



UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001
September 16, 2010

Mr. Michael J. Pacilio
President and Chief Nuclear Officer
Exelon Nuclear
4300 Winfield Road
Warrenville, IL 60555

SUBJECT: LASALLE COUNTY STATION, UNITS 1 AND 2 - ISSUANCE OF AMENDMENTS
RE: MEASUREMENT UNCERTAINTY RECAPTURE POWER UPRATE
(TAC NOS. ME3288 AND ME3289)

Dear Mr. Pacilio:

The U.S. Nuclear Regulatory Commission (the Commission) has issued the enclosed Amendment No. 198 to Facility Operating License No. NPF-11 and Amendment No. 185 to Facility Operating License No. NPF-18 for the LaSalle County Station, Units 1 and 2, respectively. The amendments are in response to your application dated January 27, 2010 (Agencywide Documents Access and Management System (ADAMS) Package No. ML100321303), as supplemented by letters dated May 12, 2010 (ADAMS Accession No. ML101330504) and May 13, 2010 (ADAMS Package No. ML101380594).

The amendments revise the Operating License and Technical Specifications to implement an increase of approximately 1.65 percent in rated thermal power from the current licensed thermal power of 3489 megawatts thermal (MWt) to 3546 MWt. The 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 is currently installed in LaSalle County Station (LaSalle), Unit 1 and will be installed in LaSalle, Unit 2 in refueling outage L2R13, currently scheduled to be completed in March 2011.

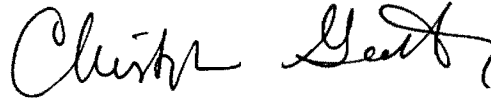
The May 12 and May 13, 2010, supplements contained clarifying information, did not expand the scope of the proposed amendment, and did not change the NRC staff's initial proposed finding of no significant hazards consideration.

M. Pacilio

-2-

A copy of the Safety Evaluation is also enclosed. The Notice of Issuance will be included in the Commission's biweekly *Federal Register* notice.

Sincerely,

A handwritten signature in black ink, appearing to read "Chris Gratton", with a stylized flourish at the end.

Christopher Gratton, Sr. Project Manager
Plant Licensing Branch III-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-373 and 50-374

Enclosures:

1. Amendment No. 198 to NPF-11
2. Amendment No. 185 to NPF-18
3. Safety Evaluation

cc w/encls: Distribution via Listserv



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

EXELON GENERATION COMPANY, LLC

DOCKET NO. 50-373

LASALLE COUNTY STATION, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 198
License No. NPF-11

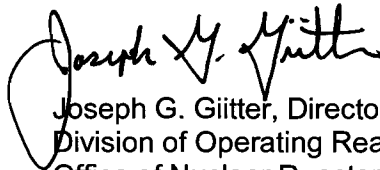
1. The U.S. Nuclear Regulatory Commission (the Commission) has found that:
 - A. The application for amendment filed by the Exelon Generation Company, LLC (the licensee), dated January 27, 2010, as supplemented by letters dated May 12, and May 13, 2010, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's regulations set forth in 10 CFR Chapter I;
 - B. The facility will operate in conformity with the application, the provisions of the Act, and the regulations of the Commission;
 - C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations set forth in 10 CFR Chapter I;
 - D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and
 - E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.
2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and paragraph 2.C.(2) of the Facility Operating License No. NPF-11 is hereby amended to read as follows:

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A, as revised through Amendment No. 198 , and the Environmental Protection Plan contained in Appendix B, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of the date of its issuance and shall be implemented within 90 days of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

A handwritten signature in black ink, appearing to read "Joseph G. Giitter". The signature is stylized with a large loop at the beginning and a cursive script.

Joseph G. Giitter, Director
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical
Specifications and Facility Operating License

Date of Issuance: September 16, 2010



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

EXELON GENERATION COMPANY, LLC

DOCKET NO. 50-374

LASALLE COUNTY STATION, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 185
License No. NPF-18

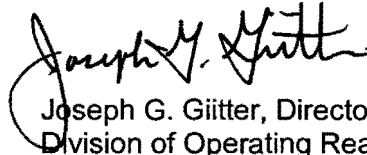
1. The U.S. Nuclear Regulatory Commission (the Commission) has found that:
 - A. The application for amendment filed by the Exelon Generation Company, LLC (the licensee), dated January 27, 2010, as supplemented by letters dated May 12, and May 13, 2010, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's regulations set forth in 10 CFR Chapter I;
 - B. The facility will operate in conformity with the application, the provisions of the Act, and the regulations of the Commission;
 - C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations set forth in 10 CFR Chapter I;
 - D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and
 - E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.
2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the enclosure to this license amendment and paragraph 2.C.(2) of the Facility Operating License No. NPF-18 is hereby amended to read as follows:

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A, as revised through Amendment No. 185 , and the Environmental Protection Plan contained in Appendix B, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of the date of its issuance and shall be implemented within 90 days of the completion of refueling outage L2R13, currently scheduled for completion in March 2011.

FOR THE NUCLEAR REGULATORY COMMISSION

A handwritten signature in black ink, appearing to read "Joseph G. Giitter". The signature is written in a cursive, flowing style with a large initial "J".

Joseph G. Giitter, Director
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical
Specifications and Facility Operating License

Date of Issuance: September 16, 2010

ATTACHMENT TO LICENSE AMENDMENT NOS. 198 AND 185

FACILITY OPERATING LICENSE NOS. NPF-11 AND NPF-18

DOCKET NOS. 50-373 AND 50-374

Replace the following pages of the Facility Operating Licenses and Appendix "A" Technical Specifications with the enclosed pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

Remove

License NPF-11
Page 3

License NPF-18
Page 3

Insert

License NPF-11
Page 3

License NPF-18
Page 3

Interim pages: Use following the implementation at Unit 1, prior to the implementation at Unit 2:

TSs

1.1-5
3.3.1.1-7
3.3.1.3-3

TSs

1.1-5
3.3.1.1-7
3.3.1.3-3

Final pages: Use following the implementation at both units:

TSs

1.1-5
3.3.1.1-7
3.3.1.3-3

TSs

1.1-5
3.3.1.1-7
3.3.1.3-3

Am. 146
01/12/01

(4) Exelon Generation Company, LLC, pursuant to the Act and 10 CFR Parts 30, 40, and 70, to receive, possess, and use in amounts as required any byproduct, source or special nuclear material without restriction to chemical or physical form, for sample analysis or instrument calibration or associated with radioactive apparatus or components; and

Am. 146
01/12/01

(5) Exelon Generation Company, LLC, pursuant to the Act and 10 CFR Parts 30, 40, and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of LaSalle County Station, Units 1 and 2.

C. This license shall be deemed to contain and is subject to the conditions specified in the Commission's regulations set forth in 10 CFR Chapter I and is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) Maximum Power Level

The licensee is authorized to operate the facility at reactor core power levels not in excess of full power (3546 megawatts thermal).

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A, as revised through Amendment No. 198 and the Environmental Protection Plan contained in Appendix B, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

Am. 194
08/28/09

(3) DELETED

Am. 194
08/28/09

(4) DELETED

Am. 194
08/28/09

(5) DELETED

Am. 194
08/28/09

(6) DELETED

Am. 194
08/28/09

(7) DELETED

Am. 34
12/08/87

(5) Pursuant to the Act and 10 CFR Parts 30, 40, and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of LaSalle County Station, Units 1 and 2.

C. The license shall be deemed to contain and is subject to the conditions specified in the Commission's regulations set forth in 10 CFR Chapter I and is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) Maximum Power Level

The licensee is authorized to operate the facility at reactor core power levels not in excess of full power (3546 megawatts thermal). Items in Attachment 1 shall be completed as specified. Attachment 1 is hereby incorporated into this license.

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A, as revised through Amendment No. 185, and the Environmental Protection Plan contained in Appendix B, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

Am. 181
08/28/09

(3) DELETED

Am. 181
08/28/09

(4) DELETED

Am. 181
08/28/09

(5) DELETED

Am. 181
08/28/09

(6) DELETED

Am. 181
08/28/09

(7) DELETED

Am. 181
08/28/09

(8) DELETED

Am. 181
08/28/09

(9) DELETED

1.1 Definitions (continued)

LINEAR HEAT GENERATION RATE (LHGR)	The LHGR shall be the heat generation rate per unit length of fuel rod. It is the integral of the heat flux over the heat transfer area associated with the unit length.
LOGIC SYSTEM FUNCTIONAL TEST	A LOGIC SYSTEM FUNCTIONAL TEST shall be a test of all logic components required for OPERABILITY of a logic circuit, from as close to the sensor as practicable up to, but not including, the actuated device, to verify OPERABILITY. The LOGIC SYSTEM FUNCTIONAL TEST may be performed by means of any series of sequential, overlapping, or total system steps so that the entire logic system is tested.
MINIMUM CRITICAL POWER RATIO (MCPR)	The MCPR shall be the smallest critical power ratio (CPR) that exists in the core for each class of fuel. The CPR is that power in the assembly that is calculated by application of the appropriate correlation(s) to cause some point in the assembly to experience boiling transition, divided by the actual assembly operating power.
MODE	A MODE shall correspond to any one inclusive combination of mode switch position, average reactor coolant temperature, and reactor vessel head closure bolt tensioning specified in Table 1.1-1 with fuel in the reactor vessel.
OPERABLE—OPERABILITY	A system, subsystem, division, component, or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified safety function(s) and when all necessary attendant instrumentation, controls, normal or emergency electrical power, cooling and seal water, lubrication, and other auxiliary equipment that are required for the system, subsystem, division, component, or device to perform its specified safety function(s) are also capable of performing their related support function(s).
RATED THERMAL POWER (RTP)	RTP shall be a total reactor core heat transfer rate to the reactor coolant of 3546 MWt (Unit 1); 3489 MWt (Unit 2).

(continued)

RPS Instrumentation
3.3.1.1

Table 3.3.1.1-1 (page 1 of 3)
Reactor Protection System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION D.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Intermediate Range Monitors					
a. Neutron Flux-High	2	3	G	SR 3.3.1.1.1 SR 3.3.1.1.4 SR 3.3.1.1.6 SR 3.3.1.1.7 SR 3.3.1.1.13 SR 3.3.1.1.15	≤ 123/125 divisions of full scale
	5 ^(a)	3	H	SR 3.3.1.1.1 SR 3.3.1.1.5 SR 3.3.1.1.13 SR 3.3.1.1.15	≤ 123/125 divisions of full scale
b. Inop	2	3	G	SR 3.3.1.1.4 SR 3.3.1.1.15	NA
	5 ^(a)	3	H	SR 3.3.1.1.5 SR 3.3.1.1.15	NA
2. Average Power Range Monitors					
a. Neutron Flux-High, Setdown	2	2	G	SR 3.3.1.1.1 SR 3.3.1.1.4 SR 3.3.1.1.7 SR 3.3.1.1.8 SR 3.3.1.1.11 SR 3.3.1.1.15	≤ 20% RTP
b. Flow Biased Simulated Thermal Power-Upscale	1	2	F	SR 3.3.1.1.1 SR 3.3.1.1.2 SR 3.3.1.1.3 SR 3.3.1.1.8 SR 3.3.1.1.9 SR 3.3.1.1.11 ^{(b) (c)} SR 3.3.1.1.14 SR 3.3.1.1.15	≤ 0.61 W + 68.2% RTP (Unit 1); ≤ 0.62W + 69.3% RTP (Unit 2) and ≤ 115.5% RTP ^(d)
c. Fixed Neutron Flux-High	1	2	F	SR 3.3.1.1.1 SR 3.3.1.1.2 SR 3.3.1.1.8 SR 3.3.1.1.9 SR 3.3.1.1.11 SR 3.3.1.1.15 SR 3.3.1.1.17	≤ 120% RTP

(continued)

- (a) With any control rod withdrawn from a core cell containing one or more fuel assemblies.
- (b) (For Unit 1 only) If the as-found channel setpoint is outside its predefined as-found tolerance, then the channel shall be evaluated to verify that it is functioning as required before returning the channel to service.
- (c) (For Unit 1 only) The instrument channel setpoint shall be reset to a value that is within the as-left tolerance around the nominal trip setpoint (NTSP) at the completion of the surveillance; otherwise, the channel shall be declared inoperable. Setpoints more conservative than the NTSP are acceptable provided that the as-found and as-left tolerances apply to the actual setpoint implemented in the surveillance procedures (field setting) to confirm channel performance. The NTSP and the methodologies used to determine the as-found and the as-left tolerances are specified in the Technical Requirements Manual.
- (d) Allowable Value is ≤ 0.54 W + 55.9% RTP (Unit 1); ≤ 0.55W + 56.8% RTP (Unit 2) and ≤ 112.3% RTP when reset for single loop operation per LCO 3.4.1, "Recirculation Loops Operating."

SURVEILLANCE REQUIREMENTS

-----NOTE-----
When a channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed for up to 6 hours provided the OPRM maintains trip capability.

SURVEILLANCE	FREQUENCY
SR 3.3.1.3.1 Perform CHANNEL FUNCTIONAL TEST.	184 days
SR 3.3.1.3.2 Calibrate the local power range monitors.	2000 effective full power hours
SR 3.3.1.3.3 -----NOTE----- Neutron detectors are excluded. ----- Perform CHANNEL CALIBRATION. The setpoints for the trip function shall be as specified in the COLR.	24 months
SR 3.3.1.3.4 Perform LOGIC SYSTEM FUNCTIONAL TEST.	24 months
SR 3.3.1.3.5 Verify OPRM is not bypassed when THERMAL POWER is $\geq 28.1\%$ RTP (Unit 1); $\geq 28.6\%$ RTP (Unit 2) and recirculation drive flow is $< 60\%$ of rated recirculation drive flow.	24 months
SR 3.3.1.3.6 -----NOTE----- Neutron detectors are excluded. ----- Verify the RPS RESPONSE TIME is within limits.	24 months on a STAGGERED TEST BASIS

1.1 Definitions (continued)

LINEAR HEAT GENERATION RATE (LHGR)	The LHGR shall be the heat generation rate per unit length of fuel rod. It is the integral of the heat flux over the heat transfer area associated with the unit length.
LOGIC SYSTEM FUNCTIONAL TEST	A LOGIC SYSTEM FUNCTIONAL TEST shall be a test of all logic components required for OPERABILITY of a logic circuit, from as close to the sensor as practicable up to, but not including, the actuated device, to verify OPERABILITY. The LOGIC SYSTEM FUNCTIONAL TEST may be performed by means of any series of sequential, overlapping, or total system steps so that the entire logic system is tested.
MINIMUM CRITICAL POWER RATIO (MCPR)	The MCPR shall be the smallest critical power ratio (CPR) that exists in the core for each class of fuel. The CPR is that power in the assembly that is calculated by application of the appropriate correlation(s) to cause some point in the assembly to experience boiling transition, divided by the actual assembly operating power.
MODE	A MODE shall correspond to any one inclusive combination of mode switch position, average reactor coolant temperature, and reactor vessel head closure bolt tensioning specified in Table 1.1-1 with fuel in the reactor vessel.
OPERABLE-OPERABILITY	A system, subsystem, division, component, or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified safety function(s) and when all necessary attendant instrumentation, controls, normal or emergency electrical power, cooling and seal water, lubrication, and other auxiliary equipment that are required for the system, subsystem, division, component, or device to perform its specified safety function(s) are also capable of performing their related support function(s).
RATED THERMAL POWER (RTP)	RTP shall be a total reactor core heat transfer rate to the reactor coolant of 3546 MWt.

(continued)

SURVEILLANCE REQUIREMENTS

-----NOTE-----
When a channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed for up to 6 hours provided the OPRM maintains trip capability.

SURVEILLANCE	FREQUENCY
SR 3.3.1.3.1 Perform CHANNEL FUNCTIONAL TEST.	184 days
SR 3.3.1.3.2 Calibrate the local power range monitors.	2000 effective full power hours
SR 3.3.1.3.3 -----NOTE----- Neutron detectors are excluded. ----- Perform CHANNEL CALIBRATION. The setpoints for the trip function shall be as specified in the COLR.	24 months
SR 3.3.1.3.4 Perform LOGIC SYSTEM FUNCTIONAL TEST.	24 months
SR 3.3.1.3.5 Verify OPRM is not bypassed when THERMAL POWER is $\geq 28.1\%$ RTP and recirculation drive flow is $< 60\%$ of rated recirculation drive flow.	24 months
SR 3.3.1.3.6 -----NOTE----- Neutron detectors are excluded. ----- Verify the RPS RESPONSE TIME is within limits.	24 months on a STAGGERED TEST BASIS

Table 3.3.1.1-1 (page 1 of 3)
Reactor Protection System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION D.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Intermediate Range Monitors					
a. Neutron Flux-High	2	3	G	SR 3.3.1.1.1 SR 3.3.1.1.4 SR 3.3.1.1.6 SR 3.3.1.1.7 SR 3.3.1.1.13 SR 3.3.1.1.15	≤ 123/125 divisions of full scale
	5 ^(a)	3	H	SR 3.3.1.1.1 SR 3.3.1.1.5 SR 3.3.1.1.13 SR 3.3.1.1.15	≤ 123/125 divisions of full scale
b. Inop	2	3	G	SR 3.3.1.1.4 SR 3.3.1.1.15	NA
	5 ^(a)	3	H	SR 3.3.1.1.5 SR 3.3.1.1.15	NA
2. Average Power Range Monitors					
a. Neutron Flux-High, Setdown	2	2	G	SR 3.3.1.1.1 SR 3.3.1.1.4 SR 3.3.1.1.7 SR 3.3.1.1.8 SR 3.3.1.1.11 SR 3.3.1.1.15	≤ 20% RTP
b. Flow Biased Simulated Thermal Power-Upscale	1	2	F	SR 3.3.1.1.1 SR 3.3.1.1.2 SR 3.3.1.1.3 SR 3.3.1.1.8 SR 3.3.1.1.9 SR 3.3.1.1.11 ^{(b) (c)} SR 3.3.1.1.14 SR 3.3.1.1.15	≤ 0.61 W + 68.2% RTP and ≤ 115.5% RTP ^(d)
c. Fixed Neutron Flux-High	1	2	F	SR 3.3.1.1.1 SR 3.3.1.1.2 SR 3.3.1.1.8 SR 3.3.1.1.9 SR 3.3.1.1.11 SR 3.3.1.1.15 SR 3.3.1.1.17	≤ 120% RTP

(continued)

- (a) With any control rod withdrawn from a core cell containing one or more fuel assemblies.
- (b) If the as-found channel setpoint is outside its predefined as-found tolerance, then the channel shall be evaluated to verify that it is functioning as required before returning the channel to service.
- (c) The instrument channel setpoint shall be reset to a value that is within the as-left tolerance around the nominal trip setpoint (NTSP) at the completion of the surveillance; otherwise, the channel shall be declared inoperable. Setpoints more conservative than the NTSP are acceptable provided that the as-found and as-left tolerances apply to the actual setpoint implemented in the surveillance procedures (field setting) to confirm channel performance. The NTSP and the methodologies used to determine the as-found and the as-left tolerances are specified in the Technical Requirements Manual.
- (d) Allowable Value is ≤ 0.54 W + 55.9% RTP and ≤ 112.3% RTP when reset for single loop operation per LCO 3.4.1, "Recirculation Loops Operating."



