the maximum allowable pressure as a function of reactor coolant temperature. As the reactor vessel is exposed to increased neutron irradiation, its fracture toughness is reduced. The P-T limits must account for the anticipated reactor vessel fluence.

The current P-T limits are based upon 32 EFPY fluence projections that were considered to represent the amount of power to be generated over 40 years of plant operation, assuming a 40-year average capacity factor of 80 percent. The current P-T limits are located in the unit-specific pressure-temperature limit reports (PTLRs). Since they were originally based upon a 40-year assumption regarding capacity factor, the P-T limits satisfy the criteria of 10 CFR 54.3(a) and have been identified as TLAAs.

In accordance with NUREG-1800, Revision 2, Section 4.2.2.1.3, P-T Limits for the period of extended operation need not be submitted as part of the LRA since P-T limits need to be updated through the Administrative Section of the Technical Specifications and the plant's PTLR process. ~~The Reactor Vessel Surveillance Program will assure that updated P-T limits based on updated ART values will be submitted to the NRC and approved prior to exceeding the current terms of applicability. 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.~~ ***The PTLR revision necessary to extend the P-T limits into the period of extended operation will consider all ferritic materials of pressure-retaining components of the reactor coolant pressure boundary, including the impact of structural discontinuities, and address the impact of neutron fluence accumulation in accordance with the requirements of 10 CFR 50, Appendix G.***

The P-T limit curves will be maintained during the period of extended operation using the Administration Section of Technical Specifications to amend the P-T limits through the PTLR process. These TLAAs are dispositioned in accordance with 10 CFR 54.21(c)(1)(iii).

As a result of the response to RAI 4.2.6-1 provided in Enclosure A of this letter, LRA Section 4.2.6, page 4.2-38, is revised as shown below. Changes are highlighted with ***bolded italics*** for inserted text and ~~strikethroughs~~ for deleted text.

4.2.6 LOW TEMPERATURE OVERPRESSURE PROTECTION (LTOP) Analyses

TLAA Description:

Low temperature overpressure protection (LTOP) system at Byron and Braidwood is required by Technical Specification Limited Condition for Operation 3.4.12. Two pressurizer power-operated relief valves (PORV) are used to provide the automatic relief capability during the design basis mass input (MI) and the design basis heat input (HI) transients to automatically prevent the reactor coolant system pressure from exceeding the pressure-temperature limit curves based on 10 CFR 50, Appendix G. The residual heat removal (RHR) suction relief valves may also be used to provide low-temperature overpressure protection (two RHR suction relief valves or one PORV and one RHR suction relief valve). The design basis MI and HI transients are defined in the Updated Final Safety Analysis Report and Technical Specifications Bases.

Since LTOP system setpoints are based on the P-T limits calculation, which is a TLAA, the calculation of the LTOP setpoints and the supporting analyses have been identified as TLAAs.

TLAA Evaluation:

Since these LTOP system analyses do not depend upon any other time-dependent values beyond the ART at the critical locations and the P-T limits, changes to the reactor coolant system P-T limits also require an evaluation of the LTOP temperature and PORV pressure setpoints, and supporting safety analyses.

The LTOP system setpoints are established in the Pressure Temperature Limit Report (PTLR) and are managed consistent with the P-T limits, which will be managed through the period of extended operation as described in Section 4.2.5, Pressure-Temperature (P-T) Limits. Therefore the LTOP setpoints will be managed through the period of extended operation.

TLAA Disposition: 10 CFR 54.21 (c)(1)(iii) – The LTOP system licensing and design basis analyses will be managed through the period of extended operation by the ~~Reactor Vessel Surveillance (B.2.1.19) program~~***Technical Specifications***. The LTOP system setpoints are ***established*** based on the P-T limits***, and are updated in accordance with the Pressure-Temperature Limit Report (PTLR) process***. The ~~Reactor Vessel Surveillance (B.2.1.19) program~~***plant Technical Specifications (5.6.6)*** will ensure that updated P-T limits based upon updated ***Adjusted Reference Temperature (***ART***)*** values will be submitted to the NRC for approval prior to exceeding the current terms of applicability for Byron and Braidwood, Units 1 and 2. The revision of the P-T limits to increase their applicability term through 57 EFPY will require re-evaluation of the LTOP system setpoints.

As a result of the response to RAI 4.2.6-1 provided in Enclosure A of this letter, LRA Appendix A, Section A.4.2.6, pages A-51 and A-52, is revised as shown below. Changes are highlighted with ***bolded italics*** for inserted text and ~~strikethroughs~~ for deleted text.

A.4.2.6 Low Temperature Overpressure Protection (LTOP) Analyses

Low temperature overpressure protection (LTOP) system at Byron and Braidwood is required by Technical Specification Limited Condition for Operation 3.4.12. Two pressurizer power-operated relief valves (PORV) are used to provide the automatic relief capability during the design basis mass input (MI) and the design basis heat input (HI) transients to automatically prevent the reactor coolant system pressure from exceeding the pressure-temperature limit curves based on 10 CFR 50, Appendix G. The residual heat removal (RHR) suction relief valves may also be used to provide low-temperature overpressure protection (two RHR suction relief valves or one PORV and one RHR suction relief valve). The design basis MI and HI transients are defined in the Updated Final Safety Analysis Report and Technical Specifications Bases.

Since LTOP system setpoints are based on the P-T limits calculation, which is a TLAA, the calculation of the LTOP setpoints and the supporting safety analyses have been identified as TLAAs.

The LTOP system setpoints are established in the PTLRs and managed consistent with the P-T limits, which will be managed through the period of extended operation as described in Section A.4.2.5, Pressure-Temperature (P-T) Limits. ***P-T Limits for the period of extended operation are not required to be submitted as part of the LRA since P-T limits are updated through the Administrative Section of the Technical Specifications and the plant's Pressure-Temperature Limit Report (PTLR) process.*** The ~~Reactor Surveillance Program~~***plant Technical Specifications (5.6.6)*** will ensure that updated P-T limits based upon updated ***Adjusted Reference Temperature (***ART***)*** values will be submitted to the NRC and approved prior to exceeding the current terms of applicability. Therefore, the LTOP system setpoints will be managed through the period of extended operation. These TLAAs are dispositioned in accordance with 10 CFR 54.21(c)(1)(iii).