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10 CFR 50.90
10 CFR 50, Appendix K

RS-10-001

January 27, 2010

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555-0001

LaSalle County Station, Units 1 and 2
Facility Operating License Nos. NPF-11 and NPF-18
NRC Docket Nos. 50-373 and 50-374

Subject: Request for License Amendment Regarding Measurement Uncertainty Recapture 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-11 and NPF-18 for LaSalle County Station (LSCS), Units 1 and 2, respectively. Specifically, the proposed changes revise the Operating License and Technical Specifications (TS) to implement an increase of approximately 1.65% in rated thermal power from the current licensed thermal power (CLTP) of 3489 megawatts thermal (MWt) to 3546 MWt.

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. LEFM instrumentation is currently installed in LSCS, Unit 1 and will be installed in LSCS, Unit 2 in refueling outage L2R13, currently scheduled to complete in March 2011.

The content of this request is consistent with the guidance contained in the referenced RIS.

The proposed changes also modify the TS and Technical Requirements Manual (TRM) for the TS setpoint (i.e., the Flow Biased Simulated Thermal Power – Upscale scram) that is revised in these proposed changes by adding requirements to assess channel performance during testing.

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, and do not require NRC approval.
- Attachment 4 provides a cross-reference between the contents of this request and the referenced RIS.
- Attachment 5 provides a summary of the regulatory commitments made in this request.
- Attachment 6 provides the General Electric-Hitachi (GEH) Nuclear Energy document NEDC-33485P, "Safety Analysis Report for LaSalle County Station, Units 1 and 2 Thermal Power Optimization," (Proprietary Version).
- Attachment 7 provides an affidavit from GEH Nuclear Energy supporting withholding of Attachment 6.
- Attachment 8 provides the GEH Nuclear Energy document NEDC-33485, "Safety Analysis Report for LaSalle County Station, Units 1 and 2 Thermal Power Optimization," (Non-Proprietary Version).
- Attachment 9 provides Cameron documents ER-629, Rev. 1, "Bounding Uncertainty Analysis for Thermal Power Determination at LaSalle Unit 1 Using the LEFM CheckPlus System," (Proprietary Version), and ER-746, Rev. 1a, "Bounding Uncertainty Analysis for Thermal Power Determination at LaSalle Unit 2 Using the LEFM CheckPlus System," (Proprietary Version).
- Attachment 10 provides affidavits from Cameron International Corporation supporting withholding of Attachment 9.
- Attachment 11 provides EGC calculation L-003445, "Core Thermal Power Uncertainty to Support MUR for LaSalle Unit 2."
- Attachment 12 provides PJM Interconnection document, "Generator Transient Stability Study for LaSalle Station," and ComEd document, "2010 Power Grid Voltage Analysis for LaSalle Generating Station (Post MUR Power Uprate)."
- Attachment 13 provides EGC calculations for the instrument setpoint that is being revised in these proposed changes.
- Attachment 14 provides drawings describing the installation of the LEFM.

The proposed changes have been reviewed by the LSCS Plant Operations Review Committee 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 November 29, 2010. The requested review period is consistent with NRC internal guidance and supports business plan initiatives to increase EGC’s generation capacity. Once approved, the amendment will be implemented within 90 days for Unit 1. This implementation period will provide adequate time for revision of the affected station documents using the appropriate change control mechanisms. For Unit 2, the amendment will be implemented within 90 days of completion of refueling outage L2R13, which is currently scheduled for completion in March 2011. This implementation period will allow for installation of the LEFM instrumentation during L2R13 and subsequent 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 6 and 9. Attachment 6 is considered proprietary by GEH Nuclear Energy. An affidavit supporting this request is included as Attachment 7 and a non-proprietary version of Attachment 6 is provided in Attachment 8. Attachment 9 is considered proprietary by Cameron International Corporation. An affidavit supporting this request is included as Attachment 10. A non-proprietary version of Attachment 9 is 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 27th day of January, 2010.

Respectfully,

Michael D. Jesse
Manager, Licensing - Power Uprate

Attachments:

1. Evaluation of Proposed Changes
2. Markup of Proposed Operating License and Technical Specifications Pages
3. Markup of Proposed Technical Specifications Bases and Technical Requirements Manual Pages
4. NRC Regulatory Issue Summary 2002-03 Cross-Reference
5. Summary of Regulatory Commitments
6. GEH Nuclear Energy Safety Analysis Report for LaSalle County Station, Units 1 and 2 Thermal Power Optimization, NEDC-33485P (Proprietary Version)
7. GEH Nuclear Energy Affidavit Supporting Withholding
8. GEH Nuclear Energy Safety Analysis Report for LaSalle County Station, Units 1 and 2 Thermal Power Optimization, NEDO-33485 (Non-Proprietary Version)
9. Cameron ER-629, Rev. 1, "Bounding Uncertainty Analysis for Thermal Power Determination at LaSalle Unit 1 Using the LEFM CheckPlus System," (Proprietary Version), and Cameron ER-746, Rev. 1a, "Bounding Uncertainty Analysis for Thermal Power Determination at LaSalle Unit 2 Using the LEFM CheckPlus System," (Proprietary Version)
10. Cameron International Corporation Affidavits Supporting Withholding
11. Exelon Generation Company, LLC Calculation L-003445, "Core Thermal Power Uncertainty to Support MUR for LaSalle Unit 2"

