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

April 8, 2011

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

SUBJECT: LIMERICK GENERATING STATION, UNITS 1 AND 2 - ISSUANCE OF AMENDMENTS RE: MEASUREMENT UNCERTAINTY RECAPTURE POWER UPRATE AND STANDBY LIQUID CONTROL SYSTEM CHANGES (TAC NOS. ME3589, ME3590, ME3591, AND ME3592)

Dear Mr. Pacilio:

The U.S. Nuclear Regulatory Commission (NRC or the Commission) has issued the enclosed Amendment No. 201 to Facility Operating License No. NPF-39 and Amendment No. 163 to Facility Operating License No. NPF-85 for Limerick Generating Station (LGS), Units 1 and 2, respectively. The amendments are in response to your application dated March 25, 2010,[1] as supplemented by additional letters.[2]

The amendments revise the facility operating license and certain technical specifications to implement an increase of approximately 1.65 percent in rated thermal power from the current licensed thermal power of 3458 megawatts thermal (MWt) to 3515 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 LGS, Unit 1 and will be installed in LGS, Unit 2 in refueling outage number 11 (Li2R11), currently scheduled to be completed in April 2011. The amendments also reflect changes to the standby liquid control system (SLCS) on each unit. The SLCS changes involve the installation of a modified hand switch that allows operators to select two pumps for the automatic start function on an Anticipated Transient Without Scram (ATWS) signal as compared to the current configuration where three pumps receive an automatic start signal. This change ensures that pressure in the discharge header of the SLCS pumps does not challenge the relief valve(s) setpoint while preserving the assumptions of the analysis for the ATWS event performed at the uprated power level.

Additionally, by letters dated August 30, 2010, and December 17, 2010, in response to requests for additional information from the NRC staff, you submitted information relating to the requirements of Title 10 of the *Code of Federal Regulations* Part 50, Appendix G, paragraph V.1.a. It was necessary to resolve these issues in order for the NRC staff to complete its review of the requested power uprate amendments. The NRC staff has reviewed your analyses and

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1. Agencywide Documents Access and Management System (ADAMS) Package No. ML100850379.
2. April 26, 2010 (ADAMS Accession No. ML101180250); June 29, 2010 (ML101820088); July 22, 2010 (ML102070196); July 28, 2010 (ML102110046); July 28, 2010 (ML102110047); August 10, 2010 (ML102240259); August 12, 2010 (ML102240330); August 12, 2010 (ML102280332); August 30, 2010 (ML102440265); December 17, 2010 (ML103560667); and January 7, 2011 (ML110100447).

concludes that all materials from the LGS, Units 1 and 2, Reactor Pressure Vessels which require evaluation in accordance with 10 CFR Part 50 Appendix G for upper-shelf energy, have been demonstrated to meet the requirements of 10 CFR Part 50, Appendix G, either by having their upper shelf energy values comply with criteria provided in the regulation, or by having an acceptable equivalent margins analysis (EMA) performed for them, as permitted by the regulation. Specifically, the NRC staff approves the EMAs submitted for the low-pressure coolant injection nozzles and their associated welds, at the uprated (3515 MWt) power level, up to 32 effective full-power years, for LGS, Units 1 and 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,

Peter Bamford

Peter Bamford, Project Manager
Plant Licensing Branch I-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-352 and 50-353

Enclosures:
1. Amendment No. 201 to License No. NPF-39
2. Amendment No. 163 to License No. NPF-85
3. Safety Evaluation

cc w/enclosures: Distribution via Listserv

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

EXELON GENERATION COMPANY, LLC

DOCKET NO. 50-352

LIMERICK GENERATING STATION, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 201
License No. NPF-39

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Exelon Generation Company, LLC (the licensee), dated March 25, 2010,[1] as supplemented by additional letters,[2] complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and 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 rules and 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;

   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 paragraphs 2.C.(1) and 2.C.(2) of Facility Operating License No. NPF-39 are hereby amended to read as follows:

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1. Agencywide Documents Access and Management System (ADAMS) Package No. ML100850379.
2. April 26, 2010 (ADAMS Accession No. ML101180250); June 29, 2010 (ML101820088); July 22, 2010 (ML102070196); July 28, 2010 (ML102110046); July 28, 2010 (ML102110047); August 10, 2010 (ML102240259); August 12, 2010 (ML102240330); August 12, 2010 (ML102280332); August 30, 2010 (ML102440265); December 17, 2010 (ML103560667); and January 7, 2011 (ML110100447).

(1) Maximum Power Level

Exelon Generation Company is authorized to operate the facility at reactor core power levels not in excess of 3515 megawatts thermal (100% rated power) in accordance with the conditions specified herein and in Attachment 1 to this license. The items identified in Attachment 1 to this license shall be completed as specified. Attachment 1 is hereby incorporated into this license.

(2) Technical Specifications

The Technical Specifications contained in Appendix A and the Environmental Protection Plan contained in Appendix B, as revised through Amendment No. 201, are hereby incorporated into this license. Exelon Generation Company shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Joseph G Giitter

Joseph 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: April 8, 2011

ATTACHMENT TO LICENSE AMENDMENT NO. 201

FACILITY OPERATING LICENSE NO. NPF-39

DOCKET NO. 50-352

Replace the following page of the Facility Operating License with the revised page. The revised page is identified by amendment number and contains marginal lines indicating the area of change.

Remove	Insert
Page 3	Page 3

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

Remove	Insert
1-6	1-6
2-4	2-4
3/4 1-19	3/4 1-19
3/4 1-20	3/4 1-20
3/4 3-5	3/4 3-5
3/4 3-7	3/4 3-7
3/4 3-8	3/4 3-8
---	3/4 3-8a
3/4 3-46	3/4 3-46
3/4 3-48	3/4 3-48
3/4 3-60	3/4 3-60
3/4 4-1	3/4 4-1
3/4 4-2	3/4 4-2

(3) Pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess and use at any time any byproduct, source and special nuclear material as sealed neutron sources for reactor startup, sealed sources for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as required;

(4) Pursuant to the Act and 10 CFR Parts 30, 40, 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

(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 the facility, and to receive and possess, but not separate, such source, byproduct, and special nuclear materials as contained in the fuel assemblies and fuel channels from the Shoreham Nuclear Power Station.

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 (except as exempted from compliance in Section 2.D. below) 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

Exelon Generation Company is authorized to operate the facility at reactor core power levels not in excess of 3515 megawatts thermal (100% rated power) in accordance with the conditions specified herein and in Attachment 1 to this license. The items identified in Attachment 1 to this license shall be completed as specified. Attachment 1 is hereby incorporated into this license.

(2) Technical Specifications

The Technical Specifications contained in Appendix A and the Environmental Protection Plan contained in Appendix B, as revised through Amendment No.201 , are hereby incorporated into this license. Exelon Generation Company shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

PURGE - PURGING

1.31 PURGE or PURGING shall be the controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration or other operating condition, in such a manner that replacement air or gas is required to purify the confinement.

RATED THERMAL POWER

1.32 RATED THERMAL POWER shall be a total reactor core heat transfer rate to the reactor coolant of 3515 MWt.

REACTOR ENCLOSURE SECONDARY CONTAINMENT INTEGRITY

1.33 REACTOR ENCLOSURE SECONDARY CONTAINMENT INTEGRITY shall exist when:

a. All reactor enclosure secondary containment penetrations required to be closed during accident conditions are either:

   1. Capable of being closed by an OPERABLE secondary containment automatic isolation system, or

   2. Closed by at least one manual valve, blind flange, slide gate damper, or deactivated automatic valve secured in its closed position, except as provided by Specification 3.6.5.2.1.

b. All reactor enclosure secondary containment hatches and blowout panels are closed and sealed.

c. The standby gas treatment system is in compliance with the requirements of Specification 3.6.5.3.

d. The reactor enclosure recirculation system is in compliance with the requirements of Specification 3.6.5.4.

e. At least one door in each access to the reactor enclosure secondary containment is closed.

f. The sealing mechanism associated with each reactor enclosure secondary containment penetration, e.g., welds, bellows, or O-rings, is OPERABLE.

g. The pressure within the reactor enclosure secondary containment is less than or equal to the value required by Specification 4.6.5.1.1a.

REACTOR PROTECTION SYSTEM RESPONSE TIME

1.34 REACTOR PROTECTION SYSTEM RESPONSE TIME shall be the time interval from when the monitored parameter exceeds its trip setpoint at the channel sensor until de-energization of the scram pilot valve solenoids. The response time may be measured by any series of sequential, overlapping or total steps such that the entire response time is measured.

RECENTLY IRRADIATED FUEL

1.35 RECENTLY IRRADIATED FUEL is fuel that has occupied part of a critical reactor core within the previous 24 hours.

REFUELING FLOOR SECONDARY CONTAINMENT INTEGRITY

1.36 REFUELING FLOOR SECONDARY CONTAINMENT INTEGRITY shall exist when:

a. All refueling floor secondary containment penetrations required to be closed during accident conditions are either:

TABLE 2.2.1-1

REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
1. Intermediate Range Monitor, Neutron Flux-High	≤ 120/125 divisions of full scale	≤ 122/125 divisions of full scale
2. Average Power Range Monitor:		
a. Neutron Flux-Upscale (Setdown)	≤ 15.0% of RATED THERMAL POWER	≤ 20.0% of RATED THERMAL POWER
b. Simulated Thermal Power - Upscale:		
- Two Recirculation Loop Operation	≤ 0.65 W + 61.7% and ≤ 116.6% of RATED THERMAL POWER	≤ 0.65 W + 62.2% and ≤ 117.0% of RATED THERMAL POWER
- Single Recirculation Loop Operation***	≤ 0.65 (W-7.6%) + 61.5% and ≤ 116.6% of RATED THERMAL POWER	≤ 0.65 (W-7.6%) + 62.0% and ≤ 117.0% of RATED THERMAL POWER
c. Neutron Flux - Upscale	118.3% of RATED THERMAL POWER	118.7% of RATED THERMAL POWER
d. Inoperative	N.A.	N.A.
e. 2-Out-Of-4 Voter	N.A.	N.A.
f. OPRM Upscale	****	N.A.
3. Reactor Vessel Steam Dome Pressure - High	≤ 1096 psig	≤ 1103 psig
4. Reactor Vessel Water Level - Low, Level 3	≥ 12.5 inches above instrument zero*	≥ 11.0 inches above instrument zero
5. Main Steam Line Isolation Valve - Closure	≤ 8% closed	≤ 12% closed
6. DELETED	DELETED	DELETED
7. Drywell Pressure - High	≤ 1.68 psig	≤ 1.88 psig
8. Scram Discharge Volume Water Level - High		
a. Level Transmitter	≤ 260' 9 5/8" elevation**	≤ 261' 5 5/8" elevation
b. Float Switch	≤ 260' 9 5/8" elevation**	≤ 261' 5 5/8" elevation
9. Turbine Stop Valve - Closure	≤ 5% closed	≤7% closed
10. Turbine Control Valve Fast Closure, Trip Oil Pressure - Low	≥ 500 psig	≥ 465 psig
11. Reactor Mode Switch Shutdown Position	N.A.	N.A.
12. Manual Scram	N.A.	N.A.

* See Bases Figure B 3/4.3-1.
** Equivalent to 25.45 gallons/scram discharge volume.
*** The 7.6% flow "offset" for Single Loop Operation (SLO) is applied for W ≥ 7.6%. For flows W < 7.6%, the (W-7.6%) term is set equal to zero.
**** See COLR for OPRM period based detection algorithm trip setpoints. OPRM Upscale trip output auto-enable (not bypassed) setpoints shall be APRM Simulated Thermal Power ≥ 29.5% and recirculation drive flow < 60%.

REACTIVITY CONTROL SYSTEMS

3/4.1.5 STANDBY LIQUID CONTROL SYSTEM

LIMITING CONDITION FOR OPERATION

3.1.5 The standby liquid control system shall be OPERABLE and consist of the following:

a. In OPERATIONAL CONDITIONS 1 and 2, two pumps and corresponding flow paths,

b. In OPERATIONAL CONDITION 3, a minimum of one pump and corresponding flow path.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3

ACTION:

a. With only one pump and corresponding explosive valve OPERABLE, in OPERATIONAL CONDITION 1 or 2, restore one inoperable pump and corresponding explosive valve to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours.

b. With standby liquid control system otherwise inoperable, in OPERATIONAL CONDITION 1, 2, or 3, restore the system to OPERABLE status within 8 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the next 24 hours.

SURVEILLANCE REQUIREMENTS

4.1.5 The standby liquid control system shall be demonstrated OPERABLE:

a. In accordance with the Surveillance Frequency Control Program by verifying that:

   1. The temperature of the sodium pentaborate solution is within the limits of Figure 3.1.5-1.

   2. The available volume of sodium pentaborate solution is at least 3160 gallons.

   3. The temperature of the pump suction piping is within the limits of Figure 3.1.5-1 for the most recent concentration analysis.

b. In accordance with the Surveillance Frequency Control Program by:

1. Verifying the continuity of the explosive charge.

2. Determining by chemical analysis and calculation* that the available weight of Boron-10 is greater than or equal to 185 lbs; the concentration of sodium pentaborate in solution is less than or equal to 13.8% and within the limits of Figure 3.1.5-1 and; the following equation is satisfied:

$$\frac{C}{13\%\ \text{wt.}} \times \frac{E}{29\ \text{atom}\ \%} \times \frac{Q}{86\ \text{gpm}} \geq 1$$

where

C = Sodium pentaborate solution (% by weight)
Q = Two pump flowrate, as determined per surveillance requirement 4.1.5.c.
E = Boron 10 enrichment (atom % Boron 10)

3. Verifying that each valve (manual, power-operated, or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position.

4. Verifying that no more than two pumps are aligned for automatic operation.

c. Demonstrating that, when tested pursuant to Specification 4.0.5, the minimum flow requirement of 41.2 gpm per pump at a pressure of greater than or equal to 1230 ± 25 psig is met.

d. In accordance with the Surveillance Frequency Control Program by:

1. Initiating at least one of the standby liquid control system loops, including an explosive valve, and verifying that a flow path from the pumps to the reactor pressure vessel is available by pumping demineralized water into the reactor vessel. The replacement charge for the explosive valve shall be from the same manufactured batch as the one fired or from another batch which has been certified by having one of the batch successfully fired. All injection loops shall be tested in 3 operating cycles.

2. Verify all heat-treated piping between storage tank and pump suction is unblocked.**

e. Prior to addition of Boron to storage tank verify sodium pentaborate enrichment to be added is ≥ 29 atom % Boron 10.

* This test shall also be performed anytime water or boron is added to the solution or when the solution temperature drops below the limits of Figure 3.1.5-1 for the most recent concentration analysis, within 24 hours after water or boron addition or solution temperature is restored.

** This test shall also be performed whenever suction piping temperature drops below the limits of Figure 3.1.5-1 for the most recent concentration analysis, within 24 hours after solution temperature is restored.

TABLE 3.3.1-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION

TABLE NOTATIONS

(a) A channel may be placed in an inoperable status for up to 6 hours for required surveillance without placing the trip system in the tripped condition provided at least one OPERABLE channel in the same trip system is monitoring that parameter.

(b) This function shall be automatically bypassed when the reactor mode switch is in the Run position.

(c) DELETED

(d) The noncoincident NMS reactor trip function logic is such that all channels go to both trip systems. Therefore, when the "shorting links" are removed, the Minimum OPERABLE Channels Per Trip System is 6 IRMs.

(e) An APRM channel is inoperable if there are less than 3 LPRM inputs per level or less than 20 LPRM inputs to an APRM channel, or if more than 9 LPRM inputs to the APRM channel have been bypassed since the last APRM calibration (weekly gain calibration).

(f) This function is not required to be OPERABLE when the reactor pressure vessel head is removed per Specification 3.10.1.

(g) This function shall be automatically bypassed when the reactor mode switch is not in the Run position.

(h) This function is not required to be OPERABLE when PRIMARY CONTAINMENT INTEGRITY is not required.

(i) With any control rod withdrawn. Not applicable to control rods removed per Specification 3.9.10.1 or 3.9.10.2.

(j) This function shall be automatically bypassed when turbine first stage pressure is equivalent to a THERMAL POWER of less than 29.5% of RATED THERMAL POWER.

(k) Also actuates the EOC-RPT system.

(l) DELETED

(m) Each APRM channel provides inputs to both trip systems.

(n) DELETED

(o) With THERMAL POWER ≥ 25% RATED THERMAL POWER. The OPRM Upscale trip output shall be automatically enabled (not bypassed) when APRM Simulated Thermal Power is ≥ 29.5% and recirculation drive flow is < 60%. The OPRM trip output may be automatically bypassed when APRM Simulated Thermal Power is < 29.5% or recirculation drive flow is ≥ 60%.

(p) A minimum of 23 cells, each with a minimum of 2 OPERABLE LPRMs, must be OPERABLE for an OPRM channel to be OPERABLE.

