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

February 24, 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, 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 needed to complete the review. The specific information requested is addressed in the enclosure to this letter. During a discussion with your staff documented in an email dated January 18, 2012, (Agencywide Documents Access and Management System ADAMS Accession No. ML120380394) 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 as supplemented on August 25, 2011 (Agencywide Documents Access and Management System (ADAMS) Accession Nos. ML111790030 and ML11255A332, respectively), 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 to complete its review:

Balance of Plant Branch

1. Technical Specification (TS) 3.7.4 for the steam generator (SG) power-operated relief valves (PORVs) currently allows 24 hours completion time to restore all but one of the four PORVs when two or more PORVs are inoperable. Hence, the TS action statement would allow all four PORVs to be inoperable for 24 hours.

   Technical Specification (TS) 3.7.4 for the SG PROVs currently allows 24 hours completion time to restore all but one of the four SG PORVs when two or more are operable. Hence, the TS action statement allows all four SG PROVs inoperable for up to 24 hours. Westinghouse Standard TS does provide some guidance on multiple SG PROVs inoperable based upon availability of the steam bypass system and main steam safety valves. However, for Byron/Braidwood these components may not be available in a loss of off-site power or may not provide the same cool down function. Hence, the licensee may have to provide alternate (safety-related or available during a LOOP) components that can perform the same function to justify continued operations with no operable SG PORVs to provide the safety function.

   Justify the current TS action statement that allows all four SG PORVs to be inoperable based on the new steam generator tube rupture (SGTR) analysis.

2. The licensee identifies the SG PORVs as being a key component in mitigating an SGTR from an overfill condition. The licensee identified an SG PORV failing to open on one of the intact SGs as the most limiting failure for the margin to overfill (MTO) analysis. The installation of an uninterruptible power supply was made to reduce the current vulnerability of a single-failure, making two SG PORVs inoperable.

In Table I-2, "Steam Generator Tube Rupture Equipment List," the licensee states, "Table I-2 identifies the systems, components, and instrumentation which are credited for accident mitigation." The Table I-2 does not list the SG PORV controllers.

Provide a description of the PORVs electrical systems to include power supplies to the controllers and circuitry, and include any other circuits that would affect the SG PORV's ability to perform its function; identify any shared components (i.e., electrical, mechanical, Instrumentation & Control, etc.); and justify not including the SG PORV controllers.

3. The licensee is making modifications to the auxiliary feedwater (AFW) flow control valves to include an air accumulator tank capable of supplying air for 30 minutes. In accordance with their analysis, AFW flow control is required longer than 30 minutes to mitigate the SGTR and for RCS cool down. In Attachment 5a, Section II.2.E, "Single Failure Considerations," the licensee states:

> In addition, since the failure of an intact SG PORV scenario assumes a loss of offsite power with an associated loss of Instrument Air (IA), the modification described in Section II.2.F, Item 1, assures that AFW flow control is maintained throughout the event.

According to the licensee's evaluation, an SGTR event continues until break flow is terminated at 3458/3258 seconds (Units 1 and 2 at both stations).

Describe the basis for selecting 30 minutes, and explain how the amount of air that is required is determined and the amount of air available to support this function.

4. Figure II-5 of Attachment 5a shows the SG water volume on Units 1 trending towards the maximum available quantity. At approximately 3200 seconds, the trend tapers off, resulting in a margin to overfill of approximately 94 cubic feet. At the same time other graphs show a sharp reduction in SG pressure, which logically corresponds to a second opening of the SG PORVs on the intact SGs. This action stops the upward trend and prevents the overfill condition. The licensee does not identify a critical operator action to open the SG PORVs a second time within a certain time period as a condition to prevent an overfill of the SG.

In the updated final safety analysis report, Section 15.6.3.2, under the section describing major operator actions, the licensee's analysis credits operators for reopening pressurizer PORVs four minutes after establishing normal charging and letdown, in order to equalize the RCS and SG pressures.

In Attachment 5a (page II-10), the licensee states that the SG PORVs on the intact SGs automatically open, as necessary, to maintain RCS subcooling margin. The above mentioned graph trend shows a sharp pressure reduction at 3200 seconds, which is not indicative of an SG PORV automatically controlling pressure at a prescribed setpoint.

Evaluate whether this operator action is credited to be performed within a specific time in order to prevent an overfill condition.

a. If operator action is required, identify the action as a critical operator action.

b. Describe whether the new analysis changes the existing UFSAR analysis, and results in the major operator action opening a SG PORV rather than a pressurizer PORV after SI termination to stop an overfill condition from occurring.

5. Calculation Westinghouse commercial atomic power (WCAP) -10698-P-A provides a general assessment of the MTO for Westinghouse type reactors. There were instances where the licensee deviated from the input parameters selected in WCAP-10698-P-A as the most conservative.

   a. Decay heat is one of the input factors that influence MTO analyses and Thermal/Hydraulic analyses during a tube rupture. For the MTO analysis, the licensee states that plant specific sensitivities were performed for Bryon and Braidwood Units 1 and 2. These studies concluded that the 1979-2σ American Nuclear Society (ANS) decay heat factor was more conservative compared to the 1971 +20% ANS decay heat model specified in WCAP-10698-P-A.

      Justify use of the 1979-2σ ANS decay heat factor was more conservative compared to the 1971 +20% ANS decay heat factor.

   b. Similar to above, in determining the most conservative input values, the licensee chose to model the minimum AFW enthalpy of 0.03 Btu/lbm; whereas, WCAP-10698-P-A models the maximum temperature of AFW (maximum enthalpy) as the most conservative parameter in the analysis for MTO.

      Justify how the use of the minimum AFW enthalpy is more conservative compared to using the maximum temperature (enthalpy) for AFW.

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Sincerely,

/RA/
Brenda Mozafari, Senior Project Manager
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Enclosure:
Request for Additional Information

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DISTRIBUTION:
PUBLIC
RidsNrrDorlLpl3-2 Resource
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ADAMS Accession No. ML120530258
NRR-088

OFFICE	LPL3-2/PE	LPL3-2/PM	LPL3-2/LA	LPL3-2/BC	LPL3-2/PM
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