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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION
RELATED TO AMENDMENT NO. 198 TO FACILITY OPERATING LICENSE NO. NPF-11
AND AMENDMENT NO. 185 TO FACILITY OPERATING LICENSE NO. NPF-18
EXELON GENERATION COMPANY, LLC
LASALLE COUNTY STATION, UNITS 1 AND 2
DOCKET NOS. 50-373 AND 50-374

1.0 INTRODUCTION

By letter to the Nuclear Regulatory Commission (NRC, the Commission) dated January 27, 2010 (Agencywide Documents Access and Management System (ADAMS) Package No. ML100321303) as supplemented by letters dated May 12, 2010 (ADAMS Accession No. ML101330504) and May 13, 2010 (ADAMS Package No. ML101380594), Exelon Generation Company, LLC (EGC, or the licensee), requested changes to the technical specifications (TSs) and facility operating license for LaSalle County Station (LSCS), Units 1 and 2. The proposed changes would revise the Operating License and TSs to implement an increase of approximately 1.65 percent in rated thermal power from the current licensed thermal power (CLTP) of 3489 megawatts thermal (MWt) to 3546 MWt. The changes are based on increased feedwater (FW) flow measurement accuracy, which will be achieved by utilizing Cameron International (formerly Caldon) CheckPlus™ Leading Edge Flow Meter (LEFM) ultrasonic flow measurement instrumentation.

The May 12 and May 13, 2010, supplements, contained clarifying information, did not expand the scope of the proposed amendment, and did not change the NRC staff's initial proposed finding of no significant hazards consideration.

2.0 BACKGROUND

Title 10 of the *Code of Federal Regulations* (10 CFR), Part 50, Appendix K, paragraph I.A, "Sources of heat during the LOCA," requires that emergency core cooling system (ECCS) evaluation models assume that the reactor has been operating continuously at a power level at least 1.02 times the licensed power level to allow for instrumentation error. A change to this paragraph, which became effective on July 1, 2000, allows a lower assumed power level, provided the proposed value has been demonstrated to account for uncertainties due to power level instrumentation error.

Utilization of the Cameron CheckPlus™ LEFM system at LSCS, Units 1 and 2 will result in reduced uncertainty in feedwater flow measurement, which reduces the total power level measurement uncertainty. As described in Section 3.2, "LEFM Ultrasonic Flow Measurement and Core Thermal Power Uncertainty," of Attachment 1 to the licensee's application, with the

utilization of the LEFM system, the core thermal power measurement uncertainty will be a maximum of 0.346 percent.

As summarized in Section 3.4.1, "Summary of Analyses," of Attachment 1, and Attachment 8, "NEDC-33485P, "GEH Nuclear Energy Safety Analysis Report for LaSalle County Station, Units 1 and 2 Thermal Power Optimization," of the application, 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 reactor thermal power (RTP) up to 1.654 percent based on the reduction in thermal power uncertainty. The 1.654 percent increase in RTP corresponds to 3546.7 MWt, which, for the purpose of this application, has been rounded down by the licensee to 3546 MWt, or an approximately 1.65 percent increase.

EGC has evaluated the effects of a bounding 1.7 percent 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. These evaluations are described in detail in Attachment 8 of the application.

The licensee stated in its application that the scope and content of the evaluations performed and described in its request for amendment 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 (Ref.) 10]."

3.0 EVALUATION

3.1 Plant Systems

3.1.1 Containment

3.1.1.1 Regulatory Evaluation

Section 3.1, "Conformance with NRC General Design Criteria," of the LSCS Final Safety Analysis Report describes compliance with those General Design Criteria (GDC) of Appendix A, "General Design Criteria for Nuclear Power Plants," to *Title 10 of the Federal Code of Regulations* (10 CFR) Part 50 related to the containment (primary and secondary) and the safety-related ventilation systems. These include GDC 16, "Containment Design," and GDC 50, "Containment Design Basis," through GDC 57, "Closed System Isolation Valves," for containment and GDC 19, "Control Room," GDC 41, "Containment Atmosphere Cleanup," GDC 60, "Control of Release of Radioactive Materials to the Environment," GDC 61, "Fuel Storage and Handling and Radioactivity Control," and GDC 64, "Monitoring Radioactivity Releases," for the habitability filtration and ventilation systems.

Regulatory Guidance for the containment systems (primary and secondary) is found in the Standard Review Plan (SRP), Sections 6.2.1, "Containment Functional Design," 6.2.2, "Containment Heat Removal Systems," 6.2.3, "Secondary Containment Functional Design," 6.2.4, "Containment Isolation System," and 6.2.5, "Combustible Gas Control in Containment."

3.1.1.2 Technical Evaluation for Containment Systems

NEDC-33485P, "GEH Nuclear Energy Safety Analysis Report for LaSalle County Station, Units 1 and 2 Thermal Power Optimization," (Attachment 8 of ADAMS Accession No. ML100321303) discusses the containment system performance at uprated conditions. A table in Section 4.1 of NEDC-33485P lists the effects on containment system performance due to the proposed uprate. These aspects of containment performance include short-term and long-term pressure and temperature responses to the loss-of-coolant accident (LOCA) and containment dynamic loads due to reactor vessel blowdown following a LOCA or discharge through the safety relief valve discharge lines to the suppression pool. The key parameters affected by the proposed uprate are the break flow and energy, and the decay heat. For both effects, the current analyses based on 102 percent of the current licensed thermal power are bounding.

NEDC-33485P states that the ability of the containment isolation valves and operators to perform their required functions is not affected because the evaluations have either been performed at 102 percent of CLTP, or the existing analyses or equipment testing remains bounding.

The hydrogen recombining function of the post-LOCA Combustible Gas Control System has been eliminated for LSCS. This change was approved by the NRC by license amendments 172 (Unit 1) and 158 (Unit 2) based on the CLTP. The change is based on revisions to 10 CFR 50.44 "Combustible gas control for nuclear power reactors," (68 FR 54141, September 16, 2003). The revision retained requirements to inert Mark II containments (such as the LSCS containments) and to ensure a mixed atmosphere. Section 4.7 of NEDC-33485P states that the blower and associated piping have not been abandoned and remain operational to maintain the drywell mixing function. In addition, the revision to 10 CFR 50.44 did not specify any restriction on power level associated with the rule change. Therefore, the current evaluation is valid for the proposed uprate.

NEDC-33485P states that the capability of the Standby Gas Treatment System to minimize offsite and control room doses and to maintain the secondary containment at a slightly negative pressure is not changed by the proposed uprate.

Therefore, the NRC staff finds the proposed uprate acceptable with respect to containment systems.

3.1.2 Heating, Ventilation Systems and Air Conditioning Systems

3.1.2.1 Regulatory Evaluation

Regulatory Guidance for the habitability, filtration and ventilation systems is found in the SRP, Sections 6.4, "Control Room Habitability System," 6.5.1, "ESF Atmosphere Cleanup Systems," 9.4.1, "Control Room Area Ventilation System," 9.4.2, "Spent Fuel Pool Area Ventilation System," 9.4.3, "Auxiliary and Radwaste Area Ventilation System," 9.4.4, "Turbine Area Ventilation System," and 9.4.5, "Engineered Safety Feature Ventilation System."

3.1.2.2 Technical Evaluation

Section 6.6 of NEDC-33485P states:

The HVAC systems that are potentially affected by the [thermal power optimization] TPO uprate consist mainly of heating, cooling supply, exhaust, and recirculation units in the turbine building, reactor building (including steam tunnel), and primary containment.

TPO results in a minor increase in the heat load caused by the slightly higher FW [feedwater] process temperature (~3°F). The increased heat load in the steam tunnel is within the capability of the Reactor Building HVAC system. In the drywell, the increase in heat load due to the FW process temperature is within the system capacity. In the turbine building, the maximum temperature increases due to the increase in the FW process temperatures and new pump motor heat loads are expected to be very low. This is because the seven chiller units provide supply air at 65°F, instead of the 83°F used in the original design calculations. In the reactor building, the increase in heat load caused by the slightly higher FW process temperature is within the capability of the area coolers. Other areas are unaffected by the TPO because the process temperatures and electrical heat loads remain constant. Therefore, the power-dependent HVAC systems are adequate to support the TPO uprate.

Based on the results of the licensee's analyses, as described above, the NRC staff finds the proposed MUR power uprate acceptable with respect to the heating, ventilation and air conditioning systems.

3.1.1.3 Conclusion

The proposed power uprate for LSCS is acceptable with respect to the containment systems and heating, ventilation and air conditioning systems based on the technical justifications provided in NEDC-33485P.

3.1.2 Cooling Water Systems

Core Standby Cooling System / Equipment Cooling Water and Ultimate Heat Sink

The safety-related Core Standby Cooling System / Equipment Cooling Water (CSCS/ECW) system provides cooling water to essential equipment during and following a design basis accident, such as a Loss-of-Offsite Power or LOCA.

The required volume, duration, and heat rejection capability of the CSCS/ECW system flow in the event of a break is determined based on the analytical and empirical models that simulate reactor and containment conditions subsequent to the postulated reactor coolant system and main steam system breaks. From these analyses, the system and component criteria necessary to demonstrate compliance with regulatory requirements at the uprated conditions are established. The licensee reviewed the CSCS/ECW system for impacts of the uprated power conditions and determined that the MUR power uprate conditions are bounded by the current licensing basis. Therefore, no further evaluation is required.

The ultimate heat sink (UHS) for LaSalle is a cooling pond that remains after the main dike of the cooling lake is breached. The UHS cooling pond is designed to hold approximately 460 acre-feet of water at a surface elevation of 690 ft. The CSCS/ECW system provides the ultimate heat sink for equipment cooling throughout the plant. As a result of operation at the MUR power uprate reactor thermal power level, the post-LOCA heat load increases slightly, primarily due to higher reactor decay heat. However, the licensee stated that the ability of the UHS to perform required safety functions is demonstrated with previous analyses based on 102 percent of the current licensed thermal power, which bounds the expected MUR power uprate reactor thermal power level. Therefore, all safety aspects of the UHS are within previous evaluations and the requirements are unchanged for the MUR power uprate conditions.

The safety-related performance of the UHS and the CWCS/ECW system during and following a LOCA, the most demanding design basis event for the CWCS/ECW system, does not change because the current LOCA analysis was based on 102 percent of the current licensed thermal power. Therefore, the UHS and CWCS/ECW system are acceptable for the MUR power uprate.

Fuel Pool Cooling and Cleanup System

The principal function of the Fuel Pool Cooling and Cleanup System (FPCC) system is to provide cooling of the spent fuel. The primary impact of a power uprate would be to the decay heat of the fuel recently discharged from the core. The MUR power uprate does not affect the heat removal capability of the FPCC. The MUR power uprate heat load is within the design basis heat load for the FPCC. The SFP cooling adequacy is maintained by controlling the timing of the discharge (fuel offload) to the spent fuel pool to ensure the capability of the FPCC to maintain adequate fuel pool cooling under MUR power uprate conditions. Therefore, the NRC staff concurs with the licensee's conclusion that the SFPC system will not be impacted by the power uprate.

3.2 Chemical Engineering

3.2.1 Protective Coating Systems (Paints) – Organic

Regulatory Evaluation

Protective coating systems (paints) protect the surfaces of facilities and equipment from corrosion and radionuclide contamination. Protective coating systems also provide wear protection during plant operation and maintenance activities. The NRC staff's review covered protective coating systems used inside containment, including the coating's suitability for, and stability under, design-basis loss-of-coolant accident (LOCA) conditions, considering radiation and chemical effects. The NRC's acceptance criteria for protective coating systems are based on (1) 10 CFR Part 50, Appendix B, "Quality Assurance Criteria For Nuclear Power Plants and Fuel Reprocessing Plants," and (2) Regulatory Guide 1.54, Revision 1, "Service Level I, II, and III Protective Coatings Applied to Nuclear Power Plants," July 2000. Specific review criteria are contained in SRP Section 6.1.2, "Protective Coating Systems (Paints) – Organic Materials Review Responsibilities."

Technical Evaluation

The licensee stated that the Service Level I coatings (protective coating) in the primary containment are qualified to withstand the existing maximum post-accident primary containment operating conditions of 340 °F, 45 psig, 100 percent relative humidity, and 2×10^8 rad total integrated dose. The licensee stated that the existing maximum post-accident conditions bound the conditions that are expected after implementation of the measurement uncertainty recapture (MUR) power uprate. The NRC staff finds this acceptable.

The NRC staff has reviewed the licensee's evaluation and has confirmed that the applicable regulatory guidance was followed. The staff concurs that the coatings will not be adversely impacted by the MUR power uprate and that temperature and pressure limits under power uprate conditions are bounded by the conditions to which the coatings were qualified.

Conclusion

The NRC staff has reviewed the licensee's evaluation of the effects of the proposed MUR on protective coating systems. The staff concludes that the licensee has appropriately addressed the impact of changes in conditions following a design-basis LOCA and their effects on the protective coating systems. The staff further concludes that the licensee has demonstrated that the protective coating systems will continue to be acceptable following implementation of the proposed MUR power uprate. Specifically, the protective coatings will continue to meet requirements of 10 CFR Part 50, Appendix B. Therefore, the staff finds the proposed MUR power uprate acceptable with respect to protective coating systems.

3.2.2 Flow-Accelerated Corrosion

Regulatory Evaluation

Flow-accelerated corrosion (FAC) is a corrosion mechanism occurring in carbon steel components exposed to single-phase or two-phase water flow. Components made from stainless steel do not experience FAC. FAC is significantly reduced in components containing small amounts of chromium or molybdenum. The rates of material loss due to FAC depend on the fluid's velocity, temperature, steam quality, oxygen content, and pH. During plant operation, flexibility to control these parameters to minimize FAC is limited, therefore, loss of material by FAC occurs. The NRC staff has reviewed the effects of the proposed MUR power uprate on FAC and the adequacy of the licensee's FAC monitoring program. The intent of the FAC monitoring program is to predict the rate of loss so that repair or replacement of damaged components can be made before they reach the critical thickness determined by the predictive method. The licensee's FAC monitoring program is based on NRC Generic Letter (GL) 89-08, "Erosion/Corrosion - Induced Pipe Wall Thinning," May 1989. The NRC's acceptance criteria are based on the structural evaluation of the minimum acceptable wall thickness for the components undergoing degradation by FAC.

Technical Evaluation

The licensee stated that carbon steel piping systems in the main steam (MS), feedwater (FW), and balance of plant (BOP) can be affected by FAC. The licensee reiterated that changes in fluid velocity, temperature, and moisture content have an effect on FAC in piping systems. The

licensee has an established FAC monitoring program for monitoring pipe wall thinning in single-phase and two-phase high energy carbon steel piping.

The licensee stated that the changes in velocity, temperature, and moisture content due to MUR conditions would have a minor effect on FAC. The licensee utilizes a FAC monitoring program that includes the use of a predictive method to calculate wall thinning of components susceptible to FAC. The licensee provided a table that showed piping line segments needing additional review under the FAC monitoring program. The listed piping components are expected to exceed recommended flow velocity guidelines under MUR power uprate conditions. Subsequently, the licensee stated that the FAC monitoring program will be updated to ensure adequate inspection frequencies.

In support of the MUR power uprate, the licensee stated that the FAC monitoring program will take into consideration adjustments to predicted material loss rates. In addition, the program will be updated to include the effects of MUR power uprate conditions. By updating the FAC monitoring program with the associated changes due to the proposed power uprate, the licensee plans to evaluate the need for maintenance or replacement of system piping components prior to reaching their minimum wall thickness limits. The licensee further stated that this program provides assurance that the proposed MUR power uprate has no adverse effect on high energy piping systems susceptible to pipe wall thinning due to FAC. The NRC staff concurs with the licensee's evaluation.

The NRC staff has reviewed the licensee's evaluation and confirms that the applicable regulatory guidance was followed. The staff has also reviewed the piping systems that will be updated to meet MUR power uprate conditions. The licensee has demonstrated that the FAC monitoring program is adequate for managing the potential effects on the piping components susceptible to FAC. The NRC staff concurs that the FAC monitoring program is adequate in predicting the rate of material loss.