TABLE 4.3.1.1-1

REACTOR PROTECTION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK(n)	CHANNEL FUNCTIONAL TEST(n)	CHANNEL CALIBRATION(a)(n)	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
1. Intermediate Range Monitors:				
a. Neutron Flux - High	(b)	 (j)		2 3(i), 4(i), 5(i)
b. Inoperative	N.A.	(j)	N.A.	2, 3(i), 4(i), 5(i)
2. Average Power Range Monitor(f):				
a. Neutron Flux - Upscale (Setdown)	(b)	(l)		2
b. Simulated Thermal Power - Upscale		(e)	(d), (g), (o), (p)	1
c. Neutron Flux - Upscale			(d)	1
d. Inoperative	N.A.		N.A.	1, 2
e. 2-Out-Of-4 Voter			N.A.	1, 2
f. OPRM Upscale		(e)	(c)(g)	1(m)
3. Reactor Vessel Steam Dome Pressure - High				1, 2(h)
4. Reactor Vessel Water Level- Low, Level 3				1, 2
5. Main Steam Line Isolation Valve - Closure	N.A.			1
6. DELETED				
7. Drywell Pressure - High				1, 2
8. Scram Discharge Volume Water Level - High				
a. Level Transmitter				1, 2, 5(i)
b. Float Switch	N.A.			1, 2, 5(i)

REACTOR PROTECTION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK(n)	CHANNEL FUNCTIONAL TEST(n)	CHANNEL CALIBRATION(a)(n)	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
9. Turbine Stop Valve - Closure	N.A.			1
10. Turbine Control Valve Fast Closure, Trip Oil Pressure - Low	N.A.			1
11. Reactor Mode Switch Shutdown Position	N.A.		N.A.	1, 2, 3, 4, 5
12. Manual Scram	N.A.		N.A.	1, 2, 3, 4, 5

(a) Neutron detectors may be excluded from CHANNEL CALIBRATION.
(b) The IRM and SRM channels shall be determined to overlap for at least 1/2 decades during each startup after entering OPERATIONAL CONDITION 2 and the IRM and APRM channels shall be determined to overlap for at least 1/2 decades during each controlled shutdown, if not performed within the previous 7 days.
(c) Calibration includes verification that the OPRM Upscale trip auto-enable (not-bypass) setpoint for APRM Simulated Thermal Power is ≥ 29.5% and for recirculation drive flow is < 60%.
(d) The more frequent calibration shall consist of the adjustment of the APRM channel to conform to the power values calculated by a heat balance during OPERATIONAL CONDITION 1 when THERMAL POWER ≥25% of RATED THERMAL POWER. Adjust the APRM channel if the absolute difference is greater than 2% of RATED THERMAL POWER.
(e) CHANNEL FUNCTIONAL TEST shall include the flow input function, excluding the flow transmitter.
(f) The LPRMs shall be calibrated at least once per 2000 effective full power hours (EFPH).
(g) The less frequent calibration includes the flow input function.
(h) This function is not required to be OPERABLE when the reactor pressure vessel head is removed per Specification 3.10.1.
(i) With any control rod withdrawn. Not applicable to control rods removed per Specification 3.9.10.1 or 3.9.10.2.
(j) If the RPS shorting links are required to be removed per Specification 3.9.2, they may be reinstalled for up to 2 hours for required surveillance. During this time, CORE ALTERATIONS shall be suspended, and no control rod shall be moved from its existing position.
(k) DELETED
(l) Not required to be performed when entering OPERATIONAL CONDITION 2 from OPERATIONAL CONDITION 1 until 12 hours after entering OPERATIONAL CONDITION 2.
(m) With THERMAL POWER ≥ 25% of RATED THERMAL POWER.
(n) Frequencies are specified in the Surveillance Frequency Control Program unless otherwise noted in the table.

TABLE 4.3.1.1-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

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

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

END-OF-CYCLE RECIRCULATION PUMP TRIP SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.4.2 The end-of-cycle recirculation pump trip (EOC-RPT) system instrumentation channels shown in Table 3.3.4.2-1 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3.4.2-2 and with the END-OF-CYCLE RECIRCULATION PUMP TRIP SYSTEM RESPONSE TIME as shown in Table 3.3.4.2-3.

APPLICABILITY: OPERATIONAL CONDITION 1, when THERMAL POWER is greater than or equal to 29.5% of RATED THERMAL POWER.

ACTION:

a. With an end-of-cycle recirculation pump trip system instrumentation channel trip setpoint less conservative than the value shown in the Allowable Values column of Table 3.3.4.2-2, declare the channel inoperable until the channel is restored to OPERABLE status with the channel setpoint adjusted consistent with the Trip Setpoint value.

b. With the number of OPERABLE channels one less than required by the Minimum OPERABLE Channels per Trip System requirement for one or both trip systems, place the inoperable channel(s) in the tripped condition within 12 hours.

c. With the number of OPERABLE channels two or more less than required by the Minimum OPERABLE Channels per Trip System requirement for one trip system and:

    1. If the inoperable channels consist of one turbine control valve channel and one turbine stop valve channel, place both inoperable channels in the tripped condition within 12 hours.

    2. If the inoperable channels include two turbine control valve channels or two turbine stop valve channels, declare the trip system inoperable.

d. With one trip system inoperable, restore the inoperable trip system to OPERABLE status within 72 hours or take the ACTION required by Specification 3.2.3.

e. With both trip systems inoperable, restore at least one trip system to OPERABLE status within one hour or take the ACTION required by Specification 3.2.3.

TABLE 3.3.4.2-1

END-OF-CYCLE RECIRCULATION PUMP TRIP SYSTEM INSTRUMENTATION

TRIP FUNCTION	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM*
1. Turbine Stop Valve - Closure	2**
2. Turbine Control Valve-Fast Closure	2**

* A trip system may be placed in an inoperable status for up to 6 hours for required surveillance provided that the other trip system is OPERABLE.

** This function shall be automatically bypassed when turbine first stage pressure is equivalent to THERMAL POWER LESS than 29.5% of RATED THERMAL POWER.

TABLE 3.3.6-2

CONTROL ROD BLOCK INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
1. ROD BLOCK MONITOR		
a. Upscale[(a)]		
1) Low Trip Setpoint (LTSP)	*	*
2) Intermediate Trip Setpoint (ITSP)	*	*
3) High Trip Setpoint (HTSP)	*	*
b. Inoperative	N/A	N/A
c. Downscale (DTSP)	*	*
d. Power Range Setpoint[(b)]		
1) Low Power Setpoint (LPSP)	28.1% RATED THERMAL POWER	28.4% RATED THERMAL POWER
2) Intermediate Power Setpoint (IPSP)	63.1% RATED THERMAL POWER	63.4% RATED THERMAL POWER
3) High Power Setpoint (HPSP)	83.1% RATED THERMAL POWER	83.4% RATED THERMAL POWER
2. APRM		
a. Simulated Thermal Power - Upscale:		
- Two Recirculation Loop Operation	≤ 0.65 W + 54.3% and ≤ 108.0% of RATED THERMAL POWER	≤ 0.65 W + 54.7% and ≤ 108.4% of RATED THERMAL POWER
- Single Recirculation Loop Operation****	≤ 0.65 (W-7.6%) + 54.1% and ≤ 108.0% of RATED THERMAL POWER	≤ 0.65 (W-7.6%) + 54.5% and ≤ 108.4% of RATED THERMAL POWER
b. Inoperative	N.A.	N.A.
c. Neutron Flux - Downscale	≥ 3.2% of RATED THERMAL POWER	≥ 2.8% of RATED THERMAL POWER
d. Simulated Thermal Power - Upscale (Setdown)	≤ 12.0% of RATED THERMAL POWER	≤ 13.0% of RATED THERMAL POWER
e. Recirculation Flow - Upscale	*	*
f. LPRM Low Count	< 20 per channel < 3 per axial level	< 20 per channel < 3 per axial level
3. SOURCE RANGE MONITORS		
a. Detector not full in	N.A.	N.A.
b. Upscale	≤ 1 x 10^5 cps	≤ 1.6 x 10^5 cps
c. Inoperative	N.A.	N.A.
d. Downscale	≥ 3 cps**	≥ 1.8 cps**

3/4.4 REACTOR COOLANT SYSTEM

3/4.4.1 RECIRCULATION SYSTEM

RECIRCULATION LOOPS

LIMITING CONDITION FOR OPERATION

3.4.1.1 Two reactor coolant system recirculation loops shall be in operation.

APPLICABILITY: OPERATIONAL CONDITIONS 1* and 2*.

ACTION:

a. With one reactor coolant system recirculation loop not in operation:

   1. Within 4 hours:

      a. Place the recirculation flow control system in the Local Manual mode, and

      b. Reduce THERMAL POWER to ≤ 74.9% of RATED THERMAL POWER, and,

      c. Limit the speed of the operating recirculation pump to less than or equal to 90% of rated pump speed, and

      d. Verify that the differential temperature requirements of Surveillance Requirement 4.4.1.1.5 are met if THERMAL POWER is ≤ 30% of RATED THERMAL POWER or the recirculation loop flow in the operating loop is ≤ 50% of rated loop flow, or suspend the THERMAL POWER or recirculation loop flow increase.

*See Special Test Exception 3.10.4.

SURVEILLANCE REQUIREMENTS

4.4.1.1.1 DELETED

4.4.1.1.2 DELETED

4.4.1.1.3 DELETED

4.4.1.1.4 With one reactor coolant system recirculation loop not in operation, in accordance with the Surveillance Frequency Control Program, verify that:

a. Reactor THERMAL POWER is ≤ 74.9% of RATED THERMAL POWER,

b. The recirculation flow control system is in the Local Manual mode, and

c. The speed of the operating recirculation pump is ≤ 90% of rated pump speed.

4.4.1.1.5 With one reactor coolant system recirculation loop not in operation, within 15 minutes prior to either THERMAL POWER increase or recirculation loop flow increase, verify that the following differential temperature requirements are met if THERMAL POWER is ≤ 30% of RATED THERMAL POWER or the recirculation loop flow in the operating recirculation loop is ≤ 50% of rated loop flow.

a. ≤ 145°F between reactor vessel steam space coolant and bottom head drain line coolant,

b. ≤ 50°F between the reactor coolant within the loop not in operation and the coolant in the reactor pressure vessel, and

c. ≤ 50°F between the reactor coolant within the loop not in operation and the operating loop.

The differential temperature requirements of Specification 4.4.1.1.5b. and c. do not apply when the loop not in operation is isolated from the reactor pressure vessel.

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

EXELON GENERATION COMPANY, LLC

DOCKET NO. 50-353

LIMERICK GENERATING STATION, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 163
License No. NPF-85

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Exelon Generation Company, LLC (the licensee), dated March 25, 2010,[1] as supplemented by additional letters,[2] complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and 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 rules and 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;

   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 paragraphs 2.C.(1) and 2.C.(2) of Facility Operating License No. NPF-85 are hereby amended to read as follows:

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1. Agencywide Documents Access and Management System (ADAMS) Package No. ML100850379.
2. April 26, 2010 (ADAMS Accession No. ML101180250); June 29, 2010 (ML101820088); July 22, 2010 (ML102070196); July 28, 2010 (ML102110046); July 28, 2010 (ML102110047); August 10, 2010 (ML102240259); August 12, 2010 (ML102240330); August 12, 2010 (ML102280332); August 30, 2010 (ML102440265); December 17, 2010 (ML103560667) and January 7, 2011 (ML110100447).

(1) Maximum Power Level

Exelon Generation Company is authorized to operate the facility at reactor core power levels of 3515 megawatts thermal (100 percent rated power) in accordance with the conditions specified herein.

(2) Technical Specifications

The Technical Specifications contained in Appendix A and the Environmental Protection Plan contained in Appendix B, as revised through Amendment No. 163, are hereby incorporated into this license. Exelon Generation Company shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of its date of issuance and shall be implemented within 90 days of the completion of refueling outage Li2R11.

FOR THE NUCLEAR REGULATORY COMMISSION

Joseph G. Giitter

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: April 8, 2011

ATTACHMENT TO LICENSE AMENDMENT NO. 163

FACILITY OPERATING LICENSE NO. NPF-85

DOCKET NO. 50-353

Replace the following page of the Facility Operating License with the revised page. The revised page is identified by amendment number and contains marginal lines indicating the area of change.

Remove	Insert
Page 3	Page 3

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

Remove	Insert
1-6	1-6
2-4	2-4
3/4 1-19	3/4 1-19
3/4 1-20	3/4 1-20
3/4 3-5	3/4 3-5
3/4 3-7	3/4 3-7
3/4 3-8	3/4 3-8
---	3/4 3-8a
3/4 3-46	3/4 3-46
3/4 3-48	3/4 3-48
3/4 3-60	3/4 3-60
3/4 4-1	3/4 4-1
3/4 4-2	3/4 4-2

(4) Pursuant to the Act and 10 CFR Parts 30, 40, 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

(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 the facility, and to receive and possess, but not separate, such source, byproduct, and special nuclear materials as contained in the fuel assemblies and fuel channels from the Shoreham Nuclear Power Station.

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 (except as exempted from compliance in Section 2.D. below) 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

Exelon Generation Company is authorized to operate the facility at reactor core power levels of 3515 megawatts thermal (100 percent rated power) in accordance with the conditions specified herein.

(2) Technical Specifications

The Technical Specifications contained in Appendix A and the Environmental Protection Plan contained in Appendix B, as revised through Amendment No. 163, are hereby incorporated into this license. Exelon Generation Company shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

(3) Fire Protection (Section 9.5, SSER-2, -4)*

Exelon Generation Company shall implement and maintain in effect all provisions of the approved Fire Protection Program as described in the Updated Final Safety Analysis Report for the facility, and as approved in the NRC Safety Evaluation Report dated August 1983 through Supplement 9, dated August 1989, and Safety Evaluation dated November 20, 1995, subject to the following provision:

The licensee may make changes to the approved fire protection program without prior approval of the Commission only if those changes would not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire.

*The parenthetical notation following the title of license conditions denotes the section of the Safety Evaluation Report and/or its supplements wherein the license condition is discussed.

PURGE - PURGING

1.31 PURGE or PURGING shall be the controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration or other operating condition, in such a manner that replacement air or gas is required to purify the confinement.

RATED THERMAL POWER

1.32 RATED THERMAL POWER shall be a total reactor core heat transfer rate to the reactor coolant of 3515 MWt.

REACTOR ENCLOSURE SECONDARY CONTAINMENT INTEGRITY

1.33 REACTOR ENCLOSURE SECONDARY CONTAINMENT INTEGRITY shall exist when:

a. All reactor enclosure secondary containment penetrations required to be closed during accident conditions are either:

1. Capable of being closed by an OPERABLE secondary containment automatic isolation system, or

2. Closed by at least one manual valve, blind flange, slide gate damper or deactivated automatic valve secured in its closed position, except as provided by Specification 3.6.5.2.1.

b. All reactor enclosure secondary containment hatches and blowout panels are closed and sealed.

c. The standby gas treatment system is in compliance with the requirements of Specification 3.6.5.3.

d. The reactor enclosure recirculation system is in compliance with the requirements of Specification 3.6.5.4.

e. At least one door in each access to the reactor enclosure secondary containment is closed.

f. The sealing mechanism associated with each reactor enclosure secondary containment penetration, e.g., welds, bellows, or O-rings, is OPERABLE.

g. The pressure within the reactor enclosure secondary containment is less than or equal to the value required by Specification 4.6.5.1.1a.

REACTOR PROTECTION SYSTEM RESPONSE TIME

1.34 REACTOR PROTECTION SYSTEM RESPONSE TIME shall be the time interval from when the monitored parameter exceeds its trip setpoint at the channel sensor until de-energization of the scram pilot valve solenoids. The response time may be measured by any series of sequential, overlapping or total steps such that the entire response time is measured.

RECENTLY IRRADIATED FUEL

1.35 RECENTLY IRRADIATED FUEL is fuel that has occupied part of a critical reactor core within the previous 24 hours.

REFUELING FLOOR SECONDARY CONTAINMENT INTEGRITY

1.36 REFUELING FLOOR SECONDARY CONTAINMENT INTEGRITY shall exist when:

a. All refueling floor secondary containment penetrations required to be closed during accident conditions are either:

TABLE 2.2.1-1

REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
1. Intermediate Range Monitor, Neutron Flux-High	≤ 120/125 divisions of full scale	≤ 122/125 divisions of full scale
2. Average Power Range Monitor:		
a. Neutron Flux-Upscale (Setdown)	≤ 15.0% of RATED THERMAL POWER	≤ 20.0% of RATED THERMAL POWER
b. Simulated Thermal Power - Upscale:		
- Two Recirculation Loop Operation	≤ 0.65 W + 61.7% and ≤ 116.6% of RATED THERMAL POWER	≤ 0.65 W + 62.2% and ≤ 117.0% of RATED THERMAL POWER
- Single Recirculation Loop Operation***	≤ 0.65 (W-7.6%) + 61.5% and ≤ 116.6% of RATED THERMAL POWER	≤ 0.65 (W-7.6%) + 62.0% and ≤ 117.0% of RATED THERMAL POWER
c. Neutron Flux - Upscale	118.3% of RATED THERMAL POWER	118.7% of RATED THERMAL POWER
d. Inoperative	N.A.	N.A.
e. 2-Out-Of-4 Voter	N.A.	N.A.
f. OPRM Upscale	****	N.A.
3. Reactor Vessel Steam Dome Pressure - High	≤ 1096 psig	≤ 1103 psig
4. Reactor Vessel Water Level - Low, Level 3	≥ 12.5 inches above instrument zero*	≥ 11.0 inches above instrument zero
5. Main Steam Line Isolation Valve - Closure	≤ 8% closed	≤ 12% closed
6. DELETED	DELETED	DELETED
7. Drywell Pressure - High	≤ 1.68 psig	≤ 1.88 psig
8. Scram Discharge Volume Water Level - High		
a. Level Transmitter	≤ 261' 1 1/4" elevation**	≤ 261' 9 1/4" elevation
b. Float Switch	≤ 261' 1 1/4" elevation**	≤ 261' 9 1/4" elevation

* See Bases Figure B 3/4.3-1.