12. PJM Interconnection document, "Generator Transient Stability Study for LaSalle Station," and ComEd document, "2010 Power Grid Voltage Analysis for LaSalle Generating Station (Post MUR Power Uprate)"
13. Exelon Generation Company, LLC Instrument Setpoint Calculations
14. Mechanical Drawings for Leading Edge Flowmeter Installation

cc: NRC Regional Administrator, Region III
NRC Senior Resident Inspector – LaSalle County 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-11 and NPF-18 for LaSalle County Station (LSCS), Units 1 and 2, respectively. Specifically, the proposed changes revise the Operating License and Technical Specifications (TS) to implement an increase of approximately 1.65% in rated thermal power (RTP) from 3489 megawatts thermal (MWt) to 3546 MWt.

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. LEFM instrumentation is currently installed in LSCS Unit 1 and will be installed in LSCS Unit 2 prior to implementation of these requested changes.

The proposed amendment would also modify the TS and Technical Requirements Manual (TRM) for the applicable TS setpoint (i.e., the Flow Biased Simulated Thermal Power – Upscale scram) that is revised in these proposed changes by adding requirements to assess channel performance during testing. This change is consistent with interim guidance proposed by the industry in a letter from the Technical Specifications Task Force (TSTF) to the NRC, "Industry Plan to Resolve TSTF-493, 'Clarify Application of Setpoint Methodology for LSSS Functions,' " (Reference 1) and the NRC's response (Reference 2).

2.0 DETAILED DESCRIPTION

The proposed changes to the Operating Licenses and TS are described below, with marked-up pages included in Attachment 2.

1. Changes related to the value of RTP
   LaSalle County Station, Units 1 and 2, Facility Operating License Numbers NPF-11 and NPF-18, Sections 2.C(1), "Maximum Power Level," are revised to increase the value of RTP from 3489 MWt to 3546 MWt.

   The definition of RTP in TS Section 1.1, "Definitions," is revised to increase the value of RTP from 3489 MWt to 3546 MWt.

2. Changes related to TS Table 3.3.1.1-1, Function 2.b, Flow Biased Simulated Thermal Power – Upscale
   In TS Table 3.3.1.1-1, "Reactor Protection System Instrumentation," Function 2.b, Flow Biased Simulated Thermal Power – Upscale, the allowable value (AV) is revised as follows.

   Current: $\leq$ 0.62 W + 69.3% RTP and $\leq$ 115.5% RTP [(b)]

   Proposed: $\leq$ 0.61 W + 68.2% RTP and $\leq$ 115.5% RTP [(d)]

   In TS Table 3.3.1.1-1, footnote (b) is revised as follows. The footnote is also renumbered to become footnote (d).

Current: Allowable value is ≤ 0.55 W + 56.8% RTP and ≤ 112.3% RTP when reset for single loop operation per LCO 3.4.1, "Recirculation Loops Operating."

Proposed: Allowable value is ≤ 0.54 W + 55.9% RTP and ≤ 112.3% RTP when reset for single loop operation per LCO 3.4.1, "Recirculation Loops Operating."

3. <u>Changes related to TS 3.3.1.3, Oscillation Power Range Monitor (OPRM) Instrumentation</u>
   The value of RTP in SR 3.3.1.3.5 is revised from ≥ 28.6% RTP to ≥ 28.1% RTP.
4. <u>Changes related to instrument channel performance during testing</u>
   In TS Table 3.3.1.1-1, for Function 2.b, the following notes are added to Surveillance Requirement (SR) 3.3.1.1.11.

   (b) If the as-found channel setpoint is outside its predefined as-found tolerance, then the channel shall be evaluated to verify that it is functioning as required before returning the channel to service.

   (c) The instrument channel setpoint shall be reset to a value that is within the as-left tolerance around the nominal trip setpoint (NTSP) at the completion of the surveillance; otherwise, the channel shall be declared inoperable. Setpoints more conservative than the NTSP are acceptable provided that the as-found and as-left tolerances apply to the actual setpoint implemented in the surveillance procedures (field setting) to confirm channel performance. The NTSP and the methodologies used to determine the as-found and the as-left tolerances are specified in the Technical Requirements Manual.

Proposed changes to the TS Bases and Technical Requirements Manual (TRM) are described below, with marked-up pages included in Attachment 3. These changes are for information only, and do not require NRC approval.

<u>TRM Changes</u>

1. The definition of RTP in TRM Section 1.1, "Definitions," is revised to increase the value of RTP from 3489 MWt to 3546 MWt.
2. In Table T3.3.c-1, "Control Rod Block Instrumentation," the AVs for Function 1.a, Flow Biased Simulated Thermal Power – Upscale are revised as follows.

   Current: < 0.62 W + 57.9% RTP
   Proposed: < 0.61 W + 56.9% RTP

   Footnote (a) is revised as follows.

   Current: Allowable value is < (0.55 W + 45.4%) RTP when reset for single loop operation per Technical Specification 3.4.1, "Recirculation Loops Operating."

