From:	Tobin, Jennifer
To:	Grabnar, John J
Cc:	Lashley, Phil H (EH)
Subject:	Beaver Valley FINAL RAIs for RCS and PTLR TS LAR (EPID: L-2020-LLA-0233)
Date:	Tuesday, March 09, 2021 3:00:00 PM

Dear Mr. Grabnar,

By letter dated October 30, 2020 (Agencywide Documents Access and Management Systems Accession No. ML20304A215), Energy Harbor Nuclear Corporation (the licensee) requested an amendment to Technical Specification 5.6.4, "Reactor Coolant System (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)," Item b, to replace the currently referenced analytical methods with more recent analytical methods found acceptable by the Nuclear Regulatory Commission (NRC) staff for calculating reactor vessel neutron fluence and reactor coolant system pressure and temperature limits when updating the reactor coolant system Pressure and Temperature Limits Report.

A clarification call was held March 9th, no changes to the draft RAIs were made as a result of the discussion.

The specific request for additional information (RAI) questions are provided below and will be made publicly available in ADAMS. A response to these RAIs is due April 23, 2021.

Please let me know if you have questions or concerns.

Thanks!
-Jenny

Regulatory Basis for Requests for Additional Information:

By letter and application dated October 30, 2020, Energy Harbor Nuclear Corp. (the licensee) submitted a license amendment request (LAR) for the Beaver Valley Power Stations (BVPS), Units 1 and 2. In this LAR, the licensee requested staff approval of proposed changes to the currently approved PTLR. The provisions in 10 CFR 50.90 require TS changes of this nature to submitted for staff approval as a facility LAR. The October 30, 2021 submittal complies with the reporting requirement in 10 CFR 50.90.

Appendix A to 10 CFR 50 establishes minimum criteria (General Design Criteria or GDC) for the safe operation of light water reactors. GDC 31, “Fracture Prevention of Reactor Coolant Pressure Boundary”, requires that the reactor coolant pressure boundary be designed with sufficient margin to assure that the boundary behaves in a nonbrittle manner and that the probability of rapidly propagating failure is minimized. Among other considerations, it specifies that the design should reflect uncertainties in determining effects of irradiation on material properties of the reactor coolant pressure boundary.

Appendix G to 10 CFR 50 establishes minimally conservative procedures for licensees to evaluate the fracture toughness of ferritic components of the reactor pressure vessel (RPV), in order to guard against brittle failure. These procedures are intended to reflect changes in material properties due to irradiation, as well as anticipated uncertainty in estimating this effect. The criteria in Section IV.A.2.b of 10 CFR Part 50, Appendix G, “Fracture Toughness Requirements,” require that pressure-temperature limits identified as "ASME Appendix G limits" in Table 1 of the rule must be at least as conservative as the limits obtained by following the methods of analysis and the margins of safety in Appendix G of Section XI of the ASME Boiler and Pressure Vessel Code, Division 1.

NRC Regulatory Guide (RG) 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence," (ADAMS Accession No. ML010890301) provides guidance for determining whether methods for estimating vessel fluence are acceptable. It notes that a vessel fluence uncertainty of 20% (1 sigma) is acceptable for RT_{PTS} and RT_{NDT} determination. RG 1.190 also provides a correction (Equation 6) that can be applied to fluence estimates when the uncertainty is greater than 20%. However, the guidance states that when fluence uncertainty is greater than 30%, the methodology of the regulatory guide is not applicable, and the application should be reviewed on an individual basis.

RAI 1

Basis for the RAI:

Table 2 of Appendix B in the submittal describes fast neutron (E > 1.0 MeV) fluence values projected to 54 effective full-power years (EFPY) for the end of license extended (EOLE) period for beltline (including "extended" beltline) materials. The submittal references WCAP-18102-NP, Beaver Valley Unit 1 Heatup and Cooldown Curves for Normal Operation. The projected fluence values in the report are projected to 50 EFPY.

Request:

Provide consistent fast neutron (E > 1.0 MeV) fluence values and corresponding EFPY for the period applicable to the LAR for beltline (and "extended" beltline) materials.

RAI 2

Basis for the RAI:

As clarified in Regulatory Information Summary (RIS) 2014-11, "Information on Licensing Applications for Fracture Toughness Requirements for Ferritic Reactor Coolant Pressure Boundary Components" (ADAMS Accession No. ML14149A165), P-T limit calculations for ferritic RV components other than those materials with the highest reference temperature, may define curves that are more limiting than those calculated for the reactor vessel (RV) beltline shell materials because the consideration of stress levels from structural discontinuities (such as nozzles) may produce a lower allowable pressure. The P-T limits calculations for ferritic reactor coolant pressure boundary components that are not RV beltline shell materials (have projected neutron fluence values less than 1×10^{17} n/cm^2, E > 1 MeV) may define P-T curves that are more limiting than those calculated for the RV beltline shell materials because RV nozzles, penetrations and other discontinuities have complex geometries that may exhibit higher stresses than those for the RV beltline shell region. These higher stresses can potentially result in more restrictive P-T limits, even if the reference temperature (RT_{NDT}) for these components is not as high as that of the RV beltline shell materials that have simpler geometries.