** Equivalent to 25.58 gallons/scram discharge volume.

*** The 7.6% flow "offset" for Single Loop Operation (SLO) is applied for W ≥ 7.6%. For flows W < 7.6%, the (W-7.6%) term is set equal to zero.

**** See COLR for OPRM period based detection algorithm trip setpoints. OPRM Upscale trip output auto-enable (not bypassed) setpoints shall be APRM Simulated Thermal Power ≥ 29.5% and recirculation drive flow < 60%.

3/4.1.5 STANDBY LIQUID CONTROL SYSTEM

LIMITING CONDITION FOR OPERATION

3.1.5 The standby liquid control system shall be OPERABLE and consist of the following:

a. In OPERATIONAL CONDITIONS 1 and 2, two pumps and corresponding flow paths,

b. In OPERATIONAL CONDITION 3, a minimum of one pump and corresponding flow path.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3

ACTION:

a. With only one pump and corresponding explosive valve OPERABLE, in OPERATIONAL CONDITION 1 or 2, restore one inoperable pump and corresponding explosive valve to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours.

b. With standby liquid control system otherwise inoperable, in OPERATIONAL CONDITION 1, 2, or 3, restore the system to OPERABLE status within 8 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the next 24 hours.

SURVEILLANCE REQUIREMENTS

4.1.5 The standby liquid control system shall be demonstrated OPERABLE:

a. In accordance with the Surveillance Frequency Control Program by verifying that:

   1. The temperature of the sodium pentaborate solution is within the limits of Figure 3.1.5-1.

   2. The available volume of sodium pentaborate solution is at least 3160 gallons.

   3. The temperature of the pump suction piping is within the limits of Figure 3.1.5-1 for the most recent concentration analysis.

b. In accordance with the Surveillance Frequency Control Program by:

1. Verifying the continuity of the explosive charge.

2. Determining by chemical analysis and calculation* that the available weight of Boron-10 is greater than or equal to 185 lbs; the concentration of sodium pentaborate in solution is less than or equal to 13.8% and within the limits of Figure 3.1.5-1 and; the following equation is satisfied:

$$\frac{C}{13\%\ \text{wt.}} \times \frac{E}{29\ \text{atom}\ \%} \times \frac{Q}{86\ \text{gpm}} \geq 1$$

where

C = Sodium pentaborate solution (% by weight)
Q = Two pump flowrate, as determined per surveillance requirement 4.1.5.c.
E = Boron 10 enrichment (atom % Boron 10)

3. Verifying that each valve (manual, power-operated, or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position.

4. Verifying that no more than two pumps are aligned for automatic operation.

c. Demonstrating that, when tested pursuant to Specification 4.0.5, the minimum flow requirement of 41.2 gpm per pump at a pressure of greater than or equal to 1230±25 psig is met.

d. In accordance with the Surveillance Frequency Control Program by:

1. Initiating at least one of the standby liquid control system loops, including an explosive valve, and verifying that a flow path from the pumps to the reactor pressure vessel is available by pumping demineralized water into the reactor vessel. The replacement charge for the explosive valve shall be from the same manufactured batch as the one fired or from another batch which has been certified by having one of the batch successfully fired. All injection loops shall be tested in 3 operating cycles.

2. Verify all heat-treated piping between storage tank and pump suction is unblocked.**

e. Prior to addition of Boron to storage tank verify sodium pentaborate enrichment to be added is ≥ 29 atom % Boron 10.

* This test shall also be performed anytime water or boron is added to the solution or when the solution temperature drops below the limits of Figure 3.1.5-1 for the most recent concentration analysis, within 24 hours after water or boron addition or solution temperature is restored.

** This test shall also be performed whenever suction piping temperature drops below the limits of Figure 3.1.5-1 for the most recent concentration analysis, within 24 hours after solution temperature is restored.

TABLE 3.3.1-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION

TABLE NOTATIONS

(a) A channel may be placed in an inoperable status for up to 6 hours for required surveillance without placing the trip system in the tripped condition provided at least one OPERABLE channel in the same trip system is monitoring that parameter.

(b) This function shall automatically be bypassed when the reactor mode switch is in the Run position.

(c) DELETED

(d) The noncoincident NMS reactor trip function logic is such that all channels go to both trip systems. Therefore, when the "shorting links" are removed, the Minimum OPERABLE Channels Per Trip System is 6 IRMs.

(e) An APRM channel is inoperable if there are less than 3 LPRM inputs per level or less than 20 LPRM inputs to an APRM channel, or if more than 9 LPRM inputs to the APRM channel have been bypassed since the last APRM calibration (weekly gain calibration).

(f) This function is not required to be OPERABLE when the reactor pressure vessel head is removed per Specification 3.10.1.

(g) This function shall be automatically bypassed when the reactor mode switch is not in the Run position.

(h) This function is not required to be OPERABLE when PRIMARY CONTAINMENT INTEGRITY is not required.

(i) With any control rod withdrawn. Not applicable to control rods removed per Specification 3.9.10.1 or 3.9.10.2.

(j) This function shall be automatically bypassed when turbine first stage pressure is equivalent to a THERMAL POWER of less than 29.5% of RATED THERMAL POWER.

(k) Also actuates the EOC-RPT system.

(l) DELETED

(m) Each APRM channel provides inputs to both trip systems.

(n) DELETED

(o) With THERMAL POWER ≥ 25% RATED THERMAL POWER. The OPRM Upscale trip output shall be automatically enabled (not bypassed) when APRM Simulated Thermal Power is ≥ 29.5% and recirculation drive flow is < 60%. The OPRM trip output may be automatically bypassed when APRM Simulated Thermal Power is < 29.5% or recirculation drive flow is ≥ 60%.

(p) A minimum of 23 cells, each with a minimum of 2 OPERABLE LPRMs, must be OPERABLE for an OPRM channel to be OPERABLE.

TABLE 4.3.1.1-1

REACTOR PROTECTION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK (n)	CHANNEL FUNCTIONAL TEST (n)	CHANNEL CALIBRATION(a)(n)	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
1. Intermediate Range Monitors:				
a. Neutron Flux - High	(b)			2
		(j)		3(i), 4(i), 5(i)
b. Inoperative	N.A.	(j)	N.A.	2, 3(i), 4(i), 5(i)
2. Average Power Range Monitor(f):				
a. Neutron Flux - Upscale (Setdown)	(b)	(l)		2
b. Simulated Thermal Power - Upscale		(e)	(d), (g), (o), (p)	1
c. Neutron Flux - Upscale			(d)	1
d. Inoperative	N.A.		N.A.	1, 2
e. 2-Out-Of-4 Voter			N.A.	1, 2
f. OPRM Upscale		(e)	(c)(g)	1(m)
3. Reactor Vessel Steam Dome Pressure - High				1, 2(h)
4. Reactor Vessel Water Level-Low, Level 3				1, 2
5. Main Steam Line Isolation Valve - Closure	N.A.			1
6. DELETED				
7. Drywell Pressure - High				1, 2
8. Scram Discharge Volume Water Level - High				
a. Level Transmitter				1, 2, 5(i)
b. Float Switch	N.A.			1, 2, 5(i)

TABLE 4.3.1.1-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK (n)	CHANNEL FUNCTIONAL TEST (n)	CHANNEL CALIBRATION(a)(n)	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
9. Turbine Stop Valve - Closure	N.A.			1
10. Turbine Control Valve Fast Closure, Trip Oil Pressure - Low	N.A.			1
11. Reactor Mode Switch Shutdown Position	N.A.		N.A.	1, 2, 3, 4, 5
12. Manual Scram	N.A.		N.A.	1, 2, 3, 4, 5

(a) Neutron detectors may be excluded from CHANNEL CALIBRATION.
(b) The IRM and SRM channels shall be determined to overlap for at least 1/2 decades during each startup after entering OPERATIONAL CONDITION 2 and the IRM and APRM channels shall be determined to overlap for a least 1/2 decades during each controlled shutdown, if not performed within the previous 7 days.
(c) Calibration includes verification that the OPRM Upscale trip auto-enable (not-bypass) setpoint for APRM Simulated Thermal Power is ≥ 29.5% and for recirculation drive flow is < 60%.
(d) The more frequent calibration shall consist of the adjustment of the APRM channel to conform to the power values calculated by a heat balance during OPERATIONAL CONDITION 1 when THERMAL POWER ≥ 25% of RATED THERMAL POWER. Adjust the APRM channel if the absolute difference is greater than 2% of RATED THERMAL POWER.
(e) CHANNEL FUNCTIONAL TEST shall include the flow input function, excluding the flow transmitter.
(f) The LPRMs shall be calibrated at least once per 2000 effective full power hours (EFPH).
(g) The less frequent calibration includes the flow input function.
(h) This function is not required to be OPERABLE when the reactor pressure vessel head is removed per Specification 3.10.1.
(i) With any control rod withdrawn. Not applicable to control rods removed per Specification 3.9.10.1 or 3.9.10.2.
(j) If the RPS shorting links are required to be removed per Specification 3.9.2, they may be reinstalled for up to 2 hours for required surveillance. During this time, CORE ALTERATIONS shall be suspended, and no control rod shall be moved from its existing position.
(k) DELETED
(l) Not required to be performed when entering OPERATIONAL CONDITION 2 from OPERATIONAL CONDITION 1 until 12 hours after entering OPERATIONAL CONDITION 2.
(m) With THERMAL POWER ≥ 25% of RATED THERMAL POWER.
(n) Frequencies are specified in the Surveillance Frequency Control Program unless otherwise noted in the table.

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

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

END-OF-CYCLE RECIRCULATION PUMP TRIP SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.4.2 The end-of-cycle recirculation pump trip (EOC-RPT) system instrumentation channels shown in Table 3.3.4.2-1 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3.4.2-2 and with the END-OF-CYCLE RECIRCULATION PUMP TRIP SYSTEM RESPONSE TIME as shown in Table 3.3.4.2-3.

APPLICABILITY: OPERATIONAL CONDITION 1, when THERMAL POWER is greater than or equal to 29.5% of RATED THERMAL POWER.

ACTION:

a. With an end-of-cycle recirculation pump trip system instrumentation channel trip setpoint less conservative than the value shown in the Allowable Values column of Table 3.3.4.2-2, declare the channel inoperable until the channel is restored to OPERABLE status with the channel setpoint adjusted consistent with the Trip Setpoint value.

b. With the number of OPERABLE channels one less than required by the

   Minimum OPERABLE Channels per Trip System requirement for one or both trip systems, place the inoperable channel(s) in the tripped condition within 12 hours.

c. With the number of OPERABLE channels two or more less than required by the Minimum OPERABLE Channels per Trip System requirement for one trip system and:

   1. If the inoperable channels consist of one turbine control valve channel and one turbine stop valve channel, place both inoperable channels in the tripped condition within 12 hours.

   2. If the inoperable channels include two turbine control valve channels or two turbine stop valve channels, declare the trip system inoperable.

d. With one trip system inoperable, restore the inoperable trip system to OPERABLE status within 72 hours or take the ACTION required by Specification 3.2.3.

e. With both trip systems inoperable, restore at least one trip system to OPERABLE status within one hour or take the ACTION required by Specification 3.2.3.

TABLE 3.3.4.2-1

END-OF-CYCLE RECIRCULATION PUMP TRIP SYSTEM INSTRUMENTATION

TRIP FUNCTION	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM*
1. Turbine Stop Valve - Closure	2**
2. Turbine Control Valve-Fast Closure	2**

* A trip system may be placed in an inoperable status for up to 6 hours for required surveillance provided that the other trip system is OPERABLE.

** This function shall be automatically bypassed when turbine first stage pressure is equivalent to THERMAL POWER LESS than 29.5% of RATED THERMAL POWER.

TABLE 3.3.6-2

CONTROL ROD BLOCK INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
1. ROD BLOCK MONITOR		
a. Upscale[(a)]		
1) Low Trip Setpoint (LTSP)	*	*
2) Intermediate Trip Setpoint (ITSP)	*	*
3) High Trip Setpoint (HTSP)	*	*
b. Inoperative	N/A	N/A
c. Downscale (DTSP)	*	*
d. Power Range Setpoint[(b)]		
1) Low Power Setpoint (LPSP)	28.1% RATED THERMAL POWER	28.4% RATED THERMAL POWER
2) Intermediate Power Setpoint (IPSP)	63.1% RATED THERMAL POWER	63.4% RATED THERMAL POWER
3) High Power Setpoint (HPSP)	83.1% RATED THERMAL POWER	83.4% RATED THERMAL POWER
2. APRM		
a. Simulated Thermal Power - Upscale:		
- Two Recirculation Loop Operation	≤ 0.65 W + 54.3% and ≤ 108.0% of RATED THERMAL POWER	≤ 0.65 W + 54.7% and ≤ 108.4% of RATED THERMAL POWER
- Single Recirculation Loop Operation****	≤ 0.65 (W-7.6%) + 54.1% and ≤ 108.0% of RATED THERMAL POWER	≤ 0.65 (W-7.6%) + 54.5% and ≤ 108.4% of RATED THERMAL POWER
b. Inoperative	N.A.	N.A.
c. Neutron Flux - Downscale POWER	≥ 3.2% of RATED THERMAL POWER	≥ 2.8% of RATED THERMAL
d. Simulated Thermal Power - Upscale (Setdown)	≤ 12.0% of RATED THERMAL POWER	≤ 13.0% of RATED THERMAL POWER
e. Recirculation Flow - Upscale	*	*
f. LPRM Low Count	< 20 per channel < 3 per axial level	< 20 per channel < 3 per axial level
3. SOURCE RANGE MONITORS		
a. Detector not full in	N.A.	N.A.
b. Upscale	≤ 1 x 10^5 cps	≤ 1.6 x 10^5 cps
c. Inoperative	N.A.	N.A.
d. Downscale	≥ 3 cps**	≥ 1.8 cps**

3/4.4 REACTOR COOLANT SYSTEM

3/4.4.1 RECIRCULATION SYSTEM

RECIRCULATION LOOPS

LIMITING CONDITION FOR OPERATION

3.4.1.1 Two reactor coolant system recirculation loops shall be in operation.

APPLICABILITY: OPERATIONAL CONDITIONS 1* and 2*.

ACTION:

a. With one reactor coolant system recirculation loop not in operation:

1. Within 4 hours:

a. Place the recirculation flow control system in the Local Manual mode, and

b. Reduce THERMAL POWER to ≤ 74.9% of RATED THERMAL POWER, and,

c. Limit the speed of the operating recirculation pump to less than or equal to 90% of rated pump speed, and

d. Verify that the differential temperature requirements of Surveillance Requirement 4.4.1.1.5 are met if THERMAL POWER is ≤ 30% of RATED THERMAL POWER or the recirculation loop flow in the operating loop is ≤ 50% of rated loop flow, or suspend the THERMAL POWER or recirculation loop flow increase.

*See Special Test Exception 3.10.4.

SURVEILLANCE REQUIREMENTS

4.4.1.1.1 DELETED

4.4.1.1.2 DELETED

4.4.1.1.3 DELETED

4.4.1.1.4 With one reactor coolant system recirculation loop not in operation, in accordance with the Surveillance Frequency Control Program, verify that:

a. Reactor THERMAL POWER is ≤ 74.9% of RATED THERMAL POWER,

b. The recirculation flow control system is in the Local Manual mode, and

c. The speed of the operating recirculation pump is ≤ 90% of rated pump speed.

4.4.1.1.5 With one reactor coolant system recirculation loop not in operation, within 15 minutes prior to either THERMAL POWER increase or recirculation loop flow increase, verify that the following differential temperature requirements are met if THERMAL POWER is ≤ 30% of RATED THERMAL POWER or the recirculation loop flow in the operating recirculation loop is ≤ 50% of rated loop flow.

a. ≤ 145°F between reactor vessel steam space coolant and bottom head drain line coolant,

b. ≤ 50°F between the reactor coolant within the loop not in operation and the coolant in the reactor pressure vessel, and

c. ≤ 50°F between the reactor coolant within the loop not in operation and the operating loop.

The differential temperature requirements of Specification 4.4.1.1.5b. and c. do not apply when the loop not in operation is isolated from the reactor pressure vessel.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 201 TO FACILITY OPERATING LICENSE NO. NPF-39

AND AMENDMENT NO. 163 TO FACILITY OPERATING LICENSE NO. NPF-85

EXELON GENERATION COMPANY, LLC

LIMERICK GENERATING STATION, UNITS 1 AND 2

DOCKET NOS. 50-352 AND 50-353

1.0 INTRODUCTION

By letter dated March 25, 2010,[1] as supplemented by additional letters,[2] Exelon Generation Company, LLC (Exelon, the licensee) requested changes to the operating license and Technical Specifications (TSs) for Limerick Generating Station (LGS), Units 1 and 2. The U.S. Nuclear Regulatory Commission (NRC or Commission) staff's original proposed no significant hazards consideration determination was published in the *Federal Register* on June 8, 2010 (75 FR 32512). The 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.

