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10 CFR 50.90
10 CFR 50, Appendix K

RS-11-099

June 23, 2011

U. S. Nuclear Regulatory Commission
ATTN: 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: Request for License Amendment Regarding Measurement Uncertainty Recapture (MUR) Power Uprate

Reference: NRC Regulatory Issue Summary (RIS) 2002-03, "Guidance on the Content of Measurement Uncertainty Recapture Power Uprate Applications," dated January 31, 2002

In accordance with 10 CFR 50.90, "Application for amendment of license or construction permit," and 10 CFR 50, Appendix K, "ECCS Evaluation Models," Exelon Generation Company, LLC (EGC) requests an amendment to Facility Operating License Nos. NPF-72, NPF-77, NPF-37 and NPF-66, and Appendix A, Technical Specifications (TS), for Braidwood Station, Units 1 and 2,and Byron Station, Units 1 and 2, respectively. The proposed changes would revise the maximum power level specified in each unit's operating license and the TS definition of rated thermal power (RTP). Specifically, the proposed change requests an increase from the current licensed thermal power (CLTP) of 3586.6 megawatts thermal (MWt) to 3645 MWt; an increase of approximately 1.63% RTP. Other TS changes associated with this license amendment request are summarized in Attachment 1, "Evaluation of Proposed Changes." Once approved, associated changes to the Braidwood Station and Byron Station Core Operating Limits Report and Technical Requirements Manual will be made in support of the uprated power conditions.

The proposed changes are based on increased feedwater (FW) flow measurement accuracy, which will be achieved by utilizing Cameron International (formerly Caldon) CheckPlus™ Leading Edge Flow Meter (LEFM) ultrasonic flow measurement instrumentation. The LEFM instrumentation has been or will be installed in each of the four units as follows:

- Braidwood Unit 1 Spring 2012 (during refueling outage A1R16)
- Braidwood Unit 2 Spring 2011 (during refueling outage A2R15 - completed)
- Byron Unit 1 Spring 2011 (during refueling outage B1R17 - completed)
- Byron Unit 2 Fall 2011 (during refueling outage B2R16)

A comprehensive evaluation has been completed for Byron Station, Units 1 and 2 and Braidwood Station, Units 1 and 2, to confirm that the requested increase in licensed rated thermal power is acceptable. The evaluations/analyses were performed to bound the requested increase in RTP to 3645 MWt (i.e., an increase of 1.63%). These evaluations addressed design transients, accidents, nuclear fuel, nuclear steam supply system (NSSS) systems and Balance of Plant (BOP) systems. Note that some of the analyses were performed utilizing the VIPRE subchannel analysis code and associated DNB correlations, ABB-NV and WLOP. Use of these codes is necessary to restore adequate DNB margin under MUR operating conditions. The results of all analyses and evaluations performed were found to be acceptable and will adequately support MUR uprated power conditions.

In addition to the above changes, a revised Steam Generator Tube Rupture (SGTR) and Margin to Overfill (MTO) Analysis is being submitted for NRC approval. This revised analysis was performed as the MTO values in the current analysis of record (AOR) are unacceptably small and revisions to the analysis assumptions are necessary. NRC approval of this reanalysis is required as the proposed changes result in more than a minimal increase in the accident dose as defined in NEI 96-07, "Guidelines for 10 CFR 50.59 Implementation," Revision 1, dated November 2000. The detailed results of the SGTR and MTO Analysis are provided in Attachment 5a, "Steam Generator Tube Rupture and Margin to Overfill Analysis Report." Note that the revised analysis did not result in any TS changes.

The content of this request is consistent with the guidance contained in NRC Regulatory Issue Summary (RIS) 2002-03, "Guidance on the Content of Measurement Uncertainty Recapture Power Uprate Applications," dated January 31, 2002.

This request is subdivided as follows.

Attachment 1: provides a description and evaluation of the proposed changes.

Attachment 2: provides a markup of the affected Operating License and TS pages.

Attachment 3: provides a markup of the affected TS Bases and Technical Requirements Manual pages. These pages are provided for information only.

Attachment 4: provides a summary of the regulatory commitments made in this request.

Attachment 5: provides the "Braidwood and Byron Stations Measurement Uncertainty Recapture Technical Evaluation," (Proprietary Version). This attachment provides the information requested in NRC Regulatory Information Summary (RIS) 2002-03.

Attachment 5a: provides the “Steam Generator Tube Rupture and Margin to Overfill Analysis Report.”

Attachment 6: provides affidavit from Westinghouse Electric Company supporting withholding of Attachment 5.

Attachment 7: provides the “Braidwood and Byron Stations Measurement Uncertainty Recapture Technical Evaluation," (Non-Proprietary Version).

Attachment 8a: provides the Cameron Engineering Report, ER-800, Rev. 1, "Bounding Uncertainty Analysis for Thermal Power Determination at Byron Unit 1 Using the LEFM CheckPlus System," (Proprietary Version) and it’s associated Appendix A.3, ER-829, Rev. 1, “Meter Factor Calculation and Accuracy Assessment for Byron Unit 1.”

Attachment 8b: provides the Cameron Engineering Report, ER-801, Rev. 1, "Bounding Uncertainty Analysis for Thermal Power Determination at Byron Unit 2 Using the LEFM CheckPlus System," (Proprietary Version) and it’s associated Appendix A.3, ER-832, Rev. 1, “Meter Factor Calculation and Accuracy Assessment for Byron Unit 2.”

Attachment 8c: provides the Cameron Engineering Report, ER-802, Rev. 1, "Bounding Uncertainty Analysis for Thermal Power Determination at Braidwood Unit 1 Using the LEFM CheckPlus System," (Proprietary Version) and it’s associated Appendix A.3, ER-843, Rev. 0, “Meter Factor Calculation and Accuracy Assessment for Braidwood Unit 1.”

Attachment 8d: provides the Cameron Engineering Report, ER-803, Rev. 1, "Bounding Uncertainty Analysis for Thermal Power Determination at Braidwood Unit 2 Using the LEFM CheckPlus System," (Proprietary Version) and it’s associated Appendix A.3, ER-844, Rev. 0, “Meter Factor Calculation and Accuracy Assessment for Braidwood Unit 2.”

Attachment 9: provides an affidavit from Cameron International Corporation supporting withholding of Attachments 8a through 8d.

Attachment 10a: provides PJM Interconnection document, "Generator Transient Stability Study for Braidwood Station," and ComEd document, "2012 Power Grid Voltage Analysis for Braidwood Station with MUR Power Uprate."

Attachment 10b: provides PJM Interconnection document, "Generator Transient Stability Study for Byron Station," and ComEd document, "2012 Power Grid Voltage Analysis for Byron Station with MUR Power Uprate."

Attachment 11: provides drawings describing the typical installation of the LEFM for all Byron and Braidwood Units.

The proposed changes have been reviewed by the Braidwood and Byron Plant Operations Review Committees and approved by the Nuclear Safety Review Board in accordance with the requirements of the EGC Quality Assurance Program.

EGC requests approval of the proposed changes by June 30, 2012. The requested review period is consistent with NRC internal guidance and supports initiatives to increase EGC's generation capacity. Once approved, the amendment will be implemented within 180 days of approval assuming a June 30, 2012 approval date.

Note that each unit may implement the amendment at different time depending on necessary outage related activities, modifications, or the potential need to promptly implement the VIPRE and associated DNB codes to obtain appropriate DNB margin. The 180 day implementation period was selected simply to bound the needs for all units. A summary of the anticipated implementation schedules are as follows:

- Braidwood Unit 1: Within 180 days after NRC approval

  This implementation period will allow for installation of the LEFM instrumentation and other MUR-related modifications during A1R16 (Spring 2012), installation of the SG PORV trim modification (after the MUR LAR approval) and revision of the affected station documents.

- Braidwood Unit 2: Within 180 days after NRC approval

  This implementation period will allow for installation of the LEFM instrumentation during A2R15 (Spring 2011 - completed), installation of other MUR-related modifications during the A2R16 (Fall 2012), and revision of the affected station documents.

- Byron Unit 1: Within 180 days after NRC approval

  This implementation period will allow for installation of the LEFM instrumentation during B1R17 (Spring 2011 - completed); implementation of VIPRE and the associated DNB codes in advance of the B1R18 outage (Fall 2012) (if necessary to restore adequate DNB margin); installation of other MUR-related modifications during B1R18; and revision of the affected station documentation. Power increase to MUR power level is planned after B1R18.

- Byron Unit 2: Within 180 days after NRC approval

  This implementation period will allow for installation of the LEFM instrumentation and other MUR-related modifications during B2R16 (Fall 2011) and revision of the affected station documents.

In accordance with 10 CFR 50.91, "Notice for public comment; State consultation," paragraph (b), EGC is notifying the State of Illinois of this application for license amendment by transmitting a copy of this letter and its attachments to the designated State Official.

In accordance with 10 CFR 2.390, "Public inspections, exemptions, requests for withholding," EGC requests withholding of Attachments 5 and 8a through 8d. Attachment 5 contains information considered proprietary by Westinghouse Electric Company, the owner of this information. An affidavit from Westinghouse supporting this request is included in Attachment 6. A non-proprietary version of Attachment 5 is provided in Attachment 7. Attachments 8a through 8d also contain information considered proprietary by Cameron International Corporation, the owner of this information. An affidavit from Cameron supporting this request is included as Attachment 9. Non-proprietary versions of Attachments 8a through 8d are not available.

Should you have any questions concerning this request, please contact Mr. Joseph A. Bauer at (630) 657-3376.

I declare under penalty of perjury that the foregoing is true and correct. Executed on the 23rd day of June, 2011.

Respectfully,

Craig Lambert
Vice President - Power Uprate
Attachments:

| | |
|---|---|
| Attachment 1: | Description and evaluation of the proposed changes. |
| Attachment 2: | Markup of the affected Operating License and TS pages. |
| Attachment 3: | Markup of the affected TS Bases and Technical Requirements Manual pages. These pages are provided for information only. |
| Attachment 4: | Summary of the regulatory commitments made in this request. |
| Attachment 5: | Braidwood and Byron Stations Measurement Uncertainty Recapture Technical Evaluation, (Proprietary Version). This attachment provides the information requested in NRC Regulatory Information Summary (RIS) 2002-03. |
| Attachment 5a: | Steam Generator Tube Rupture and Margin to Overfill Analysis Report. |
| Attachment 6: | Affidavit from Westinghouse Electric Company supporting withholding of Attachment 5. |
| Attachment 7: | Braidwood and Byron Stations Measurement Uncertainty Recapture Technical Evaluation, (Non-Proprietary Version). |

Attachment 8a: Cameron Engineering Report, ER-800, Rev. 1, "Bounding Uncertainty Analysis for Thermal Power Determination at Byron Unit 1 Using the LEFM CheckPlus System," (Proprietary Version) and it's associated Appendix A.3, ER-829, Rev. 1, "Meter Factor Calculation and Accuracy Assessment for Byron Unit 1."

Attachment 8b: Cameron Engineering Report, ER-801, Rev. 1, "Bounding Uncertainty Analysis for Thermal Power Determination at Byron Unit 2 Using the LEFM CheckPlus System," (Proprietary Version) and it's associated Appendix A.3, ER-832, Rev. 1, "Meter Factor Calculation and Accuracy Assessment for Byron Unit 2."

