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10 CFR 50.90
10 CFR 50, Appendix K

RS-10-002

March 25, 2010

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

Limerick Generating Station, Units 1 and 2
Facility Operating License Nos. NPF-39 and NPF-85
NRC Docket Nos. 50-352 and 50-353

Subject: Request for License Amendment Regarding Measurement Uncertainty Recapture Power Uprate

References: 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-39 and NPF-85 for Limerick Generating Station (LGS), 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 3458 megawatts thermal (MWt) to 3515 MWt.

The proposed changes are based on 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 will be installed in LGS, Unit 1 in refueling outage Li1R13, currently scheduled to complete in April 2010, and in LGS, Unit 2 in refueling outage Li2R11, currently scheduled to complete in April 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 Simulated Thermal Power – Upscale scram) that is revised in these proposed changes by adding requirements to assess channel performance during testing.

Additionally, the proposed changes include a modification to the Standby Liquid Control System (SLCS), to install a modified hand switch that allows operators to select two pumps for the automatic start function on an Anticipated Transient Without Scram (ATWS) signal. This proposed change requires NRC approval in accordance with 10 CFR 50.59, "Changes, tests, and experiments," paragraph (c)(2)(ii), in that it involves a change to the facility that may result in more than a minimal increase in the likelihood of occurrence of a malfunction of a structure, system, or component important to safety previously evaluated in the updated Final Safety Analysis Report (UFSAR). The proposed changes also revise the TS and TS Bases for the SLCS system to ensure that the assumptions in the SLCS analysis for an ATWS event are preserved.

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, TS pages, and UFSAR pages.
- Attachment 3 provides a markup of the affected Technical Requirements Manual and TS Bases. 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-33484P, "Safety Analysis Report for Limerick Generating 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-33484, "Safety Analysis Report for Limerick Generating Station, Units 1 and 2 Thermal Power Optimization," (Non-Proprietary Version).
- Attachment 9 provides Cameron documents ER-739, Rev. 1, "Bounding Uncertainty Analysis for Thermal Power Determination at Limerick Unit 1 Using the LEFM CheckPlus System," (Proprietary Version), and ER-745, Rev. 1, "Bounding Uncertainty Analysis for Thermal Power Determination at Limerick 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 LE-0113, Rev. 0, "Reactor Core Thermal Power Uncertainty Calculation Unit 1."
- Attachment 12 provides PJM Interconnection document, "Generator Transient Stability Study for Limerick Station," and PECO document, "Power Grid Voltage Analysis - Power Uprate Scenario for Limerick Generating Station."
- Attachment 13 provides EGC calculations for instrument setpoint revisions.
- Attachment 14 provides drawings describing the installation of the LEFM.

The proposed changes have been reviewed by the LGS 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 March 25, 2011. 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 Li2R11, which is currently scheduled for completion in April 2011. This implementation period will allow for installation of the LEFM instrumentation during Li2R11 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 Pennsylvania 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 25th day of March, 2010.

Respectfully,

Michael D. Jesse
Manager, Licensing - Power Uprate
Exelon Generation Company, LLC

Attachments:
1. Evaluation of Proposed Changes
2. Markup of Proposed Operating License, Technical Specifications, and UFSAR Pages
3. Markup of Proposed Technical Requirements Manual and Technical Specifications Bases

4. NRC Regulatory Issue Summary 2002-03 Cross-Reference
5. Summary of Regulatory Commitments
6. GEH Nuclear Energy Safety Analysis Report for Limerick Generating Station, Units 1 and 2 Thermal Power Optimization, NEDC-33484P (Proprietary Version)
7. GEH Nuclear Energy Affidavit Supporting Withholding
8. GEH Nuclear Energy Safety Analysis Report for Limerick Generating Station, Units 1 and 2 Thermal Power Optimization, NEDO-33484 (Non-Proprietary Version)
9. Cameron ER-739, Rev. 1, "Bounding Uncertainty Analysis for Thermal Power Determination at Limerick Unit 1 Using the LEFM CheckPlus System," (Proprietary Version), and ER-745, Rev. 1, "Bounding Uncertainty Analysis for Thermal Power Determination at Limerick Unit 2 Using the LEFM CheckPlus System," (Proprietary Version)
10. Cameron International Corporation Affidavits Supporting Withholding
11. Exelon Generation Company, LLC calculation LE-0113, Rev. 0, "Reactor Core Thermal Power Uncertainty Calculation Unit 1"
12. PJM Interconnection document, "Generator Transient Stability Study for Limerick Station," and PECO document, Power Grid Voltage Analysis - Power Uprate Scenario for Limerick Generating Station."
13. Exelon Generation Company, LLC Instrument Setpoint Calculations
14. Mechanical Drawings for Leading Edge Flowmeter Installation

cc: NRC Regional Administrator, Region I
NRC Senior Resident Inspector - Limerick Generating Station
Pennsylvania Department of Environmental Protection - Bureau of Radiation Protection

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-39 and NPF-85 for Limerick Generating Station (LGS), 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 3458 megawatts thermal (MWt) to 3515 MWt.