The amendments propose to revise the facility operating license and TSs to implement an increase of approximately 1.65 percent in rated thermal power (RTP) from the current licensed thermal power of 3458 megawatts thermal (MWt) to 3515 MWt. The changes are based on increased feedwater flow measurement accuracy, which will be achieved by utilizing Cameron International (formerly Caldon) CheckPlus™ (Checkplus) Leading Edge Flow Meter (LEFM) ultrasonic flow measurement instrumentation. LEFM instrumentation is currently installed in LGS, Unit 1 and will be installed in LGS, Unit 2 in refueling outage number 11 (Li2R11), currently scheduled to be completed in the spring of 2011. The amendments make changes to simulated thermal power upscale scram setpoint, including the addition of requirements to assess channel performance during testing, as well as other corresponding changes that reflect the higher licensed thermal power level. The amendments also include changes to the standby liquid control system (SLCS) on each unit. The changes involve the installation of a modified hand switch that allows operators to select two SLCS pumps for the automatic start function on an Anticipated Transient Without Scram (ATWS) signal as compared to the current configuration where three pumps receive an automatic start signal. This change ensures that pressure in the discharge header of the SLCS pumps does not challenge the relief valve setpoint, while still preserving the assumptions of the analysis for the ATWS event.

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1. Agencywide Documents Access and Management System (ADAMS) Package No. ML100850379.
2. April 26, 2010 (ADAMS Accession No. ML101180250); June 29, 2010 (ML101820088); July 22, 2010 (ML102070196); July 28, 2010 (ML102110046); July 28, 2010 (ML102110047); August 10, 2010 (ML102240259); August 12, 2010 (ML102240330); August 12, 2010 (ML102280332); August 30, 2010 (ML102440265); December 17, 2010 (ML103560667); and January 7, 2011 (ML110100447).

Further, by letters dated August 30, 2010, and December 17, 2010, the licensee also requested approval for an equivalent margins analysis (EMA) of the LGS, Unit 1 and 2, low pressure coolant injection (LPCI) nozzles and associated welds, pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, Appendix G, paragraph V.1.a, regarding fracture toughness requirements. Since the EMA was submitted in response to an NRC request for additional information (RAI) during the power uprate license amendment review, the NRC's disposition of this request will be documented in this safety evaluation (SE).

2.0 BACKGROUND

Nuclear power plants are licensed to operate at a specified maximum core thermal power, often called RTP. Part 50, Appendix K of 10 CFR, formerly required licensees to assume that the reactor has been operating continuously at a power level at least 1.02 times the licensed power level when performing loss-of-coolant accident (LOCA) and emergency core cooling system (ECCS) analyses. This requirement was included to ensure that instrumentation uncertainties were adequately accounted for in the safety analyses. In practice, many of the design bases analyses assumed a two percent power uncertainty, consistent with 10 CFR Part 50, Appendix K.

A revision to 10 CFR Part 50, Appendix K, effective July 31, 2000, allows licensees to use a power level less than 1.02 times the RTP for the LOCA and ECCS analyses, but not less than the licensed power level, based on the use of state-of-the art feedwater flow measurement devices that provide a more accurate calculation of power. Licensees can use a lower uncertainty in the LOCA and ECCS analyses provided the licensee has demonstrated that the proposed value adequately accounts for instrumentation uncertainties. Because there continues to be substantial conservatism in other Appendix K requirements, sufficient margin to ECCS performance in the event of a LOCA is preserved.

However, the final rule, by itself, did not allow increases in licensed power levels. Because the licensed power level for a plant is contained in the plant's operating license, proposals to raise the licensed power level must be reviewed and approved under the license amendment process. LGS, Units 1 and 2 are currently licensed to operate at a maximum power level of 3458 MWt, which includes a two percent margin in the ECCS evaluation model to allow for uncertainties in core thermal power measurement as was previously required by 10 CFR 50, Appendix K.

A Caldon LEFM CheckPlus™ System for feedwater flow measurement is currently installed in LGS, Unit 1 and will be installed in LGS, Unit 2 in refueling outage Li2R11, currently scheduled to be completed in April 2011. The LEFM installations are in addition to the venturi-based feedwater flow measurement system traditionally used to obtain the daily calorimetric heat balance measurements at LGS. Use of the LEFM CheckPlus System will reduce the calorimetric core power measurement uncertainty to approximately 0.31 percent. Based on this, LGS is proposing to reduce power measurement uncertainty, while meeting the requirements of 10 CFR 50, Appendix K, to permit an increase of approximately 1.65 percent in licensed power level. This proposed amendment is called a measurement uncertainty recapture (MUR) power uprate request.

Uncertainty in feedwater flow measurement is the most significant contributor to core power measurement uncertainty. The licensee states that use of the LEFM CheckPlus System provides a more accurate measurement of feedwater flow compared to the accuracy of the venturi-based instrumentation originally installed at LGS. Caldon Engineering Report ER-80P, "Improving Thermal Power Accuracy and Plant Safety While Increasing Operating Power Level Using the LEFM Check System," documents the theory, design, and operating features of the system and its ability to achieve increased accuracy of flow measurement. In an SE dated March 8, 1999, (ADAMS Legacy Library Accession No. 9903190065), ER-80P was approved by the NRC staff for use in justification of MUR power uprates up to one percent. ER-80P was supplemented by Caldon Engineering Report ER-157P, "Basis for a Power Uprate With the LEFM Check or LEFM CheckPlus System." On December 20, 2001, the NRC issued an SE (ADAMS Accession No. ML013540256) approving ER-157P for use in justifying MUR power uprates up to 1.7 percent. The NRC reviewed and approved ER-80P and ER-157P again on July 5, 2006 (ADAMS Accession No. ML061700222), as part of a generic assessment of the hydraulic aspects of ultrasonic flow meter applications to increase licensed thermal power.

The NRC issued Regulatory Issue Summary (RIS) 2002-03, "Guidance on the Content of Measurement Uncertainty Recapture Power Uprate Applications," on January 31, 2002, (ADAMS Accession No. ML013530183) to provide guidance to licensees on the scope and detail of the information that should be provided to the NRC for MUR power uprate applications. While this document does not constitute an NRC requirement, it is available to aid licensees in optimizing a MUR power uprate license amendment request (LAR), and to provide guidance to the NRC staff for conducting the review. The licensee stated in its application dated March 25, 2010, that the LAR was submitted consistent with the guidance of RIS 2002-03.

3.0 EVALUATION

3.1 MUR Power Uprate - Human Factors

3.1.1 Regulatory Evaluation

The NRC's human factors review addresses whether the licensee has adequately considered the effects of the proposed MUR on programs, procedures, training, and plant design features related to operator performance during normal and accident conditions. The NRC 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. Human factors considerations specific to the proposed SLCS changes are evaluated separately in Section 3.14 of this SE.

3.1.2 Technical Evaluation

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

Operator Actions

The licensee stated in its application that existing operator actions are not affected by the MUR power uprate. This satisfies Section VII.1 of Attachment 1 to RIS 2002-03, which requests that the licensee make 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. The NRC staff concludes that the proposed MUR power uprate will not adversely impact operator actions or their response times because there are no such changes required.

Emergency and Abnormal Operating Procedures

The licensee stated in its application that any required changes to the Emergency Operating Procedures (EOPs) will be made using their standard procedure updating process and that any required changes are expected to be negligible or have no effect. In response to a question from the NRC staff, the licensee stated that minor changes to the Abnormal Operating Procedures (AOPs) have been identified and will likewise be updated using Exelon's standard procedure updating process. The licensee's response satisfies Section VII.2.A of Attachment 1 to RIS 2002-03.

Changes to Control Room Controls, Displays, and Alarms

In its submittal, Exelon described the evaluations performed to identify control room changes in support of the proposed MUR power uprate. 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. The LEFM 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. If the LEFM system, or a portion of the system, becomes inoperable, the control room operators will be alerted by a control room alarm. Any changes to the control room as well as the associated operator training will be completed 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 licensee's response satisfies Section VII.2.B of Attachment 1 to RIS 2002-03.

Control Room Plant Reference Simulator

Exelon stated that potential simulator changes will be identified as part of the plant configuration control process. The submittal also included statements that these modifications will be evaluated, implemented and tested per the plant configuration control processes. Simulator changes and validation for the MUR power uprate will be performed in accordance with established LGS plant certification testing procedures. The licensee stated that these modifications, as well as the associated operator training, will be completed 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 completing the associated operator training prior to MUR power uprate implementation. The NRC staff finds that the licensee's response satisfies Section VII.2.C of Attachment 1 to RIS 2002-03.

Operator Training

The licensee stated in its submittals that no additional training (apart from normal training for plant changes) is required to operate the plant under uprate conditions. 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. Minor changes to the power/flow map, flow-referenced setpoint, etc., will be communicated during 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 licensee's response satisfies Section VII.2.D of Attachment 1 to RIS 2002-03.

Modifications

The licensee stated in its submittal that the LEFM system for LGS Units 1 and 2 will be installed prior to uprate implementation and that other non-safety related modifications for the power uprate, including minor equipment changes, replacements, and setpoint or alarm point changes will be implemented prior to uprate implementation. 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 licensee's response satisfies Section VII.3 of Attachment 1 to RIS 2002-03.

Temporary Operation Above Licensed Full Power Level

LGS Operating Procedure GP-5, "Steady State Operations," provides guidance to ensure that reactor power remains within the requirements of the operating license. According to the licensee this procedure is consistent with Nuclear Energy Institute (NEI) guidance for monitoring and controlling reactor power. The application references an NRC memorandum, "Safety Evaluation Regarding Endorsement of the NEI Guidance for Adhering to the Licensed Thermal Power Limit," dated October 8, 2008 (ADAMS Accession No. ML082690105), which documents the NRC endorsement of the NEI guidance for adhering to the licensed thermal power limit. The NRC staff notes that the endorsement of the NEI guidance for monitoring and controlling reactor power can be found in RIS 2007-21, "Adherence to Licensed Power Limits," Revision 1. In response to a request for additional information, the licensee provided a summary of the current version of LGS Operating Procedure GP-5, Revision 144 which states the following:

- While operating at rated power, the goal of the operator is to achieve one hour average less than or equal to maximum allowed (i.e, the maximum thermal power as stated in the plant operating license).

- At no time should reactor power be intentionally raised above maximum level.

- It is recognized that normal changes in plant parameters can cause small fluctuations in thermal power. However, operators are expected to take prompt action to reduce thermal power whenever it is found above the licensed limit.

- In no case should core thermal power average for a shift exceed 100% [percent] power, where a shift can be no longer than 12 hours.

- For pre-planned evolutions that may cause a transient increase in reactor power that could exceed 100% rated power, prudent action to reduce power prior to the evolution should be taken.

The NRC concludes that Exelon has implemented appropriate controls, in accordance with NRC-endorsed guidance, that will be applicable for operation at the new licensed power level, for the monitoring and control of reactor power level.

3.1.3 Conclusion

As described above, the NRC staff has reviewed the licensee's planned actions related to the human factors area 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 to ensure that the operators' performance is not adversely affected by the proposed MUR uprate.

3.2 MUR Power Uprate - Dose Consequence Analysis

3.2.1 Regulatory Evaluation

In Amendments 185 and 146 for LGS Units 1 and 2, respectively, which were issued on August 23, 2006 (ADAMS Accession No. ML062210214), the NRC approved a full-scope implementation of the alternative source term (AST) in accordance with 10 CFR 50.67, and following the guidance and methodology provided in applicable sections of Regulatory Guide (RG) 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors." Consistent with the licensee's current methodology, the staff conducted this evaluation to verify that the results of the licensee's LOCA, Fuel Handling Accident (FHA), Main Steam Line Break (MSLB), and Control Rod Drop Accident (CRDA) Design-Basis Accident (DBA) radiological dose consequence analyses continue to meet the dose acceptance criteria given in 10 CFR 50.67. The staff utilized the regulatory guidance provided in applicable sections of RG 1.183, and the LGS Updated Final Safety Analysis Report (UFSAR), Chapter 15, "Accident Analyses," in performing this review.

3.2.2 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 Attachments 6 and 8[3] to the March 25, 2010, application. 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 LGS UFSAR. The NRC staff reviewed all the DBAs that have the potential for a significant dose consequence as a result of the proposed changes. The specific accidents that were reviewed are as follows:

- LOCA
- FHA
- MSLB
- CRDA

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3. Attachment 6 to the licensee's application dated March 25, 2010, NEDC-33484P, is titled "Safety Analysis Report for Limerick Generating Station, Units 1 and 2, Thermal Power Optimization." This is a proprietary document. A non-proprietary version of this document, Attachment 8 to the licensee's submittal dated March 25, 2010, NEDO-33484, has the same title and is available at ADAMS Accession No. ML100850403.

In the LAR submittal, the licensee stated that the current DBA dose analyses of record for LGS, which depend on core power level, were performed at 3527 MWt, or 102 percent of the currently licensed thermal power of 3458 MWt. These DBA analyses include those for LOCA, FHA, MSLB, and CRDA. Based on the analysis provisions at 102 percent of the current licensed thermal power, the dose consequence analyses remain bounding at the proposed MUR uprated power level of 3515 MWt, with allowance for an LEFM-based 0.31 percent power measurement uncertainty. Using the licensing basis documentation, as contained in the current LGS UFSAR, in addition to information in the application dated March 25, 2010, the staff concludes that the existing radiological consequence analyses from the applicable DBAs bound the radiological consequences from DBAs at the proposed uprated power level of 3515 MWt.

3.2.3 Conclusion

The NRC staff reviewed the licensee's assessment of the impact of the proposed 1.65 percent MUR power uprate on dose consequence analyses for LGS, Units 1 and 2. As discussed above, the NRC staff determined that the results of the licensee's analyses of the radiological consequences of DBAs continue to meet the applicable acceptance criteria following implementation of the proposed MUR uprate. Therefore, the NRC staff finds the proposed MUR power uprate acceptable with respect to the dose consequences of DBAs.

3.3 MUR Power Uprate - Fire Protection

3.3.1 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) General Design Criterion (GDC)-3 of 10 CFR Part 50, Appendix A, 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, 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. According to the LGS UFSAR, Section 3.1, LGS complies with GDC-3 and GDC-5.

3.3.2 Technical Evaluation

The NRC staff's fire protection review of the licensee's March 25, 2010, application, including Attachments 6 and 8, Section 6.7, identified areas in which additional information was necessary to complete the review of the proposed application. The licensee responded to the NRC staff RAI via letter dated June 29, 2010, as discussed below.

In the RAI, the staff noted that the application states that "...There is no change in the physical plant configuration and the potential for minor changes to combustible loading as result of TPO [thermal power optimization] 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 50, Appendix R.

By letter dated June 29, 2010, the licensee provided additional information. Specifically, the licensee stated that, for Unit 1, the MUR power uprate added a small amount of flame retardant control and power cables to five fire areas, i.e., 88A, 98A, 114, 20, and 24A. Further, the licensee stated that the increased combustible load does not change the fire severity classification for these five affected fire areas. The LGS combustible loading calculations indicates that the affected areas remain below the administrative limits for fire severity classification. Section 9A.4.2 of the LGS Fire Protection Evaluation Report (FPER) defines and describes the fire area severity computation, methodology and area classification criteria. For Unit 2, the licensee stated that modification is in its design phase and, to the extent practical, will be identical to the Limerick Unit 1 modification installation. There is adequate margin to administrative limits in the combustible loading for the affected areas to accommodate a similar modification on Unit 2. Thus, the physical plant configuration changes relating to fire protection for the proposed uprate are very minor. Since these changes do not change the fire severity classification in the affected fire areas for Unit 1 and there is likely to be sufficient margin available on Unit 2 for a similar determination, the NRC staff finds the response acceptable.

In the RAI, the NRC staff noted that the application states that "...The fire safe-shutdown analysis includes consideration of equipment needed to achieve and maintain hot shutdown, fire barriers, operator actions, personnel resources, and repair activities credited to achieve and maintain cold shutdown..." The staff requested the licensee to verify that: (1) the MUR power uprate will not require any new operator actions; (2) any effects from additional heat in the plant environment from the increased power will not prevent required post-fire operator manual actions, as identified in the LGS, Units 1 and 2, fire protection program, from being performed at and within their designated time; and (3) procedures and resources necessary for systems required to achieve and maintain safe-shutdown will not change and are adequate for the MUR power uprate.

By letter dated June 29, 2010, the licensee provided additional information. The licensee stated that the existing LGS fire safe shutdown evaluation is based on a core thermal power of 3622 MWt, which bounds the proposed MUR power level of 3515 MWt. Thus, the existing fire safe shutdown evaluation is unaffected by the proposed uprate, and there are no new operator actions required for fire safe shutdown. Additional heat in the plant environment from the increased power will not prevent required post-fire operator manual actions from being performed at and within their designated time. There is no impact on the procedures and resources necessary to achieve and maintain safe shutdown, which remain adequate for the MUR power uprate. Since these changes do not impact fire protection features and post-fire safe-shutdown capability, the NRC staff finds the response acceptable.

In the RAI, the NRC staff stated that some plants credit aspects of their fire protection system for other than fire protection activities, e.g., utilizing the fire water pumps and water supply as backup cooling or inventory for non-primary reactor systems. If the LGS, Units 1 and 2, credit the fire protection system in this way, the staff indicated that 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 LGS, Units 1 and 2, do not take such credit, the staff requested that the licensee verify this as well.

