NRR-PMDAPEm Resource

From: DiFrancesco, Nicholas
Sent: Monday, September 19, 2011 5:19 PM
To: joseph.bauer@exeloncorp.com
Cc: kevin.borton@exeloncorp.com; Lisa.Schofield@exeloncorp.com; Zimmerman, Jacob; Mahoney, Michael
Subject: DRAFT RAI Braidwood/Byron - Measurement Uncertainty Recapture LAR (TAC NOs. ME6587, ME6588, ME6589, and ME6590)
Attachments: Byron-Braidwood.DRAFT.MUR.LAR.RAI.#1.pdf

Mr. Joseph Bauer,

This e-mail is in reference to the Exelon Generation Company, LLC (EGC), submittal of a license amendment request
associated with a measurement uncertainty recapture power uprate for the Braidwood Station, Units 1 and 2, and Byron Station, Unit Nos. 1 and 2, dated June 23, 2011.

The Office of Nuclear Reactor Regulation (NRR) staff in the technical branches of Instrumentation and Control, Fire Protection, Health Physics & Human Performance, Containment and Ventilation, and Vessels and Internals Integrity has reviewed EGC's submittal and identified areas requiring additional information to complete their review. The staff has developed requests for additional information (RAI). The draft RAIs are provided as an attachment to this e-mail to support RAI clarification discussions and establish a response date. Please feel free to contact me with any questions and when ready to establishing a time for the RAI clarification call.

Sincerely,

Nicholas DiFrancesco

Project Manager - Braidwood, Byron, and Clinton
U.S. Nuclear Regulatory Commission
Office of Nuclear Reactor Regulation
Division of Operating Reactor Licensing
nicholas.difrancesco@nrc.gov | Tel: (301) 415-1115

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From: DiFrancesco, Nicholas

Created By: Nicholas.DiFrancesco@nrc.gov

Recipients:
"kevin.borton@exeloncorp.com" <kevin.borton@exeloncorp.com>
Tracking Status: None
"Lisa.Schofield@exeloncorp.com" <Lisa.Schofield@exeloncorp.com>
Tracking Status: None
"Zimmerman, Jacob" <Jacob.Zimmerman@nrc.gov>
Tracking Status: None
"Mahoney, Michael" <Michael.Mahoney@nrc.gov>
Tracking Status: None
"joseph.bauer@exeloncorp.com" <joseph.bauer@exeloncorp.com>
Tracking Status: None

Post Office:

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DRAFT REQUEST FOR ADDITIONAL INFORMATION REGARDING

LICENSE AMENDMENT REQUEST

MEASUREMENT UNCERTAINTY RECAPTURE POWER UPRATE

BRAIDWOOD STATION, UNITS 1 & 2

BYRON STATION, UNIT NOS. 1 & 2

DOCKET NOS. STN 50-454, STN 50-455,

STN 50-456, AND STN 50-457

TAC NOS. ME6587, ME6588, ME6589, AND ME6590

In reviewing of the Exelon Generation Company, LLC (EGC) license amendment request dated June 23, 2011, related to a measurement uncertainty recapture power uprate, for the Braidwood Station (Braidwood), Units 1 and 2, and Byron Station (Byron), Unit Nos. 1 and 2, the NRC staff has determined that the following information is needed in order to complete its review:

Request for Additional Information - Vessels and Internals Integrity Branch

1. Section IV.1.C.vi of Attachment 5 to the June 23, 2011, submittal stated, "the current capsule withdrawal schedule shown in each Unit's PTLR [Pressure Temperature Limits Report] will be updated to reflect the latest capsule fluence, lead factor, and withdrawal EFPY [effective full power years] associated with each capsule." The updated capsule withdrawal schedules for Byron, Units 1 and 2 can be found in Tables IV.1.C.vi-1 and IV.1.C.vi-2 and for Braidwood, Units 1 and 2 in Tables IV.1.C.vi-3 and IV.1.C.vi-4 of Attachment 5 to the submittal. Although Note (b) of these Tables stated that the information was updated as part of the measurement uncertainty recapture power uprate (MUR PU), the source or reference for the latest capsule fluence, lead factor, and withdrawal EFPY associated with each capsule is not given. Provide the references for the updated fluence, lead factor, and withdrawal EFPY for the Byron and Braidwood units' surveillance capsules. If no such references exist, provide calculation details for the updated values in this application.