   Proposed: Allowable value is < (0.54 W + 44.7% RTP) when reset for single loop operation per Technical Specification 3.4.1, "Recirculation Loops Operating."
3. New TRM Limiting Condition for Operation (TLCO), 3.3.q, "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 the current licensed power level (i.e., pre-uprate power level) of 3489 MWt. A channel check of the LEFM is also specified at a 12-hour frequency.

4. TRM Appendix D is revised to include a reference to the methodology for calculating the as-found and as-left tolerances.

TS Bases Changes

1. The Bases for SR 3.3.1.1.11 are revised to incorporate explanation of the notes added to this SR regarding evaluation of instrument channel performance during testing.
2. The Bases for SR 3.3.1.3.5 are revised to reflect the reduction in the value at which the OPRM instrumentation is armed from 28.6% RTP to 28.1% RTP.

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 that emergency core cooling system (ECCS) 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 1, 2000, allows a lower assumed power level, provided the proposed value has been demonstrated to account for uncertainties due to power level instrumentation error.

Utilization of the Cameron CheckPlus™ LEFM system at LSCS, Units 1 and 2 will result in reduced uncertainty in FW flow measurement, which reduces the total power level measurement uncertainty. As described in Section 3.2, "LEFM Ultrasonic Flow Measurement and Core Thermal Power Uncertainty," with the utilization of the LEFM system, the core thermal power measurement uncertainty will be a maximum of 0.346%.

As summarized in Section 3.4.1, "Summary of Analyses," and Attachment 6, the ECCS evaluation models and other plant safety analyses currently assume a two percent thermal power uncertainty. Utilization of the LEFM system thus supports an increase in RTP up to 1.654% (i.e., 2% - 0.346%), based on the reduction in thermal power uncertainty. This increase in RTP corresponds to 3546.7 MWt, which is rounded down to the requested 3546 MWt, or approximately 1.65%.

EGC has evaluated the effects of a bounding 1.7% increase in RTP using an approach developed by General Electric-Hitachi (GEH) Nuclear Energy and approved by the NRC, which is documented in NEDC 32938P-A, "Licensing Topical Report: Generic Guidelines and Evaluations for General Electric Boiling Water Reactor Thermal Power Optimization," (Reference 3). These evaluations are described in detail in Attachment 6.

The scope and content of the evaluations performed and described in this request 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 4). Attachment 4 provides a cross-reference between the contents of this application and the RIS 2002-03 guidance.

The proposed changes would also modify the TS for the instrumentation with a revised setpoint (i.e., the Flow Biased Simulated Thermal Power – Upscale scram) related to the power uprate. The change adds new test requirements, thereby ensuring the instrument will function as required to initiate protective systems or actuate mitigating systems at the point assumed in the applicable safety analysis. This TS change is made through the addition of individual footnote requirements to the instrument function.

3.2. LEFM Ultrasonic Flow Measurement and Core Thermal Power Uncertainty

3.2.1 LEFM flow measurement

The LEFM system uses ultrasonic transit time principles to determine fluid velocity. This flow measurement method is described in topical reports ER-80P, "Improving Thermal Power Accuracy and Plant Safety While Increasing Operating Power Level Using the LEFM √™ System," (Reference 5) and ER-157P, "Supplement to Cameron Topical Report ER-80P: Basis for Power Uprates with an LEFM √ or an LEFM √Plus System," (Reference 6). These topical reports were approved by the NRC in documents titled, "Comanche Peak Steam Electric Station, Units 1 and 2 - Review of Cameron Engineering Topical Report ER 80P, 'Improving Thermal Power Accuracy and Plant Safety While Increasing Power Level Using the LEFM System,' " (Reference 7) and "Waterford Steam Electric Station, Unit 3; River Bend Station; and Grand Gulf Nuclear Station - Review of Caldon, Inc. Engineering Report ER-157P," (Reference 8).

In References 7 and 8, the NRC established criteria for use of these topical reports in requests for license amendments. EGC's response to those criteria is provided in Section 3.2.4, "Disposition of NRC Criteria for Use of LEFM Topical Reports."

This instrumentation is not safety-related. However, the LEFM system is designed and manufactured in accordance with Cameron's Quality Assurance Program, which conforms with 10 CFR 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants." Cameron's verification and validation (V&V) program fulfills the requirements of ANSI/IEEE-ANS Std. 7-4.3.2, 1993, "IEEE Standard Criteria for Digital Computers in Safety Systems of Nuclear Power Generating Stations," Annex E, and ASME NQA-2a-1990, "Quality Assurance Requirements for Nuclear Facility Applications." In addition, the program is consistent with guidance for software V&V in EPRI TR-103291, "Handbook for Verification and Validation of Digital Systems," December 1994. Specific examples of quality measures undertaken in the design, manufacture, and testing of the LEFM system are provided in Reference 5, Section 6.4 and Table 6.1.

3.2.2 Plant Implementation

The LEFM spool pieces are installed in LSCS, Unit 1 and will be installed in LSCS, Unit 2 FW piping as shown in Attachment 14. The LEFM spool pieces will be installed in two straight sections of piping. The installation location is downstream of the common FW header, which splits into two straight sections of piping. After the piping splits, each pipe has an installed flow straightener. The installation location is between the flow straightener and the originally-installed FW flow nozzle.