The proposed changes are based on 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 will be installed in LGS, Units 1 and Unit 2 prior to implementation of these requested changes.

The proposed amendment would also modify the TS and TS Bases for the applicable TS setpoint (i.e., the 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).

Additionally, the proposed changes include a modification to the Standby Liquid Control System (SLCS), to install a modified hand switch that limits the auto start function of SLCS to two pumps on an Anticipated Transient Without Scram (ATWS) signal. This proposed change requires NRC approval in accordance with 10 CFR 50.59, "Changes, tests, and experiments," paragraph (c)(2)(ii), in that it involves a change to the facility that may result in more than a minimal increase in the likelihood of occurrence of a malfunction of a structure, system, or component important to safety previously evaluated in the updated Final Safety Analysis Report (UFSAR). The proposed changes also revise the TS and TS Bases for the SLCS system to ensure that the assumptions in the SLCS analysis for an ATWS event are preserved. As described in this request, the proposed modification is consistent with NRC regulations and maintains the current assumptions, methods, and results for the ATWS event.

2.0 DETAILED DESCRIPTION

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

1. <u>Changes related to the value of RTP</u>
   Limerick Generating Station, Units 1 and 2, Facility Operating License Numbers NPF-39 and NPF-85, Sections 2.C(1), "Maximum Power Level," are revised to increase the value of RTP from 3458 MWt to 3515 MWt.

   The definition of RTP in TS Section 1.0, "Definitions," is revised to increase the value of RTP from 3458 MWt to 3515 MWt.

2. <u>Changes related to TS Table 2.2.1-1, Function 2.b, Simulated Thermal Power – Upscale</u>
   In TS Table 2.2.1-1, "Reactor Protection System Instrumentation Setpoints," Function

2.b, Simulated Thermal Power - Upscale, the trip setpoints and allowable values (AVs) are revised as follows.

Two recirculation loop operation
Trip setpoint:

Current: $\leq$ 0.66 W + 62.8% and $\leq$ 116.6% of RTP
Proposed: $\leq$ 0.65 W + 61.7% and $\leq$ 116.6% of RTP

Allowable value:

Current: $\leq$ 0.66 W + 63.3% and $\leq$ 117.0% of RTP
Proposed: $\leq$ 0.65 W + 62.2% and $\leq$ 117.0% of RTP

Single recirculation loop operation
Trip setpoint:

Current: $\leq$ 0.66 (W - 7.6%) + 62.8% and $\leq$ 116.6% of RTP
Proposed: $\leq$ 0.65 (W - 7.6%) + 61.5% and $\leq$ 116.6% of RTP

Allowable value:

Current: $\leq$ 0.66 (W - 7.6%) + 63.3% and $\leq$ 117.0% of RTP
Proposed: $\leq$ 0.65 (W - 7.6%) + 62.0% and $\leq$ 117.0% of RTP

3. Changes related to TS Table 2.2.1-1 and Table 3.3.1-1, Function 2.f, OPRM Upscale
Table 2.2.1-1 note **** and Table 3.3.1-1, note (o) are revised to require the OPRM upscale trip output be enabled with APRM simulated thermal power $\geq$ 29.5% RTP (from $\geq$ 30% RTP) and to allow the function to be bypassed when APRM simulated thermal power < 29.5% RTP.

   Table 4.3.1.1-1, "Reactor Protection System Instrumentation Surveillance Requirements," note (c) is also revised to state that calibration includes verification that the OPRM Upscale auto-enable setpoint for APRM Simulated Thermal Power is $\geq$ 29.5% (from $\geq$ 30%).

4. Changes related to TS Table 3.3.1-1, Functions 9 and 10, Turbine Stop and Control Valve Closure
Table note (j) is revised to require that this function be automatically bypassed when turbine first stage pressure is equivalent to a thermal power of less than 29.5% RTP (from < 30% RTP).

5. Changes related to TS 3.3.4.2, "End-of-Cycle Recirculation Pump Trip System Instrumentation"
The 30% RTP value for Applicability is revised to 29.5 % RTP.