2. For Byron, Unit 2, Section IV.1.C.iii of Attachment 5 to the June 23, 2011, submittal stated, "For Unit 2, the limiting ART [adjusted reference temperature] values used in the development of the current P-T limit curves at 32 EFPY are slightly lower than the MUR [PU] limiting ART values (at 32 EFPY)." The staff cannot verify this because this statement seems to contradict the information in Table IV.1.C.ii-1 of Attachment 5 where the maximum neutron fluence value of 2.06 E19 n/cm^2 (E > 1.0 MeV) on record (i.e., the 2006 PTLR for 32 EFPY) bounds the MUR PU maximum neutron fluence value of 1.76 E19 n/cm^2 (E > 1.0 MeV). Provide details regarding the calculation of the Byron, Units 1 and 2 RPV beltline material ARTs which demonstrate how the values in this submittal are different from the corresponding values in the 2006 Byron, Units 1 and 2 PTLRs. These ARTs will be considered as the licensing basis in support of the MUR PU license amendment request.

3. For the upper-shelf energy (USE) evaluation, Section IV.1.C.v of Attachment 5 to the June 23, 2011, submittal stated that the limiting projected ¼ T USE value is 65 ft-lbs for the nozzle-to-intermediate shell forging circumferential weld for Byron, Unit 1, 68 ft-lbs for the nozzle-to-intermediate shell forging circumferential weld for Byron, Unit 2, 75 ft-lbs for the intermediate-to-lower shell forging circumferential weld for Braidwood, Unit 1, and 66 ft-lbs for the intermediate-to-lower shell forging circumferential weld for Braidwood, Unit 2. However, the details regarding the calculation of these limiting USE values or appropriate references are not given in Attachment 5. The 2006 Byron PTLRs and the Braidwood PTLRs, Revision 4 contain no current USE estimates either. Please provide details regarding the calculation of the limiting USE values for the Byron and Braidwood units or references containing this information.

4. Section IV.1.A.ii of Attachment 5 to the EGC's June 23, 2011, submittal provides generic information for only a few reactor vessel (RV) internals under the MUR PU conditions. Table Matrix-1 of NRC RS-001, Revision 0, "Review Standard for Extended Power Uprates," provides the staff's basis for evaluating the potential for extended PU to induce aging effects on RV internals. Depending on the magnitude of the projected RV internals fluence, Table Matrix-1 may be applicable to the MUR application. In the Notes to Table Matrix-1, the staff states that guidance on the neutron irradiation-related threshold for irradiation-assisted stress corrosion cracking (SCC) for pressurized water reactor RV internal components are given in BAW-2248A, "Demonstration of the Management of Aging Effects for the Reactor Vessel Internals," and WCAP-14577, Revision 1-A, "License Renewal Evaluation: Aging Management for Reactor Internals." The "Notes" to Table Matrix-1 state that for thermal and neutron embrittlement of cast austenitic stainless steel, SCC, and void swelling, licensees will need to provide plant-specific degradation management programs or participate in industry programs to investigate degradation effects and determine appropriate management programs.

   The BAW-2248A report and the WCAP-14577, Revision 1-A have been superseded by the MRP-227 report, "Pressurized Water Reactor (PWR) Internals Inspection and Evaluation Guidelines," which summarized the industry's most current recommended inspection and evaluation guidelines for RV internals. The safety evaluation dated June 22, 2011, lists the limitations and conditions imposed by the staff on use of the MRP-227 report. Please confirm whether you have established an inspection plan to manage the age-related degradation in the Byron and Braidwood units RV internals, or whether you have participated in the industry's initiatives on age-related degradation of PWR RV internals and plan to submit your plant-specific program consistent with the MRP-227 report guidelines. For the former case, discuss your management of the above-mentioned aging effects on RV internals and demonstrate that the management is appropriate to ensure integrity and operability of RV internals to the end of license.

<u>Request for Additional Information - Fire Protection Branch</u>

5. The staff notes that the license amendment request (LAR), Attachment 5, MUR Technical Evaluation, Section VII.6.A, "Fire Protection Program," on page VII-5 states that, "…an analysis of the change in combustible loading determined that the overall increase in fire loading is small and does not change the fire load classification of each affected fire zone…"

It is unclear to the staff whether there are fire protection program plant modifications planned (e.g., adding new cable trays, or re-routing of existing cables) at MUR power uprate conditions. Clarify whether this request involves plant modifications, or changes to the fire protection program. If any, the staff requests the licensee to identify proposed modifications and discuss the impact of these modifications on the plant's compliance with the fire protection program licensing basis, Title 10 of the *Code of Federal Regulations,* Section 50.48, or applicable portions of Branch Technical Position CMEB 9.5-1.