The transducers will be located in the Main Steam Line Tunnel in the Auxiliary Building in an anticipated radiation field of 2.50 R/hr at full power. The electronics cabinet will be on the other side of the wall of the Main Steam Line Tunnel in an anticipated radiation field of less than 2 mR/hr at full power. No radiation damage or degradation to the instruments (including electronics) due to such exposure is anticipated.

For LSCS, Unit 2, a modification package has been developed outlining the steps to install and test the LEFM system. Once the unit has been shutdown for the refueling outage, the LEFM spool pieces will be installed, transducers installed, cables routed, and connections made to the plant process computer. Following installation, testing will include an inservice leak test, comparisons of FW flow and thermal power calculated by various methods, and final commissioning testing. Final commissioning testing is described in Appendix F of Reference 5.

3.2.3 LEFM and Core Thermal Power Measurement Uncertainty and Methodology

Attachment 9 provides the results of testing and calibration of the LEFM system at LSCS, Units 1 and 2. The results for Unit 1 indicate a FW mass flow rate uncertainty of ±0.31% with a fully functional LEFM system. The results for Unit 2 indicate a FW mass flow rate uncertainty of ±0.30% with a fully functional LEFM system. These uncertainty values were calculated using the methodology described in Reference 6, which was approved by the NRC in Reference 8.

To bound the mass flow rate uncertainty, EGC has used a FW mass flow rate uncertainty of ±0.32% to determine the core thermal power uncertainty for both units. Based on a FW mass flow rate uncertainty of ±0.32%, EGC has completed the thermal power uncertainty calculation for LSCS, which results in a total uncertainty of ±0.346% in the calculation of RTP for the site-specific installation. This calculation is provided in Attachment 11. The calculation methodology is consistent with the EGC setpoint calculation methodology. The uncertainty is at a 95% probability and 95% confidence level. Attachment 11 provides further discussion of the uncertainty in the core thermal power calculation.

3.2.4 Disposition of NRC Criteria for Use of LEFM Topical Reports

In References 7 and 8, the NRC established four criteria to be addressed by licensees incorporating the LEFM methodology into the licensing basis. The four criteria are listed below, along with a discussion of how each will be satisfied.

<u>Criterion 1</u>

Discuss maintenance and calibration procedures that will be implemented with the incorporation of the LEFM, including processes and contingencies for inoperable LEFM instrumentation and the effect on thermal power measurements and plant operation.

<u>Response to Criterion 1</u>

<u>*Calibration and Maintenance*</u>

Implementation of the power uprate license amendment will include developing the necessary procedures and documents required for maintenance and calibration of the LEFM system. Plant maintenance and calibration procedures will be revised to incorporate Cameron’s maintenance and calibration requirements prior to declaring the LEFM system operational and raising power above the current licensed thermal power (CLTP) of 3489 MWt. The incorporation of, and continued adherence to, these requirements will assure that the LEFM system is properly maintained and calibrated.

Preventive maintenance scope and frequency is based on vendor recommendations. The current vendor-recommended frequency is every refueling outage (i.e., nominally every 24 months for LSCS). Preventive maintenance activities consist of physical inspections, power supply and pressure transmitter checks, and clock verification. These preventive maintenance activities are being implemented via the associated plant modification package.

Maintenance of the LEFM system will be performed by personnel qualified on the LEFM system.

For instrumentation other than the LEFM system that contributes to the power calorimetric computation, calibration and maintenance is performed periodically using existing site procedures. Maintenance and test equipment, setting tolerances, calibration frequencies, and instrumentation accuracy were evaluated and accounted for within the thermal power uncertainty calculation.

LEFM Inoperability

The redundancy inherent in the two measurement planes of an LEFM system makes the system tolerant to component failures. The system features automatic self-checking. A continuously operating on-line test is provided to verify that the digital circuits are operating correctly and within the specified accuracy range. Failure messages are generated by the plant process computer and monitored in the control room, if system failure events are detected.

The proposed TRM specification requires an LEFM channel check every 12 hours. In addition to this confirmation of status, the plant process computer will provide a computer alarm message to the Control Room if the status of the LEFM instrumentation changes. The electronics cabinet performs on-line, continuous monitoring of system parameters; the maintenance status of the cabinet will change if this monitoring reveals problems with the instrumentation.

As noted in the TRM changes provided and discussed in Section 2.0, "Detailed Description," if the LEFM system is not repaired within 72 hours, power will be reduced and administratively controlled to remain less than or equal to the CLTP of 3489 MWt.

The 72-hour allowed outage time (AOT) for the LEFM system prior to reducing to the CLTP is acceptable. During the AOT, core thermal power will be calculated using the FW flow as measured by the existing FW flow nozzles. Although the FW flow nozzle measurements may drift slightly during this period due to fouling, fouling of the nozzles results in a higher than actual indication of FW flow. This condition results in an overestimate of the calculated calorimetric power level, which is conservative, as the reactors will actually be operating below the calculated power level. A sudden de-fouling event during the 72-hour inoperability period is unlikely and significant sudden defouling would be detected by a change in the FW flow nozzle differential pressure. Regarding potential drift in the measurement of feedwater differential pressure across the flow nozzle, Reference 5, in Table A-1, shows a typical power measurement uncertainty calculation for a two-feedwater line BWR to be approximately 1.4%. The systematic error associated with feed flow nozzle differential pressure in this calculation is shown to be approximately 1.0%. Assuming this was calculated based on an 18-month cycle, this would represent a maximum potential drift in the differential pressure measurement of less than 0.002% per day. Over a 72-hour period, this would have an insignificant effect on the feedwater flow measurement. Finally, operators routinely monitor other indications of core thermal power, including Average Power Range Monitors (APRMs), steam flow, feed flow, turbine first stage pressure, and main generator output.