   In Table 3.3.4.2-1, "End-of-Cycle Recirculation Pump Trip System Instrumentation," the 30% RTP value in footnote ** is revised to 29.5%.

6. Changes related to TS 3.3.6, "Control Rod Block Instrumentation"
In Table 3.3.6-2, "Control rod Block Instrumentation Setpoints," the trip setpoints and AVs for Function 2.a, APRM Simulated Thermal Power Upscale are revised as follows.

   Trip setpoint – two recirculation loop operation
   Current: $\leq$ 0.66 W + 55.2% and $\leq$ 108.0% of RTP
   Proposed: $\leq$ 0.65 W + 54.3% and $\leq$ 108.0% of RTP

<u>Allowable value – two recirculation loop operation</u>
Current: $\leq$ 0.66 W + 55.7 % and $\leq$ 108.4% of RTP
Proposed: $\leq$ 0.65 W + 54.7% and $\leq$ 108.4% of RTP

<u>Trip setpoint – single recirculation loop operation</u>
Current: $\leq$ 0.66 (W – 7.6%) + 55.2% and $\leq$ 108.0% of RTP
Proposed: $\leq$ 0.65 (W – 7.6%) + 54.1% and $\leq$ 108.0% of RTP

<u>Allowable value – single recirculation loop operation</u>
Current: $\leq$ 0.66 (W – 7.6%) + 55.7 % and $\leq$ 108.4% of RTP
Proposed: $\leq$ 0.65 (W – 7.6%) + 54.5% and $\leq$ 108.4% of RTP

7. <u>Changes related to TS 3.4.1.1, "Recirculation Loops"</u>
   Action a.1.b is revised to reduce thermal power to $\leq$ 74.9% of RTP (from 76.2%)

   SR 4.4.1.1.4.a is revised to verify that thermal power is $\leq$ 74.9% of RTP (from 76.2%)

8. <u>Changes related to instrument channel performance during testing</u>
   In TS Table 4.3.1.1-1, for Function 2.b, the following notes are added to the channel calibration.

   (o) 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.

   (p) The instrument channel setpoint shall be reset to a value that is within the as-left tolerance around the Trip Setpoint at the completion of the surveillance; otherwise, the channel shall be declared inoperable. Setpoints more conservative than the Trip Setpoint 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 methodologies used to determine the as-found and the as-left tolerances are specified in the associated Technical Specifications Bases.

9. <u>Changes related to the SLCS</u>

   The proposed changes include a modification to install a modified hand switch that limits the auto start function of SLCS to two pumps on an Anticipated Transient Without Scram (ATWS) signal. The modified hand switch will allow the operators to inhibit the auto-start ATWS signal to the C SLCS pump, which is initiated by Redundant Reactivity Control System (RRCS). This will limit the auto-start function of SLCS to two pumps (i.e., the A and B pumps) during an ATWS event, with the C pump available for manual start if required. This change reduces the SLCS pressure at the SLCS relief valve during a postulated ATWS event with main steam isolation valve closure, thus increasing the margin between the SLCS pressure and the relief valve setpoint. Redundancy is also maintained by this change. If either the A or B SLCS pump is found to be inoperable or taken out of service, then the C SLCS pump can be aligned for automatic start, simply by repositioning the C SLCS pump control switch from the "stop" to the "norm" position. The proposed UFSAR markup is provided in Attachment 2.

This proposed change also modifies TS Section 3/4.1.5, "Standby Liquid Control System," as follows:

> Limiting Condition for Operation (LCO) 3.1.5 is revised to remove the phrase "a minimum of" from the LCO.

Proposed changes to the Technical Requirements Manual (TRM) and TS Bases 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. New TRM Section 3.3.7.10, "Feedwater Flow Instrumentation," is added to specify the requirements for an inoperable LEFM system.

<u>TS Bases Changes</u>

1. The Bases for Section 2.2.1, "Reactor Protection System Instrumentation," are revised to incorporate the revised AV for the Simulated Thermal Power - Upscale and the revised value at which the OPRM Upscale function is enabled.
2. The Bases for Section 3/4.3.1, "Reactor Protection System Instrumentation," are revised to incorporate the revised value at which the OPRM Upscale function is enabled.
3. The Bases for Section 3/4.3.1 are also revised to incorporate discussion of the footnotes regarding evaluation of instrument channel performance during testing.
4. The Bases for Section 3/4.3.4, "Recirculation Pump Trip Actuation Instrumentation," are revised to incorporate the revised value at which the End-of-Cycle Recirculation Pump Trip function is enabled.
5. The Bases for Section 3/4.1.5, "Standby Liquid Control System," are revised to state that the SLCS system is inoperable if more than two SLCS pumps are aligned for automatic operation and to clarify the applicable action statement for this condition.