Attachment 8c: Cameron Engineering Report, ER-802, Rev. 1, "Bounding Uncertainty Analysis for Thermal Power Determination at Braidwood Unit 1 Using the LEFM CheckPlus System," (Proprietary Version) and it's associated Appendix A.3, ER-843, Rev. 0, "Meter Factor Calculation and Accuracy Assessment for Braidwood Unit 1."

Attachment 8d: Cameron Engineering Report, ER-803, Rev. 1, "Bounding Uncertainty Analysis for Thermal Power Determination at Braidwood Unit 2 Using the LEFM CheckPlus System," (Proprietary Version) and it's associated Appendix A.3, ER-844, Rev. 0, "Meter Factor Calculation and Accuracy Assessment for Braidwood Unit 2."

Attachment 9: Affidavit from Cameron International Corporation supporting withholding of Attachments 8a through 8d.

Attachment 10a: PJM Interconnection document, "Generator Transient Stability Study for Braidwood Station," and ComEd document, "2012 Power Grid Voltage Analysis for Braidwood Station with MUR Power Uprate."

Attachment 10b: PJM Interconnection document, "Generator Transient Stability Study for Byron Station," and ComEd document, "2012 Power Grid Voltage Analysis for Byron Station with MUR Power Uprate."

Attachment 11: Typical installation drawings of the LEFM for all Byron and Braidwood Units.

cc: NRC Regional Administrator, Region III
NRC Senior Resident Inspector – Braidwood Station
NRC Senior Resident Inspector – Byron Station
Illinois Emergency Management Agency – Division of Nuclear Safety

**ATTACHMENT 1**
**Evaluation of Proposed Changes**

## 1.0 SUMMARY DESCRIPTION

In accordance with 10 CFR 50.90, "Application for amendment of license or construction permit," and 10 CFR 50, Appendix K, "ECCS Evaluation Models," Exelon Generation Company, LLC (EGC) requests an amendment to Facility Operating License Nos. NPF-72, NPF-77, NPF-37 and NPF-66, and Appendix A, Technical Specifications (TS), for Braidwood Station, Units 1 and 2,and Byron Station, Units 1 and 2, respectively. The proposed changes would revise the maximum power level specified in each unit's operating license and the TS definition of rated thermal power (RTP). Specifically, the proposed change requests an increase from the current licensed thermal power (CLTP) of 3586.6 megawatts thermal (MWt) to 3645 MWt; an increase of approximately 1.63% RTP.

The proposed Measurement Uncertainty Recapture power uprate (MUR-PU) is based on a change in instrumentation error assumptions specified in 10 CFR 50, Appendix K, "ECCS Evaluation Models." Prior to the subject change, Appendix K required the following: "…it must be assumed that the reactor is operating continuously at a power level at least 1.02 times the licensed power level (to allow for instrumentation error), with the maximum peaking factor allowed by the technical specifications." The NRC approved a change to the Appendix K requirements on June 1, 2000 (effective July 31, 2000), that allowed licensees the option that states: "An assumed power level lower than the level specified in this paragraph (but not less than the licensed power level) may be used provided the proposed alternative value has been demonstrated to account for uncertainties due to power level instrumentation errors."

The reduction in the ECCS evaluation model assumed power level is justified by increased feedwater flow measurement accuracy, which will be achieved by utilizing Cameron International (formerly Caldon) CheckPlus™ Leading Edge Flow Meter (LEFM) ultrasonic flow measurement instrumentation. LEFM instrumentation has been or will be installed in each of the Braidwood Station and Byron Station units as follows:

- Braidwood Unit 1 Spring 2012 (during refueling outage A1R16)
- Braidwood Unit 2 Spring 2011 (during refueling outage A2R15 - completed)
- Byron Unit 1 Spring 2011 (during refueling outage B1R17 - completed)
- Byron Unit 2 Fall 2011 (during refueling outage B2R16)

Each unit's LEFM system will be installed prior to implementation of these requested changes.

The LEFM will be used as the normal (preferred) FW flow indication in lieu of the current venturi-based feedwater flow indication and RTD temperature indication to perform the plant calorimetric measurement calculation. The currently installed venturi-based feedwater flow instruments will continue to provide inputs to other indication, protection and control systems, and will be used if the LEFM is not functional as defined in Technical Requirements Manual, TLCO 3.3.k, "Feedwater Flow Instrumentation," described in Section 2.0 below.

Other TS changes associated with this MUR license amendment request are summarized in Section 2.0, "Detailed Description," below. Note that there are no TS setpoint changes associated with this amendment request. After the license amendment is approved, prior to implementation, associated changes to the Braidwood Station and Byron Station Core Operating Limits Reports (COLR), and Updated Final Safety Analysis Report (UFSAR) will be made in support of the uprated power conditions. The proposed amendment would also modify the Technical Requirements Manual (TRM) to add a TRM Limiting Condition for Operation (TLCO) addressing the LEFM system operability requirements.

EGC requests approval of the proposed changes by June 30, 2012. The requested review period is consistent with NRC internal guidance and supports initiatives to increase EGC's generation capacity. Once approved, the amendment will be implemented within 180 days of approval assuming a June 30, 2012 approval date.

Note that each unit may implement the amendment at different time depending on necessary outage related activities, modifications, or the potential need to promptly implement the VIPRE and associated DNB codes to obtain appropriate DNB margin. The 180 day implementation period was selected simply to bound the needs for all units. A summary of the anticipated implementation schedules are as follows

- Braidwood Unit 1: Within 180 days after NRC approval

  This implementation period will allow for installation of the LEFM instrumentation and other MUR-related modifications during A1R16 (Spring 2012), installation of the SG PORV trim modification (after the MUR LAR approval) and revision of the affected station documents.

- Braidwood Unit 2: Within 180 days after NRC approval

  This implementation period will allow for installation of the LEFM instrumentation during A2R15 (Spring 2011 - completed), installation of other MUR-related modifications during the A2R16 (Fall 2012), and revision of the affected station documents.

- Byron Unit 1: Within 180 days after NRC approval

  This implementation period will allow for installation of the LEFM instrumentation during B1R17 (Spring 2011 - completed); implementation of VIPRE and the associated DNB codes in advance of the B1R18 outage (Fall 2012) (if necessary to restore adequate DNB margin); installation of other MUR-related modifications during B1R18; and revision of the affected station documentation. Power increase to MUR power level is planned after B1R18.

- Byron Unit 2: Within 180 days after NRC approval

  This implementation period will allow for installation of the LEFM instrumentation and other MUR-related modifications during B2R16 (Fall 2011) and revision of the affected station documents.

## 2.0 DETAILED DESCRIPTION

The proposed changes to the Operating Licenses, TS, TS Bases and TRM are described below, with marked-up pages included in Attachments 2 and 3. Please note that the TS Bases and TRM changes are provided for information only.

Operating License Maximum Power Level

Item 2.C(1), “Maximum Power Level,” of the current operating licenses for Braidwood Station, Units 1 and 2 (Facility Operating License Numbers NPF-72 and NPF-77); and Byron Station, Units 1 and 2 (Facility Operating License Numbers NPF-37 and NPF-66, states: “The licensee is authorized to operate the facility at reactor core power levels not in excess of 3586.6 megawatts thermal….” This value is being increased to “3645 megawatts thermal.”

TS Section 1.1, Definition of “Rated Thermal Power (RTP)”

The definition of RTP in TS Section 1.1, "Definitions," is revised to increase the value of RTP from 3586.6 MWt to 3645 MWt.

TS Section 2.1.1, “Reactor Core SLs”

TS 2.1.1.1 currently states: “In MODE 1, the Departure from Nucleate Boiling Ratio (DNBR) shall be maintained ≥ 1.24 for the WRB-2 DNB correlation for a thimble cell and ≥ 1.25 for the WRB-2 DNB correlation for a typical cell.”

> TS 2.1.1.1 is being revised to state: “In MODE 1, the Departure from Nucleate Boiling Ratio (DNBR) shall be maintained ≥ 1.24 for the WRB-2 DNB correlation for a thimble cell, ≥ 1.25 for the WRB-2 DNB correlation for a typical cell and ≥ 1.19 for the ABB-NV DNB correlation for a thimble cell and a typical cell.”

TS 2.1.1.2 currently states: ”In MODE 2, the DNBR shall be maintained ≥ 1.17 for the WRB-2 DNB correlation, and ≥ 1.30 for the W-3 DNB correlation.”

> TS 2.1.1.2 is being revised to state: In MODE 2, the DNBR shall be maintained ≥ 1.17 for the WRB-2 DNB correlation, and ≥ 1.13 for the ABB-NV correlation and ≥ 1.18 for the WLOP DNB correlation.”

TS 3.4.1, “RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits”

LCO 3.4.1.c currently states: “RCS total flow rate ≥ 380,900 gpm and within the limit specified in the COLR.” The flow rate value is being revised to “386,000 gpm.”

SR 3.4.1.3 currently states: “Verify RCS total flow rate is ≥ 380,900 gpm and within the limit specified in the COLR.” The flow rate value is being revised to “386,000 gpm.”

SR 3.4.1.4 currently states: “Verify by precision heat balance that RCS total flow rate is ≥ 380,900 gpm and within the limit specified in the COLR.” The flow rate value is being revised to “386,000 gpm.”

TS 5.6.5, “Core Operating Limits Report (COLR)“

A new reference is being added to the list of analytical methods that are used to determine the core operating limits, specifically:

TS 5.6.5 currently states: "b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:"

TS 5.6.5 is being revised to add:

11. "WCAP-14565-P-A, "VIPRE-01 Modeling and Qualification for Pressurized Water Reactor Non-LOCA Thermal-Hydraulic Safety Analysis," October 1999.

<u>TRM Section 1.1, Definition of "Rated Thermal Power (RTP)"</u>

The definition of RTP in TS Section 1.1, "Definitions," is revised to increase the value of RTP from 3586.6 MWt to 3645 MWt.

<u>TRM Limiting Condition for Operation (TLCO) 3.3.k, "Feedwater Flow Instrumentation"</u>

A new TRM TLCO 3.3.k, "Feedwater Flow Instrumentation," is added. This TLCO allows operation at the uprated power level for up to 72 hours with an inoperable LEFM system; otherwise, power must be reduced to less than or equal to the current licensed power level (i.e., pre-uprate power level) of 3586.6 MWt which corresponds to 98.3% RTP as noted in the TLCO. A channel check of the LEFM is specified at a 24-hour frequency (note that the process computer actually performs a channel check once every 12 hours); and a channel calibration at an 18 month frequency.

## 3.0 TECHNICAL EVALUATION

### 3.1 Background and General Approach

10 CFR 50, Appendix K, paragraph I.A, "Sources of heat during the LOCA," requires the following: "…it must be assumed that the reactor is operating continuously at a power level at least 1.02 times the licensed power level (to allow for instrumentation error), with the maximum peaking factor allowed by the technical specifications." The NRC approved a change to the Appendix K requirements on June 1, 2000 (effective July 31, 2000), that allowed licensees the option that states: "An assumed power level lower than the level specified in this paragraph (but not less than the licensed power level) may be used provided the proposed alternative value has been demonstrated to account for uncertainties due to power level instrumentation errors."