As noted in Attachment 3, the limitations discussed above regarding operation with an inoperable LEFM system will be included in the TRM, which will be revised prior to implementation. The NRC has previously approved power uprate applications with AOTs of up to 72 hours.

Reactor power is calculated by the backup plant computer when the primary plant computer system is not operable. In the event that both the primary and backup computers are inoperable, the reactor power can be ascertained from multiple parameters (APRMs, steam flow, feed flow, turbine first stage pressure, main generator output), or via a hand calculation.

<u>Criterion 2</u>

For plants that currently have LEFMs installed, provide an evaluation of the operational and maintenance history of the installed installation and confirmation that the installed instrumentation is representative of the LEFM system and bounds the analysis and assumptions set forth in Topical Report ER-80P.

<u>Response to Criterion 2</u>

The LEFM system on LSCS, Unit 1 was installed during refueling outage L1R12 in 2008 and has been used to supply the FW flow input to the plant process computer core thermal power calculation since installation and commissioning. The following maintenance issues have occurred with the system since installation.

- The signal conditioner associated with one of the pressure transmitters on one of the LEFM spool pieces drifted, which placed the LEFM system in maintenance mode. The signal conditioner was replaced, and the replacement has been performing normally since replacement.
- There was a series of communication failures between the LEFM and the plant process computer, causing the LEFM to temporarily enter the maintenance mode. This problem was resolved with a software modification and an LEFM central processing unit (CPU) replacement in 2008, and has not re-occurred since.
- Following a maintenance outage in July 2009, the FW line B path 1 transducer gain increased approximately 5 dB. Cameron International determined that this condition did not affect the accuracy of the FW flow measurement.
- Finally, the B LEFM CPU experienced repeated reboots due to humidity entering the panel in early spring timeframe. An open conduit from the turbine building leading into the panel was sealed, eliminating this problem. In addition, there has been one incident in which the A LEFM CPU locked up in the first couple of months of operation. This CPU was rebooted without issue and the problem has not been seen since.

Final commissioning of the LEFM system on LSCS, Unit 1, was completed following installation; installation occurred during refueling outage L1R12 in February 2008. The commissioning process verified bounding calibration test data, as described in Appendix F of Reference 5. This step provided final confirmation that actual performance in the field meets the uncertainty bounds established for the instrumentation as described in Attachment 9.

For LSCS, Unit 2, this criterion does not apply, as the LEFM will be installed during the L2R13 refueling outage scheduled for completion in March 2011.

Criterion 3

Confirm that the methodology used to calculate the uncertainty of the LEFM in comparison to the current FW instrumentation is based on accepted plant setpoint methodology (with regard to the development of instrument uncertainty). If an alternative approach is used, the application should be justified and applied to both venturi and ultrasonic flow measurement instrumentation installations for comparison.

Response to Criterion 3

The LEFM system uncertainty calculation is based on the American Society of Mechanical Engineers PTC 19.1 methodology (Reference 9) and the Instrumentation, Systems, and Automation Society ISA-RP67.04.02-200 methodology (Reference 10.) This LEFM system uncertainty calculation methodology is based on a square-root-sum-of-squares (SRSS) calculation, which is consistent with the method used in the current core thermal power uncertainty calculation for the existing FW instrumentation, as well as the method used for the revised core thermal power uncertainty calculation using the LEFM system.

Criterion 4

For plants where the ultrasonic meter (including LEFM) was not installed and flow elements were not calibrated to a site-specific piping configuration (i.e., flow profiles and meter factors not representative of the plant specific installation), additional justification should be provided for its use. The justification should show that the meter installation is either independent of the plant specific flow profile for the stated accuracy, or that the installation can be shown to be equivalent to known calibrations and plant configurations for the specific installation including the propagation of flow profile effects at higher Reynolds numbers. Additionally, for previously installed calibrated elements, confirm that the piping configuration remains bounding for the original LEFM installation and calibration assumptions.

Response to Criterion 4

Criterion 4 does not apply to LSCS, Units 1 and 2. The calibration factors for the LSCS, Units 1 and 2 spool pieces were established by tests of these spools at Alden Research Laboratory. These tests included a full-scale model representative of the LSCS hydraulic geometry. The Alden data report for these tests is on file and Cameron engineering reports evaluating the test data for both units are provided in Attachment 9.

There is no significant difference between the FW piping configuration and the model used at Alden Research Lab for LSCS, Unit 1. Based on similar plant configuration, the Unit 2 spool pieces will also be installed to match the as-tested configuration. As discussed in Attachment 9 for LSCS, Unit 2, the LEFM testing included verifying the effects of the flow straightener orientation; therefore the LEFM uncertainty takes this effect into consideration.