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 LGS, 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 Flow Measurement and Core Thermal Power and Uncertainty," with the utilization of the LEFM system, the core thermal power measurement uncertainty will be a maximum of 0.347%.

As summarized in Section 3.4.1, "Summary of Analyses," below 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 1.653% (i.e., 2% - 0.347%), based on the reduction in thermal power uncertainty. This increase in RTP corresponds to 3515.1 MWt, which is rounded down to the requested 3515 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 summarized in Section 3.4.1, "Summary of Analyses," and 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 Simulated Thermal Power – Upscale scram) related to the power uprate. The change now formalizes 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 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 Topical Report ER-80P: Basis for Power Uprates with an LEFM √™ or LEFM CheckPlus™ 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 Caldon 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 will be installed in the feedwater piping of the three feedwater loops as shown in the attached drawings. The installations on Loop B and Loop C will be located in straight sections of pipe downstream of existing flow straighteners and upstream of existing FW flow nozzles. The installation on Loop A will be upstream of both the flow straightener and the FW flow nozzle. Pre-installation drawings are provided in Attachment 14.

The transducers will be located in the Turbine Enclosure above the number six feedwater heater rooms in an anticipated radiation field of 20 mR/hr at full power. The electronics cabinet will be in the corridor outside the Turbine Enclosure number six feedwater heater rooms in an anticipated radiation field of less than 1 mR/hr at full power. No radiation damage or degradation to the instruments (including electronics) due to such exposure is anticipated.

Modification packages have been developed outlining the steps to install and test the LEFM system on each unit. 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 LGS, Units 1 and 2. The Unit 1 and Unit 2 results indicate a feedwater mass flow rate uncertainty of ±0.28% with a fully functional LEFM system. This uncertainty was calculated using the methodology described in Reference 6, which was approved by the NRC in Reference 8.

To bound the FW mass flow rate uncertainty, EGC has used a FW mass flow rate uncertainty of ±0.32% for both units. Based on a FW mass flow rate uncertainty of ±0.32%, EGC has completed the thermal power uncertainty calculation for LGS, which results in a total uncertainty of ±0.347% in the calculation of RTP for the site-specific installation. This calculation is provided in Attachment 11. The calculation methodology is consistent with the Limerick 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.

Criterion 1

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.

Response to Criterion 1

Calibration and Maintenance

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 3458 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 LGS). 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-testing. A continuously operating on-line test is provided to verify that the digital circuits are operating correctly and within the specified accuracy range. System malfunctions will result in main control room alarms.

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

A process will be implemented to use the LEFM system feedwater mass flow and temperature to adjust or calibrate the existing feedwater flow nozzle-based signals. If the

LEFM system or a portion of the system becomes inoperable, control room operators will be promptly alerted by a control room alarm. Feedwater flow input to the core thermal power calculation would then be provided by the existing flow nozzles.

Since the feedwater flow nozzles will be calibrated to the last validated data from the LEFM system, it will be acceptable to remain at the uprated RTP of 3515 MWt for up to 72 hours to enact LEFM system repairs. As noted in the TRM changes provided, 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 3458 MWt.

The 72-hour allowed outage time (AOT) for the LEFM system prior to reducing to the CLTP is acceptable. As discussed above, during the 72-hour AOT, the existing feedwater flow nozzle-based signals will be calibrated to the last validated data from the LEFM system. 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 overestimation of the calculated calorimetric power level, which is conservative, as the reactor will actually be operating below the calculated power level. A sudden de-fouling event during the 72-hour inoperability period is unlikely. Significant sudden defouling would be detected by a change in the secondary plant parameters. 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. In addition, 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. Note that the NRC has previously approved power uprate applications with AOTs of up to 72 hours.

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.

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, a procedure currently exists for reactor engineering personnel to manually calculate core thermal power. The reactor power can also be estimated from multiple parameters (e.g., APRMs, steam flow, feed flow, turbine first stage pressure, main generator output).

<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.

Response to Criterion 2

This criterion is not applicable to LGS.

Criterion 3

Confirm that the methodology used to calculate the uncertainty of the LEFM in comparison to the current feedwater instrumentation is based on the 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-RP67.04.02-2000 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 feedwater 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 with flow elements 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 LGS, Units 1 and 2. The calibration factors for the LGS, Units 1 and 2 spool pieces were established by tests of these spools at Alden Research Laboratory. These tests were performed on a full-scale model of the LGS 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 LGS, Units 1 and 2. The test configurations model the portion of piping upstream of the LEFM spool pieces. There is a possibility that the LEFMs will need to be rotated from their tested configuration during installation to allow for transducer replacement. The Alden tests included testing the spool piece in different axial configurations to address the uncertainty associated with field installation. Therefore, the flow measurement uncertainty for the LEFM spool pieces accounts for the potential need to rotate the spool pieces during installation.