6. The staff notes that the LAR, Attachment 5, MUR Technical Evaluation, Section VII.6.A.i, "Fire Protection Systems," on page VII-6, states that the fire protection water is utilized to supply the spent fuel pool and cooling water to the centrifugal charging pumps.

   Are there any other uses of fire water pumps and water for non-fire protection uses at Braidwood Station, Units 1 and 2, and Byron Station, Unit Nos. 1 and 2? If so, the MUR power uprate LAR should identify the specific situations and discuss to what extent, if any, the MUR power uprate affects these "non-fire-protection" aspects of the plant fire protection system.

   In your response discuss how any non-fire suppression use of fire protection water will impact the need to meet the fire protection system design demands.

<u>Request for Additional Information - Health Physics & Human Performance Branch</u>

7. Section 3.4.5 of Attachment 1, (Plant Modifications) includes a list of modifications of interest. The last two bullets of that list include various Balance of Plant (BOP) instrument rescaling, setpoint and alarm changes and ATWS Mitigation System time delay changes. In addition, Section VII.2.B of attachment 5 states that the MUR power uprate modification will implement the changes that are required to certain non-safety related systems, including Control Room displays and alarms. There is no other information regarding what these changes might look like or how the licensee plans to validate them. Please provide additional information regarding what potential modifications might be included, and how they will be verified and validated.

8. Section VII.2.A of attachment 5 states that changes to emergency operating procedure (EOP) and abnormal operating procedure (AOP) will be made in conformance with the Westinghouse EOP Setpoint Methodology. There is no additional information regarding this methodology. Please provide a description of the Westinghouse EOP Setpoint Methodology, or a reference if it is an approved methodology.

9. RIS 2002-03, "Guidance on the Content of Measurement Uncertainty Recapture Uprate Applications" asks for information on this topic:

   > "A statement confirming licensee intent to revise existing plant operating procedures related to temporary operation above 'full steady-state licensed power levels' to reduce the magnitude of the allowed deviation from the licensed power level. The magnitude should be reduced from the pre-power uprate value

of 2% to a lower value corresponding to the uncertainty in power level credited by the proposed power uprate application."

This was not addressed in the submittal. Please provide information to assure that they have revised the administrative controls for preventing inadvertent excursions above the new 100% power level.

Request for Additional Information - Containment and Ventilation Branch

10. Attachment 5 Section III.15.1 - Explain the Input Modification data base and how it is used.

11. Attachment 5 Section III.15.1 - Why were corrections necessary to the reactor coolant pump homologous curves? What were these corrections? Did the corrections significantly affect the results of the mass and energy release analyses?

12. Attachment 5 Section III.15.1 - Why is it assumed that containment spray is terminated at eight hours? What is the current assumption in the mass and energy release calculations and why was this assumption changed?

13. Attachment 5 Section III.15.1 - Why is the barrel baffle metal mass only included for upflow design plants? How significant is the omission of the barrel baffle metal mass in the current analyses?

14. Attachment 5 Section III.15.1 - Please describe or reference a description of the "identified inconsistencies" in the EPITOME computer code. Did these inconsistencies significantly affect the current mass and energy release calculation results?

15. Attachment 5 Section III.15.3 - Describe or reference the modeling of the Unit 1 Babcock and Wilcox (B&W) steam generators and the Unit 2 Westinghouse Model D5 steam generators with respect to mass and energy release calculations and explain why this modeling is conservative for mass and energy release calculations.

16. Attachment 5 Section III.15.1 4 and Attachment 5 Section III.16.4 - Describe or reference the computer codes and the assumptions used to derive the electrical equipment environmental qualification temperature and pressure profiles.

17. Attachment 5 Section III.16.2 - Please explain the difference between the statements

    The break flows and enthalpies of the steam release through the steamline break inside containment are analyzed with the LOFTRAN computer code…

    and

    Blowdown mass and energy releases were also determined using LOFTRAN….

18. Attachment 5 Section III.16.3 - Please describe the modeling of the two steam generator types for the MSLB analyses, especially those characteristics affecting the MSLB results, e.g., mass of water, location of nozzle, etc.

19. Attachment 5 Table III.16-1 - What is meant by a composite curve, how is it determined?

20. Attachment 5 Section VI.1.F.ii - This section states that the heat loads in a limited number of areas did increase due to the MUR and that the increase was minimal. Please state which areas would experience an increase in heat load and the magnitude of the increase. The maximum value of these increases is sufficient.