A discussion of the impact of other plant-specific installation factors on the FW flow measurement uncertainty is provided in Attachment 9, in appendices to ER-629 and ER-746. Appendix A.3 to ER-629 contains Cameron ER-644, "LEFM Check Plus Meter Factor Calibration and Accuracy Assessment for LaSalle Nuclear Power Station," [Unit 1 version], and Appendix A.3 to ER-746 contains ER-791, "Meter Factor Calculation and Accuracy Assessment for LaSalle Unit 2." Sections 2.2, 4.2, and 4.4 of these reports provide responses to many of the previous NRC requests for additional information from NRC-approved applications listed in Section 4.2, "Precedent."

For LSCS, Unit 2, final acceptance of the site-specific uncertainty analyses will occur after the completion of the commissioning process. Final commissioning is expected to be completed in March 2011.

3.2.5 Deficiencies and Corrective Actions

Cameron has procedures to notify users of important LEFM deficiencies. LSCS also has existing processes for addressing manufacturer's deficiency reports. Such deficiencies will be documented in the LSCS corrective action program.

Problems with plant instrumentation identified by LSCS personnel are also documented in the LSCS corrective action program and necessary corrective actions are identified and implemented. Deficiencies associated with the vendor’s processes or equipment are reported to the vendor to support corrective action.

3.2.6 Reactor Power Monitoring

LaSalle General Operating Procedure LGP-3-1, "Power Changes," Revision 46, provides guidance to ensure that reactor power remains within the requirements of the operating license. LGP-3-1, Step D.9, "NEI Licensed Power Limit," specifies the following:

> No actions are allowed that would intentionally raise core thermal power above the licensed power limit for any period of time. Small, short-term fluctuations in power that are not under the direct control of a licensed reactor operator (e.g., bi-stable flow) are not considered intentional. During full power operation the Reactor Operator will closely monitor thermal power trends and take action, as required, to maintain a goal of keeping the 2 hour thermal power average at or below the license limit. If the core thermal power average for a 2 hour period exceeds the licensed power limit, take timely action to ensure that thermal power is less than or equal to the licensed power limit. The core thermal power average for a shift is not to exceed the licensed power limit. A shift can be no longer than 12 hours.

LGP-3-1 guidance is consistent with the guidance proposed by the Nuclear Energy Institute and endorsed by the NRC in Reference 11.

3.3. Evaluation of Changes to License and Technical Specifications

The proposed changes to the TS described in Section 2.0, "Description of Changes," are evaluated below. The numbering of these changes corresponds to the numbering in Section 2.0.

Section 2.0, Item 1 (change in RTP)

The proposed increase in RTP to 3546 MWt in the operating license and 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.

Section 2.0, Item 2 (revised AVs for Flow-Biased Thermal Power function)

The proposed change to the AVs for the reactor protection system Flow-Biased Simulated Thermal Power function are based on the approach described in Reference 3. The Flow-Biased Simulated Thermal Power analytical limits (ALs) and AVs, for both two-loop operation and single loop operation, are unchanged in units of absolute core thermal power versus recirculation drive flow. Because these values are expressed in percent of RTP, they decrease in proportion to the power uprate. Further discussion of the setpoint methodology is found in this document in Section 3.4.4, "Instrument Setpoint Methodology." The specific values for the ALs are provided in Attachment 6, Section 5.3, "Technical Specification Instrument Setpoints." The AV is calculated using EGC setpoint methodology; the EGC calculation is provided in Attachment 13.

Section 2.0, Item 3 (revised OPRM armed region)

LSCS is operating under the requirements of reactor stability Long-Term Solution Option III. The Option III solution monitors OPRM signals to determine when a reactor scram is required. The OPRM system will only cause a scram when plant operation is in the Option III armed region. For TPO operation, the armed region is modified to maintain the pre-TPO absolute power (MWt) and flow, and thus the power level expressed in percent RTP decreases in proportion to the power uprate.

Section 2.0, Item 4 (changes related to instrument channel performance during testing)

A discussion of these changes is provided in Section 3.4.4, "Instrument Setpoint Methodology."

A discussion of key TS values that are unaffected is provided in Attachment 6, Section 5.3.

3.4. Additional Considerations

3.4.1 Summary of Analyses

The following is a summary of the analyses performed in support of these proposed changes, along with the results and a reference to the sections of Attachment 6 providing further detail.

Topic	Conclusion	Attachment 6 Section
Normal plant operating conditions	Uprate accommodated within previously licensed power-flow map	Section 1
Reactor core and fuel performance	All fuel and core design limits met	Section 2
Reactor coolant and connected systems	Overpressure protection, fracture toughness, structural, and piping evaluations acceptable	Section 3
Engineered safety features	Acceptable based on previous analyses at 102% of current licensed power	Section 4
Instrumentation and control	Current instrumentation acceptable; changes to some TS values; some non-safety related alarm setpoints revised	Section 5
Electrical power and auxiliary systems	Minor increases in normal power system loads; emergency power systems unaffected; auxiliary systems acceptable	Section 6
Power conversion systems	Modification to high pressure turbine; remaining power conversion systems adequate without modification	Section 7
Radwaste and radiation sources	Small increases in normal operation radiation levels and effluents; accident consequences bounded by previous evaluations.	Section 8
Reactor safety performance evaluations, including design basis events and special events	Design basis events bounded by previous evaluations, special events meet acceptance criteria	Section 9
Other evaluations	All evaluation results acceptable	Section 10

3.4.2 Adverse Flow Effects

Industry experience has revealed that power uprate conditions can cause vibrations associated with acoustic resonance that can lead to steam dryer and main steam line (MSL) valve degradation. This experience has been associated with extended power uprates (EPUs), and not with smaller uprates, such as stretch or measurement uncertainty recapture uprates.