Utilization of the Cameron CheckPlus$^{TM}$ LEFM system at Braidwood Station, Units 1 and 2, and Byron Station, Units 1 and 2 will result in reduced uncertainty in FW flow measurement, which reduces the total power level measurement uncertainty. As summarized below in Section 3.2, "LEFM Ultrasonic Flow Measurement and Core Thermal Power Uncertainty Summary," and detailed in Attachments 8a through 8d, with utilization of the LEFM instrumentation system, the core thermal power measurement uncertainty will be a maximum of 0.345%. This uncertainty supports a power increase of approximately 1.63%.

EGC has evaluated the effects of a bounding 1.7% increase in RTP using an analysis approach developed by Westinghouse Electric Company. These evaluations are described in detail in Attachment 5, "Braidwood and Byron Stations Measurement Uncertainty Recapture Technical Evaluation." The scope and content of the evaluations are consistent with the guidance contained in NRC Regulatory Issue Summary (RIS) 2002-03, "Guidance on the Content of Measurement Uncertainty Recapture Power Uprate Applications," (Reference 1).

## 3.2 Evaluation of Changes to License and Technical Specifications

The proposed changes to the TS described in Section 2.0, "Detailed Description," are evaluated below.

Operating License Maximum Power Level

Item 2.C(1), “Maximum Power Level,” of the current operating licenses for Braidwood Station, Units 1 and 2 (Facility Operating License Numbers NPF-72 and NPF-77); and Byron Station, Units 1 and 2 (Facility Operating License Numbers NPF-37 and NPF-66, states: “The licensee is authorized to operate the facility at reactor core power levels not in excess of 3586.6 megawatts thermal….” This value is being increased to “3645 megawatts thermal.”

Evaluation

The proposed increase in RTP from 3586.6 MWt to 3645 MWt in the operating license is acceptable based on the decreased uncertainty in the core thermal power calculation due to the use of the LEFM system and on the evaluations provided in this amendment request.

TS Section 1.1, Definition of “Rated Thermal Power (RTP)”

The definition of RTP in TS Section 1.1, "Definitions," is revised to increase the value of RTP from 3586.6 MWt to 3645 MWt.

Evaluation

The proposed increase in RTP from 3586.6 MWt to 3645 MWt in the TS definitions is acceptable based on the decreased uncertainty in the core thermal power calculation due to the use of the LEFM system and on the evaluations provided in this amendment request.

TS Section 2.1.1, “Reactor Core SLs”

TS 2.1.1.1 currently states: “In MODE 1, the Departure from Nucleate Boiling Ratio (DNBR) shall be maintained ≥ 1.24 for the WRB-2 DNB correlation for a thimble cell and ≥ 1.25 for the WRB-2 DNB correlation for a typical cell.”

TS 2.1.1.1 is being revised to state: “In MODE 1, the Departure from Nucleate Boiling Ratio (DNBR) shall be maintained ≥ 1.24 for the WRB-2 DNB correlation for a thimble cell, ≥ 1.25 for the WRB-2 DNB correlation for a typical cell and ≥ 1.19 for the ABB-NV DNB correlation for a thimble cell and a typical cell.”

TS 2.1.1.2 currently states: ”In MODE 2, the DNBR shall be maintained ≥ 1.17 for the WRB-2 DNB correlation, and ≥ 1.30 for the W-3 DNB correlation.”

TS 2.1.1.2 is being revised to state: In MODE 2, the DNBR shall be maintained ≥ 1.17 for the WRB-2 DNB correlation, ≥ 1.13 for the ABB-NV DNB correlation and ≥ 1.18 for the WLOP DNB correlation.”

Evaluation

Under MUR conditions, the projected core DNB margins were unacceptably small. To increase the DNB margins to an acceptable value, the NRC-approved W-3 alternative correlations (i.e., the ABB-NV and WLOP correlations) are used in place of the W-3 correlation as the secondary DNB correlation for conditions where the primary DNB correlation is not applicable. In order to utilize the ABB-NV and WLOP correlations the NRC-approved VIPRE-W (VIPRE) subchannel analysis code is used in place of the THINC-IV (THINC) subchannel analysis code and the FACTRAN code for DNBR calculations.

A detailed discussion regarding the use of VIPRE and the associated DNB correlations, ABB-NV and WLOP, is presented in Attachment 5, Section III.1.A, “Core Thermal Hydraulic Analysis.”

TS 3.4.1, “RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits”

LCO 3.4.1.c currently states: “RCS total flow rate ≥ 380,900 gpm and within the limit specified in the COLR.” The flow rate value is being revised to “≥ 386,000 gpm.”

SR 3.4.1.3 currently states: “Verify RCS total flow rate is ≥ 380,900 gpm and within the limit specified in the COLR.” The flow rate value is being revised to “≥ 386,000 gpm.”

SR 3.4.1.4 currently states: “Verify by precision heat balance that RCS total flow rate is ≥ 380,900 gpm and within the limit specified in the COLR.” The flow rate value is being revised to “≥ 386,000 gpm.”

Evaluation

The departure from nucleate boiling (DNB) calculations are based on the NSSS design parameters provided in Section 3.4, “Analysis Summary,” below. The MUR power uprate DNB analyses assume a nominal core power level of 3648 MWt, which bounds the requested increase to the current Byron and Braidwood rated thermal power. The MUR power uprate DNBR calculations are based on a minimum measured flow of 386,000 gpm compared to the value of 380,900 gpm used in the current DNB analyses of record. The higher core flow is consistent with the value in the Core Operating Limits Reports (COLRs) for the current operating cycles in the Byron and Braidwood units.

TS 5.6.5, “Core Operating Limits Report (COLR)“

A new reference is being added to the list of analytical methods that are used to determine the core operating limits, specifically:

TS 5.6.5 currently states: “b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:”

TS 5.6.5 is being revised to add:

11. "WCAP-14565-P-A, "VIPRE-01 Modeling and Qualification for Pressurized Water Reactor Non-LOCA Thermal-Hydraulic Safety Analysis," October 1999.

Evaluation

The thermal-hydraulic design methods for the MUR-PU remain the same as currently described in the Byron and Braidwood UFSAR except for the following two changes:

- the NRC-approved W-3 alternative correlations, ABB-NV and WLOP correlations (Attachment 5, Reference III.1-1), are used in place of the W-3 correlation (Attachment 5, Reference III.1-3) as the secondary DNB correlation for conditions where the primary DNB correlation is not applicable; and

- the NRC-approved VIPRE-W (VIPRE) subchannel analysis code (Attachment 5, Reference III.1-4) is used in place of the THINC-IV (THINC) subchannel analysis code (Attachment 5, References III.1-5 and III.1-6) and the FACTRAN code (Attachment 5, Reference III.1-7) for DNBR calculations.

The change to the VIPRE subchannel analysis code is necessary to implement the ABB-NV and WLOP DNB correlations for use in the MUR-PU analyses as secondary DNB correlations. The NRC Safety Evaluation addressing the use of these correlations, (Attachment 5, Reference III.1-2), requires that the ABB-NV correlation for Westinghouse PWR application and the WLOP correlation must be used in conjunction with the Westinghouse version of the VIPRE-01 code since the correlations were justified and developed based on VIPRE and the associated VIPRE modeling specifications. To support the use of the VIPRE code as the licensing basis subchannel analysis code for Byron and Braidwood Units 1 and 2, DNBR calculations have been performed with the VIPRE code for all of the DNB-limited UFSAR Chapter 15 events that are currently analyzed with the THINC subchannel analysis code. The DNBR calculations performed with the VIPRE code address the increased nominal heat flux and the change in power measurement uncertainty associated with the MUR-PU.

### 3.3 LEFM Ultrasonic Flow Measurement and Core Thermal Power Uncertainty Calculation Summary

The below discussion describes the method used to determine the appropriate increase in power level allowed by the LEFM reduced measurement uncertainty. Note that the detailed responses to the RIS-2002-03 specific guidance is provided in Attachment 5, Section 1.

The proposed MUR rated themal power (RTP) is determined by subtracting the measurement uncertainty (in terms of MWt) from the safety analysis limit. The uncertainty for each unit is provided in Cameron's "Bounding Uncertainty Analysis for Thermal Power Determination at (Byron Unit 1, Byron Unit 2, Braidwood Unit 1, and Braidwood Unit 2) Nuclear Generating Station Using the LEFM √+ System" (Attachments 8a through 8d). These reports (one per unit) determine the total RTP uncertainty based on use of LEFMs for feedwater flow measurement along with plant specific data. The methodology in determining the uncertainty is based on NRC approved Cameron Topical Report ER-157P-A Rev. 8 and Rev. 8 Errata (Reference 2).

Cameron's bounding uncertainty analysis uses the plant specific uncertainty data and combines these values with additional uncertainty terms related to the LEFM. Attachments 8a – 8d, Appendix B, page 1 of 1, provides a complete list of all the uncertainty terms that contribute to the total RTP uncertainty. The total RTP uncertainty for each unit is shown below and assumes the LEFM is operating in the "normal" mode. It is appropriate to apply these uncertainties at the requested MUR power level.

| | |
|---|---|
| Byron Unit 1: | ± 0.337% |
| Byron Unit 2: | ± 0.334% |
| Braidwood Unit 1: | ± 0.345% |
| Braidwood Unit 2: | ± 0.337% |

In order to maintain consistency in the licensed thermal power for all four units, the bounding uncertainty value is selected and used in the computations below. Based on the above results, a bounding uncertainty value of ± 0.345% (from Braidwood Unit 1) will be used for all units.

The current licensed thermal power (CLTP) for each of the units at Byron Station and Braidwood Station is 3586.6 MWt. When adding the 2.0% safety analysis margin to this value, the safety analysis limit is 3658.3 MWt. This value is conservatively rounded down to the nearest whole number, i.e., 3658 MWt for determining the requested increase in power level.

Applying the measurement uncertainty to the safety analysis limit results in a power measurement of 12.62 MWt (i.e., 3658 MWt * 0.00345). Subtracting this value from the safety analysis limit results in a proposed MUR RTP of 3645.38 MWt (i.e., 3658 MWt – 12.62 MWt), which is conservatively rounded down to 3645 MWt.

The revised value of rated thermal power is therefore requested to be 3645 MWt for all units at Byron and Braidwood Stations.

## 3.4 Analyses Summary

### 3.4.1 MUR Uprate Evaluation Approach

A comprehensive engineering evaluation has been completed for Byron Station, Units 1 and 2 and Braidwood Station, Units 1 and 2, to confirm that the requested increase in rated thermal power (RTP) from 3586.6 MWt to 3645 MWt is acceptable. The evaluations/analyses were performed to bound the requested increase in RTP to 3645 MWt (i.e., an increase of 1.63%). These evaluations addressed design transients, accidents, nuclear fuel, NSSS systems and Balance of Plant (BOP) systems and are summarized in Attachment 5, "Braidwood and Byron Stations Measurement Uncertainty Recapture Technical Evaluation." This attachment provides the information requested in NRC Regulatory Information Summary (RIS) 2002-03. The results of all analyses and evaluations were found to be acceptable and will adequately support MUR uprated power conditions.

