Exelon Generation
4300 Winfield Road
Warrenville, IL 60555

www.exeloncorp.com

Exelon.
Nuclear

10 CFR 50.90

April 4, 2006

RS-06-044

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

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 to Technical Specification 5.6.5, "Core Operating Limits Report (COLR)."

References:

(1) Letter from K. R. Jury (Exelon Generation Company, LLC) to USNRC, "Request for Amendment to Technical Specifications Section 5.6.5, 'Core Operating Limits Report (COLR)' ", dated March 7, 2005

(2) Letter from S. P. Sands (USNRC) to C. M. Crane (Exelon Generation Company, LLC), LaSalle County Station, Units 1 and 2, "Issuance of Amendments Re: Technical Specifications Section 5.6.5, 'Core Operating Limits Report (COLR)' ", dated February 1, 2006

(3) Letter from H. N. Berkow (USNRC) to R. L. Gardner (Framatome), "Final Safety Evaluation for Framatome ANP Topical Report ANF-1358(P), Revision 3, 'The Loss of Feedwater Heating Transient in Boiling Water Reactors' ", dated September 20, 2005

Pursuant to 10 CFR 50.90, "Application for amendment of license or construction permit," Exelon Generation Company, LLC, (EGC), hereby requests an amendment to Appendix A, Technical Specifications (TS), of Facility Operating License Nos. NPF-11 and NPF-18 for LaSalle County Station (LSCS), Units 1 and 2. The proposed changes will add one NRC-approved topical report reference to the list of analytical methods in TS 5.6.5, "Core Operating Limits Report (COLR)," and will delete seven obsolete references from the same TS section.

In Reference 1, EGC submitted a license amendment request (LAR) to add two NRC-approved topical report references that provided: 1) a determination of fuel assembly critical power for previously loaded Global Nuclear Fuel (GNF) GE14 fuel, co-resident with reloaded Framatome ANP (FRA-ANP) ATRIUM-10 fuel; and 2) a new FRA-ANP EXEM BWR-2000 LOCA Methodology. The LAR was submitted in support of the re-introduction of ATRIUM-10 fuel in Unit 1 at LSCS. The NRC approved this LAR in Reference 2. This submittal supports the continued irradiation of ATRIUM-10 in the LSCS reactors and the use of an NRC-approved analytical methodology for evaluating loss of feedwater heater (LOFWH) transients.

The added reference, Topical Report ANF-1358(P)(A), Revision 3, "The Loss of Feedwater Heating Transient in Boiling Water Reactors," will list a FRA-ANP method for evaluating the LOFWH transient. This topical report has been previously reviewed and approved by the NRC for use by licensees in Reference 3.

The seven deleted references describe previously approved Global Nuclear Fuel (GNF) and FRA-ANP methodologies for the analyses of ATRIUM-9B and GE9 fuel. Both of these fuel types have been or will be completely discharged from both LSCS reactors after the loading of ATRIUM-10 fuel during the LSCS Unit 2 refuel outage currently scheduled to begin in February 2007 (i.e., L2R11).

This request is subdivided as follows:

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

Attachment 2 provides a mark-up of TS pages with the proposed changes indicated.

Attachment 3 provides the typed TS pages with the proposed changes incorporated.

EGC requests approval of the proposed change by February 26, 2007 to support restart following L2R11. Once approved, the amendment will be implemented within 30 days.

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.

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

Should you have any questions or require additional information, please contact Mr. John L. Schrage at (630) 657-2821.

I declare under penalty of perjury that the foregoing is true and correct. Executed on the 4th day of April 2006.

Respectfully,

Keith R. Jury

Keith R. Jury
Director, Licensing and Regulatory Affairs

Attachment 1: Evaluation of Proposed Changes

Attachment 2: Mark-up of Proposed Technical Specifications Page Changes

Attachment 3: Typed Pages for Technical Specifications Page Changes

ATTACHMENT 1
EVALUATION OF PROPOSED CHANGES

INDEX

1.0 DESCRIPTION

Pursuant to 10 CFR 50.90, Exelon Generation Company, LLC, (EGC), requests the following amendment to Appendix A, Technical Specifications (TS), of Facility Operating License Nos. NPF-11 and NPF-18 for LaSalle County Station (LSCS), Units 1 and 2. Specifically, the proposed changes will add an NRC-approved topical report reference to the list of analytical methods in TS 5.6.5, "Core Operating Limits Report (COLR)," that may be used to determine core operating limits, and delete seven obsolete references from the same TS section.