Conclusion

The NRC staff has reviewed the licensee's evaluation of the effect of the proposed MUR power uprate on the FAC analysis for the plant and concludes that the licensee has adequately addressed the impact of changes in the plant operating conditions on the FAC analysis. The licensee has demonstrated that the updated analyses will predict the loss of material by FAC, and allow for timely repair or replacement of degraded components following implementation of the proposed MUR power uprate. Therefore, the staff finds the proposed MUR power uprate acceptable with respect to FAC.

3.2.3 Reactor Water Cleanup

Regulatory Evaluation

The reactor water cleanup system (RWCS) provides a means for maintaining reactor water quality by filtration and ion exchange. Additionally, it serves as a path for removal of reactor coolant when necessary. Portions of the RWCS comprise the reactor coolant pressure boundary (RCPB). The NRC staff's review of the RWCS included component design parameters for flow, temperature, pressure, heat removal capability, and impurity removal capability. The review consisted of evaluating the adequacy of the plant's TSs in these areas under MUR power uprate conditions. The NRC staff's acceptance criteria for the RWCS are

based on (1) GDC-14, "Reactor Coolant Pressure Boundary," which requires that the RCPB be designed to have an extremely low probability of abnormal leakage, of rapidly propagating fracture, and of gross rupture; (2) GDC-60, "Control of Releases of Radioactive Materials to the Environment," which requires that the plant design include means to control the release of radioactive effluents; and (3) GDC-61, "Fuel Storage and Handling and Radioactivity Control," which requires that systems that contain radioactivity be designed with appropriate confinement. Specific review criteria are contained in SRP Section 5.4.8, "Reactor Water Cleanup System (BWR)."

Technical Evaluation

The licensee stated that the RWCS would be negligibly affected by MUR power uprate conditions. The high-pressure portion of the RWCS will not be significantly impacted by the changes in operating temperature and pressure conditions. The licensee indicated that transients are the primary source of challenge to the system, and as such, safety and operational aspects of water chemistry performance will be minimally affected by MUR power uprate conditions.

Additionally, the licensee stated that since there will be no significant effect on RWCS temperature or pressure as a result of the power uprate, the RWCS temperature based leak detection will not be effected. Since MUR power uprate conditions will result in higher levels of activation and fission products, the licensee indicated that the RWCS filter demineralizer will require more frequent backwashes. This operational change will not affect compliance with the regulatory requirements described above, and hence, the NRC staff finds the MUR uprate acceptable with respect to the RWCS operational considerations.

The licensee stated that the RWCS line breaks are the limiting breaks for structural design and equipment qualification in several areas of the plant. The licensee also stated that a small increase in recirculation temperature and no pressure increase will decrease the blowdown rate, but increase the energy. These conservatisms more than offset the effects of the temperature change, such that the original HELB analysis remains bounding. The NRC staff evaluated the licensee's disposition regarding the high energy line break analysis and determined that, because the MUR uprate will impose a small impact on the RWCS performance under postulated HELB conditions, and because the original HELB analysis remains bounding, the proposed MUR uprate is acceptable with respect to RWCS performance under postulated HELB conditions.

Because the licensee has considered the RWCS operation and its performance under postulated transient conditions, and concluded that the proposed MUR would have a negligible impact on the system and that the system operation remains bounded by existing analyses, the NRC staff concludes that the licensee has demonstrated that the RWCS will continue to maintain reactor coolant system inventory and water chemistry, consistent with GDC 60 and 61. The NRC staff finds that the RWCS will continue to meet system design requirements and that no new design transients will be created at MUR power uprate conditions, meaning that RWCS will continue to meet the intent of GDC 14 under proposed MUR uprate conditions.

Conclusion

The NRC staff has reviewed the licensee's evaluation of the effects of the proposed MUR on the RWCS and concludes that the licensee has adequately addressed changes in the temperature

of the reactor coolant and its effects on the RWCS. The staff further concludes that the licensee has demonstrated that the RWCS will continue to be acceptable and will continue to meet the requirements of GDC-14, GDC-60 and GDC-61 following implementation of the proposed MUR. Therefore, the NRC staff finds the proposed MUR acceptable with respect to the RWCS.

3.3 Vessels and Internals Integrity

The NRC staff's review in the area of reactor pressure vessel (RPV) integrity for boiling-water reactors focuses on the impact of the proposed MUR power uprate on adjusted reference temperature (ART) calculations, neutron fluence evaluations, pressure-temperature (P-T) limit curves, upper-shelf energy (USE), and surveillance capsule withdrawal schedules. This review is conducted to verify that the results of the licensee's analyses related to these areas continue to meet the requirements of Title 10 of the Code of Federal Regulations (10 CFR) 50.60, and 10 CFR Part 50, Appendices G and H following implementation of the proposed MUR power uprate. The guidance contained in RIS 2002-03, has been used by the NRC staff to conduct this review.

3.3.1 RPV Material Surveillance Program

Regulatory Evaluation

The RPV material surveillance program provides a means for determining and monitoring the fracture toughness of the RPV beltline materials to support analyses for ensuring the structural integrity of the ferritic components of the RPV. Appendix H of 10 CFR Part 50 provides the staff's requirements for the design and implementation of the RPV material surveillance program. The NRC staff's review primarily focused on the effects of the proposed MUR power uprate on the licensee's RPV surveillance capsule withdrawal schedule.

Technical Evaluation

Regarding the RPV surveillance program and capsule withdrawal schedule, the licensee concluded in Section 3.2.1(d) of Enclosure 1 to the submittal that, "TPO has no effect on the existing surveillance schedule; LaSalle will comply with the [integrated surveillance program] ISP requirements."

The licensee's RPV material surveillance program is an ISP designed by the Boiling-Water Reactor Vessel and Internals Project (BWRVIP) for operating BWR plants. The ISP is documented in BWRVIP-78, "BWR Vessel and Internals Project, BWR Integrated Surveillance Program Plan," and BWRVIP-86A, "BWR Vessel and Internals Project, Updated BWR Integrated Surveillance Program (ISP) Implementation Plan." Both were reviewed and approved by the NRC in a safety evaluation (SE) dated February 1, 2002 (ADAMS Accession No. ML020380691). The implementation of the BWRVIP ISP at LSCS was approved in an SE dated October 31, 2003 (ADAMS Accession No. ML032801461), which concluded that the BWRVIP ISP can be implemented for LSCS as the basis for demonstrating the facility's continued compliance with the requirements of Appendix H to 10 CFR Part 50. Table 3-1b of Enclosure 1 to the submittal reported the peak end-of-license (EOL), i.e., 32 effective full power years (EFPY), inside diameter (ID) fluence as 1.04×10^{18} n/cm² (Energy (E) > 1.0 MeV) for the LSCS, Unit 1 shell plates and all welds and Table 3-2b reported the peak EOL, i.e., 32 EFPYs, ID fluence as 1.11×10^{18} n/cm² (E > 1.0 MeV) for the LSCS, Unit 2 shell plates and all welds. The NRC staff confirmed that these fluence values are negligibly greater than the values,

1.02×10^{18} n/cm² (E > 1.0 MeV) and 1.09×10^{18} n/cm² (E > 1.0 MeV) for LSCS, Unit 1 and LSCS, Unit 2, respectively, used in the most recent P-T limits evaluation that was approved in an NRC staff SE dated June 21, 2004 (ADAMS Accession No. ML041900152). The NRC staff's evaluation was based on the guidance in Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials."

Based on the above, the NRC staff determined that the negligible change of the EOL ID fluence will have essentially no impact on the EOL transition temperature shift values and, therefore, on the capsule withdrawal schedule of BWRVIP ISP, currently in the LSCS licensing basis. Therefore, the NRC staff determined that the LSCS RPV surveillance program would continue to meet the requirements of 10 CFR Part 50, Appendix H under the MUR power uprate condition.

3.3.2 Pressure-Temperature Limits and Upper-Shelf Energy

Regulatory Evaluation

Part 50 of 10 CFR, Appendix G, "Fracture Toughness Requirements," provides fracture toughness requirements for ferritic (low alloy steel or carbon steel) materials in the RCPB, including requirements on the USE values used for assessing the safety margins of the RPV materials against ductile tearing and for calculating P-T limits for the plant. These P-T limits are established to ensure the structural integrity of the ferritic components of the RCPB during any condition of normal operation, including anticipated operational occurrences and hydrostatic tests. The NRC staff's review of the USE assessments covered the impact of the MUR power uprate on the neutron fluence values for the RPV beltline materials and the USE values for the RPV materials through the end of the current licensed operating period. The NRC staff's P-T limits review covered the P-T limits methodology and the calculations for the number of the EFPY specified for the proposed MUR power uprate, including neutron embrittlement effects.

Technical Evaluation

Regarding the topic of the RPV P-T limits, the licensee concluded in Section 3.2.1(c) of Attachment 6 to the submittal that, "The CLTP [current licensed thermal power] P-T curves remain unchanged for TPO operation up to 20 and 32 EFPY; sufficient margin exists to account for the approximately 2°F increase in Adjusted Ref. Temperature (ART)."

The current LSCS TSs contain P-T limit curves identified as being applicable through 20 and 32 EFPY for each unit. The 32 EFPY curves are more limiting, and are based on projected peak RPV inside diameter fluences of 1.02×10^{18} n/cm² and 1.09×10^{18} n/cm² (E > 1.0 MeV), for LSCS, Units 1 and 2, respectively, as evaluated in the SE dated June 21, 2004 (ADAMS Accession No. ML041900152). The MUR power uprate projected fluences at 32 EFPY are 1.04×10^{18} n/cm² and 1.11×10^{18} n/cm² (E > 1.0 MeV) for LaSalle, Units 1 and 2, respectively, at 32 EFPY. As shown above, the difference in neutron fluence and the change in ART were verified by staff to be very small. Hence, the NRC staff confirmed that the LSCS P-T limit curves would continue to meet the requirements of 10 CFR Part 50, Appendix G under the MUR power uprate condition.

Regarding the topic of the RPV USE, the licensee concluded in Section 3.2.1(a) of Attachment 6 to the submittal that:

The upper shelf energy (USE) for the beltline region materials remains greater than 50 ft-lb for the design life of the vessel and maintains the margin requirements of 10 CFR 50 Appendix G...

The NRC staff reviewed the licensee's calculations and performed its own confirmatory calculations on the information provided by the licensee in the submittal. For the beltline region materials, the staff found that the EOL USE values for these materials remain greater than 50 ft-lb, with a USE value of 69 ft-lb for limiting Lower Shell Plate C-5978-2 and 60 ft-lb for limiting Shell Axial Weld 3-308 (heat no. 1P3571) for LSCS, Unit 1 and 53 ft-lb for limiting Lower Shell Plate C-9434-2 and 54 ft-lb for limiting lower-to-lower intermediate Shell Axial Weld 5P6771 for LaSalle, Unit 2. Therefore, the NRC staff confirmed that the LSCS RPV materials continue to meet the USE criteria requirements of 10 CFR Part 50, Appendix G under the MUR power uprate condition.

Therefore, the NRC staff finds the proposed MUR power uprate acceptable with respect to the P-T limits and USE.

3.3.3 Conclusion

The NRC staff has reviewed the licensee's evaluation of the effects of the proposed MUR power uprate on the RPV and concludes that the licensee has addressed the surveillance capsule withdrawal schedule, the P-T limit curves and USE, satisfactorily. Hence, the staff has determined that the changes identified in the proposed license amendment request (LAR) will not impact the remaining safety margins required for the above-mentioned structural integrity assessments.

3.4 Fire Protection

Regulatory Evaluation

The purpose of the fire protection program is to provide assurance, through a defense-in-depth design, that a fire will not prevent the performance of necessary plant safe-shutdown functions, nor will it significantly increase the risk of radioactive releases to the environment. The NRC staff's review focused on the effects of the increased decay heat due to the MUR power uprate on the plant's safe-shutdown analysis to ensure that structures, systems, and components (SSCs) required for the safe-shutdown of the plant are protected from the effects of the fire and will continue to be able to achieve and maintain safe-shutdown following a fire. The NRC's acceptance criteria for the fire protection program are based on (1) 10 CFR 50.48, "Fire protection," insofar as it requires the development of a fire protection program to ensure, among other things, the capability to safely shutdown the plant; (2) GDC 3, "Fire Protection," of Appendix A to 10 CFR Part 50, insofar as it requires that [a] SSCs important to safety be designed and located to minimize the probability and effect of fires, [b] noncombustible and heat resistant materials be used, and [c] fire detection and suppression systems be provided and designed to minimize the adverse effects of fires on SSCs important to safety; and (3) GDC 5, "Sharing of Structures, Systems, and Components," of Appendix A to 10 CFR Part 50, insofar as it requires that SSCs important to safety not be shared among nuclear power units unless it can be shown that sharing will not significantly impair their ability to perform their safety functions.

A revision to 10 CFR Part 50, Appendix K, "ECCS Evaluation Models," effective July 31, 2000, allowed licensees to use a power uncertainty of less than 2 percent in design-basis LOCA analyses, based on the use of state-of-the-art FW flow measurement devices that provide for a more accurate calculation of reactor power. Appendix K to 10 CFR Part 50 did not originally require that the reactor power measurement uncertainty be determined, but instead required a 2 percent margin. The revision allows licensees to justify a smaller margin for power measurement uncertainty based on power level instrumentation error. This type of change is also commonly referred to as an MUR power uprate.

Technical Evaluation

EGC developed the LAR consistent with the guidelines in RIS 2002-03. In the LAR, the licensee re-evaluated the applicable SSCs and safety analyses at the proposed MUR core power level of 3546 MWt against the previously analyzed core power level of 3489 MWt.

The NRC staff reviewed Attachment 6, to the GEH Nuclear Energy, "Safety Analysis Report for LSCS, Units 1 and 2 Thermal Power Optimization," NEDO-33485P, Section 6.7, "Fire Protection," and Section 10 CFR Part 50 Appendix R, "Fire Event," of the LAR. The NRC staff also reviewed the licensee's commitment to 10 CFR 50.48, "Fire protection" (i.e., approved fire protection program). The review covered the impact of the proposed MUR power uprate on the results of the safe-shutdown fire analysis as noted in RIS 2002-03, Attachment 1, Sections II and III. The review focused on the effects of the MUR power uprate on the post-fire safe-shutdown capability and increase in decay heat generation following plant trips.

The NRC staff issued an RAI on April 15, 2010, noting that Attachment 6 to the GEH Nuclear Energy Safety Analysis Report for LSCS, Units 1 and 2 Thermal Power Optimization, NEDO-33485P, Section 6.7, "Fire Protection" states that *"...There is no change in the physical plant configuration and the potential for minor changes to combustible loading as result of TPO uprate..."* The NRC staff requested the licensee to summarize any changes to the combustible loading, however minor, and discuss the impact of these changes on the plant's compliance with the fire protection program licensing basis, 10 CFR 50.48, or applicable portions of 10 CFR Part 50, Appendix R.

The licensee responded in a letter to the NRC dated May 12, 2010, stating that for the proposed MUR power uprate, combustible loading calculation results for Unit 1 Fire Zones 4F3 (Auxiliary Building (AB)) and 5C11 (Turbine Building (TB)), and for Unit 2 Fire Zones 4F3 (AB) and 5C11(TB) are within the fire load limits of these fire zones. Further, the licensee indicated that the combustible loading will have an insignificant impact on the fire zone loading. The licensee identified that there will be changes to combustible loading in certain fire zones, and stated that these changes in combustible loading were evaluated and determined to be within the fuel load limits of the affected fire zones. Since these changes do not impact fire protection features and post-fire safe-shutdown capability, the NRC staff finds the response to the RAI acceptable.

The NRC staff also requested in its April 15, 2010, RAI that the licensee verify whether LSCS credits aspects of their fire protection systems for activities other than fire protection, e.g., utilizing the fire water pumps and water supply as backup cooling or inventory for non-primary reactor systems. If the LSCS credits their fire protection systems for other than fire protection activities, the MUR power uprate LAR should identify the specific situations and discuss to what

extent, if any, the MUR power uprate affects these "non-fire-protection" aspects of the plant fire protection system. If the LSCS does not take such credit, the NRC staff requested that the licensee verify this as well.

In response, the licensee confirmed that there are no design-basis accidents (DBAs) or transients that credit the use of the fire protection system at LSCS. However, the licensee did identify three uses of the fire protection system for beyond design basis events. Water from the fire protection system is also used to (1) inject into the reactor pressure vessel to control reactor pressure vessel water level if the design-basis sources of water are unavailable, (2) provide a backup source of make-up water for the spent fuel pool and/or reactor cavity when reactor building access is restricted due to high radiation levels, fires, and similar conditions, and (3) support certain security-related event scenarios. The licensee also indicated that these beyond design-basis events are unaffected by the MUR power uprate condition since the fire protection system makeup capacity significantly exceeds the capacity of the makeup sources credited in the design basis.

The licensee's response satisfactorily addressed the NRC staff's concerns, in that water from the fire protection system is used to backup the water supply for the RPV, add inventory to the spent fuel pool and reactor cavity if reactor building access is restricted, and provide a backup supply to certain security-related events. The licensee analyzed and concluded that all three beyond design-basis events crediting the fire protection system are unaffected by the MUR power uprate. Therefore, the NRC staff finds the response to the RAI acceptable because the licensee's analysis concluded that all three non-fire suppression uses of fire protection water are unaffected by the proposed MUR power uprate.

Based on the licensee's fire-related safe-shutdown assessment and responses to the RAIs, the NRC staff concludes that the licensee has adequately accounted for the effects of the 1.65 percent increase in decay heat on the ability of the required systems to achieve and maintain safe-shutdown conditions. The NRC staff finds this aspect of the capability of the associated SSCs to perform their design-basis functions at an increased core power level of 3546 MWt acceptable with respect to fire protection.