A discussion of the impact of other plant-specific installation factors on the feedwater flow measurement uncertainty is provided in Attachment 9, in appendices to ER-739 and ER-745. Appendix A.3 to ER-739 contains Cameron ER-789, "LEFM CheckPlus Meter Factor Calibration and Accuracy Assessment for Limerick Unit 1 Nuclear Power Station," and Appendix A.3 to ER-745 contains ER-797, "Meter Factor Calculation and Accuracy Assessment for Limerick 2." Sections 2.2, 4.2, and 4.4 of ER-789 and ER-797 provide responses to many of the previous NRC requests for additional information from NRC-approved applications listed in Section 4.2, "Precedent." The tested configuration of the LEFM spool pieces can be compared to the plant installation drawings by comparing the drawings in ER-789 and ER-797, Figure 3, to the pre-installation drawings in Attachment 14.

For LGS, Units 1 and 2, final acceptance of the site-specific uncertainty analyses will occur after the completion of the commissioning process and prior to the implementation of these proposed changes.

3.2.5 Deficiencies and Corrective Actions

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

Problems with plant instrumentation identified by LGS personnel are also documented in the LGS 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

Limerick procedure GP-5, "Steady State Operations," provides guidance to ensure that reactor power remains within the requirements of the operating license. Procedure Section 3.0, Notes 5 and 6, provide guidance for monitoring and controlling reactor power that 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.

<u>Section 2.0, Item 1 (change in RTP)</u>

The proposed increase of approximately 1.65% in RTP 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.

<u>Section 2.0, Item 2 (revised values for Simulated Thermal Power - Upscale function)</u>

The proposed change to the nominal trip setpoints and Allowable Values (AVs) for the Simulated Thermal Power - Upscale function are based on the approach described in Reference 3, Section F.4.2.1, "Flow Referenced APRM Trip and Alarm Setpoints." The 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. The specific values for the ALs are provided in Attachment 6, Section 5.3, "Technical Specification Instrument Setpoints." The AVs are calculated using LGS setpoint methodology; the AV calculations are provided in Attachment 13. Further discussion of the setpoint methodology is found in this document in Section 3.4.4, "Instrument Setpoint Methodology."

<u>Section 2.0, Item 3 (revised OPRM armed region)</u>

LGS 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. Based on the approach described in Reference 3, Section 5.3.4, "Thermal-Hydraulic/Neutronic Stability," the Option III armed region is rescaled to maintain the same absolute power/flow region boundaries.

<u>Section 2.0, Item 4 (revised turbine scram bypass level)</u>

Based on the guidelines in Reference 3, Section F.4.2.3, "Turbine First-Stage Pressure Signal Setpoint," the value at which the turbine stop valve closure scram and turbine control valve fast closure scram are bypassed, in percent of RTP, is reduced by the ratio of the power increase. The value does not change with respect to absolute thermal power.

<u>Section 2.0, Item 5 (revised end-of cycle-recirculation pump trip bypass level)</u>

Based on the guidelines in Reference 3, Section F.4.2.3, "Turbine First-Stage Pressure Signal Setpoint," the value at which the end-of-cycle recirculation pump trip is bypassed, in percent of RTP, is reduced by the ratio of the power increase. The value does not change with respect to absolute thermal power.

<u>Section 2.0, Item 6 (revised values for rod blocks)</u>

The proposed change to the nominal trip setpoints and AVs for the Simulated Thermal Power control rod blocks are based on the same rationale as discussed in Item 2 above for the simulated thermal power scram function. The nominal trip setpoints 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.

<u>Section 2.0, Item 7 (revised limit for single-loop operation)</u>

The proposed change to the power limitation for single-loop operation is based on the approach described in Reference 3, Section 5.2, "Power/Flow Map." The limiting value is unchanged in units of absolute core thermal power. Because this value is expressed in percent of RTP, it decreases in proportion to the power uprate.

<u>Section 2.0, Item 8 (changes related to instrument channel performance during testing)</u>

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

<u>Section 2.0, Item 9 (SLCS modification)</u>

See Section 3.5, "Evaluation of Standby Liquid Control System Modification."

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 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	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	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.

LGS 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 showed that there is no expected increase in normalized root mean square pressure in the main steam lines as flow conditions are changed from current rated thermal power to uprated power.