By letter dated June 29, 2010, the licensee provided additional information. The licensee stated that there is no DBA or transient (other than fire) that credit the use of the fire protection system at LGS. However, there are situations beyond the design basis in which LGS would use the fire protection system as an alternate water source in EOPs and other procedures for situations in which the design basis sources of water are unavailable. The licensee identified the following provisions to use the fire protection system for non-fire suppression uses: (1) use of fire protection water as an alternate water spray to suppression pool or drywell; and (2) provide a backup source of make-up water for spent fuel pool and/or reactor cavity. Further, the licensee stated that the proposed uprate has no impact on the use of the fire protection system in these postulated situations. The staff finds the response to the RAI acceptable because the licensee's analysis concluded that all above functions of non-fire suppression uses of fire protection water are beyond design basis and are not affected by the proposed MUR power uprate.

3.3.3 Conclusion

The NRC staff reviewed the licensee's fire-related safe-shutdown assessment including responses to the staff RAIs and concludes that the licensee has adequately accounted for the effects of the small increase in decay heat on the ability of the required systems to achieve and maintain safe-shutdown conditions. The NRC staff finds that the changes described do not impact conformance with 10 CFR 50.48, GDC-3 and GDC-5 and that operation at an increased core power level of 3515 MWt is acceptable with respect to fire protection.

3.4 MUR Power Uprate - Materials and Chemical Engineering

3.4.1 Protective Coating Systems (Paints) – Organic

3.4.1.1 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 staff's review covered protective coating systems used inside containment, including the coating's suitability for, and stability under, design-basis 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) RG 1.54, Revision 1, "Service Level I, II, and III Protective Coatings Applied to Nuclear Power Plants," July 2000. LGS is not committed to RG 1.54, as described in UFSAR Section 1.8, however, as described in the LGS UFSAR Section 6.1.2, protective coatings within the primary containment must be qualified for the expected service conditions, including postulated LOCA conditions.

3.4.1.2 Technical Evaluation

In Attachments 6 and 8 to the licensee's application dated March 25, 2010, Section 4.1.4, the licensee stated that the coatings in the primary containment are qualified such that they do not fail when exposed to the existing maximum post-LOCA primary containment operation conditions. The licensee stated that the existing maximum post-accident conditions bound those conditions which are expected after implementation of the MUR power uprate since the current operating conditions are based on 102 percent of the current licensed power level. Therefore, the MUR power uprate will not result in any postulated post-LOCA environmental condition changes beyond those already assumed. The staff concurs that the coatings will not be adversely impacted by the MUR power uprate and that the post-LOCA environmental conditions postulated under power uprate conditions continue to be bounded by the conditions to which the coatings were qualified.

3.4.1.3 Conclusion

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

3.4.2 Flow-Accelerated Corrosion

3.4.2.1 Regulatory Evaluation

Flow-accelerated corrosion (FAC) is a corrosion mechanism occurring in carbon steel components exposed to flowing single or two-phase water. Components made from stainless steel are immune to FAC, and FAC is significantly reduced in components containing small amounts of chromium or molybdenum. The rates of material loss due to FAC depend on flow velocity, fluid temperature, steam quality, oxygen content, and pH. During plant operation, control of these parameters is limited and the optimum conditions for minimizing FAC effects, in most cases, cannot be achieved. Loss of material by FAC will, therefore, occur.

The NRC staff 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 critical thickness. 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 meeting the structural evaluation of the minimum acceptable wall thickness for the components undergoing degradation by FAC.

3.4.2.2 Technical Evaluation

The licensee stated that carbon steel piping in the main steam (MS), feedwater (FW), and balance of plant (BOP) systems 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 and two-phase high energy carbon steel piping. The licensee stated that the changes in velocity, temperature, and moisture content due to MUR power uprate conditions would have a minor effect on the parameters affecting 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.

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, including the continuing inspection 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 will provide 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 has reviewed the licensee's evaluation and confirms that the licensee's FAC program provides reasonable assurance that the potentially-impacted piping systems will maintain an acceptable wall thickness. The staff has also reviewed the piping systems whose monitoring 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, including changes due to the proposed increase in power level.

3.4.2.3 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, provide an appropriate inspection frequency, 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.

3.4.3 Reactor Water Cleanup System

3.4.3.1 Regulatory Evaluation

The reactor water cleanup (RWCU) system provides a means for maintaining reactor water quality by filtration and ion exchange and a path for removal of reactor coolant when necessary. Portions of the RWCU system comprise the reactor coolant pressure boundary (RCPB). The NRC staff review of the RWCU system 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's acceptance criteria for the RWCU system are based on: (1) 10 CFR 50, Appendix A, GDC-14, "Reactor Coolant Pressure Boundary," as it 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," as it 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," as it requires that systems that contain radioactivity be designed with appropriate confinement. Specific review criteria are contained in NUREG-0800, Section 5.4.8, "Reactor Water Cleanup System (BWR)." The LGS UFSAR, Section 3.1 states that LGS meets the requirements of GDC-14, GDC-60, and GDC-61.

3.4.3.2 Technical Evaluation

The licensee stated that the performance requirements of the RWCU system would be negligibly affected by MUR power uprate conditions. There will be no significant effect on operating temperature and pressure conditions in the high-pressure portion of the system. The licensee indicated that transients are the primary source of challenge to the system; as such, safety and operational aspects of water chemistry performance will not be affected by operating at the slightly higher uprated steady-state power level. Based on these considerations, the NRC staff finds the MUR power uprate acceptable with respect to the RWCU system operational considerations.

Regarding RWCU system high energy line break (HELB) considerations, there will be a small decrease in recirculation temperature with a negligible increase in pressure, resulting in a slightly increased blowdown rate and slightly lower blowdown energy level. The licensee also stated that the original HELB analysis included conservative modeling assumptions that more than offset the effects of the temperature change. The NRC staff evaluated the licensee's disposition regarding the HELB analysis, and because the operating conditions resulting from MUR uprate will impose a small impact on the RWCU system HELB analysis, and because the staff judges that the original HELB analysis remains bounding, the proposed uprate is acceptable with respect to RWCU system performance under HELB conditions.

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

3.4.3.3 Conclusion

The NRC staff has reviewed the licensee's evaluation of the effects of the proposed MUR power uprate on the RWCS and concludes that the licensee has adequately addressed changes in the temperature of the reactor coolant and its effects on the RWCU system. The staff further concludes that the licensee has demonstrated that the RWCU system will continue to be acceptable and will continue to conform with GDC-14, GDC-60 and GDC-61 following implementation of the proposed MUR power uprate. Therefore, the staff finds the proposed MUR power uprate acceptable with respect to the RWCU system.

3.5 MUR Power Uprate - Mechanical and Civil Engineering

3.5.1 Regulatory Evaluation

The NRC staff's review in the civil and mechanical engineering disciplines addresses the structural and pressure boundary integrity of the SSCs affected by the proposed MUR power uprate at LGS. 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 reactor pressure vessel (RPV) and its supports; and (3) the reactor vessel internals (RVIs), including the core support and non-core support structures.

Technical areas encompassed in this review include stresses, fatigue and cumulative usage factors (CUFs), flow-induced vibration (FIV), HELB locations and jet impingement and thrust forces.

The NRC staff's evaluation is based on 10 CFR 50.55a and GDC-1, 2, 4, 10, 14 and 15, in 10 CFR 50, Appendix A. According to the LGS UFSAR Section 3.1, LGS complies with each of these GDCs. The NRC staff's review focused on verifying that the licensee has provided reasonable assurance of the structural and functional integrity of the previously mentioned 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 criterion are based on continued conformance with the following: (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 (SAFDLs) 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 being designed with sufficient margin to ensure that the design conditions are not exceeded.

For the Mechanical and Civil Engineering areas, RIS 2002-03, Section IV, documents the scope and detail of the information that licensees should submit to the NRC staff. The licensee's application states that the scope and contents of the application were developed consistent with the guidelines specified in RIS 2002-03. Further, the licensee's MUR power uprate application follows the generic format and content of topical report NEDC-32938P-A, "Generic Guidelines and Evaluations for General Electric Boiling Water Reactor Thermal Power Optimization" (ADAMS Accession No. ML003738608). A non-proprietary version of this report, NEDO-32938, same title, may be found at ADAMS Accession No. ML023170607. By letter dated April 1, 2003 (ADAMS Accession No. ML031050138), the NRC staff found the approach specified in NEDC-32938P-A acceptable for referencing in licensing applications, to the extent specified under the limitations delineated in the report, and in the NRC's associated SE.

3.5.2 Technical Evaluation

The NRC staff review in the civil and mechanical engineering disciplines focused on the effects of the proposed MUR power uprate on the structural and pressure boundary integrity of NSSS and BOP piping systems, components, and their supports, the reactor vessel and internal components, as well as FIV and HELB analyses. Minor increases in steam flow and FW flow will accompany the proposed increase in reactor power.

Table 1-2 of Attachments 6 and 8 in the licensee's March 25, 2010, application lists the temperatures, pressures, and flow rates for the current and proposed MUR power uprate conditions. At full power, the dome temperature remains at a constant 551.5 degrees Fahrenheit ($^{\circ}$F) from current to MUR power uprate conditions. At full power, the dome pressure remains at a constant 1060.0 pounds per square inch absolute (psia) from current to MUR power uprate conditions. The minimum full power core flow range increases from 81.0 to 82.9 million pounds per hour (Mlbm/hr) while the maximum full power core flow range remains constant at 110.0 Mlbm/hr under MUR power uprate conditions. The steam flow increases from 14.997 to 15.287 Mlbm/hr and the FW flow rate increases from 14.965 Mlbm/hr to 15.255

Mlbm/hr under MUR power uprate conditions. The FW temperature increases from 425.1 to 427.1 °F from current to MUR power uprate conditions. The design parameters for the RCPB at LGS are documented in Chapter 5.2.2, "Overpressure Protection," of the LGS UFSAR. The RCPB design pressure is 1250 pounds per square inch gauge (psig). Chapter 10, "Steam and Power Conversion System," of the LGS UFSAR provides design information for the portions of the MS system, FW system, and Condensate system outside containment.

Reactor Pressure Vessel

The licensee evaluated the effects of the proposed MUR power uprate on the structural integrity of the RPV in Section 3.2, "Reactor Vessel," of Attachments 6 and 8 in the March 25, 2010, application. 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 with addenda up to and including the Summer 1969 Addenda, including Figure N-462(e)(2) of the Summer 1970 Addenda. The licensee noted that the following RPV components have been modified since the original construction of LGS Units 1 and 2, and therefore may have a different Code of Record: the FW nozzle, the LPCI nozzle, the control rod drive (CRD) hydraulic system return nozzle (Unit 2), core spray (CS) nozzle, recirculation inlet nozzle, and the universal dry tube, power range detector, and in-core detector assembly. As indicated in Section 3.2 of Attachments 6 and 8 in the licensee's March 25, 2010, submittal, the design conditions of the RPV are unchanged. Further, reactor coolant temperatures and flows are unchanged from the current bounding analysis of record for normal and upset conditions. Stresses under emergency and faulted conditions are based on loads such as peak dome pressure which are unchanged in the uprated condition. As a result of these considerations, the licensee's application indicates that the RPV components subject to design basis loading conditions continue to meet ASME Code stress and fatigue requirements for 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. Note that RPV fracture toughness, including nozzles and attachment welds, is evaluated separately in Section 3.9 of this SE.

Reactor Vessel Internals

The licensee evaluated the effects of the proposed MUR power uprate on the structural integrity of the RVIs, including core support and non-core support structures. 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 housing, control rod guide tube, and the orificed fuel support. The non-core support structures evaluated include the FW sparger, jet pump, CS 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 LPCI 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 loads considered in the evaluation of the RVIs include reactor internal pressure differences (RIPDs), dead weight, seismic, Safety/Relief Valve (SRV), LOCA, annulus pressurization/jet reaction (AP/JR), acoustic and flow induced loads due to recirculation line break, fuel lift, hydraulic flow and thermal loads. RIPD loads are bounded by the existing design basis values. The licensee notes that the RIPDs are affected more by the maximum licensed core flow rate than by the power level and the maximum licensed core flow rate is not changed. Seismic, SRV, LOCA and AP/JR loads remain unchanged. Acoustic and flow- induced loads due to recirculation line break, hydraulic flow and thermal loads remain bounded because the current analysis conservatively assumes operation at an initial power level of 102 percent of the current licensed thermal power. Fuel lift loads increased in Service Level D, however, the stresses of core plate, top guide and control rod guide tube were reconciled for the increase of the fuel lift loads to demonstrate that adequate stress margins exist, and that stresses remain within allowable limits. Fuel bundle lift margins and control rod guide tube (CRGT) lift forces that are calculated to show that fuel bundles will not lift are bounded by the existing analysis which already assumes a higher power level and core flow than will exist at MUR power uprate conditions. The limiting stresses of other RPV internal components also remain bounded by the existing design basis values. The input loads to the existing flaw analysis are not impacted by the MUR power uprate and hence, the existing flaw evaluations remain valid for the MUR power uprate. Based on these evaluations, the licensee concludes that the RVIs are structurally qualified to operate at the MUR power uprate conditions.

Regarding steam separator and steam dryer performance, the licensee performed a qualitative analysis consisting of scale model testing and an analytical approach to assess the potential for steam dryer FIV at MUR power uprate conditions. The scale model testing has indicated potential propagation of acoustic resonance in two of the four MS lines at current licensed thermal power conditions. The licensee notes that the steam dryer has operated satisfactorily for approximately 14 years at current licensed thermal power with no adverse flow effects. The testing has also shown a reduction in the normalized root mean square (RMS) pressure in the MS lines, as flow conditions change from current licensed thermal power to MUR power uprate conditions. The licensee additionally determined that the normalized RMS pressure for the other two MS lines was determined to be constant and below any potential acoustic resonance propagation thresholds. Other loads applicable to the steam dryer evaluation are deadweight, seismic, RIPD, SRV, LOCA, AP/JR and fuel lift loads. Dead weight and seismic loads remain unchanged for MUR power uprate conditions. RIPDs contributing to the steam dryer load remain bounded by previous analyses. SRV, LOCA, AP/ JR, and fuel lift loads remain bounded by current licensed thermal power values for the MUR power uprate. Since all applicable loads for the steam dryer are bounded by the existing design basis for the MUR power uprate conditions, the licensee, therefore, concluded that the steam dryer remains structurally qualified for plant operation at MUR power uprate conditions.

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 power uprate 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 power uprate acceptable with respect to the design of the RVIs.

Flow-Induced Vibration

The licensee assessed FIV in accordance with NRC-approved NEDC-32938. This evaluation determined the effects of FIV on the reactor internals at 110 percent rated core flow and a power level of 101.7 percent of the current licensed thermal power. Vibration levels for the MUR power uprate conditions were estimated from vibration data recorded during startup testing of the NRC-designated prototype plant, Browns Ferry Unit 1, and during other tests. According to the licensee, predicted vibration levels were compared with established vibration acceptance limits. The licensee's calculations for MUR power uprate conditions indicate that vibrations of safety-related RVIs are within the design basis acceptance criteria. For some components, FIV is a function of core flow. Because the maximum licensed core flow is unchanged for the MUR power uprate, FIV for those components is not affected. Therefore, the licensee concluded that FIVs for all evaluated RVI components remain within acceptable limits for the MUR power uprate.

The NRC staff has reviewed the licensee's evaluations related to RVI FIV. For the reasons set forth above, which demonstrate that the FIV of 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 regarding this issue for the RVIs. Therefore, the NRC staff finds the proposed MUR acceptable with respect to the FIV impact on the RVIs.

The licensee notes that the safety-related MS and FW piping has minor increased flow rates and flow velocities that result from the MUR power uprate. The MS and FW piping exhibit increased vibration levels approximately proportional to the increase in the square of the flow velocities and proportional to any increase in fluid density. The decrease in FW fluid density for MUR power uprate conditions due to the approximately 2°F increase in FW temperature is not significant. The MS and FW piping vibration is expected to increase by about four percent. A MS and FW piping FIV test program conducted during initial plant startup demonstrated that vibration levels were within the acceptance criteria. Operating experience indicates that there are no existing vibration problems in the MS and FW lines at current full-power operating conditions.

In an RAI to the licensee dated June 3, 2010, the NRC staff noted that Appendix K of NEDC-32938, "Methods and Assumptions for Piping Evaluation of TPO Uprate," notes, in part, that "for plants whose licensing basis includes the results of MS line and/or FW line vibration measurements taken during startup testing, the effects of power uprate on vibratory displacements will also be included." The NRC staff further noted that Table 3.9-7, "Non-NSSS Piping Systems Power Ascension Testing," of the LGS UFSAR documents both dynamic transient and steady-state vibration tests for the MS and FW piping systems during plant startup. In response to the NRC staff's RAI, by letter dated June 29, 2010, the licensee stated that the evaluation for MS and FW piping vibration was performed by extrapolating the startup test vibration data at original licensed thermal power to the MUR power uprate condition (101.7 percent of current licensed thermal power). The vibrations are assumed to increase in proportion to the fluid density (ρ) and the square of the fluid velocity (V), or ρV^2. The piping vibrations at the MUR power uprate are expected to be about four percent higher than current vibration levels based on the extrapolated flow rates. For MS piping, the extrapolated vibrations were then compared against the original vibration acceptance criteria to show acceptability. For FW piping, the extrapolated vibrations were compared against the ASME OMa-S/G-1990 Standard, Part 3, Appendix D, Screening Velocity Criteria to show acceptability. The licensee's evaluation showed that there is substantial (47 percent-50 percent) margin to the acceptance criteria.