Conclusion

Based on our review, the NRC staff has concluded that the proposed MUR power uprate will not have a significant impact on the fire protection program or post-fire safe shutdown capability and, therefore, finds the proposed amendment acceptable.

3.5 Component Performance and Testing

Safety Related Valves

The NRC staff reviewed the licensee's safety-related valve analysis for LSCS. The NRC's acceptance criteria for review are based on 10 CFR 50.55a, "Codes and Standards." Additional information is also provided by the plant-specific evaluations of GL 89-10, "Safety-Related Motor-Operated Valve Testing and Surveillance, GL 95-07, "Pressure Locking and Thermal Binding of Safety-Related Power-Operated Gate Valves, and GL 96-05, "Periodic Verification of Design-Basis Capability of Safety-Related Power-Operated Valves.

In the submittal dated January 27, 2010, the licensee reviewed the impact of the proposed MUR power uprate conditions on the existing design-basis analyses for the safety-related valves. In Sections 1.3, 3.1, 3.8, 4.1, and 5.3.4 of Attachment 6, "Safety Analysis Report for LaSalle County Station, Units 1 and 2 Thermal Power Optimization," the licensee reviewed the revised design and operating conditions resulting from the MUR power uprate against previous licensing evaluations. In Section 1.3, the licensee indicates that only small changes in reactor coolant flow, operating pressure, and temperature resulted from MUR power uprate. Therefore, no changes to the functional requirements of the existing safety-related valves are identified as a result of the MUR power uprate. In Section 3.1, the licensee reviewed the MUR power uprate's impact on Nuclear System Safety Relief Valves (NSSRVs), and concluded that there was no increase in nominal operating pressure and no changes in the NSSRV setpoints were required for the MUR power uprate conditions. In Section 3.8, the licensee reviewed the MUR power uprate impact on Main Steam Isolation Valves (MSIVs) and concluded that all requirements for the MSIVs remain unchanged for the MUR power uprate conditions. In Section 5.3.4 for Safety Relief Valves (SRVs), the licensee indicates that because there is no increase in reactor operating dome pressure, the SRVs are not changed. In Section 4.1, the licensee also evaluated the MUR power uprate impact on the requirements of GL 89-10, GL-95-07, and GL 96-05. The evaluation shows that no required changes are identified, and all GL 89-10 motor-operated valves remain capable of performing their design-basis functions. Since there are insignificant changes in operating conditions and no changes to the design-basis requirements, the inservice testing (IST) program for safety-related valves will not be affected by the proposed MUR power uprate.

The review concluded that the MUR PU does not impact the design and operation of the safety-related valves since the operating ranges of pressure, temperature, and flow are bounded by previous evaluations. Therefore, the NRC staff finds that the performance of existing safety-related valves and the current IST program are acceptable with respect to the MUR power uprate.

3.6 Instrumentation and Controls

3.6.1 Background

This power uprate is based on a reduced measurement uncertainty of core thermal power resulting from the installation of a Cameron (formerly Caldon) LEFM CheckPlus System to measure feedwater flow and temperature at LSCS. The licensee's submittal referenced Cameron Topical Report (TR) ER-80P, Revision 0, "Improving Thermal Power Accuracy and Plant Safety While Increasing Operating Power Level Using the LEFM System," issued March 1997 (Ref. 3), and its supplement, TR ER-157P, Revision 5, "Supplement to TR ER-80P: Basis for a Power Uprate with the LEFM $\sqrt{t}$ or LEFM CheckPlusTM System," issued October 2001 (Ref. 4). These TRs, which are generically applicable to nuclear power plants, document the ability of the Cameron LEFM Check and CheckPlus Systems to increase the accuracy of flow measurement. The NRC approved TRs ER-80P and ER-157P in safety evaluation reports (SERs) dated March 8, 1999 (Ref. 5), and December 20, 2001 (Ref. 6), respectively.

TR ER-80P describes the LEFM technology, includes calculations of power measurement uncertainty using a Cameron LEFM Check System in a typical two-loop pressurized-water reactor (PWR) or two-feedwater-line boiling-water reactor (BWR), and provides guidelines and equations for determining the plant-specific power calorimetric uncertainties. Its supplement, TR ER-157P, describes the Cameron LEFM CheckPlus System and lists the results of a typical

PWR or BWR thermal power measurement uncertainty calculation using either the Cameron LEFM Check or LEFM CheckPlus System. Together, these two reports provide a generic analysis for an MUR power uprate.

Cameron Engineering Reports ER-629, Revision 1, "Bounding Uncertainty Analysis for Thermal Power Determination at LaSalle Unit 1 Using the LEFM CheckPlus System," issued March 2008 (Ref. 7), and ER-746, Revision 1a, "Bounding Uncertainty Analysis for Thermal Power Determination at LaSalle Unit 2 Using the LEFM CheckPlus System," issued December 2009 (Ref. 8), describe the plant-specific bases for the proposed uprate at LaSalle. The licensee provided the reports, which are non-public, with the LAR.

3.6.2 Regulatory Evaluation

Nuclear power plants are licensed to operate at a specified core thermal power. Appendix K, "ECCS Evaluation Models," to 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities" (Ref. 9), requires analyses of LOCAs and emergency core cooling systems (ECCSs) to assume "that the reactor has been operating continuously at a power level at least 1.02 times the licensed thermal power level (to allow for instrumentation error)." Alternatively, Appendix K allows such analyses to assume a value lower than the specified 102 percent (i.e., 1.02 times the licensed thermal power level), but not less than the licensed thermal power level, "provided the proposed alternative value has been demonstrated to account for uncertainties due to power level instrumentation error." This allowance gives licensees the option to justify a power uprate with reduced margin between the licensed power level and the power level assumed in the ECCS analysis by using more accurate instrumentation to calculate the reactor thermal power.

Because the maximum power level of a nuclear plant is a licensed limit, the NRC must review and approve a proposal to raise the licensed power level under the license amendment process. The LAR should include a justification for the reduced power measurement uncertainty to support the proposed power uprate.

TR ER-80P and its supplement, TR ER-157P, describe the Cameron LEFM CheckPlus System for the measurement of feedwater flow and provide a basis for the proposed uprate of approximately 1.6 percent of the licensed reactor thermal power. The NRC staff also considered the guidance in RIS 2002-03 (Ref. 10), in its review of the licensee's submittals for the proposed power uprate request.

3.6.3 Technical Evaluation

Neutron flux instrumentation is calibrated to the core thermal power, which is determined by an automatic or manual calculation of the energy balance around the plant's nuclear steam supply system. The accuracy of this calculation depends primarily on the accuracy of feedwater flow and feedwater net enthalpy measurements. Feedwater flow is the most significant contributor to the core thermal power uncertainty. More accurate measurements of this parameter will result in a more accurate determination of core thermal power.

The feedwater flow rate is typically measured using a venturi. This device generates a differential pressure proportional to the feedwater velocity in the pipe. Because of the high cost of calibrating the venturi and the need to improve flow instrumentation measurement uncertainty,

the industry evaluated other flow measurement techniques and found the Cameron LEFM Check and LEFM CheckPlus ultrasonic flow meters (UFMs) to be viable alternatives.

3.6.3.1 Leading Edge Flow Meter Technology and Measurement

Both the Cameron LEFM Check and LEFM CheckPlus systems use transit time methodology to measure fluid velocity. The basis of the transit time methodology for measuring fluid velocity and temperature is that ultrasonic pulses transmitted through a fluid stream travel faster in the direction of the fluid flow than opposite the flow. The difference in the upstream and downstream traversing times of the ultrasonic pulse is proportional to the fluid velocity in the pipe, and the temperature is determined using a pre-established correlation between the mean propagation velocity of the ultrasound pulses in the fluid and the fluid pressure.

Both systems use multiple diagonal acoustic paths instead of a single diagonal path, allowing velocities measured along each path to be numerically integrated over the pipe cross-section to determine the average fluid velocity in the pipe. This fluid velocity is multiplied by a velocity profile correction factor, the pipe cross-section area, and the fluid density to determine the feedwater mass flow rate in the piping. The mean fluid density may be obtained using the measured pressure and the derived mean fluid temperature as inputs to a table of thermodynamic properties of water. The velocity profile correction factor is derived from calibration testing of the LEFM Check or CheckPlus System in a plant-specific piping model at a calibration laboratory.

The Cameron LEFM Check System consists of a spool piece with eight transducers, two on each of the four acoustic paths in a single plane of the spool piece. The velocity measured by any one of the four acoustic paths is the vector sum of the axial and the transverse components of fluid velocity as projected onto the path. The Cameron LEFM CheckPlus System uses 16 transducers, 8 each in two orthogonal planes of the spool piece. In the Cameron LEFM CheckPlus System, when the fluid velocity measured by an acoustic path in one plane is averaged with the fluid velocity measured by its companion path in the second plane, the transverse components of the two velocities are canceled and the result reflects only the axial velocity of the fluid. This makes the numerical integration of four pairs of averaged axial velocities and the computation of volumetric flow inherently more accurate than a result obtained using four acoustic paths in a single plane. Also, because there are twice as many acoustic paths and there are two independent clocks to measure the transit times, errors associated with uncertainties in path length and transit time measurements are reduced.

The NRC staff's review in the area of instrumentation and control covers the proposed plant specific implementation of the feedwater flow measurement technique and the power increase gained as a result of implementing this technique, in accordance with the guidelines (A through H) in Section I of Attachment 1 to RIS 2002-03. The NRC staff conducted its review to confirm that the licensee's implementation of the proposed feedwater flow measurement device is consistent with NRC staff-approved TRs ER-80P and ER-157P and that the licensee adequately addressed the four additional requirements listed in the NRC staff's SERs, discussed in Item D in Section 3.6.3.2 of this SER. The NRC staff also reviewed the power measurement uncertainty calculations to ensure that (1) the conservatively proposed uncertainty value of 0.35 percent correctly accounts for all uncertainties associated with power level instrumentation errors and (2) the uncertainty calculations meet the relevant requirements of Appendix K to 10 CFR Part 50.

The licensee provided the following information regarding the Cameron LEFM CheckPlus System feedwater flow measurement technique and its implementation at LSCS, Units 1 and 2.

The licensee has installed LEFM spool pieces in LSCS Unit 1 and will install them in the LSCS Unit 2 feedwater piping. The LEFM spool pieces will be installed in two straight sections of piping. The installation location is downstream of the common feedwater header, which splits into two straight sections of piping. After the piping splits, each pipe has an installed flow straightener. The installation location is between the flow straightener and the originally installed feedwater flow nozzle. The licensee will locate the transducers in the main steamline tunnel in the auxiliary building in an anticipated radiation field of 2.50 Rad/hour at full power. The electronics cabinet will be on the other side of the wall of the main steamline tunnel in a radiation field expected to be less than 2 mRad/hour at full power. No radiation damage or degradation to the instruments (including electronics) caused by such exposure is anticipated.

For LSCS Unit 2, the licensee has developed a modification package outlining the steps to install and test the LEFM system. Once the unit has been shut down 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 feedwater flow and thermal power calculated by various methods, and final commissioning tests.

3.6.3.2 LAR Conformance with RIS 2002-03, Attachment 1, Section I, Guidance A through H

Items A through C

Items A, B, and C in Section I of Attachment 1 to RIS 2002-03 guide licensees to identify the approved topical reports, provide references to the NRC's approval of the measurement technique, and discuss the plant-specific implementation of the guidelines in the topical report and the NRC staff's approval of the feedwater flow measurement technique.

In the LAR, the licensee identified TRs ER-80P, Revision 0, and ER-157P, Revision 5, as applicable to the Cameron LEFM CheckPlus System. The licensee also referenced NRC SERs for TRs ER-80P and ER-157P.

The licensee stated in its submittal that it installed the Cameron LEFM CheckPlus System in LSCS Unit 1 and will install it in Unit 2. The licensee will install LEFM CheckPlus Systems in straight sections of piping. Installation locations are downstream of the common feedwater header/flow straightener upstream of the original feedwater flow nozzle. The licensee plans to install the Cameron LEFM CheckPlus System in LSCS Unit 2 during a refueling outage scheduled for spring 2011.

Based on its review of the licensee's submittals, as reflected in the above discussion, the NRC staff finds that the licensee has addressed the plant-specific implementation of the Cameron LEFM CheckPlus System using proper topical report guidelines. Therefore, the licensee's description of the feedwater flow measurement technique and implementation of the power uprate using this technique follows the guidance in Items A through C of Section I of Attachment 1 to RIS 2002-03.

Item D

Section I of Attachment 1 to RIS 2002-03 provides guidance on dispositioning the criteria that should be addressed when implementing the feedwater flow measurement uncertainty technique. Item D of Section I directs applications to address the four criteria listed below, which were originally identified in the NRC staff's SERs for TRs ER-80P and ER-157P, when referencing these topical reports for a power uprate. EGC's submittal addresses each of the four criteria as follows:

- (1) The licensee should discuss the maintenance and calibration procedures that it will implement with the incorporation of the LEFM. These procedures should include processes and contingencies for an inoperable LEFM and the effect on thermal power measurement and plant operation.

Licensee Response:

Implementation of the power uprate license amendment will include developing the necessary procedures and documents required to maintain and calibrate the new LEFM CheckPlus System. The licensee will revise the plant maintenance and calibration procedures to incorporate Cameron's maintenance and calibration requirements before using the LEFM to raise the power above the current 3,489 MWt limit.

Preventive maintenance activities for the Cameron LEFM CheckPlus System at LSCS will include the following:

- physical inspections
- power supply and pressure transmitter checks
- clock verification

Personnel qualified to work on the LEFM system will perform the maintenance.

For equipment used for the calorimetric computation in the event the LEFM is nonfunctional, the licensee will continue to perform calibration and maintenance using existing site procedures.

The NRC staff evaluates additional details on contingencies for an inoperable LEFM as part of the section of this safety evaluation covering RIS 2002-03, Attachment 1, Section I, Items G and H. Based on the review of the licensee submittals, the NRC staff concludes that the licensee adequately addressed Item D, Criterion 1.

- (2) For plants that currently have LEFMs installed, the licensee should evaluate the operational and maintenance history of the installed instrumentation and confirm that the installed instrumentation is representative of the LEFM system and bounds the analysis and assumptions set forth in TR ER-80P.

Licensee Response:

The licensee installed the Cameron LEFM CheckPlus System in LSCS Unit 1 during its refueling outage in early 2008. The licensee submittal identified four separate maintenance actions for the LEFM in this unit since its installation. Maintenance actions appear to have addressed each

issue, as none of the issues recurred. The licensee has completed final commissioning of the Unit 1 LEFM and verified bounding calibration test data, as described in ER-80P.

The licensee will not install the Cameron LEFM CheckPlus System in LSCS Unit 2 until early in 2011.

The NRC staff finds the licensee's response adequate to address Item D, Criterion 2.

- (3) The licensee should confirm that the methodology used to calculate the uncertainty of the LEFM in comparison to the current feedwater instrumentation is based on accepted plant setpoint methodology (with regard to the development of instrument uncertainty). If the licensee uses an alternative approach, its application should be justified and applied to both venturi and UFM instrumentation installations for comparison.

Licensee Response:

The licensee stated that it based the LEFM uncertainty calculation on American Society of Mechanical Engineers PTC 19.1 (Ref. 11) methodology and Instrument Society of America RP67.04.02-2000 (Ref. 12), which is consistent with the methodology used in TRs ER-80P and ER-157P. The licensee based the LEFM system uncertainty calculation methodology on a square-root-sum-of-squares calculation, which is consistent with its current core thermal power uncertainty calculation for the existing feedwater instrumentation. This methodology is consistent with the vendor determination of the Cameron LEFM CheckPlus System uncertainty, as described in the referenced topical reports, and is consistent with the guidelines in Regulatory Guide (RG) 1.105, "Setpoints for Safety-Related Instrumentation."

Based on the stated calculation methodology, the NRC staff concludes that the licensee adequately addressed Item D, Criterion 3.

- (4) For plants that did not install the ultrasonic meter (including LEFM) with flow elements calibrated to a site-specific piping configuration (flow profiles and meter factors are not representative of the plant-specific installation), licensees should provide additional justification 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, licensees should confirm that the piping configuration remains bounding for the original LEFM installation and calibration assumptions.

Licensee Response:

The licensee calibrated the Cameron LEFM CheckPlus System using a site specific piping configuration at Alden Research Laboratories. As part of the LAR, the licensee submitted ER-644, Revision 0, "LEFM CheckPlus Meter Factor Calibration and Accuracy Assessment for LaSalle Nuclear Power Station," issued December 2007 (Ref. 13) (for Unit 1), and ER-791, Revision 0, "LEFM CheckPlus Meter Factor Calibration and Accuracy Assessment for LaSalle Unit 2," issued November 2009 (Ref. 14).

According to ER-644, the meter factor uncertainty for Unit 1 is 0.24 percent. ER-791 gives the meter factor uncertainty for Unit 2 as 0.23 percent.

For Unit 1, final commissioning, as discussed in this safety evaluation under RIS 2002-03, Attachment 1, Section I, Item D.2, confirmed that performance was consistent with the uncertainty bounds contained in Engineering Report ER-629, Revision 1 (which included the plant-specific factors contained in ER-644). For Unit 2, the licensee stated that acceptance of the final site-specific uncertainty analyses will occur following completion of the commissioning process. The commissioning process will verify bounding calibration test data and confirm that actual field performance meets the uncertainty bounds established for the instruments. The licensee expects to complete final commissioning for Unit 2 early in 2011.

