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

February 14, 2012

Mr. Michael J. Pacilio
President and Chief Nuclear Officer
Exelon Nuclear
4300 Winfield Road
Warrenville, IL 60555

SUBJECT: BYRON STATION, UNIT NOS. 1 AND 2, AND BRAIDWOOD STATION, UNITS 1 AND 2 - REQUEST FOR ADDITIONAL INFORMATION RE: MEASUREMENT UNCERTAINTY RECAPTURE POWER UPRATE REQUEST (TAC NOS. ME6587, ME6588, ME6589, AND ME6590)

Dear Mr. Pacilio:

By letter to the U.S. Nuclear Regulatory Commission (NRC) dated June 23, 2011, as supplemented on August 25, 2011, Exelon Generation Company, LLC (EGC) submitted a 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.

The NRC staff is reviewing your submittal and has determined that additional information is required to complete the review. The specific information requested is addressed in the enclosure to this letter. During a discussion with your staff on January 18, 2012, it was agreed that you would provide a response by February 20, 2012.

The NRC staff considers that timely responses to requests for additional information help ensure sufficient time is available for staff review and contribute toward the NRC's goal of efficient and effective use of staff resources. If circumstances result in the need to revise the requested response date, please contact me at (301) 415-2020.

Sincerely,

Brenda Mozafari

Brenda Mozafari, Senior Project Manager
Plant Licensing Branch III-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

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

Enclosure:
Request for Additional Information

cc w/encl: Distribution via Listserv

REQUEST FOR ADDITIONAL INFORMATION

BYRON STATION, UNIT NOS. 1 AND 2, AND BRAIDWOOD STATION, UNITS 1 AND 2,

REGARDING MEASUREMENT UNCERTAINTY RECAPTURE POWER UPRATE

LICENSE AMENDMENT REQUEST

DOCKET NOS. STN 50-454, STN 50-455, STN 50-456 AND STN 50-457

TAC NOS. ME6587, ME6588, ME6589, ME6590

In reviewing the Exelon Generation Company, LLC (EGC) submittal to the U.S. Nuclear Regulatory Commission (NRC) dated June 23, 2011, and supplemented on August 25, 2011 (Agencywide Documents Access and Management System (ADAMS) Accession Nos. ML111790030 and ML11255A332, respectively), related to a measurement uncertainty recapture (MUR) 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:

Mechanical and Civil Engineering Branch

1. Section IV.1.A.ii.f of Attachment 7 to the license amendment request (LAR) discusses the structural evaluation of the lower and upper core support assemblies for the effects of increased heat generation rates. Provide further information and confirm that:

    a. the proposed MUR power uprate only affects the design loads associated with heat generation rates and all other design loads associated with the design of the reactor vessel internals are unaffected by the proposed MUR power uprate;
    b. all design loading conditions, as noted in Section 3.9.5.2 of the Byron and Braidwood updated final safety analysis report (UFSAR), were considered in the structural re-evaluation of the reactor vessel internal components to assess the impact of the proposed MUR power uprate; and
    c. the original design codes of record were utilized in the structural re-evaluation of the reactor vessel internal components.

    Provide the maximum calculated stresses and cumulative fatigue usage factor for the most limiting component of the reactor vessel internals and their respective comparison with the Byron and Braidwood design basis acceptance criteria.

2. Section 3.9.5.1 of the Byron and Braidwood UFSAR describes the reactor vessel internals as three parts consisting of the lower core support structure, the upper core support structure, and the incore instrumentation support structure. Section IV of Attachment 7 to the LAR does not discuss the incore instrumentation support structures. Provide further information relative to the impact of the design conditions associated with the proposed MUR power uprate on the incore instrumentation support structures.

Enclosure

3. Provide further information and confirm that, for the proposed MUR power uprate conditions, the maximum deflection values allowed for the reactor vessel internal support structures, as noted in Table 3.9-4 of the Byron and Braidwood UFSAR, are maintained.

4. Section IV.1.B.iv.1 of Attachment 7 to the LAR states that there is an approximate 1.2 °F increase in temperature difference across the core (Thot increases approximately 0.6 °F and Tcold decreases approximately 0.6 °F) from current operating conditions due to the MUR power uprate. Section IV.1.A.i of Attachment 7 to the LAR discusses reactor vessel structural evaluation and states that due to operational restrictions, the MUR minimum vessel inlet and maximum vessel outlet temperatures are limited to 538.2 °F and 618.4 °F, respectively. Provide further clarification on temperature effects relative to the values in Tables 3-1 and 3-2 of Attachment 1 to the LAR, the statements in Sections IV.1.B.iv.1 and IV.1.A.i of the LAR, and the temperatures used in the analysis of record.

   Furthermore, the lifting lug loads and evaluation are discussed in Section IV.1.A.i of Attachment 7 to the LAR. The terminology of "lifting lug" and its relation to and its inclusion in the proposed MUR power uprate license amendment is not clear. Provide further information to clarify which reactor vessel component is referred to as "lifting lug." Also, regarding the affected reactor vessel component,

   a. provide a table summarizing the comparison of design parameters for the current operation conditions, MUR power uprate conditions, and design basis conditions; and
   b. provide the maximum calculated stresses and cumulative fatigue usage factors at the most critical location of the affected component and their respective comparison with the Byron and Braidwood design basis acceptance criteria.

5. Section IV.1.A.iii of Attachment 7 to the LAR discusses the control rod drive mechanism (CRDM). In this section, it is stated that updated seismic and loss-of-coolant accident (LOCA) loads remain less than the allowable loads provided in the analysis of record. This statement implies that the seismic loads have been updated. Also, this statement is not consistent with Section IV.1.A.ii.e of Attachment 7 to the LAR where it is stated that the proposed MUR power uprate conditions do not affect the current design basis for seismic and LOCA loads. Provide further clarification.