Section 3.4.2, below, presents the NSSS design thermal and hydraulic parameters used in support of the requested increase in power level. These parameters serve as the basis for the NSSS analyses and evaluations. The reactor core thermal power and/or NSSS thermal power are used as inputs to most plant safety, component, and system analyses. These NSSS analyses typically address the core and/or NSSS thermal power in one of three ways.

1. Analyses that apply a 2.0% increase to the initial power level to account for the power measurement uncertainty. These analyses have not been re-performed for the MUR uprate conditions because the sum of the proposed core power level and the decreased power measurement uncertainty falls within the previously analyzed conditions. The existing 2.0% uncertainty is reallocated so a portion is applied to uprate power and the remainder is retained to accommodate the power measurement uncertainty.
2. Analyses that are performed at 0% power conditions. These analyses would normally not have to be re-performed to address MUR uprate conditions because they are not dependent on power; however, as discussed in Attachment 5, Sections III.1.A.5.5 and III.1.A.5.9, the hot zero power steam line break and the uncontrolled rod withdrawal from subcritical respectively, were reanalyzed using the VIPRE subchannel analysis code. The rod ejection at hot zero power was not reanalyzed. The results of these analyses were shown to be acceptable.
3. Analyses that employ a nominal power level. These analyses have either been evaluated or re-performed for the proposed MUR power level and were shown to be acceptable.

A revised Steam Generator Tube Rupture (SGTR) and Margin to Overfill (MTO) Analysis is presented in Attachment 5a, "Byron and Braidwood Stations, Steam Generator Tube Rupture Analysis Report." A reanalysis of this event was required as the MTO results in the current analysis of record were unacceptably small prompting revisions to the analysis assumptions. A summary of the results of the revised analysis is presented in Section 3.4.4 below while the detailed description of the revised analysis is presented in Attachment 5a as noted above.

As previously noted, the MUR analyses assume a maximum 1.7% increase in core thermal power (i.e., 1.017 * 3586.6 MWt = 3647.5 MWt). Each section in Attachment 5, where appropriate, lists the power level assumed in the associated analysis. The power level may include an additional allowance of 14 MWt for reactor coolant pump (RCP) heat addition (if applicable).

The power level assumed for each respective transient or accident analysis is summarized in Attachment 5, Table II-2.

Note that some analyses are evaluated as part of every core reload. These analyses will continue to be evaluated in a manner consistent with the core reload methodology.

### 3.4.2 NSSS Design Parameters

#### Introduction

The NSSS design parameters are the fundamental parameters used as input in the NSSS analyses. They provide the primary and secondary side system conditions (thermal power, temperatures, pressures, and flows) that are used as the basis for all the NSSS analyses and evaluations. As a result of the MUR power uprate, the Byron Station, Units 1 and 2 and Braidwood Station Units 1 and 2 NSSS design parameters have been revised as shown in Tables 3-1 and 3-2. Tables 3-1 and 3-2 provide information for the eight cases (i.e., four cases for Unit 1; four cases for Unit 2) associated with the Byron Station and Braidwood Station MUR power uprate program. These parameters have been incorporated, as required, into the applicable NSSS systems and components evaluations, as well as safety analyses, performed in support of the MUR power uprate.

**Input Parameters and Assumptions**

The major input parameters and assumptions used in the calculation of the eight cases provided in Tables 3-1 and 3-2 are summarized by the following:

- A bounding reactor core power level of 3658 MWt (NSSS power of 3672 MWt) was used for analyses. (Note that 3658 MWt is 102% of the current licensed power level of 3586.6 MWt; and the NSSS power of 3672 MWt includes 14 MWt for RCP heat).
- The thermal design flow (TDF) of 92,000 gpm/loop was used for the DNB non-limiting event analyses utilizing the Standard Thermal Design Procedure.
- A total reactor coolant system (RCS) minimum measured flow of 386,000 gpm was used for the DNB limiting event analyses utilizing the Revised Thermal Design Procedure.
- The parameters that are applicable to Babcock & Wilcox (B&W) replacement steam generators (RSGs) for Units 1 are given in Table 3-1.
- The parameters that are applicable to Westinghouse Model D5 steam generators (SGs) for Units 2 are given in Table 3-2.
- A steam generator tube plugging (SGTP) range between 0 and 5 percent for Units 1, and between 0 and 10 percent for Units 2, was evaluated.
- A feedwater temperature range of 433.0°F to 449.2°F for Units 1, and 435.0°F to 449.2°F for Units 2 was evaluated.
- An SG moisture carryover value of 0.10 percent was utilized for Units 1 and a value of 0.25 percent was utilized for Units 2.
- The parameters considered 17x17 Vantage+ fuel with thimble plugs removed (TPR) or thimble plugs installed (TPI) and intermediate flow mixing vanes (IFMs).
- Two design core bypass flows were used: 8.3 percent, which accounts for TPR and IFMs; and 6.3 percent, which accounts for TPI and IFMs.

**Parameter Cases**

The eight cases evaluated are shown in Tables 3-1 and 3-2 and are summarized as follows:

- Cases 1 and 2 in Tables 3-1 and 3-2 represent parameters based on minimum reactor vessel average temperature ($T_{avg}$) of 575.0°F. Case 2, which is based on an average 5% SGTP level for Units 1 and an average 10% SGTP level for Units 2, yields the minimum secondary side steam generator pressure and temperature. Note that all primary side temperatures are identical for Cases 1 and 2 in Table 3-1 and all primary side temperatures are identical for Cases 1 and 2 in Table 3-2.

- Cases 3 and 4 in Tables 3-1 and 3-2 represent parameters based on the maximum $T_{avg}$ of 588.0°F. Case 3, which is based on an average 0% SGTP for Units 1 and Units 2, yields the maximum secondary side steam pressure and temperature. Note that all primary side temperatures are identical for Cases 3 and 4 in Table 3-1 and all primary side temperatures are identical for Cases 3 and 4 in Table 3-2. The data provided in Note 4 of both Tables 3-1 and 3-2 were used in those NSSS analyses and evaluations that require an absolute upper-limit steam pressure. These more limiting secondary side data are based on the Case 3 parameters with an assumed steam generator fouling factor of zero.

**Table 3-1**

**NSSS Design Parameters for Byron Unit 1 and Braidwood Unit 1 MUR Power Uprate Program**

| **Thermal Design Parameters** | **Case 1** | **Case 2** | **Case 3** | **Case 4** |
|---|---|---|---|---|
| NSSS Power, MWt | 3672 | 3672 | 3672 | 3672 |
| $10^6$ Btu/hr | 12,529 | 12,529 | 12,529 | 12,529 |
| Reactor Power, MWt | 3658 | 3658 | 3658 | 3658 |
| $10^6$ Btu/hr | 12,482 | 12,482 | 12,482 | 12,482 |
| Thermal Design Flow, gpm/loop | 92,000 | 92,000 | 92,000 | 92,000 |
| Reactor $10^6$ lb/hr | 140.0 | 140.0 | 137.4 | 137.4 |
| Reactor Coolant Pressure, psia | 2250 | 2250 | 2250 | 2250 |
| Core Bypass, % | 8.3(1,2) | 8.3(1,2) | 8.3(1,3) | 8.3(1,3) |
| Reactor Coolant Temperature, °F | | | | |
| Core Outlet | 614.0(2) | 614.0(2) | 626.1(3) | 626.1(3) |
| Vessel Outlet | 608.6 | 608.6 | 620.9 | 620.9 |
| Core Average | 579.6(2) | 579.6(2) | 592.8(3) | 592.8(3) |
| Vessel Average | 575.0 | 575.0 | 588.0 | 588.0 |
| Vessel/Core Inlet | 541.4 | 541.4 | 555.1 | 555.1 |
| Steam Generator Outlet | 541.1 | 541.1 | 554.8 | 554.8 |
| Steam Generator | | | | |
| Steam Outlet Temperature, °F | 531.5 | 530.8 | 545.6(4) | 544.9 |
| Steam Outlet Pressure, psia | 897 | 891 | 1008(4) | 1002 |
| Steam Outlet Flow, $10^6$ lb/hr total | 15.98/16.36 | 15.98/16.35 | 16.06/16.43(4) | 16.05/16.43 |
| Feed Temperature, °F | 433.0/449.2 | 433.0/449.2 | 433.0/449.2 | 433.0/449.2 |
| Steam Outlet Moisture, % max. | 0.10 | 0.10 | 0.10 | 0.10 |
| Tube Plugging Level, % | 0 | 5 | 0 | 5 |
| Zero Load Temperature, °F | 557 | 557 | 557 | 557 |
| **Hydraulic Design Parameters** | | | | |
| Mechanical Design Flow, gpm/loop | 107,000 | | | |
| Minimum Measured Flow, gpm total | 386,000 | | | |

**Notes:**
1. Core bypass flow accounts for thimble plugs removed and IFMs.
2. If thimble plugs are installed, the core bypass flow is 6.3%, core outlet temperature is 612.7°F, and core average temperature is 578.8°F.
3. If thimble plugs are installed, the core bypass flow is 6.3%, core outlet temperature is 624.8°F, and core average temperature is 592.1°F.
4. If a high steam pressure is more limiting for analysis purposes, a greater steam pressure of 1019 psia, steam temperature of 546.9ºF, and total steam flow of 16.44 x $10^6$ lb/hr should be assumed. This is to envelop the possibility that the plant could operate with better than expected steam generator performance.

**Table 3-2**

**NSSS Design Parameters for Byron Unit 2 and Braidwood Unit 2 MUR Uprate Program**

| **Thermal Design Parameters** | **Case 1** | **Case 2** | **Case 3** | **Case 4** |
|---|---|---|---|---|
| NSSS Power, MWt | 3672 | 3672 | 3672 | 3672 |
| $10^6$ Btu/hr | 12,529 | 12,529 | 12,529 | 12,529 |
| Reactor Power, MWt | 3658 | 3658 | 3658 | 3658 |
| $10^6$ Btu/hr | 12,482 | 12,482 | 12,482 | 12,482 |
| Thermal Design Flow, gpm/loop | 92,000 | 92,000 | 92,000 | 92,000 |
| Reactor $10^6$ lb/hr | 140.0 | 140.0 | 137.4 | 137.4 |
| Reactor Coolant Pressure, psia | 2250 | 2250 | 2250 | 2250 |
| Core Bypass, % | 8.3(1,2) | 8.3(1,2) | 8.3(1,3) | 8.3(1,3) |
| Reactor Coolant Temperature, °F | | | | |
| Core Outlet | 614.0(2) | 614.0(2) | 626.1(3) | 626.1(3) |
| Vessel Outlet | 608.6 | 608.6 | 620.9 | 620.9 |
| Core Average | 579.6(2) | 579.6(2) | 592.8(3) | 592.8(3) |
| Vessel Average | 575.0 | 575.0 | 588.0 | 588.0 |
| Vessel/Core Inlet | 541.4 | 541.4 | 555.1 | 555.1 |
| Steam Generator Outlet | 541.1 | 541.1 | 554.8 | 554.8 |
| Steam Generator | | | | |
| Steam Outlet Temperature, °F | 523.4/522.4 | 519.4/518.4 | 538.8/537.8(4) | 534.9/533.8 |
| Steam Outlet Pressure, psia | 837/829 | 809/802 | 953/945(4) | 923/914 |
| Steam Outlet Flow, $10^6$ lb/hr total | 16.01/16.34 | 16.00/16.32 | 16.09/16.42(4) | 16.07/16.39 |
| Feed Temperature, °F | 435.0/449.2 | 435.0/449.2 | 435.0/449.2 | 435.0/449.2 |
| Steam Outlet Moisture, % max. | 0.25 | 0.25 | 0.25 | 0.25 |
| Tube Plugging Level, % | 0 | 10 | 0 | 10 |
| Zero Load Temperature, °F | 557 | 557 | 557 | 557 |
| **Hydraulic Design Parameters** | | | | |
| Mechanical Design Flow, gpm/loop | 107,000 | | | |
| Minimum Measured Flow, gpm total | 386,000 | | | |

**Notes:**
1. Core bypass flow accounts for thimble plugs removed and IFMs.
2. If thimble plugs are installed, the core bypass flow is 6.3%, core outlet temperature is 612.7°F, and core average temperature is 578.8°F.
3. If thimble plugs are installed, the core bypass flow is 6.3%, core outlet temperature is 624.8°F, and core average temperature is 592.1°F.
4. If a high steam pressure is more limiting for analysis purposes, a greater steam pressure of 982 psia, steam temperature of 542.3ºF, and total steam flow of 16.11 x $10^6$ lb/hr should be assumed. This is to envelop the possibility that the plant could operate with better than expected steam generator performance.