Request:

a. Describe how the P-T limit curves for BVPS, Units 1 and 2, consider all ferritic pressure boundary components of the reactor vessel that are predicted to experience a neutron fluence exposure greater than $1x10^{17}$ n/cm^2 (E > 1 MeV) at the end of the licensed operating period.

b. If the current P-T limit curves do not consider all ferritic pressure boundary components of the reactor vessel that are predicted to experience a neutron fluence exposure greater than $1x10^{17}$ n/cm^2 (E > 1 MeV) at the end of the licensed operating period, provide appropriately revised P-T limit curves for review.

RAI 3

Basis for the RAI:

The submittal references WCAP-18102-NP, Beaver Valley Unit 1 Heatup and Cooldown Curves for Normal Operation, which provides a technical basis for the P-T limit curves for BVPS, Unit 1. The submittal states that for BVPS, Unit 2, fracture toughness evaluations for the nozzle shell, nozzle shell longitudinal welds and the nozzle shell to intermediate shell circumferential weld should be considered "since some of these materials have initial RT_{NDT} values and chemical compositions which could result in these materials becoming limiting compared to the beltline materials."

Request:

Provide the technical basis for BVPS, Unit 2, analogous to the information for BVPS, Unit 1, contained in WCAP-18102-NP, Rev. 0. Provide inputs and analysis of the methodology and results for the generation of heatup and cooldown curves for normal operation for BVPS, Unit 2.

RAI 4

Basis for the RAI:

The submittal refers to WCAP-18102-NP, Revision 0, "Beaver Valley Unit 1 Heatup and Cooldown Limit Curves for Normal Operation," which provides a technical basis for the P-T curves and cited U.S. NRC Technical Letter Report TLR-RES/DE/CIB-2013-01, "Evaluation of the Beltline Region for Nuclear Reactor Pressure Vessels," Office of Nuclear Regulatory Research [RES], dated November 14, 2014 as a basis for not considering the shift due to irradiation for RV materials for which the predicted shift in the reference temperature (ΔRT_{NDT}) is less than 25 degrees Fahrenheit (° F).

Discounting the shift in adjusted reference temperature (ART) due to irradiation if the predicted shift is less than 25°F does not meet the requirements of Appendix G to 10 CFR Part 50. Further, 10 CFR 50.61(a)(4) states that for the RV beltline materials, the ART must account for the effects of neutron radiation.10 CFR 50.61(c) details how ΔRT_{NDT} must be calculated. The staff notes that RIS 2014-11, "Information on Licensing Applications For Fracture Toughness Requirements for Ferritic Reactor Coolant Pressure Boundary Components," clarifies that the 10 CFR 50 Appendix G and 10 CFR 50 Appendix H define the beltline as including all RV materials that will receive a neutron fluence greater than or

equal to $1x10^{17}$ n/cm^2 (E> 1
MeV).

Request:

The staff therefore requests that, unless it plans to submit an exemption, the licensee:

1. Revise the ART calculations to include ART and margin values for all RV beltline and extended beltline materials for BVPS, Units 1 and 2.
2. If WCAP-18102-NP, Rev. 0 is maintained as a reference for BVPS, Unit 1, correct the associated tables and revise the document to remove the reference to TLR-RES/DE/CIB-2013-01.

RAI 5

Basis for the RAI:
Section 3 of Enclosure B to the LAR discusses the justification for applying the method described in WCAP-18124-NP-A to Beaver Valley Units 1 and 2, based on meeting regulatory positions in RG 1.190. Based on a combination of benchmarking and analytical uncertainty analysis, the licensee assesses the uncertainty of the fluence in the Beaver Valley Unit 2 nozzle shell, nozzle shell longitudinal welds, and the nozzle shell to intermediate circumferential weld of the reactor pressure vessel to be approximately 30%. Since this uncertainty is greater than the 20% threshold discussed in RG 1.190, the licensee proposes to use equation 6 of this guide to increase the fluence estimate in these regions. However, because the licensee states that the uncertainty is approximately 30%, it is unclear whether equation 6 of RG 1.190 applies. Further, although the benchmarking analysis is described in detail in the enclosure, the licensee does not provide details on the plant-specific analytical uncertainty analysis used to estimate the fluence uncertainty.

The licensee also indicates that the Beaver Valley Unit 2 nozzle shell, nozzle shell longitudinal welds, and the nozzle shell to intermediate circumferential weld of the reactor pressure vessel may be limiting with respect to traditional beltline materials, meaning that they will be used to determine pressure and temperature limits meant to ensure the integrity of the reactor pressure vessel. Without a listing of what sensitivities were accounted for in the uncertainty analysis, the NRC staff cannot conclude that uncertainties were appropriately considered when determining the effects of irradiation on material properties as per GDC 31 in 10 CFR 50 Appendix A.

Request:
Please indicate (a) whether the uncertainty analysis referred to in LAR Enclosure B deviated from that presented in WCAP-18124-NP-A, section 4.4, and if so, please indicate how. Additionally, for each of the parameters specified below, please indicate (b) whether each of the below potential contributors to uncertainty were taken into account in the uncertainty estimation or provide justification for not explicitly including them.

i. Approximations (such as homogenization) made in modeling core insulation, biological shield, and supplementary shield
ii. Uncertainties in width of gap between RPV and biological shield
iii. Homogenized density of water and structural materials between top of core and (including) upper grid plate
iv. Order of angular quadrature and anisotropic scattering treatment