The proposed changes will:

1. add an NRC-approved Framatome Advanced Nuclear Power, Inc. (FRA-ANP) topical report reference for an updated methodology for evaluating loss of feedwater heating (LOFWH) transients; and

2. delete seven references that were used in the analyses of ATRIUM-9B and GE9 fuel. These fuel types will no longer be irradiated in the LSCS reactors following the upcoming Unit 2 refueling outage, which is currently scheduled to begin February 2007 (i.e., L2R11).

In Reference 1, EGC submitted a license amendment request (LAR) to add two NRC-approved topical report references that provided: 1) a determination of fuel assembly critical power for previously loaded Global Nuclear Fuel (GNF) GE14 fuel, co-resident with reloaded Framatome ANP (FRA-ANP) ATRIUM-10 fuel; and 2) a new FRA-ANP EXEM BWR-2000 LOCA Methodology. The LAR was submitted in support of the re-introduction of ATRIUM-10 fuel in Unit 1 at LSCS. The NRC approved this LAR in Reference 2. The current submittal supports the continued irradiation of ATRIUM-10 in the LSCS reactors and the use of an NRC-approved analytical methodology for evaluating LOFWH transients.

ATRIUM-10 fuel is manufactured and licensed by FRA-ANP, a subsidiary of the AREVA group. This organization has previously been known as Siemens Power Corporation (SPC). Framatome ANP will analyze the LSCS Unit 2 Cycle 12 core reload.

2.0 PROPOSED CHANGES

The added reference to TS 5.6.5 will list a FRA-ANP method for analyzing the LOFWH transient for various mixtures of Global Nuclear Fuels (GNF) and FRA-ANP fuel assemblies (i.e., ANF-1358(P)(A), Revision 3, "The Loss of Feedwater Heating Transient in Boiling Water Reactors"). This topical report has been previously reviewed and approved by the NRC for use by licensees in Reference 3.

The proposed changes will modify TS 5.6.5 as follows:

1. The current reference numbers 1, 2, 8, 12, 17, 18 and 21 are deleted.

2. The remaining references are renumbered from 1 to 17.

3. Topical Report ANF-1358(P)(A), Revision 3, "The Loss of Feedwater Heating Transient in Boiling Water Reactors," is added as a new Reference 18.

The marked-up and retyped TS pages are provided in Attachments 2 and 3.

3.0 BACKGROUND

The analytical methods currently listed in TS 5.6.5 support the determination of core operating limits for both units by using the GNF or FRA-ANP methodology. LSCS Unit 1 currently uses a mixture of ATRIUM-10 and GE14 fuel in the core. The determination of GE14 and ATRIUM-10 fuel assembly critical power for the current operating cycle is determined with a FRA-ANP critical power correlation. Reference 2 provides the NRC approval for the use of the FRA-ANP critical power correlation for the GE14 fuel. Overall core operating limits for LSCS Unit 1 are determined using a FRA-ANP methodology, including a cycle-specific FRA-ANP evaluation of LOFWH transients.

LSCS Unit 2 currently uses a mixture of ATRIUM-9B, ATRIUM-10 and GE-14 fuel. The determination of fuel assembly critical power for the mixture of ATRIUM-9B, ATRIUM-10 and GE14 fuel is determined with a GNF critical power correlation, and overall core operating limits are determined using GNF methodology.

EGC will continue to load FRA-ANP ATRIUM-10 fuel and completely discharge ATRIUM-9B during the upcoming Unit 2 Refueling Outage (L2R11) currently scheduled to begin February 2007. All GE9 fuel was completely discharged during a previous refueling outage. LSCS will use the most recent FRA-ANP methodologies to determine overall core operating limits for future core configurations, vice the current GNF methodology. This change will require the listing of an additional analytical methodology for evaluating LOFWH conditions with the FRA-ANP methodology. Thus, the proposed change will allow LSCS to use FRA-ANP's most recent LOFWH analytical method to determine core operating limits for LSCS Unit 1 and Unit 2.

4.0 TECHNICAL ANALYSIS

TS 5.6.5 requires that a COLR be established and that the analytical methods used to determine the core operating limits be those previously reviewed and approved by the NRC. The approved analytical methods are listed in TS 5.6.5.b. The analytical methods listed in this TS section support operation of certain types of fuel contained in the reactor core, and list the analytical codes that are used to calculate operating parameters. These analytical codes are utilized to predict core behavior under normal and accident conditions. The proposed addition of the FRA-ANP topical report will incorporate the analytical code necessary to use the most recent FRA-ANP methodologies to determine overall core operating limits for future core configurations.