Based on the foregoing, the NRC staff concludes that the licensee adequately addressed Criterion 4. In addition, the licensee committed to confirming that the in situ test data are bounded by the Engineering Report ER-746 and ER-791 calibration test data after final commissioning of the Cameron LEFM CheckPlus System in LSCS Unit 2.

Item E

Item E in Section I of Attachment 1 to RIS 2002-03, provides guidance to licensees for the submittal of a plant-specific total power measurement uncertainty calculation, explicitly identifying all parameters and their individual contributions to the power uncertainty.

To address Item E of RIS 2002-03, the licensee provided Cameron Engineering Reports ER-629, Revision 1, and ER-746, Revision 1a.

The NRC staff reviewed the calculations and determined that the licensee identified all the parameters associated with the thermal power measurement uncertainty, provided individual measurement uncertainties (including those discussed in Item D.(4) above), and calculated the overall thermal power uncertainty. The thermal power uncertainty for each unit is calculated to be 0.346 percent.

The licensee's calculations arithmetically summed uncertainties for parameters that are not statistically independent, and statistically combined them with other parameters. The licensee combined random uncertainties using the square-root-sum-of-squares approach and added systematic biases to the result to determine the overall uncertainty. This methodology is consistent with the vendor determination of the Cameron LEFM CheckPlus System uncertainty, as described in the referenced topical reports, and is consistent with the guidelines in RG 1.105, Revision 3, "Setpoints for Safety-Related Instrumentation," issued December 1999 (Ref. 15).

The NRC staff finds that the licensee has provided calculations of the total power measurement uncertainty at the plant, explicitly identifying all parameters and their individual contribution to the power uncertainty. Therefore, the NRC staff concludes that the licensee has adequately addressed the guidance in Item E of Section I of Attachment 1 to RIS 2002-03.

Item F

Item F in Section I of Attachment 1 to RIS 2002-03, guides licensees to provide information to address the specified aspects of the calibration and maintenance procedures related to all instruments that affect the power calorimetric.

In the LAR, the licensee addressed each of the five aspects of the calibration and maintenance procedures listed in Item F of RIS 2002-03:

(1) Maintaining Calibration

The licensee stated that it will perform calibration using procedures based on the appropriate LEFM CheckPlus requirements. The response to Item D.1 above addresses the preventive maintenance program and the maintenance and calibration of existing instrumentation used in the calorimetric calculation.

(2) Controlling Hardware and Software Configuration

The Cameron LEFM CheckPlus System is designed and manufactured in accordance with the vendor's quality assurance program, which meets the requirements of Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," to 10 CFR Part 50. The licensee stated that, following installation, it will maintain the LEFM CheckPlus system software configuration using existing LSCS procedures and processes. The plant computer software configuration is maintained in accordance with the Exelon Nuclear change control process, which includes the verification and validation of changes to the software configuration. The licensee will maintain the configuration of the hardware associated with the LEFM CheckPlus system and the calorimetric process instrumentation in accordance with Exelon Nuclear configuration control processes.

(3) Performing Corrective Actions

The licensee documented problems with plant instrumentation identified by LSCS personnel in its corrective action program and identified and implemented the necessary corrective actions as part of that program.

(4) Reporting Deficiencies to the Manufacturer

The licensee stated that it reports any conditions associated with a vendor's processes or equipment to the vendor to support any corrective action. For one of the four maintenance items noted since the installation of the Unit 1 LEFM (as part of the discussion of Criterion D.2 of RIS 2002-03, Attachment 1, Section I), the licensee stated that it had consulted Cameron International as part of the resolution of the issue.

(5) Receiving and Addressing Manufacturer Deficiency Reports

The licensee stated that it has existing processes to address the receipt of the manufacturer's deficiency reports. It will document such deficiencies in the LSCS corrective action program.

The NRC staff's review of the above statements found that the licensee addressed the calibration and maintenance aspects of the Cameron LEFM CheckPlus System and all other instruments affecting the power calorimetric. Thus, the NRC staff concludes that the licensee meets the guidance in Item F of Section I of Attachment 1 to RIS 2002-03.

Items G and H

Items G and H in Section I of Attachment 1 to RIS 2002-03 guide licensees to provide a proposed allowed outage time (AOT) for the instrument and to propose actions to reduce power if the AOT is exceeded.

In the LAR, the licensee proposed a 72-hour AOT for operating above 3,489 MWt (i.e., the current licensed thermal power limit) if the UFM becomes inoperable. In the May 12, 2010, submittal, the licensee stated that the LEFM system performs online self-diagnostics to verify that the system operation is within design-basis uncertainty limits. Any out-of-specification condition will result in a self-diagnostic alarm condition, either for "alert" status (i.e., increased flow measurement uncertainty) or "failure" status. In either of these cases, the licensee will consider the LEFM inoperable and will apply the proposed Technical Requirements Manual limiting condition for operation (TLCO) required actions. Additionally, if the communications link between the LEFM system and the plant computer fails (i.e., LEFM CPU Link A and B failed), the licensee will consider the LEFM inoperable and will apply the proposed TLCO required actions.

In the event that the LEFM is declared inoperable while the affected unit is below 3,489 MWt (the current licensed thermal power limit), the power level will not be raised above 3,489 MWt until the LEFM operability is restored. Additionally, if a unit's power level drops below 3,489 MWt during an AOT caused by an inoperable LEFM, the licensee will not raise the power level above 3,489 MWt until the LEFM is operable.

During the 72-hour AOT in the event of a LEFM failure, the plant would use alternate plant instruments (i.e., the existing feedwater flow nozzles) for the calorimetric calculation. The licensee stated that it will continuously monitor the ratio between the existing feedwater venturi flow measurement and the LEFM system flow measurement. In the event that the LEFM becomes inoperable, the licensee will apply a correction factor based on this ratio to the feedwater venturi flow measurement.

Feedwater flow nozzle fouling and transmitter drift were also considered as potential sources of error within the AOT window. The licensee indicated that it had observed feedwater venture fouling in the Unit 1 feedwater flow venturis. It addressed the cause of the fouling in February 2008, and there has been no evidence of an increase or decrease in the degree of fouling of the Unit 1 venturis since. The licensee also evaluated the behavior of the Unit 2 venturis from 2000 through 2006. The data indicated the possibility of a minor amount of fouling that could cause a decline in indicated feedwater flow of no more than approximately 0.5 percent of rated flow. There has been no evidence of a decrease or increase in the degree of fouling of the Unit 2 venturis since that time. Because the degree of fouling of the Unit 1 and the Unit 2 venturis has been stable for a considerable period of time, a defouling event during the 72-hour AOT is considered unlikely. If a sudden defouling event did occur (which could lead to a false low reading of flow), the event would be detected by a change in secondary plant parameters.

The LAR referred to errors in existing feedwater flow measurement techniques assumed as part of the TR ER-80P analysis. These errors, when extrapolated over a 72-hour period, would be negligible. The licensee uses Rosemount 1151 differential pressure transmitters for feedwater flow venturi measurement at the LSCS units. The feedwater flow measurement is a non-safety-related application; therefore, the licensee maintains no instrument loop drift data. However, it has analyzed instrument loop drift data for the 1151 series of Rosemount differential pressure transmitters that are used in safety-related applications at LSCS. These drift data support a total uncertainty caused by a drift of 1.6 percent for a 24-month cycle, which is comparable to the values cited from TR ER-80P.

On the basis of its review of the licensee's submittals, the NRC staff finds that the licensee has provided sufficient justification for the proposed AOT and the proposed actions to reduce the power level if the AOT is exceeded. Therefore, the staff concludes that the licensee has followed the guidance in Items G and H of Section I of Attachment 1 to RIS 2002-03.

3.6.3.3 Technical Specification Change

The licensee proposed to modify TS Table 3.3.1.1-1, Function 2.b, "Flow Biased Simulated Thermal Power-Upscale."

The licensee stated that it used the setpoint methodology documented in NES-EIC-20.04, Revision 5, "Analysis of Instrument Channel Setpoint Error and Instrument Loop Accuracy" (Ref. 16), for the calculations. The NRC previously reviewed Revision 3 of this methodology document in its letter dated March 30, 2001 (Ref. 17). In its May 12, 2010, submittal, the licensee indicated that Revisions 4 and 5 to NES-EIC-20.04 did not make changes to the setpoint methodology. The revisions consisted of administrative changes and clarifications, updated references, and editorial changes and incorporated consideration of potential effects from the high static pressure process measurement for Rosemount flow transmitters. In addition, the NRC staff reviewed the summary calculations and found them to be consistent with RG 1.105, "Setpoints for Safety-Related Instrumentation."

Surveillance Requirement 3.3.1.1.11 in the same table that contains the "Flow Biased Simulated Thermal Power-Upscale" change also adopts a footnote related to channel setpoint surveillance, which is consistent with TS Task Force (TSTF)-493, Revision 4 (Ref. 18).

The licensee proposed to modify TS Table 3.3.13, which addresses the oscillation power range monitor. The modification involves changing the value in Surveillance Requirement 3.3.1.3.5 from ≥ 28.6 percent rated thermal power to ≥ 28.1 percent rated thermal power. With the measurement uncertainty recapture's increase in power, this change will allow the value to remain equivalent in terms of absolute power (i.e., 997.9 MWt to 996.4 MWt). Based on review of the setpoint calculation summary and adherence to the guidelines of TSTF-493, Revision 4, the NRC staff finds that the setpoint changes are acceptable.

3.6.4 Conclusion

The NRC staff reviewed the licensee's proposed plant-specific implementation of the feedwater flow measurement device and the power uncertainty calculations and determined that the licensee's proposed license amendment is consistent with the NRC staff approved TR ER-80P and its supplement, TR ER-157P. The NRC staff also determined that the licensee adequately

accounted for all instrumentation uncertainties in the reactor thermal power measurement uncertainty calculations and demonstrated that the calculations meet the relevant requirements of 10 CFR Part 50, Appendix K, as described in Section 2.0 of this safety evaluation. The licensee has committed to both of the following actions:

- Verify bounding calibration test data.
- Confirm that actual field performance meets the uncertainty bounds established for the instruments in Unit 2 following installation and final commissioning (per Item D.4 above).

Therefore, the staff finds the instrumentation and control aspects of the proposed thermal power uprate of approximately 1.65 percent to be acceptable.

3.7 Electrical Engineering

3.7.1 Regulatory Evaluation

The licensee developed the LAR consistent with the guidelines in NRC RIS 2002-03, "Guidance on the Content of Measurement Uncertainty Recapture Uprate Applications."

The regulatory requirements which the staff applied in its review of the application include:

- GDC 17, "Electric power systems," 10 CFR Part 50, Appendix A requires that an onsite power system and an offsite electrical power system be provided with sufficient capacity and capability to permit functioning of SSCs important to safety.
- The regulation at 10 CFR 50.63 requires that all nuclear plants have the capability to withstand a loss of all alternating current (AC) power (station blackout (SBO)) for an established period of time, and to recover there from.
- The regulation at 10 CFR 50.49, "Environmental Qualification (EQ) of Electric Equipment Important to Safety for Nuclear Power Plants," requires licensees to establish programs to qualify electric equipment important to safety.

3.7.2 Technical Evaluation

The NRC staff reviewed the licensee evaluation of the impact of MUR power uprate on following electrical systems/components:

- AC Distribution System
- Power Block Equipment (Generator, Transformers, Isolated-phase bus duct)
- Direct Current (DC) system
- Emergency Diesel Generators
- Switchyard
- Grid Stability
- Station Blackout
- Equipment Qualification Program

3.7.2.1 AC Distribution System

The AC Distribution System is the source of power for the non-safety-related buses and for the safety-related emergency buses. It consists of the 6.9 kilovolt (kV), 4.16 kV, 480 volt (V), and 120 V systems (not including the emergency diesel generators). The loads that will be affected are the condensate pumps, condensate booster pumps, and heater drain pumps. The licensee stated that the increase in these loads will not exceed their motor nameplate ratings and the electrical supply and distribution components will have sufficient capacity.

In its May 13, 2010, letter (ADAMS Accession No. ML1013805891), the licensee stated that the Class 1E bus design loading, for the power uprate, remains unchanged. The load increase due to the power uprate is bounded by the bus design load due to the use of conservative loads for the components.

The NRC staff reviewed the licensee response and concludes that the AC power system will only experience minor load changes; therefore, the AC system has adequate capacity to operate the plant equipment within the design to support the MUR power uprate.

3.7.2.2 Power Block Equipment (Generator, Transformers, Iso-phase bus duct)

As a result of the power uprate, the rated thermal power will increase to 3546 MWt from the previously analyzed core power level of 3489 MWt.

The generator is rated at 1300.3 mega volt-ampere (MVA) and 300 kilovolt ampere (kVA) at a 0.90 power factor. The NRC staff requested additional information regarding the quantity mega voltampere reactive (MVAR) support necessary to maintain post-trip loads and minimum voltage levels. The licensee stated in its May 13, 2010, letter that the current main generator output is 486.6 MVAR and the post uprate output is 433.8 MVAR. This decrease in MVAR will be credited by the Transmission System Operator following the power uprate. Also the assessment performed by ComEd Transmission and Planning shows that the grid is capable of maintaining the design basis minimum switchyard voltage. The licensee stated in the LAR, that for summer operations the new operating point of the generator is below the main generator maximum capability, while for winter conditions the MWe output will have to be decreased if the plant is required to supply more MVARs. The licensee, in its response to the NRC staff's RAI, discussed the various power levels cited in the LAR. The current maximum generator gross output is 1190 MWe. The uprated value is 1210 MWe, which includes 20 MWe output increase due to the power uprate. Based on this information, the NRC staff finds that the generator is capable of operation at uprated conditions.

The iso-phase bus is rated for 32,000 ampere (A) for forced air cooling and 17,750 A for self cooling. In its May 13, 2010, letter, the licensee stated that the maximum of 1.65 percent increase from the MUR power uprate will increase the current on the iso-phase bus to 31,609 A. Therefore, the NRC staff finds that the iso-phase bus is capable of operation at uprated conditions since the increase from the MUR power uprate remains below the iso-phase bus rating.

Each main generator feeds electric power to two parallel connected half-size main power transformers, stepping the generator voltage of 25 kV up to the transmission voltage of 345 kV. These main (step-up) transformers are rated for 1400 MVA. The licensee stated that the

uprated loadings of the main transformers are 1300.3 MVA, which is below the rating of the main transformers (1400 MVA). Therefore, the NRC staff finds that the main (step-up) transformers are capable of operation at uprated conditions.

System auxiliary transformers, two per unit, are connected to two 345 kV ring buses and supply normal auxiliary power and startup auxiliary power for each unit. Each unit only needs one system auxiliary transformer for its power requirements. The ratings of the system auxiliary transformers are 40.3 MVA for Unit 1 and 32.5 MVA for Unit 2. The current loading is 35.27 MVA for Unit 1 and 26.32 MVA for Unit 2. Furthermore, at uprated conditions, the maximum calculated load for the system auxiliary transformers is 35.33 MVA for Unit 1 and 26.41 MVA for Unit 2, which is within the design rating of the system auxiliary transformers. Therefore, the NRC staff finds that the system auxiliary transformers are capable of operation at uprated conditions.

The unit auxiliary transformers are connected to the generator leads and supply the normal auxiliary power for each unit. The ratings of the unit auxiliary transformers are 40.3 MVA for Unit 1 and 32.5 MVA for Unit 2. The current loading is 35.27 MVA for Unit 1 and 26.32 MVA for Unit 2. Furthermore, at uprated conditions, the maximum calculated load for the unit auxiliary transformers is 35.33 MVA for Unit 1 and 26.41 MVA for Unit 2, which is within the design rating of the unit. Therefore, the NRC staff finds that the unit auxiliary transformers are capable of operation at uprated conditions.

3.7.2.3 DC System

The 125V DC power source in each unit consists of three independent batteries, three battery chargers and three distribution buses. Each unit is provided with a 250 V battery for power to the turbine emergency bearing pumps, generator emergency seal oil pumps, backup feed to the computer and Reactor Core Isolation Cooling system.

The licensee stated that the MUR power uprate does not affect any DC powered indication, control, or protection equipment. Furthermore, in the response to the staff's RAI, the licensee stated that the loads for the Class 1E batteries and battery chargers do not change due to the MUR power uprate. The NRC staff reviewed the LAR and updated final safety analysis report (UFSAR) and confirmed that the power uprate does not impact DC system loads. Therefore, the NRC staff finds that the analyses for DC system bound the MUR power uprate conditions.

3.7.2.4 Emergency Diesel Generators (EDGs)

The standby emergency AC power source for each LSCS unit consists of three diesel generators with one of the diesel generators shared between the two units. The EDG system automatically supplies the safety-related equipment (engineered safeguards and balance of plant emergency loads) when preferred power is unavailable. The licensee stated and provided data that shows that the design basis EDG loads remain unchanged by the MUR power uprate since it is based on analytical power levels of at least 102 percent of the current licensed thermal power. Hence, the EDG system has adequate capacity and capability to power the safety-related loads at MUR power uprate conditions.

Based on the above, the NRC staff, after reviewing the LAR the RAI responses and UFSAR, finds that the power uprate does not impact EDG system loads. Therefore, the staff finds that

the analyses for the EDG system bounds the MUR power uprate conditions and the onsite power system will continue to meet the requirements of GDC 17.

3.7.2.5 Switchyard

The switchyard equipment and associated components are classified as non-safety related. The switchyard serves four 345 kV lines and two 138 kV lines. The primary function of the switchyard and distribution system is to connect the station electrical system to the transmission grid. The current to the switchyard is bounded by the main transformers capability.

The small increase in plant output does not significantly impact the switchyard equipment. Therefore, the NRC staff agrees that the analyses for switchyard system for LSCS reasonably bound the MUR power uprate conditions.