3.4.3 Plant Modifications

With the exception of the SLCS modification described in Section 3.5, "Evaluation of Standby Liquid Control System Modification," the evaluations performed to support the power uprate identified that modifications are required to certain non-safety related systems, including minor equipment changes, replacements, and setpoint or alarm point changes. These modifications 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 Simulated Thermal Power - Upscale function. The nominal trip setpoints and AVs for this function are calculated using LGS setpoint methodology described in procedure CC-MA-103-2001, "Setpoint Methodology for Peach Bottom Atomic Power Station and Limerick Generating Station." This methodology is consistent with the methodology described in NEDC-31336P-A, "General Electric Instrument Setpoint Methodology," September 1996. The EGC setpoint calculation for the Simulated Thermal Power - Upscale function is included in Attachment 13.

In accordance with Reference 1, the 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. Discussion of the notes and the methodology for determining the as-found and as-left tolerances is added to the TS bases associated with this function. The associated TS bases 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.

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 PECO and neighboring control areas. The analysis assumed a 1,245 MWe output for each LGS main generator and a light load base case based on 2013 projections. The 1,245 MWe output bounds the expected output of each main generator under uprated conditions. 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. All of the breaker failure scenarios considered in the study were found to be stable. Further details regarding this study are provided in Attachment 12.

PECO Transmission Planning completed a study to determine if the capacity and capability of the preferred power supply ensures the design and licensing basis for the Limerick Generating Station under uprated conditions. Adequacy of the preferred power supply is determined by verification of the transmission system's capability to maintain the post-trip voltage drops and voltages at the safety buses to remain above the reset value of the degraded voltage relay on a steady-state basis. The study assumed a 1,240 MWe output for each LGS main generator, which is the expected maximum output of the main generators, as well as maximum MVAR output, for both summer and winter conditions. Power flow simulations were performed using 2010 transmission grid models. Two independent offsite sources are required to be operable in accordance with Limerick TS 3.8.1.1, "AC Sources-Operating." The two primary offsite sources are the #10 Startup Transformer and the #20 Startup Transformer. The alternate third offsite source is the 8A/8B transformer. The study demonstrates that the post-trip voltage drops are within the limits established in site procedures to maintain operability. The study also shows that the transmission system is capable of providing adequate voltage as required for operability of the offsite sources. The study results are compared to the acceptance criteria in the table below, with values expressed in per unit (pu) for voltage drops, and values expressed in voltage units and pu for minimum voltages. Further details regarding this study are provided in Attachment 12.

Off-Site Source	Post-Trip Voltage Drop Administrative Limit (acceptance criteria)	MUR Study Post-Trip Voltage Drop	Minimum Evaluated Post-Trip Supply Voltage (acceptance criteria)	MUR Study Post-Trip Minimum Supply Voltage
#10 (230kV)	0.025pu	0.0150pu	219.2kV/0.953pu	227.7kV/0.990pu
#20 (13.2kV)	0.025pu	0.0102pu	12.38kV/0.938pu	13.1kV/0.9917pu
8A/8B (69kV)	0.029pu	0.0180pu	65.5kV/0.949pu	66.9kV/0.9706pu

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."

3.5. Evaluation of Standby Liquid Control System Modification

3.5.1 Background and Description

The LGS SLCS is designed to be capable of shutting the reactor down from full power to cold shutdown and maintaining the reactor in a subcritical state at atmospheric temperature and pressure conditions by pumping sodium pentaborate, a neutron absorber, into the reactor. The boron injection capability of the SLCS meets the requirements of 10 CFR 50.62, "Requirements for reduction of risk from anticipated transients without scram (ATWS) events for light-water-cooled nuclear power plants." The system is also capable of maintaining suppression pool pH at a level of 7.0 or greater following a loss of coolant accident (LOCA).

The SLCS is a backup method of shutting down the reactor to cold subcritical conditions by independent means other than the normal method using the control rods. Thus, the system is considered a safe shutdown system. Although this system has been designed to achieve a high degree of reliability with many safety system features, it is not required to meet single failure criteria, as noted in UFSAR Section 7.4.2.2.2, "SLCS Specific Regulatory Requirements Conformance."

The LGS SLCS system consists of three independent loops, including pumps, discharge valves, and piping. Each positive displacement pump is sized to inject sodium pentaborate solution into the reactor at 43 gpm. The pump and system design pressure between the explosive valves and the pump discharge is 1400 psig. The set pressure for the three relief valves is set at 1400 psig, with a 1% setpoint tolerance.