The NRC, in Reference 3, approved topical report ANF-1358(P), Revision 3, "The Loss of Feedwater Heating Transient in Boiling Water Reactors." This FRA-ANP topical report describes revisions made to the methodology for evaluation of LOFWH conditions. LSCS has determined that the use of the FRA-ANP topical report, described in Reference 3, and the LOFWH evaluation methodology contained in this report is appropriate for LSCS Units 1 and 2, and provides an equivalent level of protection as that currently provided.

The seven references that will be deleted from TS 5.6.5 describe previously approved Global Nuclear Fuel (GNF) and FRA-ANP methodologies for the analyses of ATRIUM-9B and GE9 fuel. Both of these fuel types have been or will be completely discharged from both LSCS reactors after the loading of ATRIUM-10 fuel during the LSCS Unit 2 refuel outage currently scheduled to begin in February 2007 (i.e., L2R11). Thus, these references are obsolete and unnecessary.

5.0 REGULATORY ANALYSIS

5.1 No Significant Hazards Consideration

Pursuant to 10 CFR 50.90, Exelon Generation Company, LLC, (EGC), requests an amendment to Appendix A, Technical Specifications (TS), of Facility Operating License Nos. NPF-11 and NPF-18 for LaSalle County Station (LSCS), Units 1 and 2. The proposed amendment will add an NRC-approved topical report reference to the list of analytical methods in TS 5.6.5, "Core Operating Limits Report (COLR)," that may be used to determine core operating limits, and delete seven obsolete references from the same TS section.

According to 10 CFR 50.92(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 an 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.

In support of this determination, an evaluation of each of the three criteria set forth in

ATTACHMENT 1
EVALUATION OF PROPOSED CHANGES

10 CFR 50.92 is provided below regarding the proposed license amendment.

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

   TS 5.6.5 lists NRC-approved analytical methods used at LaSalle County Station (LSCS) to determine core operating limits. The proposed changes will add an NRC-approved topical report reference to the list of administratively controlled analytical methods in TS 5.6.5, "Core Operating Limits Report (COLR)," that can be used to determine core operating limits, and delete seven obsolete references.

   The addition of a Framatome ANP (FRA-ANP) methodology to determine overall core operating limits for future LSCS core configurations was approved by the NRC in Reference 2. LSCS Unit 2 will continue to load Framatome ANP ATRIUM-10 fuel during the Unit 2 Refueling Outage 11 currently scheduled for February 2007. The proposed change to TS Section 5.6.5 will add a FRA-ANP methodology as a reference to determine core operating limits for loss of feedwater heater (LOFWH) conditions. Thus, the proposed change will allow LSCS to use the most recent FRA-ANP methodology for analysis of LOFWH conditions.

   The addition and deletion of approved analytical methods in TS Section 5.6.5 has no effect on any accident initiator or precursor previously evaluated and does not change the manner in which the core is operated. The NRC-approved methods ensure that the output accurately models predicted core behavior, have no effect on the type or amount of radiation released, and have no effect on predicted offsite doses in the event of an accident. Additionally the NRC-approved methods do not change any key core parameters that influence any accident consequences. Thus, the proposed changes do not have any effect on the probability of an accident previously evaluated.

   The methodology conservatively establishes acceptable core operating limits such that the consequences of previously analyzed events are not significantly increased.

   The proposed changes in the list of analytical methods do not affect the ability of LSCS to successfully respond to previously evaluated accidents and does not affect radiological assumptions used in the evaluations. Thus, the radiological consequences of any accident previously evaluated are not increased.

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

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

   The proposed changes to TS Section 5.6.5 do not affect the performance of any LSCS structure, system, or component credited with mitigating any accident previously evaluated. The NRC-approved analytical methodology for evaluating LOFWH transients will not affect the control parameters governing unit operation or the response of plant equipment to transient conditions. The proposed changes do not introduce any new modes of system operation or failure mechanisms.

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

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

   The proposed changes will add a reference to the list of analytical methods in TS 5.6.5 that can be used to determine core operating limits and delete seven obsolete references. The proposed changes do not modify the safety limits or setpoints at which protective actions are initiated and do not change the requirements governing operation or availability of safety equipment assumed to operate to preserve the margin of safety. Therefore, the proposed changes provide an equivalent level of protection as that currently provided.

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

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

5.2 Applicable Regulatory Requirements/Criteria

TS 5.6.5 lists the NRC-approved analytical methods used at LSCS to determine core operating limits. The listed NRC-approved analytical methods provide the necessary administrative controls to ensure operation of the facility in a safe manner and thus are required for inclusion in the LSCS Technical Specifications in accordance with 10 CFR 50.36, "Technical specifications," paragraph (c)(5). The deleted references describe analytical methodologies for fuel that will no longer be used at LSCS Units 1 and 2; as such, these references should be deleted from the TS.