LSCS is committed to examining the steam dryers in accordance with Boiling Water Reactor Vessel Internals Project (BWRVIP)-139, "BWR Vessel and Internals Project Steam Dryer Inspection and Flaw Evaluation Guidelines," April 2005. In addition, an evaluation was conducted to determine the potential for acoustic resonance at uprated conditions, as described in Attachment 6, Section 3.3.2, "Reactor Internals Structural Evaluation." The evaluation concluded that there are no expected effects due to acoustic resonance at uprated conditions.

3.4.3 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 will be made in accordance with the requirements of 10 CFR 50.59, "Changes, tests, and experiments," and will be implemented prior to implementation of the proposed power uprate.

3.4.4 Instrument Setpoint Methodology

As described in Section 2.0, "Detailed Description," the only proposed change to TS Limiting Safety System Setpoints is for the Flow-biased Simulated Thermal Power – Upscale scram function. Although this function is not specifically credited in the safety analysis, the associated AV provides additional margin from transient-induced fuel damage beyond that provided by the Average Power Range Monitors Fixed Neutron Flux - High function.

The AV for this function is calculated using EGC's setpoint methodology described in Nuclear Engineering Standard NES-EIC-20.04, Rev. 5, "Analysis of Instrument Channel Setpoint Error and Instrument Loop Accuracy." This methodology was approved by the NRC in Reference 12. The EGC calculation for this function is included in Attachment 13.

In accordance with Reference 1, the Flow-Biased Simulated Thermal Power – Upscale function is to be included in functions requiring TS SR controls to provide adequate assurance that instruments will actuate safety functions at the point assumed in the applicable safety analysis. Thus, the footnotes described in Section 2.0 are applied to the SR for channel calibration for this function. The nominal trip setpoints and methodology for determining the as-found and as-left tolerances are added to the TRM. TRM changes are included in Attachment 3. Plant procedures ensure that the requirements of these footnotes are implemented.

3.4.5 Grid Studies

Two grid studies have been completed to support the proposed uprate. The studies were performed using a 1,225 MWe output for each LSCS main generator. This value was chosen for the studies to bound the highest expected electrical output of the main generator under uprated conditions. Using this bounding value provides conservative results for the two studies performed.

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 1,225 MWe output for each LSCS main generator and a light load base case based on 2013 projections.

The analysis conclusions are as follows:

1. All of the primary-clearing scenarios were found to be stable.
2. All of the primary-clearing scenarios with maintenance outages considered were found to be stable.
3. Of the twenty breaker failure scenarios studied with fault detector logic, three are unstable. The study provided remediation measures for these three scenarios, involving modifications to two 345 kV breakers in the LaSalle switchyard. EGC will ensure that any modifications required by PJM are completed prior to uprate implementation. Further details regarding this study are provided in Attachment 12.

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 1,225 MWe output for each LSCS main generator. Power flow simulations were performed using 2010 transmission grid models for four system load conditions. The assessment concluded that the lowest post-contingency voltage is 356.8 kV, which remains above the minimum required switchyard voltage of 352 kV. Further details regarding this study are provided in Attachment 12.

3.4.6 Operator Training, Human Factors, and Procedures

Operator response to transients, accidents, and special events is unaffected by the proposed changes. 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.

3.4.7 Testing

Plant testing for the proposed changes will be completed as described in Attachment 6, Section 10.4, "Testing."

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 1, 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," provides criteria for the content of license amendment requests involving power uprates based on measurement uncertainty recapture.

This application is consistent with the requirements and criteria described in 10 CFR 50, Appendix K, 10 CFR 50.90, and RIS 2002-03.

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
Cooper Nuclear Station	231	June 30, 2008
Davis Besse Nuclear Power Station	278	June 30, 2008
Calvert Cliffs, Units 1 and 2	291/267	July 22, 2009
North Anna, Units 1 and 2	257/238	October 22, 2009

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-11 and NPF-18 for LaSalle County Station (LSCS), Units 1 and 2, respectively. Specifically, the proposed changes revise the Operating License and Technical Specifications (TS) to implement an increase of approximately 1.65% in RTP from 3489 megawatts thermal (MWt) to 3546 MWt. 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 also revise the TS by adding test requirements to TS instrument functions related to those variables that have a significant safety function to ensure that instruments will function as required to initiate protective systems or actuate mitigating systems at the point assumed in the applicable safety analysis.

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 reviews and evaluations performed to support the proposed uprate conditions included all components and systems that could be affected by this change. All systems will function as designed, and all performance requirements for these systems have been evaluated and were found acceptable.