Request for Additional Information - Instrumentation and Control Branch

Regulatory Basis

Nuclear power plants are licensed to operate at a specified core thermal power. Appendix K, "ECCS Evaluation Models," to Title 10 of the Code of Federal Regulations (10 CFR) Part 50, "Domestic Licensing of Production and Utilization Facilities," requires loss-of-coolant accident and emergency core cooling system (ECCS) analyses to assume "that the reactor has been operating continuously at a power level at least 102% of the licensed thermal power level to allow for instrumentation uncertainties." Alternatively, Appendix K allows such analyses to assume a value lower than the specified 102 percent, but not less than the licensed thermal power level, "provided the proposed alternative value has been demonstrated to account for uncertainties due to power level instrumentation error." This allowance gives licensees the option of justifying a power uprate with reduced margin between the licensed power level and the power level assumed in the ECCS analysis by using more accurate instrumentation to calculate the reactor thermal power.

Because the maximum power level of a nuclear plant is a licensed limit, the NRC must review and approve a proposal to raise the licensed power level under the license amendment process. The LAR should include a justification for the reduced power measurement uncertainty to support the proposed power uprate.

The licensee developed the format of their submittal for the proposed power uprate based on the guidance of Regulatory Issue Summary (RIS) 2002-03, "Guidance on the Content of Measurement Uncertainty Recapture Power Uprate Applications," dated January 31, 2002 and an approved topical report – ER-157(P-A) Rev. 8 and Rev. 8 Errata, "Supplement to Caldon Topical Report ER-80P: Basis for Power Uprates with an LEFM Check or an LEFM CheckPlus System," (ML102950246). The NRC staff is evaluating the LAR using the provisions outlined in RIS 2002-03 as guidance.

Request for Additional Information

21. Engineering Report ER-800 Rev. 1 (ML111790063) contains several appendices labeled "A.1, A.2, A.3, A.4, & A.5." These appendices contain detailed calculations, the results of which appear to be summarized in Appendix C Table I. The NRC staff is having trouble identifying the equations in the approved topical report that correspond to the calculation in these appendices.

    Please provide a detailed and explicit cross reference between the June 23, 2011, letter (ML111790030) Attachment 8a (ML111790063) and the associated approved topical report equations (i.e., between ER-800 Rev. 1 Appendix A.1 and ML102950246).

22. Please confirm that the assumptions listed in Cameron Caldon Ultrasonics Engineering Report No. ER-157(P-A) Rev. 8 and Rev. 8Errata (ML102950246) Appendix A are valid for the Byron and Braidwood application.

23. Regulatory Guide 1.150 Rev. 3, “Setpoints for Safety-Related Instrumentation,” dated December 1999, describes a method for combining individual uncertainty terms in quadrature; this method assumes that the individual term each meet the 95/95 criteria. Please describe how each individual uncertainty term meets the 95/95 criteria.

24. Table I, “Reconciliation of Byron Unit 1 Nuclear Generating Station Uncertainties with Cameron Reports,” of Appendix C (page 5) of Cameron Engineering Report ER-800 Rev. 1 (ML111790063) was compared with Table A-1, “Representative Thermal Power Uncertainties for a Total Feedwater Flow Measurement in a PWR or BWR Using Chordal LEFM Check and LEFM CheckPlus” of ER-157(P-A) Rev. 8 and Rev.8Errata (ML102950246).

    The Byron document seems to misquote the numbers in the approved topical report in some places, for example:

    > Table I identifies the ER-157P value for the Hydraulics Profile Factor as being “+/- 0.25%” while the value in ER-157(P-A) Rev.8 And Rev.8Errata is “+/- 0.22%.”
    >
    > Table I identifies the ER-157P value for the Time Measurements as being “+/- 0.05%” while the value in ER-157(P-A) Rev.8 And Rev.8Errata is “+/- 0.06%.”

    The Byron document also seems to indicate that in some cases the Byron Unit 1 system is credited as being better than the bounding topical report, for example:

    > Table I identifies the Byron Unit 1 value for the Subtotal mass Flow Uncertainty as being “+/- 0.26%” while the value in ER-157(P-A) Rev.8 And Rev.8Errata is “+/- 0.28%.”
    >
    > Table I identifies the Byron Unit 1 value for the Feedwater Density and Feedwater Enthalpy as being “+/- 0.14%” while the value in ER-157(P-A) Rev.8 And Rev.8Errata is “+/- 0.15%.”

    Please explain.