6.0 ENVIRONMENTAL EVALUATION

EGC has evaluated this proposed license amendment consistent with the criteria for identification of licensing and regulatory actions requiring environmental assessment in accordance with 10 CFR 51.21, "Criteria for and identification of licensing and regulatory actions requiring environmental assessments." EGC has determined that this proposed change meets the criteria for categorical exclusion set forth in paragraph (c)(9) of 10 CFR 51.22, "Criterion for categorical exclusion; identification of licensing and regulatory actions eligible for categorical exclusion or otherwise not requiring environmental review," and has determined that no irreversible consequences exist in accordance with paragraph (b) of 10 CFR 50.92, "Issuance of amendment." This determination is based on the fact that this change is being processed as an amendment to the license issued pursuant to 10 CFR 50, "Domestic Licensing of Production and Utilization Facilities," which changes a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, "Standards for Protection Against Radiation," or which changes an inspection or surveillance requirement and the amendment meets the following specific criteria:

(i) **The amendment involves no significant hazards consideration.**

As demonstrated in Section 5.1 above, "No Significant Hazards Consideration," the proposed change does not involve any significant hazards consideration.

(ii) **There is no significant change in the types or significant increase in the amounts of any effluent that may be released offsite.**

The proposed changes will add a reference to the list of administratively controlled analytical methods that can be used to determine core operating limits and delete seven obsolete references. The proposed change does not result in an increase in power level, does not increase the production nor alter the flow path or method of disposal of radioactive waste or byproducts; thus, there will be no change in the amounts of radiological effluents released offsite.

Based on the above evaluation, the proposed change will not result in a significant change in the types or significant increase in the amounts of any effluent released offsite.

7.0 REFERENCES

1. Letter from K. R. Jury (EGC) to USNRC, "Request for Amendment to Technical Specifications Section 5.6.5, 'Core Operating Limits Report (COLR)' ", dated March 7, 2005

2. Letter from S. P. Sands (USNRC) to C. M. Crane (EGC), "LaSalle County Station, Units 1 and 2 – Issuance of Amendments Re: Technical Specifications Section 5.6.5, 'Core Operating Limits Report (COLR)' ", dated February 1, 2006

3. Letter from H. N. Berkow (USNRC) to R. L. Gardner (Framatome), "Final Safety Evaluation for Framatome ANP Topical Report ANF-1358(P), Revision 3, 'The Loss of Feedwater Heating Transient in Boiling Water Reactors' " dated September 20, 2005

ATTACHMENT 2

Mark-up of Proposed
Technical Specifications Page Changes

LaSalle Country Station, Units 1 and 2

<u>REVISED TS PAGES</u>

5.6-3
5.6-4

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

4. The Rod Block Monitor Upscale Instrumentation Setpoint for the Rod Block Monitor—Upscale Function Allowable Value for Specification 3.3.2.1.

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

1. ANF-1125(P)(A), "ANFB Critical Power Correlation."

2. Letter, Ashok C. Thadani (NRC) to R.A. Copeland (SPC), "Acceptance for Referencing of ULTRAFLOW™ Spacer on 9x9-IX/X BWR Design," July 28, 1993.

1. ~~3.~~ XN-NF-524(P)(A), "ANF Critical Power Methodology for Boiling Water Reactors."

2. ~~4.~~ ANF-913(P)(A), "COTRANSA 2: A Computer Program for Boiling Water Reactor Transient Analysis."

3. ~~5.~~ ANF-CC-33(P)(A), "HUXY: A Generalized Multirod Heatup Code with 10 CFR 50, Appendix K Heatup Option."

4. ~~6.~~ XN-NF-80-19(P)(A), "Advanced Nuclear Fuel Methodology for Boiling Water Reactors."

5. ~~7.~~ XN-NF-85-67(P)(A), "Generic Mechanical Design for Exxon Nuclear Jet Pump BWR Reload Fuel." e

8. ANF-89-014(P)(A), "ANF Corporation Generic Mechanical Design for ANF Corporation 9x9-IX and 9x9-9X BWR Reload Fuel."

6. ~~9.~~ EMF-CC-074(P)(A), Volume 4 - "BWR Stability Analysis: Assessment of STAIF with input from MICROBURN-B2."

7. ~~10.~~ XN-NF-81-58(P)(A), "RODEX2 Fuel Rod Thermal-Mechanical Response Evaluation Model."

8. ~~11.~~ XN-NF-84-105(P)(A), "XCOBRA-T: A Computer Code for BWR Transient Thermal-Hydraulic Core Analysis."