3.7.2.6 Grid Stability

The grid stability impact of the power uprate is discussed in the LAR, and the licensee concludes that there is no significant effect on grid stability or reliability. The licensee stated that a system stability study was performed by PJM Interconnection to determine the impacts of the expected increase in MWe. The system reliability impact study provided in appendix 12 of the LAR indicates that an increase of 20 MWe on Unit 1 and 20 MWe on Unit 2 does not affect the stability of the grid. Thus, the impact study bounds the increase in MWe of each LSCS unit from the power uprate.

The NRC staff reviewed the grid stability study, and finds that the LSCS MUR power uprate allows for continued stable and reliable grid operation.

3.7.2.7 Station Blackout (SBO)

The regulation at 10 CFR 50.63 requires that each light water cooled nuclear power plant be able to withstand and recover from a loss of all AC power, referred to as an SBO.

LSCS's SBO coping duration is 4 hours. This is based on the licensee's evaluation of the offsite power design characteristics, emergency AC power system configuration, and EDG reliability, in accordance with the evaluation procedure outlined in NUMARC 87-00 and RG 1.155. The offsite power design characteristics include the expected frequency of a grid-related loss of offsite power, the estimated frequency of loss of offsite power from severe and extremely severe weather, and the independence of offsite power.

GE TR NEDC-32938P, "General Electric Boiling-Water Reactor Thermal Power Optimization", evaluated the impact on SBO for boiling-water reactors. This evaluation included the adequacy of the condensate/reactor coolant inventory, the capacity of the Class 1E batteries, the SBO compressed nitrogen requirements, the ability to maintain containment integrity and effect of loss of ventilation. Per NEDC-32938P, no plant-specific evaluation is required if the uprated thermal power is enveloped by the current SBO analysis, and if the plant has 1120 gallons of additional condensate storage inventory available and a minimum 2°F peak containment temperature margin. In the LAR, the licensee stated that LSCS currently has margins of 621,358 gallons to the available suppression pool inventory volume and 3.5°F to the containment peak temperature limit.

The NRC staff previously approved the use of the suppression pool instead of the condensate storage tank for the High Pressure Core Spray. The available suppression pool inventory greatly exceeds the condensate storage inventory requirement.

Based on this information, the NRC staff finds that the MUR power uprate will have no impact on LSCS's SBO coping duration. Therefore, the NRC staff finds that LSCS will continue to meet the requirements of 10 CFR 50.63 under power uprate conditions.

3.7.2.8 Equipment Qualification Program

In its LAR, the licensee stated that the MUR power uprate increase in power level raises the radiation levels experienced by equipment during normal operation and accident conditions. Inside containment, the post MUR power uprate environmental qualification (EQ) for safety-related electrical equipment remains bounded by current plant environmental envelope since it is based on accident conditions developed from ≥ 102 percent of current licensed thermal power. Outside containment, the qualification envelope bounds the small changes due to the MUR power uprate. Furthermore, in its response to the RAIs, the licensee stated that the existing high-energy line break analysis and associated accident profile for the main steamline break is bounding for the MUR power uprate conditions.

Based on this information, the NRC staff finds that the current EQ parameters remain bounding for the MUR power uprate. Therefore, the NRC staff finds that the MUR power uprate will have no impact on LSCS's EQ Program and continue to meet the requirements of 10 CFR 50.49.

3.7.3 Conclusion

Based on the technical evaluation provided above, the NRC staff finds that LSCS will continue to meet GDC 17, 10 CFR 50.63, and 10 CFR 50.49. Therefore, the NRC staff finds the MUR power uprate acceptable regarding its impact on electrical systems and components.

3.8 Mechanical and Civil Engineering

3.8.1 Regulatory Evaluation

The NRC staff's review in the areas of civil and mechanical engineering covers the structural and pressure boundary integrity of a number of SSCs affected by the proposed MUR power uprate at LSCS. Specifically, this review focuses on the impact of the proposed MUR power uprate on the structural integrity of the (1) nuclear steam supply system (NSSS) piping, components, and supports, including the pressure retaining portions of the RCPB, the Reactor Recirculation System (RRS) and BOP piping; (2) the RPV and its supports; and (3) the reactor vessel internals (RVIs), including the core support and non-core support structures. Technical areas covered by this review include stresses, fatigue and corresponding cumulative usage factors (CUFs), flow-induced vibration (FIV), high-energy line break (HELB) locations and any corresponding jet impingement and thrust forces.

The NRC staff's evaluation considered 10 CFR 50.55a and GDC 1, 2, 4, 10, 14 and 15 which are located in 10 CFR Part 50, Appendix A. The NRC staff's review focused on verifying that the licensee has provided reasonable assurance of the structural and functional integrity of the aforementioned piping systems, components, component internals and their supports under

normal and vibratory loadings, including those due to fluid flow, postulated accidents, and natural phenomena such as earthquakes.

The acceptance criteria are based on continued conformance with the requirements of the following regulations: (1) 10 CFR 50.55a, and GDC 1 as they relate to structures and components being designed, fabricated, erected, constructed, tested, and inspected to quality standards commensurate with the importance of the safety function to be performed; (2) GDC 2 as it relates to structures and components important to safety being designed to withstand the effects of earthquakes combined with the effects of normal or accident conditions; (3) GDC 4 as it relates to structures and components important to safety being designed to accommodate the effects of, and to be compatible with, the environmental conditions of normal and accident conditions and these structures and components being appropriately protected against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids; (4) GDC 10 as it relates to the reactor core being designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences; (5) GDC 14 as it relates to the RCPB being designed, fabricated, erected, and tested to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture; and (6) GDC 15 as it relates to the reactor coolant system (RCS) being designed with sufficient margin to ensure that the design conditions are not exceeded.

The primary guidance used by Exelon and other licensees for LARs involving MUR power uprates is found in RIS 2002-03, which provides licensees with a guideline for organizing LAR submittals for MUR power uprates. Section IV of RIS 2002-03, "Mechanical/Structural/Material Component Integrity and Design," provides information to licensees on the scope and detail of the information which should be submitted to the NRC staff regarding the components and technical areas listed in the first paragraph of this section.

The licensee's power uprate LAR follows the generic format and content of the NRC staff-approved BWR power uprate licensing TR NEDC-32938P-A, "Generic Guidelines and Evaluations for General Electric [GE] Boiling-Water Reactor [BWR] Thermal Power Optimization [TPO]," May 2003. While the licensee utilized Revision 2 of this TR for the purposes of the LSCS MUR power uprate LAR, the sections pertinent to the NRC staff's review in the areas of the civil and mechanical engineering remain unchanged from the first revision (Ref. 19). This TR is the basis for Attachments 6 and 8 of the LAR, which are, respectively, the proprietary and non-proprietary site-specific analyses performed in support of the proposed MUR power uprate at LSCS by GE-Hitachi. As written, the TR provides guidance for evaluating the effects of MUR power uprates up to 1.5 percent above CLTP. This value was based on the current state of instrumentation technology at the time of the composition of the TR. However, advances in instrumentation technology have demonstrated that MUR power uprates up to 1.7 percent above CLTP are achievable, such that power level uncertainties of 0.3 percent are possible. As such, the licensee confirmed the applicability of the TR to the proposed MUR power uprate at LSCS throughout the license amendment application. The NRC staff's SER documenting the aforementioned acceptability of this TR for evaluating the effects of an MUR power uprate for GE BWR nuclear power plants is documented in Ref. 20.

3.8.2 Technical Evaluation

The NRC staff's technical review in the areas of civil and mechanical engineering focused on the effects of the proposed MUR power uprate on the structural and pressure boundary integrity of NSSS piping systems, components and their supports, the RPV, RVIs and applicable technical areas, including FIV and HELB analyses. The proposed 1.65 percent power uprate will increase the rated thermal power level from 3489 MWt to 3546 MWt at LSCS. The power uprate will be achieved by an increase in reactor power along the current rod and core flow control lines. An increase in steam flow and FW flow will accompany this increase in reactor power.

Table 1-2 of Attachments 6 and 8 in Ref. 1 shows the pertinent temperatures, pressures, and flow rates for the current and proposed (uprated) conditions. At full power, the dome temperature remains at a constant 547.0 °F from current to uprated conditions. At full power, the dome pressure remains at a constant 1020.0 pounds per square inch absolute from current to uprated conditions. The minimum full power core flow range increases from 87.9 to 89.8 million pounds per hour (Mlbm/hr) while the maximum full power core flow range remains constant at 113.9 Mlbm/hr under the uprated conditions. The steam flow increases from 15.145 to 15.435 Mlbm/hr while the FW flow rate increases from 15.113 Mlbm/hr to 15.403 Mlbm/hr under the uprated conditions. Additionally, the FW temperature increases from 426.5 to 428.5 °F from the current to uprated conditions. The design parameters for the RCPB at LSCS are found in Chapter 5 of the LSCS UFSAR. The RCPB design pressure is indicated as 1250 pounds per square inch gauge (psig). Chapter 10 of the LSCS UFSAR provides design information for the portions of the MS system, FW system and Condensate system outside of containment.

3.8.2.1 Reactor Pressure Vessel

The licensee evaluated the effects of the proposed power uprate on the structural integrity of the RPV, including its nozzles, in Section 3.2.2 of Attachments 6 and 8 in Ref. 1. The design code of record for the RPV is the American Society of Mechanical Engineers (ASME) Boiler & Pressure Vessel (B&PV) Code, Section III, 1968 Edition up to and including the Winter 1970 Addenda. Section 3.2.2 of Attachments 6 and 8 in Ref. 1 also includes the design codes of record for a number of components which have undergone design modifications since the components were evaluated against the original design code of record. The licensee compared the expected temperatures and pressures for the proposed power uprate condition against the analyses of record. As indicated in Section 5.5.1.2 of Ref. 3, the structural impact of an MUR power uprate on the reactor vessel is minimal due to the vessel pressure remaining constant, coupled with miniscule changes in the temperatures associated with the FW and recirculation flow throughout the vessel. For the normal and upset conditions, the licensee confirmed that the design code stress requirements for all components will continue to be satisfied following the implementation of the proposed power uprate. Additionally, the licensee noted that the loads associated with the emergency and faulted conditions are not affected by the proposed power uprate and, therefore, the stress requirements for these two conditions remain valid. Consistent with the guidance described in Ref. 19, the licensee indicated that fatigue analyses were performed for limiting components to demonstrate their acceptability at the proposed power level due to the MUR uprate. Based on its evaluations, the licensee confirmed that the RPV and its associated components will continue to meet the stress and fatigue design requirements of the design codes of record for these components following the implementation of the MUR power uprate.

The NRC staff has reviewed the licensee's evaluations related to the structural integrity of the RPV and its associated components, including nozzles. For the reasons set forth above, which demonstrate that the RPV will continue to meet its design basis acceptance criteria under the conditions of the proposed MUR power level, the NRC staff concludes that the licensee has adequately addressed the effects of the proposed MUR on these components. Based on the above, the NRC staff further concludes that the licensee has demonstrated that the RPV and its associated components will continue to meet the applicable regulatory requirements, described above, following implementation of the proposed MUR. Therefore, the NRC staff finds the proposed MUR acceptable with respect to the structural integrity of the RPV.

3.8.2.2 Reactor Vessel Internals

The licensee evaluated the effects of the proposed power uprate on the structural integrity of the RVIs, including core support and non-core support structures, in Section 3.3.2 of Attachments 6 and 8 in Ref. 1. The core support structures evaluated in support of the proposed MUR power uprate include the shroud support, shroud, core plate, top guide, control rod drive housings, control rod guide tubes, and the orificed fuel supports. The non-core support structures evaluated include the FW sparger, jet pumps, core spray line and sparger, access hole cover, shroud head and steam separator assembly, in-core housing and guide tube, vessel head cooling spray nozzle, core differential pressure and liquid control line, the low pressure coolant injection coupling, and the steam dryer. The licensee noted that while the RVIs are not designed to the requirements of the ASME B&PV Code, this code is used as a guideline in the design and analysis of the RVIs. The evaluations and stress reconciliations performed in support of the MUR power uprate were performed in accordance with the design basis requirements for the RVIs found in Chapter 3 of the LSCS UFSAR.

In parallel with the guidance provided for the structural evaluation of the RPV, Section 5.5.1.2 of Ref. 19 notes that the structural impact of an MUR power uprate on the RVIs is minimal due to the vessel pressure remaining constant, coupled with miniscule changes in the temperatures associated with the FW and recirculation flow throughout the vessel. The changes in the thermal-hydraulic loadings induced on the RVIs, due to the proposed implementation of the MUR power uprate, were evaluated to determine their impact on the current analyses of record for the normal, upset, emergency, and faulted RVI loading conditions. Loads which increased due to the increased power level were accounted for by linearly increasing the applicable stresses associated with the increased loads and then comparing these stresses to corresponding design code allowable stress limits. Based on its evaluations as described above, the licensee confirmed that the RVIs will continue to meet the design basis stress requirements for these components following the implementation of the MUR power uprate. Additionally, the licensee confirmed that the fatigue evaluations performed for the RVIs demonstrate that all CUFs for the affected RVIs remain valid for the proposed MUR power uprate.

With regards to the effects of FIV on the RVIs due to MUR power uprates, Section 5.5.1.3 of Ref. 19 notes that, due to the absence of an increase in the maximum core flow rate at MUR power uprate conditions, a majority of the RVIs in a BWR reactor system are unaffected by the implementation of an MUR power uprate. In Section 3.4 of Attachments 6 and 8 in Ref. 1, the licensee acknowledges the guidance provided by Ref. 19 and indicates that the effects of FIV on the limiting RVIs, due to the proposed MUR power uprate, were evaluated based on vibration data from prototype plants; this is consistent with the guidance found in Ref. 19. Based on the

operating experience data provided by the prototype plants' vibration data, the licensee compared the expected FIV levels due to the MUR power uprate with the vibration acceptance limits and determined that the increased vibration levels induced on the RVIs, due to the power uprate, are acceptable. In response to an NRC staff RAI regarding the acceptance criterion used by the licensee to demonstrate the acceptability of the RVI vibration levels, the licensee's response in Ref. 2 indicates that the vibration acceptance limits established by GEH for monitoring FIV are more stringent than the NRC staff-approved levels found in Section III of the ASME B&PV Code. Appendix I of the ASME B&PV Code, Section III, limits the single amplitude stress limit to 13,600 psi at 10^{11} stress cycles, compared to the GEH established limit of 10,000 psi at 10^{11} stress cycles, which is more conservative.

The NRC staff has reviewed the licensee's evaluations related to the structural integrity of the RVIs. For the reasons set forth above, which demonstrate that the RVIs will continue to meet their design basis acceptance criteria under the conditions of the proposed MUR power level, the NRC staff concludes that the licensee has adequately addressed the effects of the proposed MUR on these components. The NRC staff further concludes that the licensee has demonstrated that the RVIs will continue to meet the applicable regulatory requirements following implementation of the proposed MUR. Therefore, the NRC staff finds the proposed MUR acceptable with respect to the design of the RVIs.

3.8.2.3 Reactor Coolant Pressure Boundary Piping and Supports

The licensee evaluated the effects of the proposed power uprate on the structural integrity of the RCPB piping, including RCPB pipe supports, in Section 3.5.1 of Attachments 6 and 8 in Ref. 1. As indicated in Section 5.1 of the LSCS UFSAR, the RCPB piping includes those portions of piping systems which are part of the RCS or connected to the reactor coolant system, up to and including any and all of the following: the outermost containment isolation valve in system piping which penetrates the primary containment, the second of the two valves normally closed during normal reactor operation in system piping which does not penetrate the primary containment, and the reactor coolant system safety/relief valves; this includes the portions of the MS and FW system piping which are inside containment. The design codes of record for the portions of the RCPB evaluated in support of the proposed MUR power uprate are located in Section 5.2 of the LSCS UFSAR. Accordingly, the evaluations and stress reconciliations performed in support of the MUR power uprate were performed in accordance with the design basis requirements for the RCPB piping found in this section of the UFSAR.

The guidance provided for the structural evaluation of the RCPB piping, found in Section 5.5.2 and Appendix K of Ref. 19, notes that the structural impact of an MUR power uprate on the RCPB piping is minimal due to the vessel pressure remaining constant, coupled with a minor increase in the FW temperatures associated with the power uprate. Portions of the RCPB piping not connected to the FW and MS system piping, including the RRS piping, have negligible changes in piping stresses and pipe support loads. Therefore, the licensee evaluated the critical portions of the FW and MS piping inside containment, in addition to the piping connected to these two systems.

For these portions of the RCPB piping, Appendix K of Ref. 19 provides methods for evaluating the increased piping stresses due to MUR power uprates. These methods have been applied to piping evaluations associated with power uprates much greater than the increased power levels realized at MUR conditions (up to 20 percent, for extended power uprates (EPUs)). Appendix K

directs licensees to evaluate critical locations on the piping runs with the highest calculated stress to allowable ratios. The calculated stress is then increased according to the appropriate equation percent increase and then compared to the allowable stress prescribed by the design code of record. Additionally, fatigue considerations are accounted for and evaluated in a manner similar to the stresses to develop the CUFs at the uprated conditions for the critical piping locations. The licensee stated that its evaluations, which were performed based on the guidance described above from Appendix K of Ref. 19, confirmed that the stresses and CUFs for the affected portions of the MS and FW piping systems and their supports remain within the allowable limits prescribed by their respective design codes of record.