The licensee does not plan specific validation of predicted vibration levels at MUR power uprate conditions because of the small predicted increase in vibration levels, the substantial margin remaining to the established acceptance criteria, and the lack of vibration problems existing at LGS. Based on the licensee's conclusion that the MS and FW piping vibration levels will remain within the acceptance criteria with significant margin for MUR power uprate conditions, and the successful operating experience for LGS at power levels just below the uprated power level, the NRC staff finds the proposed MUR acceptable with respect to the FIV impact on the MS and FW piping.

Reactor Coolant Pressure Boundary Piping and Supports

In the March 25, 2010, submittal, Attachments 6 and 8, Section 3.5.1, the licensee notes that the effect of the MUR power uprate with no nominal vessel dome pressure increase is negligible for the RCPB portion of all piping except for segments of the FW lines, MS lines, and piping connected to the FW and MS lines. The licensee summarized an evaluation of the piping inside containment for the effects of the MUR power uprate, concluding that there is a negligible effect on the majority of the RCPB piping inside containment. For the MS and FW lines, supports and connected lines, the methods outlined in Section 5.5.2 and Appendix K of NEDC-32938P were used to determine the percent increases in applicable ASME Code stresses, displacements, CUFs, and pipe interface component loads (including supports) as a function of percentage increase in pressure, temperature, and flow due to MUR power uprate conditions. The percentage increases were applied to the highest calculated stresses, displacements, and the CUFs at applicable piping system node points to conservatively determine the maximum MUR power uprate calculated stresses, displacements and cumulative usage factors. The licensee indicates that this approach is conservative since the MUR power uprate does not affect dead weight or building-filtered loads such as seismic loads. The factors were also applied to nozzle loads, support loads, penetration loads, valves, pumps, heat exchangers and anchors, so that these components could be evaluated for acceptability as required. No new computer codes were used and no new assumptions were employed for this evaluation.

In the application the licensee indicates that the MS and attached piping inside containment was evaluated for compliance with the ASME Code stress criteria, and for the effects of thermal displacements on the piping snubbers, hangers, and struts. The licensee also evaluated piping interfaces with RPV nozzles, penetrations, flanges and valves. The evaluation indicated that the increase in flow associated with the MUR power uprate does not result in load limits being exceeded for the MS piping system or for the RPV nozzles. The temperature of the MS piping inside containment is unchanged for the MUR power uprate. The licensee notes that the current licensing basis design analyses have sufficient design margin between the calculated stresses and ASME Code allowable limits to justify operation at MUR power uprate conditions. The design adequacy evaluation results demonstrate that the requirements of ASME Code, Section III, Subsection NB/NC/ND (as applicable) requirements are satisfied for the evaluated piping systems. The licensee, therefore, determined that the MUR power uprate does not have an adverse effect on the MS piping design. The licensee also reviewed the current licensing basis MS piping for the effects of transient loading on the piping snubbers, hangers, struts, and pipe whip restraints. The licensee's review of the increases in MS flow for the MUR power uprate did not indicate that any pipe support load limits were exceeded.

The licensee reviewed the FW piping system inside containment for compliance with the ASME Section III Code stress criteria, and for the effects of thermal expansion displacements on the piping snubbers, hangers, and struts. Piping interfaces with RPV nozzles, penetrations, and valves were also evaluated. The licensee's review of the change in temperature, pressure, and flow associated with the MUR power uprate indicates that piping load changes do not result in load limits being exceeded for the FW piping system or for the RPV nozzles. The current licensing basis design analyses have adequate design margin between calculated stresses and ASME Code allowable limits to justify operation at the MUR power uprate conditions. The design adequacy evaluation shows that the requirements of the ASME Code Section III, Subsection NB/NC/ND-3600 requirements remain satisfied. The licensee, therefore, determined that the MUR power uprate will not have an adverse effect on the FW piping design. The licensee also found that the MUR power uprate will not affect the FW piping snubbers, hangers and struts. A review of the increase in FW temperature and flow associated with the MUR power uprate indicated that piping load changes do not result in any load limit being exceeded at MUR power uprate conditions.

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 and fatigue 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, 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.

BOP Piping Systems

In its application, the licensee evaluated the BOP piping for the effects of the MUR power uprate. According to the licensee, for the condensate, FW, extraction steam, heater drain, and MS systems, operating system pressures and temperatures under the MUR power uprate will remain within design ratings. Because there is no change in the MS operating temperature from the RPV to the MS stop valves, there is no change in the thermal expansion stress for the MUR power uprate. For systems with increased operating temperatures, i.e., MS downstream of the stop valves, condensate, feedwater, extraction steam, and heater drains, changes to thermal expansion stresses are small and acceptable. Pipe support loads will experience a less than one percent increase in thermal load. In combination with other loads that are not affected by the MUR power uprate, such as dead weight, the combined support load increase is not significant. The licensee has analyzed the BOP piping systems with increased operating temperatures to conditions that envelope operations under the MUR power uprate.

For the MS system outside containment, the turbine stop valve (TSV) closure transient was reviewed against conditions that bound operations under the MUR power uprate. The licensee stated that available stress and support load margins are adequate to accommodate the increase in loading associated with this fluid transient. In an RAI, the NRC staff asked the licensee to indicate how the TSV closure transient was reviewed for the MUR power uprate conditions, and if the licensee's review considered the results of any testing of the TSV closure transient conducted during original plant startup. In response to the staff's RAI, the licensee noted that for the previous five percent uprate approved by the NRC (Unit 1 - ADAMS Accession No. ML011560244 and Unit 2 - ML011560773), pipe stress and support load margins were reviewed for fluid transient loading. Loads were evaluated at 102 percent of the current licensed thermal power. Scaling factors were developed to bound limiting changes under uprated conditions for material properties, internal pressure, operating temperatures, and fluid transient loading. These conservative scaling factors were then applied considering appropriate load combinations with values of 15 percent applied to pipe stress and 18 percent applied to fluid transient loads. All piping and supports were determined to have a design margin to accommodate for this increase in applied stress and load. This approach and the development of the scaling factors were reviewed for the MUR power uprate evaluation and found to be appropriate and bounding, specifically for the TSV closure transient and design of MS piping and supports. The licensee noted that the TSV closure load definition was based upon measured pressure responses in the LGS Unit 1 piping. These responses are based on recorded data from TSV testing early in the operating life of the station. This load was used in the analysis of both Unit 1 and Unit 2 MS piping. Based on the licensee's response to the NRC staff's RAI, specifically the evaluation performed at 102 percent of the current licensed thermal power level, the staff concludes that the licensee has adequately reviewed the TSV closure transient for the MUR power uprate conditions.

For the FW system piping outside containment, changes to fluid transient loading such as feed pump trip are small. The licensee reviewed the LGS design for fluid transients and concluded that no changes are required for the 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.

HELB Locations

The licensee's application states that HELB evaluations were performed for all systems addressed in the LGS UFSAR. At the MUR power uprate RTP level, HELBs outside the drywell would result in an insignificant change in the sub-compartment pressure and temperature profiles. The licensee's application further states that MUR power uprate system operating temperatures and pressures change only slightly, so that there is no significant change in HELB mass and energy releases. Vessel dome pressure and other portions of the RCPB remain at current operating pressure or lower. The changes do not have a significant effect on the line break calculations of record. Therefore, the consequences of any postulated HELB would not significantly change. The postulated break locations remain the same because the piping configuration does not change due to the MUR power uprate. The licensee concludes that the affected building and cubicles that support safety-related functions are designed to withstand the resulting pressure and thermal loading following an HELB at MUR power uprate RTP conditions.

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.

3.5.3 Conclusion

The NRC staff has reviewed the licensee'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 regulatory requirements related to civil and mechanical engineering 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 LGS, Units 1 and 2.

3.6 MUR Power Uprate - Safety-Related Valves

3.6.1 Regulatory Evaluation

The NRC staff reviewed the licensee's safety-related valve analysis for LGS, Units 1 and 2. The NRC's acceptance criteria for review are based on 10 CFR 50.55a, "Codes and Standards," which contains requirements for inservice testing (IST) of ASME Code class 1, 2 and 3 valves. 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."

3.6.2 Technical Evaluation

In the submittal dated March 25, 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 the submittal, Attachments 6 and 8, Sections 3.1, 3.8, 4.1, and 5.3.4, the licensee reviewed the revised design and operating conditions resulting from the MUR power uprate against previous licensing evaluations. In Section 3.1, the licensee reviewed the MUR power uprate 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 MS 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 SRV analytical limits are not changed. In Section 4.1, the licensee concludes that no changes to the functional requirements of the existing safety-related valves are identified as a result of the MUR power uprate. The licensee also evaluated the MUR power uprate impact on the requirements of GL 89-10, GL 95-07, and GL 96-05. Regarding GL 89-10, the licensee indicated that because previous analyses were either based on 102 percent of the current licensed power level or are consistent with the plant conditions expected to result from the MUR power uprate, there are no increases in the pressure or temperature at which MOVs are required to operate, and that no changes to the functional requirements of the GL 89-10 MOVs are required. Regarding GL 95-07, the criteria for susceptibility to pressure locking or thermal binding were reviewed by the licensee and it was determined that the slight changes in operating or environmental conditions that are expected to result from the MUR power uprate would have no impact on the performance of the power-operated gate valves within the scope of GL 95-07. Regarding GL 96-06, the containment design temperatures and pressures in the current

GL 96-06 evaluation are not exceeded under post-accident conditions for the MUR power uprate. The licensee, therefore, concluded that its response to GL 96-06 remains valid under MUR power uprate conditions.

The NRC review of the licensee's application concludes that the MUR power uprate does not impact the designs and operations of the safety-related valves since the operating ranges of pressure, temperature, and flow are bounded by previous evaluations. Since there are insignificant changes in operating conditions and no changes to the design basis requirements, the IST program for safety-related valves will not be affected by MUR power uprate.

3.6.3 Safety-Related Valves Conclusion

Based on the reasons set forth above, 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.

3.7 MUR Power Uprate - Reactor Systems LEFM Analysis

3.7.1 Regulatory Evaluation

As described in the background section of this SE, MUR power uprates may be authorized by the NRC staff based on the current wording of 10 CFR 50 Appendix K, provided that the licensee has demonstrated that the proposed instrumentation adequately accounts for instrument uncertainties. In this case, the licensee has referred to NRC-approved Cameron Topical Reports ER-80P and ER-157P to provide this justification. The NRC staff reviewed thermal-hydraulic aspects of the LEFM CheckPlus system installation, including its laboratory calibration, and the effects of system changes such as transducer replacement.

3.7.2 Technical Evaluation

Feedwater Flow Measurement Device Installation

The licensee installed the LEFM in the LGS, Unit 1 feedwater piping during their Li1R13 refueling outage which was completed in the spring of 2010. The licensee plans to install the LGS Unit 2 LEFM in their feedwater piping during their Li2R11 refueling outage scheduled for the spring of 2011. For both units, the installation on Loop A will be upstream of both the flow straightener and the FW flow nozzle. The installation on Loops B and C will be located in straight sections of pipe downstream of existing flow straighteners and upstream of existing FW flow nozzles. The LGS Unit 1 and Unit 2 installations will be comparable.

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. After plant installation, testing will include an inservice leak test, comparisons of FW flow and thermal power calculated by various methods, and final commissioning testing.

Transducer Replacement

Uncertainty associated with transducer replacement was addressed in the LGS application for the MUR power uprate. The licensee will use the vendor's recommendation for preventive maintenance and has tested the ultrasonic flow meter at Alden Laboratories to account for differences between testing and installation or maintenance, including transducer replacement. The NRC staff finds that transducer installation variability has been acceptably addressed.

CheckPlus Calibration

CheckPlus calibration factors were established by testing accomplished at Alden Laboratories. The piping configuration used during testing included a full scale model of the LGS hydraulic geometry. The NRC staff reviewed drawings and schematics provided and confirmed that, insofar as configuration is concerned, the laboratory configuration largely matched the piping configuration.

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 ultrasonic flow meter for LGS, Units 1 and 2.

3.7.3 Conclusion

The proposed license amendment is based on the use of the Caldon LEFM CheckPlus system that would decrease the uncertainty in the FW flow rate, thereby decreasing the power level measurement uncertainty.

The NRC staff finds that the hydraulic aspects of the Caldon LEFM CheckPlus UFM system have been accurately described in applicable documentation and that there is a firm theoretical and operational understanding of behavior. The NRC staff further finds that the calibration accomplished at Alden Labs is appropriate for the LEFM installation at LGS and meets the hydraulic requirements for justification of reduced power measurement uncertainty, as required by 10 CFR 50 Appendix K . Therefore, the NRC staff concludes, based on the considerations discussed above, that the proposed changes are acceptable with respect to the hydraulic aspects of the CheckPlus ultrasonic flow meter when installed at LGS, Units 1 and 2.

3.8 MUR Power Uprate - Accident And Transient Analyses

3.8.1 Regulatory Evaluation

The NRC staff reviewed the accident and transient analyses using the guidance contained in NRC RIS 2002-03, which the licensee's application is based upon. RIS 2002-03 states the following regarding the general approach to the 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 the licensee's LGS 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 NEDC-32938. The only exception was for the anticipated transients without scram (ATWS) analysis, for which the existing licensing basis analyses was not bounding of the requested operation. For this event the licensee performed and presented additional evaluations and analyses, and the NRC staff, therefore, performed a detailed review.

3.8.2 Technical Evaluation

Attachment 4 of the March 25, 2010, application 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 Attachments 6 and 8, 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 Uprated Power Level," was also addressed in Attachments 6 and 8, Section 9.0. In Attachments 6 and 8, the licensee characterized the design basis accidents and transients by sorting them into three categories: anticipated operational occurrences, design basis accidents, and special events.

Anticipated Operational Occurrences (AOOs)

For AOOs, the licensee stated that the evaluation and conclusions of NEDC-32938 are applicable to the LGS MUR power uprate request. As a result, the licensee did not present results for plant-specific transient analyses because the changes are expected to be within normal cycle-to-cycle variations or are currently performed with a 2 percent overpower assumption. The licensee also stated that standard reload analyses will be performed at the first fuel cycle that implements the MUR power uprate. The licensee stated that this approach is consistent with the thermal power optimization guidelines contained in NEDC-32938, which has been generically-approved by the NRC. Because the licensee will re-analyze 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 NEDC-32938, the NRC staff finds the requested uprate acceptable with respect to AOOs.

Design Basis Accidents

Regarding the design basis accidents, the licensee stated that the design basis accident events either have been previously analyzed at 102-percent of the current licensed thermal power 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.

Special Events

The licensee identified two classes of special events requiring evaluation: the ATWS events, and the Station Blackout (SBO). The NRC's SBO evaluation is documented in Section 3.10.2 of this SE.

Anticipated Transients Without Scram

The licensee concluded that the ATWS analysis of record performed at the current licensed thermal power level did not demonstrate the required margins for generic evaluation. Therefore, consistent 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 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 Injection (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 gallons per minute (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; and,

3. Operator actions specified in the plant's EOPs are consistent with the generic emergency procedure guidelines/severe accident guidelines (EPGs/SAGs), 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 (PCT) is within the 10 CFR 50.46 limit of 2200°F;

3. The peak suppression pool temperature is less than the limits of the applicable methodology; and,

4. The peak containment pressure is less than the containment design pressure.

Specific review criteria are provided in NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants" (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 Attachments 6 and 8 to the licensee's March 25, 2010, application, Section 9.3.

The licensee stated that LGS meets the ATWS requirements defined in 10 CFR 50.62 because: (a) an 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, an 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 exceed its 190°F limit; and (c) the peak containment pressure does not exceed its 55 psig limit.

In its application, the licensee described an ATWS analyses for an equilibrium core at the proposed, uprated operating condition to demonstrate that LGS meets the ATWS acceptance criteria. The licensee's analyses indicated that the predicted peak vessel bottom pressure increased from 1458 psig in the current licensing basis analysis to 1473 psig for operation at the proposed, uprated condition. Peak suppression pool temperature and containment pressure remained within their limits at 182°F and 10.6 psig, respectively. The licensee referenced NEDC-32938 for its evaluation of PCT and fuel oxidation ATWS evaluations. NEDC-32938 states that PCT and local clad oxidation are not required to be explicitly analyzed for an MUR power uprate. The basis for this disposition has been reviewed and generically approved by the NRC staff, and the NRC staff further agrees with the licensee that the NEDC-32938 evaluation is applicable to the LGS MUR power uprate.

The licensee referenced the NEDC-33004-A, "Constant Pressure Power Uprate Licensing Topical Report (CLTR)," for the requested uprate, as well as NEDO-32164, "Mitigation of BWR Core Thermal-Hydraulic Instabilities in ATWS" (ADAMS Accession No. ML102350204), and NEDO-32047, "ATWS Rule Issues Relative to BWR Core Thermal-Hydraulics Stability" (ADAMS Accession No. ML102230093). Generally, the bases for the CLTR dispositions are:

1. The requested uprate will not affect the initial condition of the ATWS instability, since it starts from natural circulation; and,

2. The same mitigation strategy will be relied upon to terminate the ATWS instability event.

The requested uprate is of small enough magnitude to be judged by the NRC staff as having no significant effect on the mitigation strategy described in the application references. The SLCS is automatically initiated upon receipt of a signal from the redundant reactivity control system (RRCS) logic. The NRC staff finds the applicability of the CLTR is appropriate and that LGS meets all of the CLTR dispositions. Therefore, the NRC staff finds the requested uprate is acceptable with respect to ATWS instability.