(continued)

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

12. ANF-91-048(P)(A), "ANF Corporation Methodology for Boiling Water Reactors EXEM BWR Evaluation Model."

9. ~~13.~~ EMF-2209(P)(A), "SPCB Critical Power Correlation."

10. ~~14.~~ ANF-89-98(P)(A), "Generic Mechanical Design Criteria for BWR Fuel Designs."

11. ~~15.~~ NEDE-24011-P-A, "General Electric Standard Application for Reactor Fuel."

12. ~~16.~~ NFSR-0091, "Benchmark of CASMO/MICROBURN BWR Nuclear Design Methods."

17. EMF-1125(P)(A), "ANFB Critical Power Correlation Application for Co-Resident Fuel."

18. ANF-1125(P)(A), "ANFB Critical Power Correlation Determination of ATRIUM-9B Additive Constant Uncertainties."

13. ~~19.~~ EMF-85-74(P)(A), "RODEX2A (BWR) Fuel Rod Thermal-Mechanical Evaluation Model."

14. ~~20.~~ EMF-2158(P)(A), "Siemens Power Corporation Methodology for Boiling Water Reactors: Evaluation and Validation of CASMO-4/MICROBURN-B2."

21. NEDC-32981P(A), "GEXL96 Correlation for Atrium-9B Fuel."

15. ~~22.~~ NEDC-33106P, "GEXL97 Correlation for Atrium-10 Fuel."

16. ~~23.~~ EMF-2245(P)(A), "Application of Siemens Power Corporation's Critical Power Correlations to Co-Resident Fuel."

17. ~~24.~~ EMF-2361(P)(A), "EXEM BWR-2000 ECCS Evaluation Model."

INSERT 1

The COLR will contain the complete identification for each of the TS referenced topical reports used to prepare the COLR (i.e., report number, title, revision, date, and any supplements).

(continued)

INSERT 1

18. ANF-1358(P)(A), "The Loss of Feedwater Heating Transient in Boiling Water Reactors."

ATTACHMENT 3

Typed Pages
Technical Specifications Page Changes

LaSalle Country Station, Units 1 and 2

<u>REVISED TS PAGES</u>

5.6-3
5.6-4

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

4. The Rod Block Monitor Upscale Instrumentation Setpoint for the Rod Block Monitor—Upscale Function Allowable Value for Specification 3.3.2.1.

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

1. XN-NF-524(P)(A), "ANF Critical Power Methodology for Boiling Water Reactors."

2. ANF-913(P)(A), "COTRANSA 2: A Computer Program for Boiling Water Reactor Transient Analysis."

3. ANF-CC-33(P)(A), "HUXY: A Generalized Multirod Heatup Code with 10 CFR 50, Appendix K Heatup Option."

4. XN-NF-80-19(P)(A), "Advanced Nuclear Fuel Methodology for Boiling Water Reactors."

5. XN-NF-85-67(P)(A), "Generic Mechanical Design for Exxon Nuclear Jet Pump BWR Reload Fuel."

6. EMF-CC-074(P)(A), Volume 4 – "BWR Stability Analysis: Assessment of STAIF with input from MICROBURN-B2."

7. XN-NF-81-58(P)(A), "RODEX2 Fuel Rod Thermal-Mechanical Response Evaluation Model."

8. XN-NF-84-105(P)(A), "XCOBRA-T: A Computer Code for BWR Transient Thermal-Hydraulic Core Analysis."

9. EMF-2209(P)(A), "SPCB Critical Power Correlation."

10. ANF-89-98(P)(A), "Generic Mechanical Design Criteria for BWR Fuel Designs."

11. NEDE-24011-P-A, "General Electric Standard Application for Reactor Fuel."

12. NFSR-0091, "Benchmark of CASMO/MICROBURN BWR Nuclear Design Methods."

(continued)

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

13 EMF-85-74(P)(A), "RODEX2A (BWR) Fuel Rod Thermal-Mechanical Evaluation Model."

14 MF-2158(P)(A), "Siemens Power Corporation Methodology for Boiling Water Reactors: Evaluation and Validation of CASMO-4/MICROBURN-B2."

15 NEDC-33106P, "GEXL97 Correlation for Atrium-10 Fuel."

16 EMF-2245(P)(A), "Application of Siemens Power Corporation's Critical Power Correlations to Co-Resident Fuel."

17 MF-2361(P)(A), "EXEM BWR-2000 ECCS Evaluation Model."

18 ANF-1358(P)(A), "The Loss of Feedwater Heating Transient in Boiling Water Reactors."

The COLR will contain the complete identification for each of the TS referenced topical reports used to prepare the COLR (i.e., report number, title, revision, date, and any supplements).

(continued)