In Section 3.4 of Attachments 6 and 8 in Ref. 1, the licensee discusses the effects that the increased steam and FW flow rates have on the safety-related portions of the MS and FW piping systems. Additionally, the slight effects of the recirculation system thermal-hydraulic parameter changes, due to the proposed MUR power uprate, are also discussed with regards to the possible effects of FIV on the RRS. Based on operational experience and analytical evaluations, the licensee stated that these portions of the affected piping systems would meet the vibration acceptance criteria at MUR power uprate conditions. In response to an NRC staff RAI regarding the licensee's evaluations performed to demonstrate acceptable FIV performance of these piping systems, the licensee indicated in Ref. 2 that the vibration levels gathered from start up data and operating experience were linearly extrapolated, proportional to the fluid densities and velocities. The extrapolations revealed that vibration levels would be roughly 4 percent higher than the vibration levels under current operating conditions. The technique of extrapolating results to provide expected vibration behavior is the standard technique for predicting vibration levels at higher flow rates, with varying fluid densities. Given the small range of variance between the current and uprated thermal hydraulic parameters at the MUR power level, there is reasonable assurance that the extrapolation used in support of the LSCS MUR power uprate yields an adequate amount of accuracy in order to determine the acceptability of the FIV levels at the proposed power level. As previously discussed in SER Section 3.2, the licensee indicated in Ref. 2 that the vibration acceptance limits established by GEH for monitoring FIV are more stringent than the NRC staff-approved levels found in Section III of the ASME B&PV Code. The licensee also noted in Ref. 2 that the aforementioned piping systems are well within the vibration acceptance criteria at MUR power uprate conditions and maintain a margin of approximately 70 percent.

The NRC staff has reviewed the licensee's evaluations related to the structural integrity of the RCPB piping and supports. For the reasons set forth above, which demonstrate that the RCPB and supports will continue to meet their design basis stress, fatigue and FIV acceptance criteria under the conditions of the proposed MUR power level, the NRC staff concludes that the licensee has adequately addressed the effects of the proposed MUR power uprate on these components. Based on the above, the NRC staff further concludes that the licensee has demonstrated that the RCPB piping and supports will continue to meet the applicable regulatory requirements, described above, following implementation of the proposed MUR power uprate. Therefore, the NRC staff finds the proposed MUR acceptable with respect to the structural integrity of the RCPB piping and supports.

3.8.2.4 BOP Piping Systems

The licensee evaluated the effects of the proposed power uprate on the structural integrity of the BOP piping, including BOP pipe supports, in Section 3.5.2 of Attachments 6 and 8 in Ref. 1. The BOP piping evaluations performed by the licensee in support of the proposed MUR power uprate are similar to the evaluations performed for the RCPB piping, described above in SER Section 3.3 (i.e., utilizing the methodology described in Appendix K of Ref. 19). The primary focus of the BOP piping evaluations is on the FW and MS piping and pipe supports located outside containment. As indicated above and in Ref. 19, the pipe stresses and support loads accompanied with an MUR power uprate do not vary significantly from the current operating conditions. Information regarding the design codes of record for the BOP piping SSCs is located in Chapter 3 of the LSCS UFSAR.

The licensee noted that, based on the lack of a steam temperature change from current to uprated conditions, the MS pipe support thermal loads do not change. Additionally, while the FW temperature increase slightly elevates the FW pipe support loads, the supports remain acceptable and the load change is insignificant in comparison to the other loads within the design basis load combinations. The licensee also noted that applicable portions of BOP piping were evaluated for revised transient loads and were shown to be acceptable under MUR power uprate conditions. Consistent with Section 5.10.10 of Ref. 19, the licensee demonstrated that the effects of higher steam and FW flow rates, coupled with any thermal gradients, do not have a significant impact on the BOP piping. Additionally, the current analyses of record for the BOP piping and supports remain valid for the proposed MUR power uprate.

The NRC staff has reviewed the licensee's evaluations related to the structural integrity of the BOP piping and supports. For the reasons set forth above, which demonstrate that the BOP piping and supports will continue to meet their design basis acceptance criteria under the conditions of the proposed MUR power level, the NRC staff concludes that the licensee has adequately addressed the effects of the proposed MUR on these components. Based on the above, the NRC staff further concludes that the licensee has demonstrated that the BOP piping and supports will continue to meet the applicable regulatory requirements, described above, following implementation of the proposed MUR. Therefore, the NRC staff finds the proposed MUR acceptable with respect to the structural integrity of the BOP piping and supports.

3.8.2.5 High-Energy Line Break Locations

Section 10.1 of Attachment 8 in Ref. 1 summarizes the results of the HELB analyses performed in support of the proposed MUR power uprate at LSCS. The licensee stated in Section 10.1 (identified above) that the changes in the NSSS thermal-hydraulic system parameters (e.g., temperature, pressure, mass flow rates) were reviewed to determine whether the changes in these parameters would have a significant impact on the current analyses of record regarding the dynamic effects associated with HELBs. Based on the insignificant changes in these parameters, which are also described in SER Section 3.0, the licensee concluded that postulated HELB effects, currently found in the analyses of record, remain unchanged by the proposed MUR power uprate. Additionally, the licensee stated that no piping configuration modifications would be necessary to support the implementation of the proposed power uprate. Therefore, the postulated HELB locations in the current analyses of record would remain valid following the implementation of the proposed power uprate.

The NRC staff has reviewed the licensee's evaluations related to determinations of rupture locations and associated dynamic effects. For the reasons set forth above, which demonstrate that the current HELB analyses will remain valid under the conditions of the proposed MUR power level, the NRC staff concludes that the licensee has adequately addressed the effects of the proposed MUR on these analyses of record. Based on the above, the NRC staff further concludes that the licensee has demonstrated that SSCs important to safety will continue to meet the regulatory requirements applicable to HELB analyses following the implementation of the proposed MUR power uprate. Therefore, the NRC staff finds the proposed MUR power uprate acceptable with respect to the determination of rupture locations and dynamic effects associated with the postulated rupture of piping.

3.8.3 Conclusion

The NRC staff has reviewed Exelon's assessment of the impact of the proposed MUR power uprate on the NSSS and BOP SSCs with regards to stresses, CUFs, FIV, HELB locations, and jet impingement and thrust forces. Based on the review described above, the NRC staff finds the MUR power uprate acceptable with respect to the structural integrity of the aforementioned SSCs affected by the power uprate. This acceptance is based on the licensee's demonstration that the SSCs affected by the proposed uprate will maintain their structural integrity following the implementation of the MUR power uprate. Additionally, the licensee has also demonstrated that the intent of the aforementioned regulatory requirements related to the civil and mechanical engineering purview has been met. Therefore, there is reasonable assurance that these SSCs will be able to maintain their structural integrity in order to perform their intended functions following the implementation of the MUR power uprate at LSCS.

3.9 Reactor Systems and Thermal Hydraulics

3.9.1 Regulatory Evaluation

Early revisions of 10 CFR 50.46, and Appendix K to 10 CFR 50, required licensees to base their LOCA analysis on an assumed power level of at least 102-percent of the licensed thermal power level to account for power measurement uncertainty. The NRC later modified this requirement to permit licensees to justify a smaller margin for power measurement uncertainty. Licensees may apply the reduced margin to operate the plant at a level higher than the previously licensed power. The licensee proposed to use a Cameron LEFM CheckPlus system to decrease the uncertainty in the measurement of feedwater flow, thereby decreasing the power level measurement uncertainty from 2.0-percent to 0.35-percent.

The licensee developed its LAR consistent with the guidelines in NRC RIS 2002-03.

3.9.2 Technical Evaluation

The NRC staff reviewed thermal-hydraulic aspects of the LEFM CheckPlus system installation, including its laboratory calibration, the effects of system changes such as transducer replacement, and the impact the system installation will have, if any, on the applicable plant safety analyses, as discussed in the following subsections.

3.9.2.1 Feedwater Flow Measurement Device Installation

EGC has installed the LEFM in the Unit 1 feedwater piping at LSCS and will install the Unit 2 LEFM in their feedwater piping prior to implementing the MUR power uprate. The Unit 1 LEFM is installed in a straight section of feedwater piping downstream of the main feedwater header. The LEFM is situated between a flow straightener and the originally installed feedwater flow nozzle. Unit 2 installation will be comparable to Unit 1's installation.

The devices are installed or will be installed in accordance with the requirements in the approved Cameron Topical Reports ER-80P and ER-157P related to the LEFM Check and LEFM CheckPlus Systems (Ref.s 3 and 4) . After plant installation, the licensee compared the Unit 1 "as-installed" measurement uncertainties with that of the measurement uncertainties obtained during testing and calibration at Alden Labs and the measurement uncertainties were found to be consistent. The Unit 2 spool pieces will also be installed to match the as-tested configuration.

3.9.2.2 CheckPlus Inoperability

To operate above the presently licensed power of 3489 MWt, the licensee proposes that the CheckPlus cannot have been out-of-service for more than 72 hours. If the LEFM cannot be returned to operable status within 72 hours, the plant power level is to be reduced to no greater than 3489 MWt.

Power during the 72 hours without an operational CheckPlus will be monitored using existing plant instrumentation, such as the feedwater flow nozzles. The licensee justifies this operation on the basis that feedwater flow nozzle measurements become more conservative if fouling occurs. Reactor power can also be calculated via the backup computer or other methods such as Average Power Range Monitors, turbine first stage pressure, or main generator output.

The actions are to be covered in the Technical Requirements Manual. The NRC staff finds that operation with an inoperable CheckPlus has been acceptably addressed.

3.9.2.3 Transducer Replacement

Uncertainty associated with transducer replacement was addressed in the LSCS application for the MUR power uprate. Since the transducer installation uncertainty is incorporated in the LSCS Unit 1 and Unit 2 UFM System uncertainty, and no additional uncertainty terms need to be applied whenever a transducer is replaced, the NRC staff finds that transducer installation variability has been acceptably addressed.

3.9.2.4 CheckPlus Calibration

CheckPlus calibration was accomplished at Alden Laboratories. The NRC staff reviewed drawings and schematics provided and confirmed that, insofar as configuration is concerned, the laboratory configuration largely matched the in-situ configuration.

The piping configuration used during testing included a full scale model of Unit 1 and Unit 2 hydraulic geometry. The calibration factor used for the UFM is based on the reports prepared by Alden Labs after the testing.

The tests were completed using previously applied procedures and laboratory measurement elements traceable to the National Institute of Standards and Technology. The NRC staff finds that the licensee's laboratory calibration was sufficiently fabricated to provide meaningful data based on the modeling of piping geometry of the UFM at LSCS Units 1 and 2.

3.9.2.5 Nuclear Steam Supply System Parameters

The NSSS design parameters provide the RCS and secondary system conditions (pressures, temperatures, and flow) that are used as the basis for the design transients and for systems, components, accidents and transient analyses and evaluations. The parameters are established using conservative assumptions to provide bounding conditions to be used in the NSSS analyses.

3.9.2.6 Accident and Transient Analyses

3.9.2.6.1 Review Approach

The NRC staff reviewed the accident and transient analyses using the guidance contained in NRC RIS 2002-03. RIS 2002-03 states the following regarding the general approach to the NRC staff review:

- In areas for which the existing analyses of record do not bound the plant operation at the proposed uprated power level, the staff will conduct a detailed review.
- In areas for which the existing analyses of record do bound plant operation at the proposed uprated power level, the staff will not conduct a detailed review.
- In areas that are amenable to generic disposition, the staff will utilize such dispositions.

The NRC staff followed this approach to review EGC's evaluation of the LSCS licensing basis with respect to accident and transient analyses and their acceptability for power uprate. In general, the licensee provided information to demonstrate that certain accident/transient analyses were performed at a power level bounding of plant operation at the requested power level; remaining items referenced the generic disposition contained in NEDO-32938-A, Revision 2, "Generic Guidelines and Evaluations for General Electric Boiling-Water Reactor Thermal Power Optimization." The only exception was for the anticipated transients without scram (ATWS), the licensing basis analyses for which were not bounding of the requested operation, and the licensee performed and presented additional evaluations and analyses. For this class of events, the NRC staff performed a detailed review.

3.9.2.6.2 Technical Evaluation

Attachment 4 of the January 27, 2010, letter, requesting implementation of the MUR uprate contained a cross-reference to Sections II and III of NRC RIS 2002-03. Attachment 4 stated that Section II, Accidents and Transients for which the Existing Analyses of Record Bound Plant Operation at the Proposed Uprated Power Level, were addressed in Attachment 6, Section 9.0, "Reactor Safety Performance Evaluations." Similarly, Attachment 4 stated that Section III, Accidents and Transients for which the Existing Analyses of Record Do Not Bound Plant

Operation At the Proposed Up-rated Power Level, was also addressed in Attachment 6, Section 9.0. In Attachment 6, the licensee characterized the DBAs and transients by sorting them into three categories: anticipated operational occurrences, DBAs, and special events.

3.9.2.6.2.1 Anticipated Operational Occurrences

For anticipated operational occurrences, the licensee stated that the effect of the requested MUR uprate on the limiting transient events is small. As a result, the licensee did not present results for plant-specific transient analyses. The licensee also stated that cycle-specific analyses for the potentially limiting transients will be performed using the cycle-specific core loading and previous exposure history. The licensee stated that this approach is consistent with the Thermal Power Optimization guidelines contained in NEDO-32938-A, Revision 2, "Generic Guidelines and Evaluations for General Electric Boiling-Water Reactor Thermal Power Optimization," which has been generically approved by the NRC. Because the licensee will reanalyze the limiting transients on a cycle-specific basis in accordance with NRC-approved reload licensing methodology, and because this disposition is consistent with the generically approved approach documented in NEDO-32938-A, the NRC staff finds the requested uprate acceptable with respect to the transient analyses.

3.9.2.6.2.2 Design-Basis Accidents

The licensee stated that the DBA events either have been previously analyzed at 102 percent of the CLTP level and are, hence, bounding of operation at the proposed, uprated power level with reduced uncertainty, or are not dependent on core thermal power. Because the licensee analyzed the DBA events at a bounding power level, the NRC staff finds this disposition acceptable.

3.9.2.6.2.3 Special Events

The licensee identified two classes of special events requiring evaluation: the ATWS events, and the SBO.

Anticipated Transient Without Scram

EGC concluded that the ATWS analysis of record was neither bounding of plant operation at the proposed, uprated power level, nor amenable to generic disposition. Therefore, in accordance with the guidance contained in RIS 2002-03, the NRC staff performed a detailed review of the ATWS analyses described by the licensee. The NRC staff relied on the guidance contained in Review Standard (RS) 001, "Review Standard for Extended Power Uprates," to complete this review. The NRC staff determined that RS-001 was applicable because it describes an acceptable review approach for specific accidents and transients affected by power uprates, and contains specific review guidance for ATWS analyses.

Regulatory Evaluation

ATWS is defined as an anticipated operational occurrence (AOO) followed by the failure of the reactor portion of the protection system specified in GDC-20. The regulation at 10 CFR 50.62 requires that:

- Each BWR have an alternate rod insertion (ARI) system that is designed to perform its function in a reliable manner and be independent (from the existing reactor trip system) from sensor output to the final actuation device;
- Each BWR have a standby liquid control system (SLCS) with the capability of injecting into the reactor vessel a borated water solution with reactivity control at least equivalent to the control obtained by injecting 86 gpm of a 13 weight-percent sodium pentaborate decahydrate solution at the natural boron-10 isotope abundance into a 251-inch inside diameter reactor vessel. The system initiation must be automatic; and
- Each BWR have equipment to trip the reactor coolant recirculation pumps automatically under conditions indicative of an ATWS.

The NRC staff's review was conducted to ensure that:

1. The above requirements are met;
2. Sufficient margin is available in the setpoint for the SLCS pump discharge relief valve such that SLCS operability is not affected by the proposed MUR power uprate;
3. Operator actions specified in the plant's Emergency Operating Procedures are consistent with the generic emergency procedure guidelines/severe accident guidelines, insofar as they apply to the plant design.

In addition, the NRC staff reviewed the licensee's ATWS analysis to ensure that:

1. The peak vessel bottom pressure is less than the ASME Service Level C limit of 1500 psig;
2. The peak clad temperature is within the 10 CFR 50.46 limit of 2200°F;
3. The peak suppression pool temperature is less than the design limit;
4. The peak containment pressure is less than the containment design pressure.

Specific review criteria are provided in SRP Section 15.8 and additional guidance is provided in Matrix 8 of RS-001.

Technical Evaluation

The analysis of the ATWS in support of the requested power uprate is described in Section 9.3 of Attachment 6 to EGC's January 27, 2010, letter.

The licensee stated that LSCS meets the ATWS requirements defined in 10 CFR 50.62 because (a) an alternate rod insertion (ARI) system is installed, (b) the boron injection capability is equivalent to 86 gpm, and (c) there is an automatic Recirculation Pump Trip (RPT) logic (i.e. ATWS-RPT). In addition, a plant-specific ATWS analysis was performed at MUR power uprate conditions to confirm that (a) the peak vessel bottom pressure is less than ASME Service Level C limit of 1500 psig, (b) the peak suppression pool temperature does not increase over the value predicted in the current analysis of record, and (c) the peak containment pressure does not increase over the value predicted in the current analysis of record. Since the current analyses of record are acceptable, no change in the results demonstrates that the peak suppression pool temperature and containment pressure are less than their design limits.

The licensee stated that substantial generic experience with BWR power uprates has demonstrated that the two limiting ATWS events are the main steam line isolation valve closure with failure to scram, and the pressure regulator failure open-maximum steam demand with failure to scram. The licensee, therefore, evaluated these two scenarios to establish compliance with the acceptance criteria described above.