### 3.4.3 Subchannel Analysis Code (VIPRE) and DNB Correlations (ABB-NV and WLOP)

The thermal-hydraulic design methods for the MUR-PU remain the same as currently described in the Byron and Braidwood UFSAR except for two changes:

- the NRC-approved W-3 alternative correlations in Attachment 5, Reference III.1-1 (the ABB-NV and WLOP correlations) are used in place of the W-3 correlation (Attachment 5, Reference III.1-3) as the secondary DNB correlation for conditions where the primary DNB correlation is not applicable;

- the NRC-approved VIPRE-W (VIPRE) subchannel analysis code (Attachment 5, Reference III.1-4) is used in place of the THINC-IV (THINC) subchannel analysis code (Attachment 5, References III.1-5 and III.1-6) and the FACTRAN code (Attachment 5, Reference III.1-7) for DNBR calculations.

These changes are needed to restore adequate DNB margin under MUR operating conditions. The primary DNB correlation used in the analysis of the VANTAGE+ fuel at MUR-PU conditions remains the WRB-2 DNB correlation (Attachment 5, Reference III.1-8). The secondary DNB correlation, which supplements the primary DNB correlation for conditions where the primary DNB correlation is not applicable, is changed for the MUR-PU. The W-3 correlation, which is the current secondary DNB correlation for the Byron and Braidwood Units, is inadequate to provide the DNBR margin necessary to support the MUR-PU conditions. For the MUR-PU DNB analyses, the NRC-approved W-3 alternative DNB correlations noted in Attachment 5, Reference III.1-1 (the ABB-NV and WLOP correlations) are used as secondary DNB correlations.

The change to the VIPRE subchannel analysis code is necessary to implement the ABB-NV and WLOP DNB correlations for use in the MUR-PU analyses as secondary DNB correlations. The NRC Safety Evaluation addressing the use of these correlations, (Attachment 5, Reference III.1-2), requires that the ABB-NV correlation for Westinghouse PWR application and the WLOP correlation must be used in conjunction with the Westinghouse version of the VIPRE-01 code since the correlations were justified and developed based on VIPRE and the associated VIPRE modeling specifications. To support the use of the VIPRE code as the licensing basis subchannel analysis code for Byron and Braidwood Units 1 and 2, DNBR calculations have been performed with the VIPRE code for all of the DNB-limited UFSAR Chapter 15 events that are currently analyzed with the THINC subchannel analysis code. The DNBR calculations performed with the VIPRE code address the increased nominal heat flux and the change in power measurement uncertainty associated with the MUR-PU.

The DNB analyses of the VANTAGE+ fuel in Byron and Braidwood Units 1 and 2 at MUR-PU conditions continue to be based on the Revised Thermal Design Procedure (RTDP) (Attachment 5, Reference III.1-9). With the RTDP methodology, uncertainties in plant operating parameters, nuclear and thermal parameters, fuel fabrication parameters, computer codes, and DNB correlation predictions are considered statistically to obtain the overall DNB uncertainty factors. For the MUR-PU, the current plant operating parameter uncertainties remain applicable with the exception of the power measurement uncertainty. The Byron and Braidwood MUR-PU is based on a reduced power measurement uncertainty associated with the use of the LEFM CheckPlus system to measure feedwater flow. Proprietary DNBR sensitivity factors, which are used to develop the DNB uncertainty factors, are calculated using the VIPRE code for ranges of conditions which bound the events for which RTDP methodology is applied. Based on the DNB uncertainty factors, RTDP design limit DNBR values are determined which meet the DNB acceptance criterion. In addition to the above considerations for uncertainties, DNBR margin is retained by performing the safety analyses to DNBR limits higher than the RTDP design limit DNBR values. Sufficient DNBR margin is conservatively maintained in the safety analysis DNBR limits to offset the rod bow DNBR penalty and to provide flexibility in design and operation of the plant.

The Standard Thermal Design Procedure (STDP) methodology continues to be used for those DNB analyses where RTDP is not applicable. For the STDP, the initial condition uncertainties are accounted for deterministically by applying the uncertainties to the nominal conditions. The DNBR limit for STDP is the appropriate DNB correlation limit with consideration for applicable DNBR penalties.

Additional information regarding the implementation and results of using the VIPRE subchannel analysis code, and the ABB-NV and WLOP DNB correlations is presented in Attachment 5, Section III.1.A, "Core Thermal and Hydraulic Analysis." Section III.1.A also addresses the NRC Safety Evaluation conditions for implementation of the VIPRE code and ABB-NV and WLOP correlations.

### 3.4.4 Steam Generator Tube Rupture Analysis and Margin to Overfill Analysis Summary

The results of a revised Steam Generator Tube Rupture (SGTR) and Margin to Overfill (MTO) Analysis is being submitted for NRC approval. This revised analysis was performed as the MTO values in the current analysis of record (AOR) are unacceptably small and revisions to the analysis assumptions were necessary. NRC approval of this reanalysis is required as the proposed changes result in more than a minimal increase in the accident dose as defined in NEI 96-01, "Guidelines for 10 CFR 50.59 Implementation," Revision 1, dated November 2000. The detailed results of the SGTR and MTO Analysis are provided in Attachment 5a, "Steam Generator Tube Rupture and Margin to Overfill Analysis Report." Note that the revised analysis did not prompt any TS changes. This analysis addressed three major areas.

- SGTR Margin to Steam Generator Overfill
- SGTR Thermal and Hydraulic Analysis for Radiological Consequences
- SGTR Radiological Consequences

A summary of the results for these three analyses, as shown in Attachment 5a, Section V, "Overall Conclusions," is given below.

**SGTR Margin to Steam Generator Overfill (SGTR MTO) Analysis**

The SGTR MTO analysis was performed to determine the margin to SG overfill for a design basis SGTR event for the Byron and Braidwood units. The SGTR MTO accident analysis demonstrated that SG overfill does not occur.

The analysis was performed using the LOFTTR2 program and the methodology developed in Attachment 5a, Reference 1, with modifications to address NSAL-07-11 (Attachment 5a, Reference 3) consistent with WCAP-16948-P (Attachment 5a, Reference 4), and using plant-specific parameters. The MTO analysis assumed a core power of 3658.3 MWt, or 102% of 3586.6 MWt. Therefore, the analyzed RTP power bounds the MUR power uprate conditions.

<u>SGTR MTO Analysis Single Failure Assumptions</u>

A single failure analysis was conducted for the SGTR MTO event to determine the most limiting single failure. This analysis is summarized in Attachment 5a, Sections I.1.E and II.2.E. It was determined that the most limiting failure regarding SG MTO was the failure of an intact SG PORV. It should be noted that the assumptions in this scenario necessitate installation of plant modifications. These modifications are summarized below and will be installed and made operational prior to increasing power above the current licensed power level.

<u>SGTR MTO Modifications</u>

Byron and Braidwood Stations will be implementing plant modifications to support the SGTR MTO analysis single failure assumptions. These modifications are described below.

- Install safety-related air accumulator tanks to support AFW valve flow control
- Increase the capacity of the SG Power Operated Relief Valves (PORV's) (on Unit 1 only)
- Install Uninterruptible Power Supplies (UPS) on two of the four SG PORVs
- Install a manual isolation valve upstream of each High Head Safety Injection valve (1/2SI8801A/B)

A description of these modifications is provided in Attachment 5a, Section II.2.F, "Modifications to Support MTO Single Failure Considerations." As noted above, these modifications will be installed and made operational prior to increasing power above the current licensed power level. The safety-related air accumulator tanks for AFW valve flow control, the UPS to the PORVs, and the manual SI isolation valve are planned to be installed in accordance with 10 CFR 50.59; however, installation of the modification to increase the Unit 1 SG PORVs flow capacity requires NRC approval prior to installation as this modification in conjunction with the SGTR MTO methodology change results in more than a minimal increase in the accident dose.

Note that the modification to install uninterruptible power supplies to the SG PORVs is prompted by the resolution of Unresolved Items (URIs) from the 2009 Component Design Bases Inspection (CDBI) at Byron Station (URI 05000454/2009007-03; URI 05000455/2009007-03). The URIs involved a concern with respect to the single failure assumptions used in Byron Station's analysis for a Steam Generator Tube Rupture (SGTR) event. The NRC documented its position regarding these URIs in Reference 3. The NRC verified that this same SGTR-related concern was also applicable to Braidwood Station as documented in Reference 5. Byron Station responded to the NRC in Reference 4; and Braidwood Station responded to the NRC in Reference 6. In these letters, both Byron Station and Braidwood Station committed to installing the UPS modification to resolve the single failure concern. This modification places the SGTR analysis in compliance with NRC regulations and preserves the assumptions in the SGTR analysis.

**SGTR Thermal and Hydraulic Analysis for Radiological Consequences**

The thermal and hydraulic analyses were performed using the LOFTTR2 program and the methodology developed in Attachment 5a, References 1 and 2, and using the plant-specific parameters. From these predictions, the RCS and SG water masses, the ruptured SG break flow, the fraction of this break flow that flashes directly to steam, and the steam releases from the ruptured and intact SGs through the MSSVs and PORVs are calculated for input to the dose analyses. The thermal-hydraulic analyses assumed a core power of 3658.3 MWt, or 102% of

3586.6 MWt to generate this data. Therefore, the analyzed RTP power bounds the MUR power uprate conditions.

**SGTR Radiological Consequences Analysis**

The SGTR radiological analyses are based upon the alternative source term (AST) as defined in Regulatory Guide (RG) 1.183, with acceptance criteria as specified in RG 1.183 for offsite doses and in 10 CFR 50.67 for the control room. The analyses involve the transfer of activity from the primary to the secondary side of the SGs and then to the environment. The RCS iodine and noble gas source terms are scaled to the Technical Specification Dose Equivalent (DE) Iodine-131 and Xenon-133 limits in the primary coolant, which removes the power dependence from the analysis. The various parameters from the thermal-hydraulic analyses are consistent with a core power of 3658.3 MWt, or 102% of 3586.6 MWt. The resulting doses at the exclusion area boundary (EAB), low population zone (LPZ), and in the control room (CR) remain within the applicable limits as shown in Attachment 5a, Table IV-6; therefore, the results of the SGTR radiological analyses are acceptable under MUR power uprate conditions.