The SLCS is manually initiated from the control room, when the operator determines that normal reactivity control systems have not shutdown the reactor as required, by turning the key-lock switch to the run position for Loop A, B, or C. All three loops of the SLCS are also automatically initiated by the RRCS after a time delay, in response to an ATWS. This automatic initiation overrides the manual initiation signal; however, stopping the pump manually at the hand switch overrides the automatic initiation signals.

Two SLCS pumps are required to meet the requirements of 10 CFR 50.62. If a component in one of the three redundant trains is found to be inoperable, there is no threat to shutdown capability, and reactor operation may continue during repairs. TS 3.1.5, "Standby Liquid Control System," requires that a minimum of two SLCS pumps and corresponding flow paths be operable in Operational Conditions 1 and 2, and allows a seven-day period during which one of the two required pumps can be inoperable.

In preparation for the proposed power uprate, a plant-specific ATWS analysis was performed for LGS at the uprated power level to determine the effects of the expected increase in reactor pressure during the ATWS event on the SLCS system. The results of that analysis for the main steam isolation valve (MSIV) closure ATWS event indicated that, at the uprated power level, with all three pumps receiving an auto start signal and operating concurrently, using conservative values for relief valve setting and pump pressure pulsations, the SLCS discharge pressure will exceed the SLCS relief valve setting. This would divert flow through the relief valve to the pump suction, instead of to the reactor vessel, thus potentially impacting the boron injection capability of the system.

The proposed change will install a modified hand switch for the C SLCS pump, which will allow the operators to inhibit the auto-start ATWS signal to the C SLCS pump. The current hand switch has a stop position; however, this switch is presently spring return to "norm" (center position) from either the "run" or "stop" positions. By modifying the C hand switch to maintain the switch in the "stop" position, the operator can selectively inhibit the auto-start signal to the C SLCS pump. This will limit the auto-start function of SLCS to two pumps (i.e., the A and B pumps) during an ATWS event, with the C pump available for manual start if required. This change reduces the SLCS pressure at the SLCS relief valve during a postulated ATWS event, to ensure that injection assumptions of the ATWS analysis are met by establishing margin between the SLCS pressure and the relief valve setpoint.

The proposed revision to TS Section 3/4.1.5, "Standby Liquid Control System," modifies the Limiting Condition for Operation (LCO) for Operational Conditions 1 and 2 to remove the phrase "a minimum of" from the LCO. The proposed modification to the TS Bases discusses that no more than two pumps shall be aligned for automatic operation in order to ensure that the SLCS relief does not lift following an ATWS event. The Bases further state that if three pumps are aligned for automatic operation, the system is inoperable and that Action statement "b" applies.

3.5.2 Technical Evaluation

The proposed modification preserves all of the assumptions of the applicable safety analyses related to the ATWS event. The A and B SLCS pumps are capable of providing the required flow at the required pressure assumed in the ATWS analysis. The ATWS analysis for the proposed power uprate was performed assuming that only two pumps are available, as discussed in Attachment 6, Section 9.3.1, "Anticipated Transients Without Scram." A single failure does not need to be postulated for the ATWS event. However, if a single failure does occur during a postulated ATWS event, the operators can simply reposition the C SLCS pump hand switch to the "norm" position, and the pump will auto start if the ATWS signal is still present, or the pump can be manually started.

With two pumps operating, the calculated margin between the SLCS relief valve and the SLCS system pressure during a limiting ATWS event is increased as shown in the table below. As noted in the table, with two pumps operating, there is a minimum of 19.7 psi margin beyond that required to account for relief valve tolerance, piping line loss, elevation differences, and pump pressure pulsations. This provides adequate margin to ensure that the relief valve does not lift during postulated ATWS events.

Item	Parameter	Limerick Unit 1	Limerick Unit 2
1	SLCS relief valve setpoint (psig)	1400	1400
2	Relief valve tolerance (% / psi)	1.0% / 14 psi	1.0% / 14 psi
3	Pulsating margin (psi)	30.0	30.0
4	SLCS piping loss (psi) – 3 pump operation	159.3	145.4
5	SLCS piping loss (psi) – 2 pump operation	112.3	106.2
6	Pressure at point of SLCS injection (psig)	1224	1224
7	Pump discharge pressure (psig) – 3 pump operation (sum of items 4 and 6)	1383.3	1369.4
8	Pump discharge pressure (psig) – 2 pump operation (sum of items 5 and 6)	1336.3	1330.2
9	Margin to relief valve setting – 3 pump operation (psi) [item 1 - (sum of items 2, 3, 7)]	None	None
10	Margin to relief valve setting – 2 pump operation (psi) [item 1 - (sum of items 2, 3, 8)]	19.7	25.8

The control and auto-start circuitry for the A and B SLCS pumps is not affected by this design change.