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

   The nuclear steam supply systems will continue to perform their intended design functions during normal and accident conditions. The balance of plant systems and components continue to meet their applicable structural limits and will continue to perform their intended design functions. Thus, there is no increase in the probability of a failure of these components. The safety relief valves and containment isolation valves meet design sizing requirements at the uprated power level. Because the integrity of the plant will not be affected by operation at the uprated condition, EGC has concluded that all structures, systems, and components required to mitigate a transient remain capable of fulfilling their intended functions.

   A majority of the current safety analyses remain applicable, since they were performed at power levels that bound operation at a core power of 3546 MWt. Other analyses previously performed at the current power level have either been evaluated or re-performed for the increased power level. The results demonstrate that acceptance criteria of the applicable analyses continue to be met at the uprated conditions. As such, all applicable accident analyses continue to comply with the relevant event acceptance criteria. The analyses performed to assess the effects of mass and energy releases remain valid. The source terms used to assess radiological consequences have been reviewed and determined to bound operation at the uprated condition.

   The proposed changes to add test requirements to the revised TS instrument function ensure that instruments will function as required to initiate protective systems or actuate mitigating systems at the point assumed in the applicable safety analysis. Surveillance tests are not an initiator to any accident previously evaluated. As such, the probability of any accident previously evaluated is not affected. The added test requirements ensure that the systems and components required by the TS are capable of performing any mitigation function assumed in the accident analysis.

Therefore, the proposed changes 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 the proposed changes. All systems, structures, and components previously required for the mitigation of a transient remain capable of fulfilling their intended design functions. The proposed changes have no adverse effects on any safety-related system or component and do not challenge the performance or integrity of any safety-related system.

   The proposed changes to add test requirements to the revised TS instrument function do not involve a physical alteration of the plant (i.e., no new or different type of equipment will be installed, nor will there be a change in the methods governing normal plant operation). The change does not alter assumptions made in the safety analysis, but ensures that the instruments behave as assumed in the accident analysis. The proposed change is consistent with the safety analysis assumptions.

   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 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 proposed changes to add test requirements to the revised TS instrument function establish instrument performance criteria in TS that are currently required by plant procedures. The testing methods and acceptance criteria for systems, structures, and components, specified in applicable codes and standards (or alternatives approved for use by the NRC) will continue to be met as described in the plant licensing basis including the updated final safety analysis report. There is no impact to safety analysis acceptance criteria as described in the plant licensing basis because no change is made to the accident analysis assumptions.

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));

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

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

The proposed changes do not involve a significant hazards consideration. The reviews and evaluations performed to support the proposed uprate conditions concluded that all systems will function as designed, and 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 does not involve a significant reduction in a margin of safety.

There is no significant change in the types or significant increase in the amounts of any effluents. Evaluations of the effects of the proposed changes on effluent sources concluded that the increase in effluents will be small and will continue to be bounded by those described in the Final Environmental Statement for LaSalle County Station, Units 1 and 2.

There is no significant increase in individual or cumulative occupational radiation exposure. Evaluations of projected radiation exposure concluded that normal operation radiation levels increase slightly for the proposed uprate, but that occupational exposure is controlled by the plant radiation protection program and is maintained well 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.

6.0 REFERENCES

1 Letter from the Technical Specifications Task Force (TSTF), "Industry Plan to Resolve TSTF-493, 'Clarify Application of Setpoint Methodology for LSSS Functions,' " dated February 23, 2009

2 Letter from the NRC to the Technical Specifications Task Force (TSTF), "Reply to Industry Plan to Resolve TSTF-493, 'Clarify Application of Setpoint Methodology for LSSS Functions,' " dated March 9, 2009

3 NEDC 32938P-A, "Licensing Topical Report: Generic Guidelines and Evaluations for General Electric Boiling Water Reactor Thermal Power Optimization," dated May 2003

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

5 Cameron Topical Report ER-80P, "Improving Thermal Power Accuracy and Plant Safety While Increasing Operating Power Level Using the LEFM √™ System," Rev. 0, dated March 1997

6 Cameron Topical Report ER-157P, "Supplement to Topical Report ER-80P: Basis for Power Uprates with an LEFM √™ or an LEFM CheckPlus™ System," Rev. 5, dated October 2001

7 Letter from NRC to C. Lance Terry, "Comanche Peak Steam Electric Station, Units 1 and 2 - Review of Caldon Engineering Topical Report ER 80P, 'Improving Thermal Power Accuracy and Plant Safety While Increasing Power Level Using the LEFM System,' " dated March 8, 1999

8 Letter from NRC to Michael A. Krupa, "Waterford Steam Electric Station, Unit 3; River Bend Station; and Grand Gulf Nuclear Station - Review of Caldon, Inc. Engineering Report ER-157P, " dated December 20, 2001

9 ASME PTC 19.1-1998, "Test Uncertainty, Instruments and Apparatus," American Society of Mechanical Engineers, 1998

10 ISA-RP67.04.02-2000, "Methodologies for Determination of SetPoints for Nuclear Safety-Related Instrumentation," Instrumentation, Systems, and Automation Society, January 1, 2000

11 Memorandum to Mike Case, NRC, from Timothy Kobetz, NRC, "Safety Evaluation Regarding Endorsement of NEI Guidance for Adhering to the Licensed Thermal Power Limit," dated October 8, 2008

12 Letter from NRC to O. D. Kingsley, "Issuance of Amendments," dated March 30, 2001