ATWS – Conclusions

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 ARI), SLCS performance requirements (see Section 3.14 for a further discussion of SLCS performance), and system effects of the limiting postulated ATWS events, the NRC staff finds the proposed power uprate acceptable with respect to ATWS.

3.8.3 Accident and Transient Analyses – Conclusion

Based on the NRC staff's review of the 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 MUR Power Uprate - Vessel and Internals Integrity

The NRC staff's review in the area of RPV integrity for boiling water reactors focuses on the impact of the proposed MUR power uprate on neutron fluence calculations, the RPV surveillance capsule withdrawal schedules, RPV pressure-temperature (P-T) limits, and upper shelf energy (USE) evaluations. This review was conducted, consistent with the guidance contained in RIS 2002-03, to verify that the results of licensee analyses related to these areas meets the requirements of 10 CFR 50.60 and 10 CFR Part 50, Appendices G and H following implementation of the proposed MUR power uprate.

3.9.1 RPV Material Surveillance Program

Regulatory Evaluation

The surveillance program requirements in Appendix H to 10 CFR Part 50 were established to monitor the radiation-induced changes in the mechanical and impact properties of the RPV materials. Appendix H to 10 CFR Part 50 requires licensees to monitor changes in the fracture toughness properties of ferritic materials in the RPV beltline region of light-water nuclear power reactors. Appendix H to 10 CFR Part 50 states that the design of the surveillance program and the withdrawal schedule must meet the requirements of the edition of American Standard for Testing of Materials (ASTM) E185-82, "Standard Practice for Conducting Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessels," that is current on the issue date of the ASME Code to which the RPV was purchased. Later editions of ASTM E-185 may be used including those editions through 1982 (i.e., ASTM E-185-82).

Technical Evaluation

By letter dated March 25, 2010, the licensee stated that LGS, Units 1 and 2 participate in the Boiling Water Reactor Vessel and Internals Project (BWRVIP) Integrated Surveillance Program (ISP), which is administered by the Electric Power Research Institute. Regarding the RPV surveillance program and capsule withdrawal schedule, the licensee concluded in Section 3.2.1(d) of Attachments 6 and 8 to the submittal that, "TPO [thermal power optimization] has no effect on the existing surveillance schedule." LGS, Units 1 and 2 are not designated as representative plants in the ISP, therefore capsules are not currently scheduled for removal.

The licensee's RPV material surveillance program is part of the ISP designed by the BWRVIP for operating BWR plants. The BWRVIP ISP was submitted for NRC staff review and approval in proprietary topical reports BWRVIP-78, "BWR Vessel and Internals Project, BWR Integrated Surveillance Program Plan," dated December 22, 1999, and BWRVIP-86, "BWR Vessel and Internals Project, BWR Integrated Surveillance Program Implementation Plan," dated December 22, 2000. Additional information necessary to establish the technical basis for, and proposed implementation of, the BWRVIP ISP was provided in letters from the BWRVIP to the

NRC dated December 15, 2000, and May 30, 2001. The NRC staff approved the proposed BWRVIP ISP in an SE dated February 1, 2002 (ADAMS Accession No. ML020380691). The proposed ISP was consolidated into an approved topical report dated October 2002 and designated as BWRVIP-86-A (ADAMS Accession No. ML023190491), "Updated BWR ISP Implementation Plan," and found acceptable per NRC letter dated December 16, 2002.

The NRC staff's SE required that plant-specific information be provided by BWR licensees who wish to implement the BWRVIP ISP for their facilities. The plant-specific information must demonstrate that each reactor has an adequate dosimetry program and that there is an adequate arrangement for sharing data between plants. LGS, Units 1 and 2 participation in the ISP was approved by letter dated November 4, 2003, "Limerick Generating Station Units 1 and 2 - Issuance of Amendment Re: Revision to the Reactor Pressure Vessel Material Surveillance Program," (ADAMS Accession No. ML032310540)," which concluded that the participation of LGS, Units 1 and 2 in the ISP can be implemented as the basis for demonstrating the facility's continued compliance with the requirements of Appendix H to 10 CFR Part 50.

The ISP meets the requirements of 10 CFR 50, Appendix H and provides several advantages over the original program. The surveillance materials in many plant-specific programs do not represent the best match with the limiting vessel beltline materials since some were established prior to 10 CFR 50, Appendix H requirements. Also, the ISP allows for better comparison to unirradiated material data to determine actual shifts in toughness. Finally, for many plants, ISP data will be available sooner to factor into plant operations since there are more sources of data. The current withdrawal schedule for both units is based on the latest NRC-approved revision of BWRVIP-86-A. Based on this schedule, LGS, Units 1 and 2 are not scheduled to withdraw any additional material specimens. Instead, limiting weld and plate materials for LGS, Units 1 and 2 are monitored through representative surveillance material source capsules in the ISP, as indicated in Table 4-2 of BWRVIP-86-A. The end of license (EOL) fluence value used in the development of the ISP surveillance capsule test plan schedule, 1.3×10^{18} n/cm^2 at ¼ thickness (1/4T), remains unchanged in the proposed MUR; therefore, no changes are required to the currently-approved surveillance capsule program.

Based on the above, the staff determined that with no change of the EOL fluence, there will be no impact on the EOL transition temperature shift values and consequently no change to the capsule withdrawal schedule of the BWRVIP ISP, currently in the LGS, Units 1 and 2 licensing basis. As described in the next section regarding P-T curves, there is significant margin between the actual plant fluence values and the EOL fluence value used in the supporting analysis to account for any increase in fluence due to the small power level increase. Therefore, the staff determined that the LGS, Units 1 and 2 RPV surveillance program would continue to meet the requirements of 10 CFR Part 50, Appendix H, under the MUR power uprate condition.

3.9.2 P-T Limits and Use

Regulatory Evaluation

10 CFR Part 50, Appendix G 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 AOOs and hydrostatic tests. The 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 effective full-power years (EFPY) specified for the proposed MUR power uprate considering neutron embrittlement effects.

Technical Evaluation

The current P-T Limits for LGS, Units 1 and 2 were developed in April 2000 and July 2000, respectively, with curves generated for 22 and 32 EFPY. By letter dated January 2, 2003 (ADAMS Accession No. ML030030022), NRC staff approved the submitted information to increase the licensed period of applicability of the P-T curves to 32 EFPY for LGS, Units 1 and 2. In addition, the SE stated the following:

> The original peak pressure vessel inside surface fluence was calculated to be 1.7×10^{18} n/cm^2. This was subsequently increased by 10% (for future power uprates) and was assumed to apply from the reactor startup rather than from the actual power uprate. In addition, the present uprate is only 5%. The 10% increase and its application from the beginning of power operation give the 32 EFPY value of the vessel fluence a significant margin of conservatism. The final fluence value for 32 EFPYs used for the calculation of the P-T curves was 1.9×10^{18} n/cm^2 or 1.3×10^{18} n/cm^2 at the 1/4T vessel thickness. The recalculated fluence value at 1/4T is 0.89×10^{18} n/cm^2, which gives the current P-T curves a significant margin of conservatism.

The NRC staff noted that the current thermal licensed power P-T curves, contained in "Pressure-Temperature Curves for [Philadelphia Electric Company] PECO Energy Company Limerick 1 (GE-NE-B11-00836-00-01, Rev. 0)," and "Pressure-Temperature Curves for PECO Energy Company Limerick 2 (GE-NE-B11-00836-00-02, Rev. 0)," did not include an analysis of the water level instrumentation nozzles within the RPV beltline region. The staff requested that the licensee provide a technical basis to support the statement in Attachment 6 to the March 25, 2010, submittal, Section 3.2.1, "Fracture Toughness," that the water level instrumentation nozzle that occurs within the RPV beltline region is bounded by the current licensed thermal power P-T curves. By letter dated August 12, 2010, the licensee provided a description of the application of the methodology defined in Appendix J of NEDC-33178P-A, "[General Electric] GE Hitachi Nuclear Energy Methodology for Development of Reactor Pressure Pressure-Temperature Curves," for evaluating the curve representing the LGS, Units 1 and 2 water level instrumentation nozzles. This methodology was approved in an SE dated April 27, 2009 (ADAMS Accession No. ML091100117). These results were compared to the currently licensed LGS, Units 1 and 2 P-T limit curves. The results for LGS, Unit 1 demonstrate a 20 °F margin for Curve A and a 6 °F margin for Curves B and C, up through 32 EFPY. For LGS, Unit 2, the results demonstrate a 60 °F margin for Curve A and a 20 °F margin for Curves B and C, up through 32 EFPY. Therefore, it was concluded that the curves for the LGS, Units 1 and 2 water level instrumentation nozzles are bounded by the licensed LGS, Units 1 and 2 P-T limit curves. In Attachments 6 and 8 of the licensee's submittal, adjusted reference temperature (ART) calculations were performed for the LGS, Units 1 and 2 RPV materials using the 32 EFPY neutron fluence values. Since the fluence values did not change from the values in the current P-T limits submittals for LGS, Units 1 and 2, the ART values remain unchanged from the ART values contained in the current P-T limits submittals for LGS, Units 1 and 2, unless other factor(s) used in determining ART also have changed. The licensee's submittal dated August 30, 2010, "Upper Shelf Energy Evaluation for LPCI Nozzle Forging Material," stated in Table 5-3 for LGS, Unit 2 LPCI nozzle heat Q2Q33W, "as no copper data were reported, the copper content for this material was determined using heats of materials used for beltline nozzles at other plants. The mean from nine nozzles plus one standard deviation was used to obtain a value of 0.15% [weight percent]." Table 5-1 and Table 5-2, for LGS, Unit 1 LPCI nozzle heats Q2Q25W and Q2Q35W, respectively, also stated that "as no copper data were reported for the nozzle forging materials, so the copper content for this material was determined using heats of materials for beltline nozzles at other plants. The mean plus one standard deviation value was determined to be 0.18%."

The copper content weight percent value assigned to the SA508-2 RPV nozzle forgings lacking measured values should be consistent for the different LPCI nozzle forging material heats. NRC staff analysis of copper values for SA508-2 RPV nozzle forgings concluded that heats of SA508-2 RPV nozzle forgings with no measured copper values should use a best-estimate value for copper content of 0.18 weight percent. This is consistent with LGS, Unit 1 ART data and calculations. The licensee's submitted value of 0.15 weight percent copper for LGS, Unit 2 LPCI nozzle heat Q2Q33W is non-conservative, and a best-estimate value for copper content of 0.18 weight percent should have been used. However, even with this modification, the nozzle forging ART values are not limiting with respect to the determination of the LGS, Unit 2 P-T limits. The limiting ART material for LGS, Unit 2 remains Lower Plate 14-2, heat B3416-1, with a calculated ART value of 122 °F at 32 EFPY.

Since the projected 32 EFPY neutron fluence values, which reflect the effects of the MUR power uprate, are consistent with the previously submitted fluence values in the current 32 EFPY P-T limits for LGS, Units 1 and 2, the ART value for the limiting material and P-T limits remain bounded by the current analyses. Therefore, the NRC staff finds the proposed MUR power uprate to be acceptable with respect to the P-T limits.

3.9.3 USE Technical Evaluation

Appendix G to 10 CFR Part 50 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. Appendix G to 10 CFR Part 50 states:

> Reactor vessel beltline materials must have Charpy upper-shelf energy in the transverse direction for base material and along the weld for weld material according to the ASME Code, of no less than 75 [foot-pounds] ft-lb initially and must maintain Charpy upper-shelf energy throughout the life of the vessel of no less than 50 ft-lb, unless it is demonstrated in a manner approved by the Director, Office of Nuclear Reactor Regulation, that lower values of Charpy upper-shelf energy will provide margins of safety against fracture equivalent to those required by Appendix G of Section XI of the ASME Code.

In Section 3.2.1, "Fracture Toughness," of Attachments 6 and 8 to the submittal, the licensee stated:

> The USE will remain greater than 50 ft-lb for the design life of the vessel or maintain the margin requirements of 10 CFR 50, Appendix G as defined in RG 1.99, Rev. 2, "Radiation Embrittlement of Reactor Vessel Materials." The minimum USE for the LGS-1 beltline materials is 24 ft-lb for 32 EFPY and for the LGS-2 beltline materials is 25 ft-lbs for 32 EFPY. Many of the Limerick RPV materials do not have sufficient unirradiated USE data, and Charpy data from low temperature tests were used to develop an initial USE. Therefore, Equivalent Margin Analyses were performed for the limiting beltline plate, weld, and nozzle

forging materials to assure qualification. These values are provided in Tables 3-1 and 3-2 for Limerick Units 1 and 2, respectively.

Tables 3-1 and 3-2 of Attachments 6 and 8 to the submittal contained USE calculations for 32 EFPY for LGS, Units 1 and 2, respectively. As noted in Section 2.2.2 above, the weight percent copper for the LGS, Unit 2 LPCI nozzle forgings should be revised to reflect the change from 0.15 weight percent to 0.18 weight percent copper. The calculated 32 EFPY USE values in Table 3-2 of Attachment 6 to the submittal did not change as a result of the revised copper weight percent values for the LGS, Unit 2 LPCI nozzle forging materials.

Tables 3-1 and 3-2 of Attachments 6 and 8 to the submittal, identified RPV beltline materials with projected USE values below 50 ft-lb:

From Table 3-1: LGS-1	From Table 3-2: LGS-2
Lower Plate C7688-1	Lower Plate B3312-1
Lower-Intermediate Plate C7677-1	Lower Plate B3416-1
Vertical Weld BF, S3986/RUN934	Lower Plate C9621-2
Girth Weld AB, 07L857/B101A27A	Vert. Weld BA,BB,BD,BE,BF, 32A2671/H019A27A
Girth Weld AB, 09M057/C109A27A	Vert. Weld BC,BD,BE,BF, 07L669/K004A27A
Girth Weld AB, 03M014/C118A27A	Girth Weld AB, 07L857/B101A27A
LPCI Forging 45°, Q2Q25W	Girth Weld AB, 03M014/C118A27A
LPCI Forging 135°, Q2Q35W	LPCI Weld KA, C3L46C/J020A27A
LPCI Forging 225°, Q2Q25W	LPCI Weld KA, 422B7201/L030A27A
LPCI Forging 315°, Q2Q35W	LPCI Forging 892L-1, Q2Q33W
	LPCI Forging 892L-2, Q2Q33W
	LPCI Forging 892L-3, Q2Q33W
	LPCI Forging 892L-4, Q2Q33W

The following materials are identified in Tables 3-1 and 3-2 of Attachments 6 and 8 to the submittal, as having initial transverse USE values below 75 ft-lb:

From Table 3-1: LGS, Unit 1	From Table 3-2: LGS, Unit 2
Lower Plate C7698-2	Lower-Intermediate Plate C9569-2
Lower Plate C7688-2	Lower-Intermediate Plate C9526-1
Lower-Intermediate Plate C7689-1	Lower-Intermediate Plate C9526-2
Lower-Intermediate Plate C7698-1	Vert. Weld BA, BC, 03R728/L910A27A
Vert. Weld BE, 411A3531/H004A27A	
Vert. Weld BA,BB,BD,BF, 06L165/F017A27A	
Girth Weld AB, 411A3531/H004A27A	
LPCI Nozzle Weld 07L669/K004A27A	

The equivalent margins analysis (EMA) analyses were performed using the methodology of NEDO-32205, Revision 1, "10 CFR Part 50 Appendix G Equivalent Margins Analysis for Low Upper Shelf Energy in BWR/2 through BWR/6 Vessels." The NRC staff approved the use of this report for LGS, Units 1 and 2, under conditions described in the SE dated September 7, 1995 (ADAMS Legacy Library Accession No. 9509130067). For the proposed MUR power uprate the NRC staff confirmed the accuracy and applicability of this methodology and associated calculations for the weld and plate materials.

To confirm compliance with 10 CFR Part 50, Appendix G, NRC staff applied the methodology of NEDO-32205, Revision 1 to the above materials. USE decrease was obtained from Regulatory Guide 1.99, Revision 2, Figure 2 and weld materials required adjustment because the measured USE decrease based on surveillance material testing exceeded the predicted USE decrease from Figure 2. NRC staff concluded that with the exception of the LPCI forgings, LGS, Unit 2 LPCI Weld KA heat C3L46C/J020A27A, LGS, Unit 2 LPCI Weld KA heat 422B7201/L030A27A, and LGS, Unit 1 LPCI Nozzle Weld heat 07669/K004A27A, the materials for LGS, Units 1 and 2, with initial USE values below 75 ft-lb, and predicted USE values below 50 ft-lb, are bounded by the EMA methodology of NEDO-32205, Revision 1.