Redundancy is also maintained by this change. If either the A or B SLCS pump is found to be inoperable or taken out of service, then the C SLCS pump can be aligned for automatic start, simply by repositioning the C SLCS pump control switch from the "stop" to the "norm" position.

The post-LOCA function of the SLCS pumps to maintain suppression pool pH is unaffected by this proposed modification, since the SLCS pumps are manually started for the post-LOCA function.

The start of any SLCS pump also causes a valve in the Reactor Water Cleanup (RWCU) system to close, stopping flow to prevent that system from removing sodium pentaborate from the reactor coolant. The start of the A SLCS pump closes the RWCU inlet inboard isolation valve; the start of the B SLCS pump closes the RWCU inlet outboard isolation valve; and the start of the C SLCS pump closes the RWCU inlet inboard isolation valve. The modification to the C SLCS pump control switch does not affect this capability. Therefore, this modification does not reduce the effectiveness of SLCS.

Regarding human factors, the modified switch will maintain the same form, fit, and function as the existing switch with the exception that the "stop" position will not spring return to "norm" position, but will be maintained in the "stop" position, when required. Administrative controls will be in place to ensure that the modified switch is maintained in the stop position.

The revisions to TS Section 3.1.5 and the associated Bases ensure that the assumptions in the SLCS analysis for an ATWS event are preserved, by ensuring only two pumps are aligned for automatic operation. The removal of the phrase “a minimum of” from TS LCO 3.1.5, combined with the explanation in the Bases that only two pumps shall be aligned for automatic operation provides adequate guidance. Changes to the Bases are controlled in accordance with a formal TS Bases control program in accordance with TS 6.8, "Procedures and Programs."

For the condition in which three pumps are aligned for automatic operation, action statement "b" applies, and requires that the system be restored to operable status within eight hours or be in at least hot shutdown within the following twelve hours. This allowed outage time provides a minimal amount of time to correct the condition before requiring a plant shutdown.

This modification is scheduled to be installed in LGS, Unit 1 during Li1R13, which is scheduled for completion in April 2010, and in LGS, Unit 2 during Li2R11, scheduled for completion in April 2011. After the modification is installed, the switch position for the modified switch will be administratively controlled in the "norm" position, thus preserving the current design until these proposed changes are approved. The modifications will be implemented prior to power uprate implementation.

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.

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 Recapture 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 the guidelines of 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-39 and NPF-85 for Limerick Generating Station (LGS), 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 3458 megawatts thermal (MWt) to 3515 MWt. These changes are based on 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.

The proposed changes also modify the standby liquid control system (SLCS) controls and the TS for the SLCS system to allow automatic start of only two of the three redundant SLCS pumps. This change preserves the analysis assumptions for the SLCS system during an anticipated transient without scram event under uprated conditions.

The proposed changes also revise the TS by adding test requirements to the TS instrument function affected by the power uprate to ensure that the instrument function will actuate as required to initiate protective 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 does not involve a significant hazard 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 hazard. The following information is provided to support a finding of no significant hazard.

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

   Response: No

   The proposed changes do not affect system design or operation and thus do not create any new accident initiators or increase the probability of an accident previously evaluated. All accident mitigation systems will function as designed, and all performance requirements for these systems have been evaluated and were found acceptable. The SLCS performance requirements will be met with completion of the SLCS modification described in the proposed changes.

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 3515 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. The anticipated transient without scram event criteria will be met with completion of the SLCS modification described in the proposed changes. 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 add 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. Surveillance tests are not an initiator to any accident previously evaluated. As such, the probability of any accident previously evaluated is not significantly increased. 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.

The SLCS modification does not affect the probability of an accident, as the control system is not an initiator in any accident. The modification maintains all of the assumptions in the analyses of events for which the system is designed. Thus, the response to these events is unaffected.

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 regarding instrument testing 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.

   The SLCS system is not an initiator of any accidents.

   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 add test requirements that 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.

The SLCS modification maintains all of the assumptions in the analyses of events for which the system is designed. Thus, the response to these events is unaffected.

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 Limerick Generating 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. Caldon Topical Report ER-80P, “Improving Thermal Power Accuracy and Plant Safety While Increasing Operating Power Level Using the LEFM √TM System,” Rev. 0, dated March 1997

6. Caldon Topical Report ER-157P, “Supplement to Caldon Topical Report ER-80P: Basis for Power Uprates with an LEFM √ TM or an LEFM CheckPlus TM 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