### 3.4.5 Plant Modifications

The evaluations performed to support the proposed changes identified that changes are required to certain non-safety related systems, including minor equipment changes, replacements, and setpoint or alarm point changes. These changes are planned to be made in accordance with the requirements of 10 CFR 50.59, "Changes, tests, and experiments," and will be completed as necessary prior to implementation of the proposed power uprate.

Modifications of interest include:

- Potential HP turbine nozzle block replacement
- Feedwater heater drain valve modifications
- Switchyard modifications
- Modifications supporting the revised SGTR and Margin to Overfill analysis single failure assumptions as noted above in Section 3.4.4
- Various Balance of Plant (BOP) instrument rescaling, setpoint and alarm point changes
- ATWS Mitigation System time delay change

### 3.4.6 Technical Specification Instrument Setpoint Changes

The are no Technical Specification related instrument setpoint changes being proposed as part of this license amendment request.

It should be noted that the pressure coefficient constant, $K_3$, in the overtemperature delta-T (OT$\Delta$T) setpoint equation is being revised from 0.00181 to 0.00135. To support operation at MUR power uprate conditions, new core thermal limits were generated as discussed in Attachment 5, Section III.I.A.5.1, “Core Thermal Limits.” The revision to $K_3$ was required to ensure that the revised core thermal limits were fully protected and to ensure that necessary DNB margin was maintained. The $K_3$ constant is maintained in the Byron Station and Braidwood Station Core Operating Limits Report (COLR), and does not require a change to Technical Specifications.

### 3.4.7 Grid Stability

Byron Station

Two grid studies have been completed to support the proposed uprate. The studies were performed using a 1295 (1265) MWe output for Byron Unit 1(2) main generator. These values were chosen for the studies to bound the highest expected electrical output of the main generator under uprated conditions. Using these bounding values provides conservative results for the two studies performed. The scenarios studied in these grid assessments are consistent with the transmission service provider requirements and include a single unit trip at the station under study, loss of the largest unit on the grid, loss of the most critical transmission circuit, and loss of load.

PJM Interconnection (PJM), the grid operator, completed a system stability analysis to assess the impact of the uprate on the rotor angle stability of generating plants in the Commonwealth Edison (ComEd) and neighboring control areas. The analysis assumed a 1295 (1265) MWe for Byron Unit 1(2) main generator and a light load flow base case based on 2013 projections. The results of the analysis are as follows:

1. All of the primary-clearing scenarios were found to be stable.

2. All of the maintenance outage (prior outage) scenarios considered in this study were found to be stable.

3. All of the breaker failure scenarios considered in this study were found to be stable.

ComEd Transmission Planning completed an assessment of the capability of the grid to ensure adequate post-trip and LOCA voltage levels. The analysis assumed a 1295 (1265) MWe output for Byron Unit 1(2) main generator. Power flow simulations were performed using 2012 transmission grid models for four system load conditions. The assessment concluded that with one exception, the lowest post-contingency voltage is 349.1 kV, which remains above the minimum required switchyard voltage of 339.8 kV.

The scenario that analyzes a unit trip, with the other unit in shutdown, and with a system load level equal to 75% of the 50/50 load forecast resulted in a post contingency voltage of 331.9 kV, which is lower than the minimum required voltage of 339.8 kV. This low post contingency voltage for this scenario is an existing (pre MUR) condition and is not related to the MUR uprate. PJM real-time state estimator continuously monitors and predicts grid voltages under various contingencies (e.g., unit trips). If the state estimator predicts an inadequate voltage at Byron's switchyard, the Station is notified and the appropriate Station abnormal operating procedure is entered. Further details regarding this study are provided in Attachment 10b.

Braidwood Station

Two grid studies have been completed to support the proposed uprate. The studies were performed using a 1295 (1265) MWe output for Braidwood Unit 1(2) main generator. These values were chosen for the studies to bound the highest expected electrical output of the main generator under uprated conditions. Using these bounding values provides conservative results for the two studies performed. The scenarios studied in these grid assessments are consistent with the transmission service provider requirements and include a single unit trip at the station under study, loss of the largest unit on the grid, loss of the most critical transmission circuit, and loss of load.

PJM Interconnection (PJM), the grid operator, completed a system stability analysis to assess the impact of the uprate on the rotor angle stability of generating plants in the Commonwealth Edison (ComEd) and neighboring control areas. The analysis assumed a 1295 (1265) MWe for Braidwood Unit 1(2) main generator and a light load flow base case based on 2013 projections. The results of the analysis are as follows:

1. All of the scenarios considered for baseline instability were found to be stable.

2. All of the primary-clearing scenarios were found to be stable.

3. All of the prior outage scenarios considered in this study were found to be stable.

4. Of all breaker failure scenarios studied, three are unstable. The study provided remediation measures for these three scenarios involving adjustment of the critical clearing time. EGC will ensure that any modifications required by PJM are completed prior to uprate implementation. Further details regarding this study are provided in Attachment 10a

ComEd Transmission Planning completed an assessment of the capability of the grid to ensure adequate post-trip and LOCA voltage levels. The analysis assumed a 1295 (1265) MWe output for Braidwood Unit 1(2) main generator. Power flow simulations were performed using 2012 transmission grid models for four system load conditions. The assessment concluded that the lowest post-contingency voltage is 349.5 kV, which remains above the minimum required switchyard voltage of 349.2 kV. Further details regarding this study are provided in Attachment 10a.

### 3.4.8 Operator Training, Human Factors, and Procedures

Operator response to transients, accidents, and special events is unaffected by the proposed changes with the exception of the operator response times associated with the Steam Generator Tube Rupture as discussed in Attachment 5a, “Steam Generator Tube Rupture and Margin to Overfill Analysis Report.” Necessary procedure revisions will be completed prior to implementation of the proposed changes. The plant simulator will be modified for the uprated conditions and the changes will be validated in accordance with plant configuration control processes. Operator training will be completed prior to implementation of the proposed changes.

### 3.4.9 NRC Requested Information During the May 18, 2011 Pre-Application Teleconference

EGC and the NRC conducted a pre-application meeting regarding the proposed Byron Station and Braidwood Station MUR LAR on November 4, 2010. In this meeting EGC presented the proposed content of the Byron and Braidwood Station MUR LAR. The NRC documented a summary of this meeting in a letter from Marshall J. David (NRC) to Exelon Generation Company, LLC, "Summary of November 4, 2010, Meeting with Exelon Generation Company, LLC, Pre-Application Discussion on Forthcoming Byron and Braidwood Measurement Uncertainty Recapture (MUR) License Amendment Request," dated November 29, 2010.

In followup to this meeting, on May 18, 2011, EGC and NRC representatives held a conference call to further discuss and clarify the content of the proposed LAR. The NRC documented a summary of this call in a letter from Nicholas J. DiFrancesco (NRC) to Exelon Generation Company, LLC, "Summary of May 18, 2011, Meeting with Exelon Generation Company, LLC, Pre-Application Discussion on Forthcoming Byron and Braidwood Measurement Uncertainty Recapture (MUR) License Amendment Request," dated June 9, 2011.

In this letter, the NRC provided feedback in several technical areas based on the proposed request and recent NRC experience with similar reviews. The following specific information was requested in NRC letter items 2.a through 2.d:

*2.a NRC staff reiterated the need for EGC to address the specific limitations and conditions associated with NRC approval of any Topical Reports utilized in any new or updated analysis methodologies (e.g., VIPRE).*

Response:

The limitations and conditions associated with NRC-approved topical reports are discussed for the following documents:

1. Cameron LEFM CheckPlus System

   As discussed in Attachment 5, Section I.1.A, the applicable Topical Reports are:

   - Cameron Engineering Report ER-80P, Revision 0, *Improving Thermal Power Accuracy and Plant Safety While Increasing Operating Power Level using the LEFM Check System,* Caldon Inc., March 1997 (Attachment 5, Reference I-1)
   - Topical Report (TR) Engineering Report ER-157P, Revision 8, *Caldon Ultrasonics Engineering Report ER-157P, "Supplement to Topical Report ER-80P: Basis for a Power Uprate with the LEFM Check or CheckPlus System,"* dated May 11, 2009 (Attachment 5, Reference I-2)

   In approving Cameron Engineering Report 157-P the NRC stated that licensees can reference TR ER-80P and follow the example of ER-157P, Revision 8, for their plant-specific analyses subject to meeting five qualifications (Attachment 5, Reference I-4). The five qualifications are discussed in Attachment 5, Section I.1.C.

   In approving Cameron Topical Reports ER-80P (Reference I-3) and ER-157P (Reference I-4), and also in Reference I-6 the NRC established four criteria each licensee must address. EGC's response to those criteria is provided in Attachment 5, Section I.1.D.

2. VIPRE, W-3 Alternative DNB Correlation, and RTDP

   The NRC Staff reviewed Westinghouse WCAP-14565, “VIPRE-01 Modeling and Qualification for Pressurized Water Reactor Non-LOCA Thermal/Hydraulic Safety Analysis,” and concluded in a Staff SER (Attachment 5, Reference III.1-19) that the generic topical report was an acceptable reference to support plant-specific applications for use of VIPRE-01, provided four Conditions identified in the SER were addressed by the licensees. These four conditions were considered for Byron and Braidwood Stations at MUR power uprate conditions. Compliance with these four SER Conditions is addressed in Attachment 5, Section III.1.A.4.1.

   The NRC Staff reviewed Westinghouse WCAP-14565-P-A, Addendum 2, “Addendum 2 to WCAP-14565-P-A, Extended Application of ABB-NV Correlation and Modified ABB-NV Correlation WLOP for PWR Low Pressure Applications,” and concluded in a Staff SER (Attachment 5, Reference III.1-2) that the generic topical report was acceptable for licensing applications, subject to the four limitations and conditions identified in the SER being addressed by the licensees. These four conditions were considered for Byron and Braidwood Stations at MUR power uprate conditions. Compliance with these four Limitations and Conditions is addressed in Attachment 5, Section III.1.A.4.2.

   The NRC Staff reviewed Westinghouse WCAP-11397, “Revised Thermal Design Procedure,” and concluded in a Staff SER (Attachment 5, Reference III.1-20) that the generic topical report was an acceptable reference to support plant-specific applications for use of RTDP, provided seven Conditions identified in the SER were addressed by the licensees. These seven conditions were considered for Byron and Braidwood Stations at MUR power uprate conditions. Compliance with these seven SER Conditions is addressed in Attachment 5, Section III.1.A.4.3.

3. Steam Generator Tube Rupture: WCAP-10698-P-A and its Supplement 1

   Section D of Enclosure 1 of the Staff’s SER approving WCAP-10698 required additional plant specific input (for five items) for each licensee referencing this topical report. These five items are addressed in Attachment 5a, Sections I.1.A through E.