   Furthermore, Section IV.1.A.iii of Attachment 7 to the LAR states that CRDM is subjected to Tcold temperatures and reactor coolant system pressures and these are the only design parameters considered in the CRDM evaluation. Elaborate and confirm that:

   a. the design basis loading conditions and operational requirements, as noted in Section 3.9.4 of the Byron and Braidwood UFSAR, have been considered in the structural evaluation of the control rod drive system for the proposed MUR power uprate conditions; and

b. the control rod drive system will continue to be in compliance with the Byron and Braidwood design basis acceptance criteria under the proposed MUR power uprate conditions.

6. Provide further information and confirm that the design basis pressure and temperatures (normal operating and accident temperatures) used in the design of the containment structure, including the steel liner plate and its internal structures, remain bounding following the proposed MUR power uprate.

7. Section IV.1.A.iv "Reactor Coolant Piping and Supports" of Attachment 7 to the LAR discusses the effects of the proposed MUR power uprate mostly on a qualitative basis and the term "no significant changes" has been used in several areas to describe the impact of the proposed MUR power update. Discuss in more detail the information relative to the revised design conditions, before and after the proposed MUR power uprate, for those components evaluated under Section IV.1.A.iv of Attachment 7 to the LAR. Summarize the results of any additional evaluations performed for the affected components and indicate whether these components remain bounded by the current analysis of record. For those components that were not bounded by the analysis of record:

    a. provide the maximum calculated stresses and cumulative fatigue usage factors at the most critical location; and

    b. provide further clarification that the re-evaluation was performed in accordance with the design basis code of record and the affected components continue to remain in compliance with the Byron and Braidwood stations design basis acceptance criteria.

8. Section IV.1.A.v of Attachment 7 to the LAR discusses the evaluation of balance of plant (BOP) piping systems. Confirm that other BOP piping systems (e.g., chemical and volume control, auxiliary feedwater, fuel pool cooling, containment spray, essential service water, safety injection) that may be affected by the MUR uprate conditions have been evaluated and provide a complete list of BOP piping systems evaluated in support of MUR power uprate. Discuss the methodology used for evaluating BOP piping, including pipe supports, and provide further information relative to the design conditions in each BOP piping system, before and after the proposed MUR power uprate. Summarize the results of the additional evaluations performed for the affected piping systems and indicate whether these piping systems remain bounded by the current analysis of record. For those BOP piping systems not bounded by the current analysis of record:

    a. provide the maximum calculated stresses and cumulative fatigue usage factors at the most critical location in each unbounded piping system; and

    b. provide further clarification that the re-evaluation of the piping system, including pipe supports, was performed in accordance with the design basis code of record and in compliance with the Byron and Braidwood stations design basis

c. acceptance criteria. Furthermore, state whether any piping or pipe support modifications are required to support the proposed MUR power uprate.

9. Section IV.1.A.viii of Attachment 7 to the LAR discusses the pressurizer structural evaluation. In this section of the LAR, it is stated that the revised design parameters have an insignificant impact on the fatigue analysis results. It is also stated that the proposed MUR power uprate has a negligible impact on the qualification of the pressurizer surge, spray, safety, and relief nozzle structural weld overlay designs. Provide further information to support the above qualitative statements and to demonstrate compliance with the Byron and Braidwood design basis acceptance criteria. Also, provide a table summarizing the comparison of pressurizer design parameters for the current operation conditions, MUR power uprate conditions, and design basis conditions.

10. Section IV.1.B.iii of Attachment 7 to the LAR discusses the evaluation of the reactor vessel internal components for flow induced vibration (FIV) impact under MUR power uprate conditions. Also, Section IV.1.A.ii.e of Attachment 7 to the LAR states that the FIV stress levels on the core barrel assembly and upper internals are below the material high-cycle fatigue endurance limit and the proposed MUR uprated conditions do not affect the structural margin for FIV. Provide further information relative to those design parameters, before and after MUR power uprate, which could potentially influence FIV response of the reactor internals. Also, Discuss the comparison of alternating stress intensities to design basis allowable limits for the most critical components demonstrating compliance with the Byron and Braidwood design basis acceptance criteria.

11. Discuss further information and confirm that the nuclear team supply system component supports, as discussed in Section 3.9.3.4 of the Byron and Braidwood UFSAR, will continue to be in compliance with the Byron and Braidwood design basis acceptance criteria at the proposed MUR power uprate conditions. Also, confirm that the operating temperatures for support elements, as defined in Table 3.9-17 of the Byron and Braidwood UFSAR, are not affected by the MUR power uprate.

12. Section IV.1.A.vi.1.b of Attachment 7 to the LAR discusses the structural evaluation of Byron and Braidwood Unit 1 replacement steam generators and states that a reconciliation analysis was performed to address the structural integrity of the entire steam generator pressure boundary for the MUR power uprate conditions. Discuss further information relative to, before and after uprate, the maximum stress intensity and the cumulative fatigue usage factors for the critical components of the primary and secondary sides, including nozzles, of the replacement steam generators and the respective service conditions. Also, confirm that the reconciliation analysis was performed in accordance with the original design code of record and in compliance with the Byron and Braidwood design basis acceptance criteria.

13. Discuss further information to demonstrate that, for the expected post-uprate conditions, the spent fuel pool (SFP) structure, including SFP liner and the spent fuel racks, remain capable of performing their intended design functions and will continue to be in compliance with the Byron and Braidwood design basis code of record(s) and acceptance criteria.

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/RA/

Brenda Mozafari, Senior Project Manager
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