The licensee performed a plant-specific ATWS analyses for an equilibrium core at the proposed uprated operating condition to demonstrate that LSCS meets the ATWS acceptance criteria. The licensee's analyses indicated that the predicted peak vessel bottom pressure increased from 1474 psig in the current licensing basis analysis to 1491 psig for operation at the proposed uprated condition. As stated above, peak suppression pool temperature and containment pressure did not change, remaining at 204°F and 14.6 psig, respectively. The licensee referenced the generic disposition contained in NEDO-32938-A for peak cladding temperature and fuel oxidation. The generic disposition states that power uprate has a negligible effect on these parameters, and the basis for this disposition has been reviewed and generically-approved by the NRC staff.

The licensee provided a discussion of ATWS with core instability events. The discussion relied heavily on reference to generic studies performed for General Electric fuel designs which have previously been asserted to be applicable to General-Electric BWRs operating with fuel designs supplied by alternative vendors. During uprate application reviews, the NRC staff has thoroughly questioned the applicability of these analyses to the alternative fuel designs, not issuing a definitive conclusion of the acceptability of using the generic dispositions for any fuel design other than General Electric fuel.

In this case, the NRC staff did not review the information provided for ATWS instability events for two reasons:

1. The requested uprate will not affect the initial condition of the ATWS instability, since it starts from natural circulation, and
2. The same operator actions will be relied upon to terminate the ATWS instability event.

Therefore, the NRC staff makes no finding with regard to ATWS instability events, since the requested uprate is of small enough magnitude to be judged by the NRC staff as having no significant effect on the instability mitigating actions that are used to terminate the event.

Conclusion - ATWS

Because the licensee's ATWS analyses and hardware system evaluations demonstrate compliance with the acceptance criteria regarding ATWS mitigating systems actuation (i.e., recirculation pump trip and alternate rod insertion), standby liquid control system performance requirements, and system effects of the limiting postulated ATWS events, the NRC staff finds the proposed power uprate acceptable with respect to anticipated transients without scram.

Station Blackout

The licensee referenced the generic disposition contained in NEDO-32938-A for the SBO and confirmed its applicability to LSCS at the proposed, uprated operating condition. Because the

licensee confirmed that the generic disposition is applicable, the NRC staff finds the proposed power uprate acceptable with respect to the SBO event.

3.9.2.6.3 Accident and Transient Analyses - Conclusion

Based on the NRC staff's review of licensee's evaluations, analyses, and dispositions concerning the accident and transient analyses, the NRC staff finds the proposed power uprate acceptable with respect to the accident and transient analyses.

3.9.3 Conclusion for Section 3.9

The NRC staff reviewed the reactor systems and thermal-hydraulic aspects of the proposed MUR power uprate amendment. Based on the considerations discussed above, the NRC staff determined that the results of the licensee's analyses related to these areas continue to meet applicable acceptance criteria following implementation of the MUR power uprate. The proposed amendment is based on the use of a Cameron LEFM Check Plus system that would decrease the uncertainty in the feedwater flow, thereby decreasing the power level measurement uncertainty from 2.0 percent to 0.35 percent. In these cases, the proposed MUR power uprate rated thermal power of 3546 MWt is bounded by the current analyses of record.

3.10 Accident Dose Assessment

3.10.1 Regulatory Evaluation

This evaluation has been conducted to verify that the results of the licensee's DBA radiological dose consequence analyses continue to meet the dose acceptance criteria given in 10 CFR 50.67. Previously, in Amendments 197 and 184 dated September 6, 2010 (ADAMS Accession No. ML101750625), for Units 1 and 2 respectively, the licensee was granted implementation of a full-scope alternative source term in accordance with 10 CFR 50.67, and following the guidance of RG 1.183, "Alternative Radiological Source Terms for Evaluating Design-Basis Accidents at Nuclear Power Reactors." Except where the licensee proposed a suitable alternative, the NRC staff utilized the regulatory guidance provided in applicable sections of RG 1.183 and NUREG-0800, "Standard Review Plan [SRP] for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition," Chapter 15, for DBAs, and SRP Chapter 6.4, for control room habitability, as well as other applicable guidance in performing this review.

3.10.2 Technical Evaluation

The NRC staff reviewed the regulatory and technical analyses performed by the licensee in support of its proposed MUR power uprate license amendment, as they relate to the radiological consequences of DBA analyses. Information regarding these analyses was provided by the licensee in Attachment 6 to the January 27, 2010, submittal (Ref. 1). The findings of this safety evaluation are based solely on the descriptions and results of the licensee's analyses and other supporting information docketed by the licensee.

The NRC staff reviewed the impact of the proposed 1.65 percent MUR power uprate on DBA radiological consequence analyses, as documented in Chapter 15 of the LSCS UFSAR. The specific DBA analyses that were reviewed were as follows:

- LOCA
- Fuel-Handling Accident
- Steam System Pipe Break Outside Containment/Main Steam Line Break Accident
- Control Rod Drop Accident

In the LAR submittal, the licensee stated that each of the current DBA dose analyses of record for LSCS, which depend on core power level, were performed at 102 percent of the currently licensed thermal power of 3489 MWth (~3559 MWth). Therefore, the current analyses bound any analyses that would be performed at the proposed MUR power uprate level of 3546 MWth, as the currently analyzed power is 100.36 percent of the proposed uprated power. This margin is within the assumed uncertainty associated with advanced flow measurement techniques, including use of the International CheckPlus™ LEFM system credited by the licensee.

The licensee has accounted for the potential for an increase in measurement uncertainty should the LEFM system experience operational limitations. In the event system failures are detected, failure messages are generated by the plant process computer and alarm in the control room. Core power during LEFM system AOT will be calculated using feedwater flow from existing feedwater flow nozzles. If the LEFM system is not repaired within 72 hours, power will be reduced to ensure that the dose consequence analyses that support the current licensing basis will remain bounding.

Using the licensing basis documentation, as contained in the current LSCS UFSAR, in addition to information in the January 27, 2010, LAR submittal letter, the NRC staff verified that the existing LSCS UFSAR Chapter 15 radiological analyses' source term and release assumptions bound the conditions for the proposed 1.65 percent power uprate to 3546 MWth, considering the higher accuracy of the proposed feedwater flow measurement instrumentation.

3.10.3 Conclusion

As described above, the staff reviewed the assumptions, inputs, and methods used by the licensee to reassess the radiological consequences of the postulated DBA with the proposed uprated power level. The NRC staff finds that the licensee will continue to meet the applicable dose acceptance criteria, as identified in Section 3.10.1 of this evaluation, following implementation of the proposed 1.65 percent MUR power uprate. The NRC staff further finds reasonable assurance that LSCS, as modified by this approved license amendment, will continue to provide sufficient safety margins, with adequate defense-in-depth, to address unanticipated events and to compensate for uncertainties in accident progression, analysis assumptions, and input parameters. Therefore, the proposed license amendment is acceptable with respect to the radiological dose consequences of the DBAs.

3.11 Health Physics and Human Performance Branch Review

3.11.1 Regulatory Evaluation

The NRC's human factors review addresses whether the licensee has adequately considered the effects of the proposed MUR power uprate on programs, procedures, training, and plant design features related to operator performance during normal and accident conditions. The NRC's human factors evaluation is conducted to confirm that the licensee has analyzed the effects of the MUR and properly concluded that operator performance will not be adversely

affected as a result of system and procedure changes made to implement the proposed MUR power uprate. The scope of this review included licensee-identified changes to operator actions, human-system interfaces, procedures, and training needed for the proposed MUR power uprate.

3.11.2 Technical Evaluation

The NRC staff has developed a standard set of questions for human factors reviews of MUR power uprates (RIS 2002-03, Attachment 1, Section VII, Items 1 through 4). The following sections evaluate the licensee's response to these questions in the LAR.

3.11.2.1 Operator Actions

Has the licensee made a statement confirming that operator actions that are sensitive to the power uprate, including any effects on the time available for operator actions, have been identified and evaluated?

Yes, Exelon stated in its January 27, 2010, submittal, that existing operator actions are not affected by the MUR power uprate. In addition, they stated that there is no reduction in the time required for the operator actions. The licensee's review included AOOs, DBAs, and other special events such as fire, ATWS, and SBO. Reviews were also done on a system functional basis to assure that safety-related functions would not be affected, whether they require manual actions or not.

The NRC staff has reviewed the licensee's statements in their submittal relating to any impacts of the MUR power uprate on operator actions. The NRC staff concludes that the proposed MUR power uprate will not adversely impact operator actions or their response times because no changes are required. The NRC staff finds that the statements provided by Exelon conform to Section VII.1 of Attachment 1 to RIS 2002-03.

3.11.2.2 Emergency and Abnormal Operating Procedures

Has the licensee made a statement confirming that it has identified all required changes to the current emergency operating procedures (EOPs) and abnormal operating procedures (AOPs) to ensure that changes to the EOPs and /or AOPs do not adversely affect defense in depth or safety margins?

Yes, Exelon stated in its submittal that the EOP action thresholds are plant unique and will be addressed using standard procedure updating processes. The licensee expects that the MUR uprate will have a negligible or no effect on the operator action thresholds and to the EOPs in general. Necessary procedure revisions for the power uprate will be completed prior to uprate implementation. Exelon stated that operator training on the procedure changes will be provided as a part of the MUR implementation.

The NRC staff concludes that the proposed MUR power uprate does not present any adverse impacts to the EOPs and AOPs. This conclusion is based upon two licensee statements: (1) Exelon will, as necessary, revise the EOPs and AOPs to reflect the new power level and revised setpoints, and (2) any changes made to the EOPs and AOPs will be reflected in the operator training program prior to MUR implementation.

The NRC staff finds that the statements provided by Exelon are in conformance with Section VII.2.A of Attachment 1 to RIS 2002-03.

3.11.2.3 Changes to Control Room Controls, Displays, and Alarms

Has the licensee made a statement confirming that it has identified all required changes to the control room controls, displays, and alarms (including the safety parameter display system) to ensure that any required changes do not adversely affect defense in depth or safety margins?

Yes, in its submittal, Exelon described the evaluations performed to identify control room changes in support of the proposed MUR. The licensee identified that changes are required to certain non-safety related systems, including minor equipment changes, replacements, and setpoint or alarm point changes. These changes will be made in accordance with the requirements of 10 CFR 50.59, "Changes, tests, and experiments," and will be implemented prior to implementation of the proposed power uprate. 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 maintenance and calibration requirements prior to declaring the LEFM system operational and raising power above the CLTP of 3489 MWt. The system features automatic self-checking. A continuously operating on-line test is provided to verify that the digital circuits are operating correctly and within the specified accuracy range. Failure messages are generated by the plant process computer and monitored in the control room, if system failure events are detected. Any changes to the control room will be reflected in the operator training program prior to MUR implementation.

The NRC staff has reviewed the licensee's evaluation of the proposed changes to the control room. The NRC staff concludes that the proposed changes are minimal and do not present any adverse effects to the operators' functions in the control room. Exelon committed to making all modifications to the control room and providing training on these changes prior to MUR power uprate implementation. The NRC staff finds that the statements provided by Exelon are in conformance with Section VII.2.B of Attachment 1 to RIS 2002-03.

3.11.2.4 Control Room Plant Reference Simulator

Has the licensee made a statement confirming that it has identified all required changes to the control room plant reference simulator to ensure that any required changes do not adversely affect defense in depth or safety margins?

Yes, Exelon stated that potential simulator changes will be identified as part of the plant modification process. The submittal also included statements that these modifications will be evaluated, implemented and tested per the approved plant procedures. Simulator changes and validation for the MUR uprate will be performed in accordance with established LaSalle plant certification testing procedures. The licensee stated that these modifications will be completed in time to support operator training prior to the MUR power uprate implementation.

The NRC staff has reviewed Exelon's evaluation of proposed changes to the plant simulator related to the MUR power uprate. Exelon committed to making all modifications to the plant simulator and incorporating these changes into the operator training program prior to MUR

power uprate implementation. The NRC staff finds that the statements provided by Exelon are in conformance with Section VII.2.C of Attachment 1 to RIS 2002-03.

3.11.2.5 Operator Training

Has the licensee made a statement confirming that it has identified all required changes to the operator training program to ensure that any required changes do not adversely affect defense in depth or safety margins?

Yes, the licensee stated in its submittal that no additional training (apart from normal training for plant changes) is required to operate the plant under the conditions resulting from the MUR power uprate. For uprate conditions, operator response to transient, accident, and special events is not affected. Operator actions for maintaining safe shutdown, core cooling, and containment cooling, also do not change. Changes related to the LEFM system and minor setpoint changes will be identified as part of the plant modification process and communicated through normal operator training.

The NRC staff has reviewed the licensee's evaluation of the proposed changes to the operator training program. The staff concludes that the proposed changes are appropriate and do not present any adverse effects to the operators' functions in the control room. Exelon committed to providing training on these changes prior to MUR power uprate implementation. The NRC staff finds that the statements provided by Exelon are in conformance with Section VII.2.D of Attachment 1 to RIS 2002-03.

3.11.2.6 Modifications

Has the licensee made a statement confirming its intent to complete the modifications identified in Items 3.11.2.2 through 3.11.2.5 above (including the training of operators), prior to implementation of the power uprate?

Yes, the licensee stated in its submittal, that the LEFM system for LSCS, Unit 2 will be installed prior to the implementation of the proposed MUR power uprate, and that other non-safety related modifications for the power uprate, including switchyard modifications, will be implemented prior to implementation of the power uprate. Further, the licensee has committed to revise plant maintenance and calibration procedures, modify the plant simulator for the uprated conditions and validate the changes in accordance with plant configuration control processes. Maintenance personnel will be qualified on LEFM and operator training will be completed. Exelon has committed that all of the above actions will be completed prior to implementation of the proposed power uprate. The NRC staff finds that the statements provided by Exelon are in conformance with Section VII.3 of Attachment 1 to RIS 2002-03.

3.11.3 Conclusion

The NRC staff has completed its human factors review of Exelon's LAR per RIS 2002-03, and concludes that the licensee has adequately considered the impact of the proposed MUR power uprate on operator actions, EOPs and AOPs, control room components, the plant simulator, and operator training programs.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Illinois State official was notified of the proposed issuance of the amendments. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendments change requirements with respect to the installation or use of the facilities components located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration, and there has been no public comment on such finding (75 FR 26289 dated May 11, 2010). Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

6.0 REGULATORY COMMITMENTS

To support the proposed LSCS MUR power uprate, the licensee made the following commitments (as stated):

COMMITMENT	COMMITTED DATE OR "OUTAGE"	COMMITMENT TYPE	
		ONE-TIME ACTION (Yes/No)	PROGRAMMATIC (Yes/No)
Limitations regarding operation with an inoperable LEFM system will be included in the TRM.	Prior to uprate implementation	Yes	No
The LEFM system for LSCS, Unit 2 will be installed prior to uprate implementation.	Prior to uprate implementation	Yes	No
Plant maintenance and calibration procedures will be revised to incorporate Cameron's maintenance and calibration requirements.	Prior to uprate implementation	Yes	No
For LSCS, Unit 2, final acceptance of the site-specific uncertainty analyses will occur after the completion of the commissioning process.	Prior to uprate implementation	Yes	No
Non-safety related modifications for the power uprate, including switchyard modifications required by PJM, will be implemented.	Prior to uprate implementation	Yes	No
Necessary procedure revisions for the power uprate will be completed.	Prior to uprate implementation	Yes	No

COMMITMENT	COMMITTED DATE OR "OUTAGE"	COMMITMENT TYPE	
		ONE-TIME ACTION (Yes/No)	PROGRAMMATIC (Yes/No)
The plant simulator will be modified for the uprated conditions and the changes will be validated in accordance with plant configuration control processes.	Prior to uprate implementation	Yes	No
Maintenance personnel will be qualified on LEFM system.	Prior to work on system	No	Yes
Operator training will be completed prior to implementation of the proposed power uprate changes.	Prior to uprate implementation	Yes	No
Plant testing for the proposed power uprate changes will be completed as described in Attachment 6, Section 10.4, "Testing." (Ref. 1)	As described	Yes	No
Plant-specific analyses for all potentially limiting events will be performed on a cycle-specific basis as part of the reload licensing process,	Prior to uprate implementation	Yes	No

7.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (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 amendments will not be inimical to the common defense and security or to the health and safety of the public.

8.0 REFERENCES

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4. Cameron Topical Report ER-157P, "Supplement to Topical Report ER-80P: Basis for a Power Uprate with the LEFM $\sqrt{\text{TM}}$ or LEFM CheckPlusTM System," Revision 5, October 2001 (Non-public).

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15. NRC, "Setpoints for Safety-Related Instrumentation," Regulatory Guide 1.105, Revision 3, December 1999.
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18. Technical Specification Task Force (TSTF)-493, "Clarify Application of Setpoint Methodology for LSSS Functions," Revision 4.
19. Licensing Topical Report, General Electric Nuclear Energy, NEDO-32938, "Generic Guidelines and Evaluations for General Electric Boiling Water Reactor Thermal Power Optimization, Revision 1," November 2002. (ADAMS Accession No. ML023170607 (nonproprietary)).

20. Letter from H. N. Berkow, NRC, to J. F. Klapproth, [General Electric] Nuclear Energy, "Review of General Electric Nuclear Energy Licensing Topical Report NEDC-32938P, 'Generic Guidelines and Evaluations for General Electric Boiling-Water Reactor Thermal Power Optimization' (TAC No. MA9537)," dated April 1, 2003. (ADAMS Accession No. ML031050138 (non-proprietary)).

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Date: September 16, 2010

A copy of the Safety Evaluation is also enclosed. The Notice of Issuance will be included in the Commission's biweekly *Federal Register* notice.

Sincerely,

/RA/

Christopher Gratton, Sr. Project Manager
Plant Licensing Branch III-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-373 and 50-374

Enclosures:

1. Amendment No. 198 to NPF-11
2. Amendment No. 185 to NPF-18
3. Safety Evaluation

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