*2.b* *Ensure justification for duration of revised and new operator manual actions is provided (e.g., SGTR MTO analysis). The justification should include demonstration of timed operator actions and key tasks. The staff noted similar reviews where NRC has conducted site audits to verify analysis assumptions.*

<u>Response:</u>

As noted in Section 3.4.8, “Operator Training, Human Factors and Procedures,” and in Attachment 5, Section VII.1, “Operator Actions,” and VII.2.A, “Emergency and Abnormal Operating Procedures,” required operator actions and response times to transients, accidents, and special events are unaffected by the proposed changes associated with the increase to MUR power level; however, some of the operator response times associated with the Steam Generator Tube Rupture discussed in Attachment 5a, Steam Generator Tube Rupture and Margin to Overfill Analysis Report,” have been revised.

Operator actions and associated response times specific to the SGTR and MTO analysis are addressed in Attachment 5a, Section I.1.A, "Operator Response Time," Table 1-1, "Observed Operator Response Time Summary," and Section II.2.D, "Operator Action Times." In addition, one of the modifications being installed to support the SGTR MTO single failure assumptions Section II.2.F, "Modifications to Support MTO Single Failure Considerations," Item 4, discusses a manual operator action to locally isolate a manual isolation valve. As noted in Section II.2.F, this manual is not replacing an automatic function but is simply an equivalent manual action for locally isolating high head safety injection flow into the RCS. The manual isolation of this valve is not time sensitive. Procedure changes will be made to dispatch an operator to the valve location upon identification of a SGTR accident, well in advance of potential need for manual valve isolation.

Demonstration runs have been performed to validate the response times assumed in the SGTR and MTO analyses. The results are presented in Attachment 5a, Section I.1.A and Table 1-1.

There are no automatic functions or actuations that are being replaced by manual actions due to changes proposed in this LAR.

*2.c* *Ensure that revised dose analysis inputs and assumptions are well documented within the LAR.*

Response:

The input parameters and assumptions used to analyze the radiological consequences of the SGTR event, as well as the calculated results are discussed in Attachment 5a, Section IV.2,"Input Parameters and Assumptions," and the associated tables.

*2.d* *Ensure licensing basis review of the planned steam generator power operated relief valve power supply modifications is conducted. Licensee should evaluate how changes to the electric power supply affect the facilities Updated Final Safety Analysis Report and ensure these are reflected in the MUR LAR as appropriate. NRC staff suggested that the licensee explain the interface between the MUR LAR and the EGC's response to the NRC verification inspection report letter containing several regulatory commitments dated March 2, 2011 (ADAMS Accession No. ML110620089).*

Response:

The licensing basis for all modifications associated with the MUR and the modifications to support the single failure assumptions in the SGTR MTO analysis have been reviewed and are planned to be installed under the provisions of 10 CFR 50.59, "Changes, tests, and experiments," and in accordance with the EGC design change process, with the exception of the Unit 1 SG PORV trim modification (which required NRC approval) as discussed in Attachment 5a, Section II.2.F. All UFSAR revisions associated with these modifications will be completed and implemented as part of the MUR amendment implementation.

A brief listing of the MUR related modifications is presented in Section 3.4.5, "Plant Modifications," and in Attachment 5, Sections, VII.2.B, "Control Room Controls, Displays and Alarms," Section VII.2.C, "Control Room Plant Reference Simulator,", Section VII.3, "Intent to Complete Modifications," and Section VIII.2, "Protective System Settings Changes."

The licensing basis for planned SG PORV power supply modifications will be thoroughly evaluated as part of the modification's 10 CFR 50.59 evaluation. The modification to install uninterruptible power supplies to the SG PORVs was prompted by the resolution of Unresolved Items (URIs) from the 2009 Component Design Bases Inspection (CDBI) at Byron Station (URI 05000454/2009007-03; URI 05000455/2009007-03). The URIs involved a concern with respect to the single failure assumptions used in Byron Station's analysis for a SGTR event. The NRC verified that the same SGTR-related concern was also applicable to Braidwood Station.

The resolution of these issues is discussed in Section 3.4.4, "Steam Generator Tube Rupture Analysis and Margin to Overfill Analysis Summary," and in Attachment 5a, Section II.2.F (Item 3).

In addition, the NRC staff discussed several best practices when developing power uprate licensing submittals documented in NRC letter items 3.a through 3.d:

*3.a* *The NRC staff recommended that the licensee identify all related/supporting Title 10 of the Code of Federal Regulations (10 CFR) Section 50.59 reviews prior to submission of the LAR to prevent unplanned changes to the NRC staff review.*

Response:

As noted above in Item 2.d, the licensing basis for all modifications associated with the MUR and the modifications to support the single failure assumptions in the SGTR MTO analysis have been reviewed and are planned to be installed under the provisions of 10 CFR 50.59, and in accordance with the EGC design change process, with the exception of the Unit 1 SG PORV trim modification which required NRC approval. No supplemental MUR-related submittals requesting NRC approval of design changes are anticipated.

*3.b* *The NRC staff recommended that the licensee seek prior review and approval of complex analyses (e.g., best-estimate loss-of-coolant accident [as approved in December 2010, for Braidwood and Byron], DNB analysis). This best practice minimizes the potential of NRC staff questions, resulting in reanalysis being performed during NRC staff review. When similar situations have occurred, applications are often delayed or withdrawn.*

Response:

There are no license amendments related to Byron Station and/or Braidwood Station currently under NRC review that would impact the MUR application.

EGC understands the NRC's position that licensees, in general, should seek prior NRC review and approval of complex analyses that are used in a subsequent license application. However, as discussed in the November 4, 2010 NRC/EGC pre-application meeting and subsequent teleconference on May 18, 2011 (noted above), there is an understanding that the Byron and Braidwood MUR application, in addition to the MUR-related analyses, would also contain changes related to the use of the VIPRE and associated DNB codes and revised SGTR and MTO analyses. EGC appreciates the NRC's consideration of these issues and, as discussed in Item 4 below, understands that this MUR submittal is "non-standard."

3,c *The NRC regulations at 10 CFR Part 50, Appendix B, Criterion XVI, require that licensees complete corrective actions promptly. The NRC staff noted that disposition of present facility corrective actions are most effectively addressed in NRC reviews independent of a LAR for power uprate.*

Response:

The non-conservative assumptions that have been identified in the current SGTR MTO analysis have been captured in the stations corrective action program and are being addressed through administrative controls. Correction of these issues necessitates performing a revised analysis that requires NRC approval as the accident dose results in more than a minimal increase as defined in NEI 96-01, "Guidelines for 10 CFR 50.59 Implementation," Revision 1, dated November 2000. Since increasing power level to MUR levels has an impact on the results of the SGTR MTO analysis, including the revised SGTR MTO analysis with the MUR LAR is an appropriate and expeditious method to resolve analysis issues.

3.d *The NRC staff discussed recent power uprate reviews where licensees had requested power uprates which included multiple dependent safety analyses which complicated NRC staff reviews (e.g., Point Beach extended power uprate LAR, approved on May 3, 2011 (ADAMS Accession No. ML111170513).*

Response:

EGC has reviewed numerous previous power uprate applications from other licensees. Lessons learned and previous responses to requests for additional information (RAIs) have been incorporated, where appropriate, into the Byron and Braidwood Stations MUR application. EGC considers this application to be complete and thorough.

Lastly, the requested approval date for the proposed LAR was discussed in NRC letter item 4.

4. *The licensee inquired into the NRC staff concerns associated with the planned Braidwood and Byron MUR application. The NRC staff noted that NRR resources assume a MUR application typically requires 6 months to review (following NRC staff acceptance of the application). Additional analyses make the proposed MUR request more complex, requiring additional resources, and introduces uncertainty into review templates and schedules.*

Response:

EGC recognizes that the MUR application for Byron and Braidwood Stations is more complex than a "standard" MUR application due to the inclusion of the VIPRE and associated DNB codes; and the SGTR analysis. EGC acknowledges that additional NRC resources will be required to review this MUR application. Rather than the NRC's typical 6 month review time for a standard MUR application, EGC is requesting NRC approval for this application in 12 months.

## 4.0 REGULATORY EVALUATION

### 4.1. Applicable Regulatory Requirements/Criteria

10 CFR 50, Appendix K, "ECCS Evaluation Models," requires that emergency core cooling system evaluation models assume that the reactor has been operating continuously at a power level at least 1.02 times the licensed power level to allow for instrumentation error. A change to this paragraph, which became effective on July 31, 2000, allows a lower assumed power level, provided the proposed value has been demonstrated to account for uncertainties due to power level instrumentation error.

The revision to 10 CFR 50, Appendix K does not permit licensees to utilize a lower uncertainty and increase thermal power without NRC approval. 10 CFR 50.90 requires that licensees desiring to amend an operating license file an amendment with the NRC. RIS 2002-03, Guidance on the Content of Measurement Uncertainty Power Uprate Applications," (Reference 1) provides NRC guidance for the content of license amendment requests involving power uprates based on measurement uncertainty recapture.

The proposed amendment would utilize the NRC-approved W-3 alternative correlations (i.e., the ABB-NV and WLOP correlations) noted in Attachment 5, Reference III.1-1. These correlations will be used in place of the W-3 correlation as the secondary DNB correlation for conditions where the primary DNB correlation is not applicable for calculating reactor core safety limits.

Also, as noted above in Section 3.4.4, the SGTR analysis was performed utilizing previously NRC-approved methodologies. The SGTR MTO analyses were performed using the LOFTTR2 program and the methodology developed in Attachment 5a, Reference 1, with modifications to address NSAL-07-11 (Attachment 5a, Reference 3) consistent with WCAP-16948-P (Attachment 5a, Reference 4). The thermal and hydraulic analyses were performed using the LOFTTR2 program and the methodology developed in Attachment 5a, References 1 and 2. The SGTR radiological analyses are based upon the alternative source term (AST) as defined in Regulatory Guide (RG) 1.183, with acceptance criteria as specified in RG 1.183 for offsite doses and in 10 CFR 50.67 for the control room.

This application is consistent with the requirements and criteria described in 10 CFR 50, Appendix K, 10 CFR 50.90, RIS 2002-03 and previously NRC-approved methodologies as noted above.

### 4.2. Precedent

The following facilities have recently received NRC approval for power uprates based on use of the LEFM system.

| Facility | Amendment #(s) | Approval Date |
|---|---|---|
| Calvert Cliffs, Units 1 and 2 | 291/267 | July 22, 2009 |
| North Anna, Units 1 and 2 | 257/238 | October 22, 2009 |
| Prairie Island, Units 1 and 2 | 197/186 | August 18, 2010 |

| | | |
|---|---|---|
| LaSalle, Units 1 and 2 | 198/185 | September 16, 2010 |
| Surry, Units 1 and 2 | 269/268 | September 24, 2010 |

## 4.3. No Significant Hazards Consideration

In accordance with 10 CFR 50.90, "Application for amendment of license or construction permit," and 10 CFR 50, Appendix K, "ECCS Evaluation Models," Exelon Generation Company, LLC (EGC) requests an amendment to Facility Operating License Nos. NPF-72, NPF-77, NPF-37 and NPF-66, and Appendix A, Technical Specifications (TS), for Braidwood Station, Units 1 and 2, and Byron Station, Units 1 and 2, respectively. The proposed changes would revise the maximum power level specified in each unit's operating license and the TS definition of rated thermal power (RTP). Specifically, the proposed change requests an increase from the current licensed thermal power (CLTP) of 3586.6 megawatts thermal (MWt) to 3645 MWt; an increase of approximately 1.63% RTP. The proposed changes are based on increased FW flow measurement accuracy, which will be achieved by utilizing Cameron International (formerly Caldon) CheckPlus™ Leading Edge Flow Meter (LEFM) ultrasonic flow measurement instrumentation.

The proposed amendment would utilize the NRC-approved W-3 alternative correlations (i.e., the ABB-NV and WLOP correlations). These correlations will be used in place of the W-3 correlation as the secondary DNB correlation for conditions where the primary DNB correlation is not applicable for calculating reactor core safety limits. This change is requested to increase the DNB operating margin under Measurement Uncertainty Recapture (MUR) power uprate conditions.

This amendment also proposes to increase the required RCS flow rate to be consistent with the assumptions in the revised thermal hydraulic analysis supporting operations at the MUR power level.

In addition, the steam generator tube rupture (SGTR) and margin to overfill (MTO) analysis has been revised and submitted for NRC approval as part of this amendment request, although no specific Technical Specification changes are directly associated with the revised analysis. The SGTR analysis has been revised with updated assumptions to gain additional MTO during a SGTR event. This revised analysis requires NRC approval as the results show more than a minimal increase in the accident dose as defined in NEI 96-01, "Guidelines for 10 CFR 50.59 Implementation," Revision 1, dated November 2000; however, the radiological consequences remain within the regulatory limits.

According to 10 CFR 50.92, "Issuance of amendment," paragraph (c), a proposed amendment to an operating license involves no significant hazards consideration if operation of the facility in accordance with the proposed amendment would not:

(1) Involve a significant increase in the probability or consequences of any accident previously evaluated; or

(2) Create the possibility of a new or different kind of accident from any accident previously evaluated; or

(3) Involve a significant reduction in a margin of safety.

EGC has evaluated the proposed changes, using the criteria in 10 CFR 50.92, and has determined that the proposed changes do not involve a significant hazards consideration. The following information is provided to support a finding of no significant hazards consideration.

1. Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

   Response: No

   The nuclear steam supply system and balance-of-plant systems, components and analyses that could be affected by the proposed change to the rated thermal power (RTP) level were evaluated using revised design parameters. The evaluations determined that these structures, systems and components are capable of performing their design function at the proposed uprated RTP of 3645 MWt. A portion of the current safety analyses remain bounding, as they were performed at 102% of the current power level which exceeds the requested MUR power level. Other analyses were previously performed at the current RTP level and have either been evaluated as acceptable or re-performed at the increased power level. The results demonstrate that acceptance criteria of the applicable analyses continue to be met at the uprated power conditions. As such, all applicable accident analyses continue to comply with the relevant acceptance criteria. Power level is an input assumption to equipment design and accident analyses; however, it is not a transient or accident initiator, and therefore does not increase the probability of an accident. Plant safety barriers are not challenged by the proposed changes.

   The source terms used to assess radiological consequences for each transient or accident have been reviewed. The radiological consequences are either bounded by the current analysis or have been evaluated to remain within regulatory limits at the uprated condition. Specifically, the SGTR and MTO analysis has been revised with updated assumptions to gain additional margin to overfill during a SGTR event. Appropriate modifications will be added to the plant in support of the SGTR analysis single failure assumptions. Although the revised analysis results in more than a minimal increase in the accident dose, as defined in NEI 96-01, "Guidelines for 10 CFR 50.59 Implementation," Revision 1, dated November 2000, the dose results remain within the limits specified in the Standard Review Plan (SRP), Section 15.6.3, "Radiological Consequences of Steam Generator Tube Failure (PWR)."

   The primary loop components (e.g., reactor vessel, reactor internals, control rod drive housings, piping and supports, and reactor coolant pumps) remain within their applicable structural limits and will continue to perform their intended design functions. Thus, there is no significant increase in the probability of a structural failure of these components.

   In addition, the proposed use of the LEFM, the NRC-approved W-3 alternative correlations, (i.e., the ABB-NV and WLOP correlations) and the increase in required RCS flow, serve to facilitate operations at the uprated power level and have no impact on the probability or consequences of any accident previously evaluated.

   Therefore, the proposed changes described above do not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

   Response: No

   No new accident scenarios, failure mechanisms, or limiting single failures are introduced as a result of any proposed changes. LEFM system failures will not adversely affect any safety-related system or any structures, systems or components required for transient mitigation. Structures, systems and components previously required for transient mitigation continue to be capable of fulfilling their intended design functions. The proposed changes have no significant adverse affect on any safety-related structures, systems or components and do not significantly change the performance or integrity of any safety-related system.

   The proposed changes do not adversely affect any current system interfaces or create any new interfaces that could result in an accident or malfunction of a different kind than previously evaluated. Operating at RTP of 3645 MWt does not create any new accident initiators or precursors. Credible malfunctions are bounded by the current accident analyses of record or recent evaluations demonstrating that applicable criteria are still met with the proposed changes.

   The proposed changes to replace the W-3 DNB correlation with the NRC approved ABB-NV and WLOP correlations, the revision to the required RCS flow rate, and the assumptions used in the revised SGTR and MTO analysis would not prompt a new or different kind of accident.

   Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does the proposed change involve a significant reduction in a margin of safety?

   Response: No

   Operation at the uprated power condition does not involve a significant reduction in a margin of safety. Analyses of the primary fission product barriers have concluded that relevant design criteria remain satisfied, both from the standpoint of the integrity of the primary fission product barrier, and from the standpoint of compliance with the required regulatory and analysis acceptance criteria. As appropriate, all evaluations have been performed using methods that have either been reviewed or approved by the Nuclear Regulatory Commission, or that are in compliance with regulatory review guidance and standards.

   The margins of safety associated with the power uprate are those pertaining to core thermal power. These include fuel cladding, reactor coolant system pressure boundary, and containment barriers. Core analyses demonstrate that operation at the proposed uprated power level will continue to meet the nuclear design basis acceptance criteria. Impacts to components associated with the reactor coolant system pressure boundary structural integrity, and factors such as pressure-temperature limits, vessel fluence, and pressurized thermal shock were found to be acceptable under MUR operating conditions. The proposed changes will have minimal effect on operating parameters and the noted components remain capable of performing their intended safety functions following implementation of the MUR power uprate.

The revised SGTR and MTO analysis show acceptable results. The resultant SGTR dose remains within the limits specified in the Standard Review Plan (SRP), Section 15.6.3, “Radiological Consequences of Steam Generator Tube Failure (PWR).” The analysis also shows an improvement (i.e., a larger margin) in the MTO results. The results of all other associated safety analyses remain acceptable.

The proposed changes to use the NRC-approved W-3 alternative correlations, (i.e., the ABB-NV and WLOP correlations) and the increase in the required minimum RCS flow value verify that appropriate nuclear and thermal hydraulic margins to safety are maintained.

Therefore, the proposed changes do not involve a significant reduction in a margin of safety.

### 4.4. Conclusions

Based on the above evaluation, EGC concludes that the proposed amendment presents no significant hazards consideration under the standards set forth in 10 CFR 50.92, paragraph (c), and accordingly, a finding of no significant hazards consideration is justified.

In conclusion, based on the considerations discussed above, (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or the health and safety of the public.

## 5.0 ENVIRONMENTAL CONSIDERATION

10 CFR 51.22, "Criterion for categorical exclusion; identification of licensing and regulatory actions eligible for categorical exclusions or otherwise not requiring environmental review," addresses requirements for submitting environmental assessments as part of licensing actions. 10 CFR 51.22, paragraph (c)(9) states that a categorical exclusion applies for Part 50 license amendments that meet the following criteria:

i. No significant hazards consideration (as defined in 10 CFR 50.92(c)).

The proposed changes do not involve a significant hazards consideration. The reviews and evaluations performed to support the proposed uprate conditions and the revised Steam Generator Tube Rupture (SGTR) and Margin to Overfill (MTO) Analysis concluded that all systems will function as designed. All performance requirements for these systems have been evaluated and found acceptable. No new accident scenarios, failure mechanisms, or limiting single failures are introduced as a result of the proposed changes. Operation at the uprated power condition and the SGTR and MTO reanalysis results do not involve a significant reduction in a margin of safety.

ii. No significant change in the types or significant increase in the amounts of any effluents that may be released offsite.

There is no significant change in the types or significant increase in the amounts of gaseous, liquid or solid effluents. Evaluations of the effects of the proposed changes related to the increase in reactor power on effluent sources concluded that, at most, the increase in radiological effluents is proportional to or slightly greater than the requested power increase. The radiological effluent calculations in the revised SGTR MTO analysis show more than a minimal increase in the accident dose, as defined in NEI 96-01, "Guidelines for 10 CFR 50.59 Implementation," Revision 1, dated November 2000. This "more than minimal increase" is not considered a significant increase as the revised SGTR accident dose values remain within the limits specified in the Standard Review Plan (SRP), Section 15.6.3, "Radiological Consequences of Steam Generator Tube Failure (PWR)" and significant margin to the regulatory limits is maintained.

Non-radiological effluent releases are either unaffected (i.e., not power dependent) or insignificantly affected (i.e., increase by approximately 2% or less) by the proposed changes and continue to be bounded by those described in the Final Environmental Statement for Byron Station, Units 1 and 2; and Braidwood Station, Units 1 and 2.

iii. No significant increase in individual or cumulative occupational radiation exposure.

There is no significant increase in individual or cumulative occupational radiation exposure. Evaluations of projected radiation exposure due to liquid, gaseous and solid radwaste concluded that normal operation radiation levels increase slightly, (approximately 2.0%) for the proposed uprate. The occupational exposure is controlled by the plant radiation protection program and is maintained within values required by regulations.

Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22, paragraph (c)(9). Therefore, pursuant to 10 CFR 51.22, paragraph (b), no environmental impact statement or environmental assessment needs to be prepared in connection with the proposed amendment.

REFERENCES

1 NRC Regulatory Issue Summary 2002-03, "Guidance on the Content of Measurement Uncertainty Recapture Power Uprate Applications," dated January 31, 2002

2. Cameron Topical Report ER-157P, "Supplement to Topical Report ER-80P: Basis for Power Uprates with an LEFM √ TM or an LEFM CheckPlus TM System," Rev. 8, dated May 11, 2009

3. Letter from Steven A. Reynolds (USNRC) to Michael J. Pacilio (Exelon Generation Company, LLC), "Byron Station, Units 1 and 2 Follow Up Inspection of an Unresolved Item; 05000454/201 1010; 050004552011010," dated January 19, 2011

4. Letter from Timothy J. Tulon (Exelon Generation Company, LLC) to the USNRC, "Response to NRC Follow Up Inspection Report; 05000454/2011010; 05000455/2011010," dated February 18, 2011

5. Letter from Steven A. Reynolds (USNRC) to Michael J. Pacilio (Exelon Generation Company, LLC), "Braidwood Station, Units 1 and 2 Verification Inspection Related to Analysis of Steam Generator Tube Rupture Event Margin to Overfill; 05000456/2011009; 050004572011009," dated February 1, 2011

6. Letter from Daniel J. Enright (Exelon Generation Company, LLC) to the USNRC, "Response to NRC Verification Inspection Report; 05000456/2011009; 050004572011009," dated March 2, 2011