The LPCI forgings are SA-508-2 materials. Since this material was not included in the NEDO-32205, Revision 1 EMA methodology, the NRC staff issued the following RAI regarding the licensee's EMA analyses for the LGS, Units 1 and 2 LPCI forging materials:

> Appendix G to 10 CFR Part 50, "Fracture Toughness Requirements," states that RPV beltline materials... must maintain USE throughout the life of the RPV of no less than 50 foot-pounds (ft-lbs) unless it is demonstrated in a manner approved by the Director, Office of Nuclear Reactor Regulation, that lower values of Charpy USE will provide margins of safety against fracture equivalent to those required by Appendix G of Section XI of the ASME Code. The submittal states that the minimum USE for LGS, Unit 1 is 24 ft-lbs at 32 EFPY for LPCI nozzle forging Q2Q35W, and is 25 ft-lbs for LGS, Unit 2 at 32 EFPY for LPCI nozzle forging Q2Q33W, and therefore "EMAs were performed for the limiting [RPV] beltline plate, weld and nozzle forging materials." The licensee states that NEDO-32205-A, Rev. 1, "10 CFR 50 Appendix G EMA for Low USE in [Boiling Water Reactor] BWR/2 through BWR/6 Vessels," was used with a bounding peak fluence of 1.9 x 10^{18} n/cm^2 [neutrons per square centimeter] to evaluate the vessel against the requirements of 10 CFR Part 50, Appendix G. In NEDO-32205-A, Rev. 1, the materials addressed in the analysis included: SA302 Grade B and Grade B Modified low allow steel plate, SA533 Grade B Class 1 low alloy steel plate, Shielded Metal Arc Welds, Electroslag Welds, Submerged Arc Welds (SAW) made with non-Linde 80 flux, and SAW with Linde 80 flux. The nozzle materials were not included in the NEDO-32205-A, Rev. 1 analysis. Therefore, the staff does not find the application of NEDO-32205-A, Rev. 1 acceptable for demonstrating compliance with Appendix G to 10 CFR Part 50 for LGS, Units 1 and 2 nozzle materials. The methodology contained in NEDO-32205-A, Rev. 1, is applicable only to the materials analyzed in the report. For all [RPV] beltline materials with USE values below 50 ft-lbs at 32 EFPY, the licensee must submit analyses to demonstrate that the lower values of Charpy USE will provide margins of safety against fracture equivalent to those required by Appendix G of Section XI of the ASME Code.

By letter dated August 30, 2010 (ADAMS Accession No. ML102440265), the licensee responded with additional information to support review of the proposed changes. Charpy V-notch (CVN) specimen testing is conducted to develop a brittle-ductile transition and USE for a material. CVN tests are considered to exhibit upper-shelf behavior when the fracture appearance is 100 percent shear. The initial CVN USE values reported for LGS, Units 1 and 2 LPCI nozzle forgings were based on data that were the result of low temperature CVN specimen testing (-20 °F), and 100 percent shear was not obtained (licensee specimens showed 40 percent shear). In addition, orientation of the CVN specimens was not indicated, therefore in accordance with MTEB 5-2, "NRC Branch Technical Position for Fracture Toughness Requirements," the licensee applied a correction factor of 65 percent for conversion of the longitudinal properties to transverse properties to the limited CVN data. Therefore, the licensee identified the minimum initial USE for the LGS, Unit 1 LPCI forging as 26.7 ft-lbs, and 28 ft-lbs for the LGS, Unit 2 LPCI forging. For this review, the NRC staff concluded, based on the limited CVN testing data and absence of data at 100 percent shear, that for the LGS, Units 1 and 2 LPCI forgings the reported minimum initial USE values do not accurately represent the USE values for these materials.

Since the EOL USE values were below 50 ft-lbs, the licensee evaluated precedents with regard to how similar situations had been addressed previously by other licensees. The Brunswick Steam Electric Plant (BSEP), Units 1 and 2, similarly lacked initial USE values for SA-508-2 instrument nozzle forgings. On April 4, 1997 (ADAMS Legacy Library Accession No. 9704170412), the licensee for BSEP, Carolina Power & Light (CP&L), submitted Altran Technical Report 96124-TR-01, in which an extensive database search and analysis showed that the BSEP, Units 1 and 2 nozzle forgings should have had an initial USE of at least 70 ft-lb. By letter dated October 16, 1998, "Evaluation of the January 17, 1992, Operating Transient at the Brunswick Steam Electric Plant, Unit 1, and Evaluation of Carolina Power & Light Company's Equivalent Margins Analysis of the N-16A/B Instrument Nozzles at the Brunswick Steam Electric Plant, Units 1 and 2" (ADAMS Legacy Library Accession No. 9810230079), the NRC staff concluded that CP&L's method for establishing the initial USE value of 70 ft-lb for the nozzle forgings, when coupled with the results of CP&L's EMA for the nozzle forgings, was sufficiently conservative and therefore, acceptable.

Exelon updated the data search of initial USE values for SA-508-2 forging materials. Table 5-5 of the licensee's August 30, 2010, submittal contains several additional points not included in the Altran Technical Report 96124-TR-01 database. In both databases, the lowest initial USE value for an SA-508-2 material remains the Sequoyah, Unit 1 lower shell forging 04 with an initial USE value of 72 ft-lb. This was reported as 70 ft-lb in the Altran Technical Report 96124-TR-01, and revised to 72 ft-lb based upon a reanalysis of the data in WCAP-15224, "Analysis of Capsule Y from Tennessee Valley Authority Sequoyah Unit 1 Reactor Vessel Radiation Surveillance Program," June 1999 (ADAMS Legacy Library Accession No. 9909160101). The NRC staff compared the initial LGS, Units 1 and 2 LPCI nozzle forging CVN USE test data with the initial CVN USE test data from WCAP-15224 for lower shell forging 04, and found them to be reasonably consistent. Exelon chose to use the lower CVN initial USE value of 70 ft-lb, since it was the more conservative value and consistent with the initial USE previously accepted by the NRC for BSEP, Units 1 and 2.

Exelon provided a plant-specific EMA evaluation of the LGS, Units 1 and 2 LPCI nozzle forgings to demonstrate compliance with the USE requirements of 10 CFR Part 50, Appendix G. The plant-specific EMA analysis for LGS, Units 1 and 2 LPCI nozzle forging materials followed the methodologies of:

- ASME Code Case N-512-1, "Assessment of RPVs with Charpy USE Less than 50 ft-lb," August 24, 1995,

- Appendix K of ASME Section XI, "Assessment of RPVs with Low USE Charpy Impact Energy Levels," December 1993, and

- Regulatory Guide 1.161, "Evaluation of RPVs with USE less than 50 ft-lb," June 1995.

Although Regulatory Guide 1.161 does not address forging materials, Appendix K of ASME Section XI, Section K-4210 indicates that it is applicable to SA-508-2 materials. The EMA evaluation found the Level A/B Condition is governing. Based on an initial USE of 70 ft-lb, and considering a 10.8 percent decrease in USE at 32 EFPY, as prescribed by Regulatory Guide 1.99, Revision 2, the NRC staff concludes that the LGS, Units 1 and 2 LPCI nozzle forgings are predicted to have a USE of 62.4 ft-lb at 32 EFPY. Therefore, the NRC staff concludes that Exelon's method for establishing the initial USE and 32 EFPY USE values (70 ft-lb and 62.4 ft-lb, respectively) for the LGS, Units 1 and 2 LPCI nozzle forging materials, when coupled with the results of Exelon's EMA for those materials, is sufficiently conservative, and consequently demonstrates that LGS, Units 1 and 2 LPCI nozzle forging materials meet the USE requirements of 10 CFR Part 50, Appendix G.

In Table 3-1 of the submittal, the LGS, Unit 1 LPCI nozzle weld (heat number 07L669/ K004A27A) was identified as having an initial USE value of 54 ft-lb. In Table 3-2 of the submittal, LGS, Unit 2 LPCI nozzle weld KA (heat number C3L46C/J020A27A) and weld KA (heat number 422B7201/L030A27A) were identified as having initial USE values of 40 ft-lb and 38 ft-lb, respectively. By letter dated August 30, 2010, the licensee in discussing the LPCI forgings stated that, "All other reactor vessel beltline materials with USE values less than 50 ft-lb at 32 EPY have been evaluated using EMAs in accordance with NEDO-32205-A, Rev. 1." The NRC staff determined that the methodology contained in NEDO-32205-A, Rev. 1 is applicable only to the materials analyzed in the report. The LGS, Units 1 and 2 LPCI nozzle weld materials were not included in the NEDO-32205-A analysis.

By letter dated December 17, 2010, the licensee provided additional information to demonstrate compliance of the LPCI nozzle weld materials with 10 CFR Part 50, Appendix G requirements. The licensee provided Certified Material Test Reports (CMTRs) for the three heats of shielded metal arc welds (SMAWs) in the LPCI nozzle welds. All CVN impact tests were conducted at a lower temperature than the 40 °F reference temperature of NEDO-32205. The corresponding values of percent shear were less than 100 percent. As noted previously, CVN tests are considered to exhibit upper-shelf behavior when the fracture appearance is 100 percent shear. The NRC staff concluded, based on the limited CVN testing data for the LGS, Units 1 and 2 LPCI nozzle welds, and the absence of data at 100 percent shear, that the reported minimum initial USE values do not accurately represent the USE values for these materials.

By letter dated December 17, 2010, the licensee submitted a methodology of applying a ratio factor based on a survey of SMAW data. By applying the ratio factor, the lower temperature CVN data was projected to equivalent 40 °F CVN data in order to demonstrate applicability of the NEDO-32205-A report. The licensee attached SMAW materials CMTRs for all available RPV purchase orders. Additional data was obtained from GE reports that identified average CVN results at the two respective temperatures. A ratio was developed using the low CVN data at each temperature for the 11 CMTR data sets, and the six CVN averages from the GE reports. The NRC staff review did not support the selection of only the low CVN data from each CMTR set in development of the overall ratio. In addition, the NRC staff found significant variation in the ratios for each individual CMTR data set, and two CMTR CVN data sets had ratios of less than one, which would represent highly atypical CVN curves. Therefore, the NRC staff did not accept the development of a ratio to project the lower temperature CVN data to 40 °F for the LGS, Units 1 and 2 LPCI nozzle weld material.

The NRC staff reviewed the CMTR CVN data for the LGS, Units 1 and 2, LPCI nozzle weld materials at the lower temperature. The range of data was bounded by the range of data submitted from the other plants at that same temperature. Since the lower temperature CVN data for the LGS, Units 1 and 2 SMAW welds was bounded by the available lower temperature CVN data for SMAW materials from other plants, the NRC staff concludes that the 40 °F data would similarly be bounded. This conclusion is consistent with respect to the expected behavior of material CVN curves. The lowest value of 40 °F CVN data for the LGS SMAW materials from the CMTR reports is bounded by the data in Figure 8-5 of the NEDO-32205-A report. Therefore, the NRC staff concludes that use of the EMA methodology in the NEDO-32205-A report is acceptable for the LGS, Units 1 and 2 SMAW materials.

The lowest initial USE for an RPV beltline SMAW material is 73 ft-lb. The NRC staff concluded that since the lower temperature CVN data for the LGS, Units 1 and 2 SMAW materials is bounded by the lower temperature CVN data for other SMAW materials, that it would be conservative to assume that the initial USE values would be bounded as well, and an initial USE value of 73 ft-lb may be assumed for the LGS, Units 1 and 2 SMAW materials. Since the initial USE value is below 75 ft-lb, in accordance with 10 CFR Part 50, Appendix G, the NRC staff ensured that an EMA was performed. The percentage decrease in USE for the LGS, Units 1 and 2 SMAW materials predicted by Regulatory Guide 1.99, Rev. 2 is significantly below the predicted 32 EFPY percent decrease of 34 percent from the NEDO-32205-A report. Therefore, the NRC staff concludes that the LGS, Units 1 and 2 SMAW materials meet the requirements of 10 CFR Part 50, Appendix G.

Hence, the NRC staff concludes that all materials from the LGS, Units 1 and 2 RPVs which require evaluation in accordance with 10 CFR Part 50, Appendix G for upper-shelf energy have been demonstrated to meet the requirements of 10 CFR Part 50, Appendix G either by having their USE values comply with criteria provided in the regulation, or by having an acceptable EMA performed for them as permitted by the regulation.

3.9.4 Conclusion

The NRC staff has reviewed the licensee's proposed LAR to increase the rated core thermal power for LGS, Units 1 and 2, by approximately 1.65 percent and has evaluated the impact that the MUR power uprate conditions will have on the structural integrity assessments for the RPV. The staff has determined that the changes identified in the proposed LAR will not impact the remaining safety margins required for the following structural integrity assessments: (1) RPV surveillance program; (2) P-T limits; and (3) RPV USE assessment. Therefore, the staff finds that sufficient information has been provided to support the requested MUR power uprate through 32 EFPY.

3.10 MUR Power Uprate - Electrical Systems

3.10.1 Regulatory Evaluation

The regulatory framework which the staff applied in its review of the application includes:

- GDC-17, "Electric Power Systems," of 10 CFR Part 50, Appendix A states, in part, that an onsite power system and an offsite electrical power system be provided with sufficient capacity and capability to permit functioning of structures, systems, and components important to safety. LGS has onsite and offsite electrical power systems provided in accordance with GDC-17, as described in the LGS UFSAR, Section 3.1.

- 10 CFR 50.63 requires that all nuclear plants have the capability to withstand a loss of all alternating current [AC] power (SBO) for an established period of time, and to recover such an event.

- 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.

- RIS 2002-03, which the licensee states the application is based upon.

3.10.2 Technical Evaluation

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

- AC Distribution System
- Power Block Equipment (Main Generators, Transformers, Isolated-phase bus ducts)
- Direct Current (DC) System
- Emergency Diesel Generators (EDG)
- Switchyard
- Grid Stability
- SBO
- EQ Program

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 13.8 kilovolt (kV), 4.16 kV, 480 volt (V), and 120 V systems (not including the EDGs). The licensee stated that the proposed MUR power uprate will cause a small increase in condensate pump motor horsepower (hp) duty. The NRC staff asked the licensee to explain the small increase in the condensate pump hp loading in an RAI and assess its impact on the degraded voltage relay allowable values. By letter dated August 12, 2010, the licensee responded, explaining that the projected increase of 43 hp for each of the three running pump motors per operating unit from 4094 to 4137 hp is bounded by the electrical load flow analysis which provides the degraded voltage relaying bases assuming the nameplate motor rating of 4410 hp. In addition, a 120V electrical load panel associated with the implementation of LEFM Checkplus system will be added feeding an air conditioning unit, power conditioning equipment, computer boards, and the LEFM meter cabinet. This panel will be supplied from other non-safety related panels. The licensee stated that the increased loads on the panels were bounded by the transformer ratings used in the evaluation and analysis. Based on above, the licensee concluded that there are no required changes to the electrical load flow analysis associated with degraded voltage since analytical values used as input remain bounding and there is no required change to the degraded voltage relay allowable value. The staff reviewed the application and licensee response to the staff RAI and concludes that the AC system has adequate capacity to support the plant loading for the uprated condition.

Power Block Equipment

In the submittal dated March 25, 2010, the licensee stated that an increase in each unit main generator gross electrical output remains bounded by the maximum design ratings of the generator and that the new operating point of the generator is below and within the main generator maximum capability curve. In addition, the licensee's review of the existing offsite electrical equipment concluded that the gross generator Megawatts-electric (MWe) output for summer and winter operations will be on the existing generator capability curve with 0.9 power factor. By letter dated August 12, 2010, in response to an NRC staff RAI, the licensee provided a copy of the main generator capability curve to demonstrate that the generator will be operating within its capability curve. The NRC staff reviewed the information provided and concurs with the licensee that the predicted generator gross output due to MUR power uprate is within the generator's capability.

In the submittal dated March 25, 2010, the licensee discussed the ratings of the main generator step up transformers but did not provide the maximum loadings due to the proposed MUR power uprate. As such, the staff requested in an RAI to clarify the expected maximum loading on these transformers to demonstrate that the uprated loadings of the Unit 1 and Unit 2 main generator step-up transformers will be within the design MVA rating. In its response to the staff's RAI, dated August 12, 2010, the licensee clarified that the maximum MVA the generator can provide is 1264.97, which is less than the 1266/1575 MVA ratings of the main transformers on Unit 1 and Unit 2, respectively. Based on the information provided, the staff agrees with the licensee that the main generator step-up transformers are capable of operation at uprated conditions. In Attachments 6 and 8 of the submittal dated March 25, 2010, the licensee provided a discussion on isolated-phase bus ratings and concluded that the isolated phase bus duct is adequate for both rated voltage and low voltage current output due to proposed MUR. The staff's review of the LAR finds that the isolated-phase bus duct is capable of operation at uprated condition and is acceptable.

DC System

LGS UFSAR Section 8.3.2.1.1 states that each 125/250 V DC system is comprised of two 125 V batteries, each with its own charger, a fuse box for protection of each of the several 125 V power distribution circuits supplying 125/250 V motor control centers (MCCs) (one for Division I and two for Division II), and three 125 V power distribution panels. The DC power distribution system provides control and motive power for various systems and components. These loads are used as inputs for the computation of load, voltage drop, and short circuit current values. In the application, the licensee stated that DC loading requirements documented in the UFSAR and station load calculations were reviewed, and no reactor power-dependent loads were identified. According to the licensee, operation at the uprated power level does not increase any loads or revise control logic. Therefore, there are no changes to the load, voltage drop or short circuit current values. The staff reviewed the LAR and the analysis and loads presented in the LGS UFSAR and finds that the proposed power uprate does not impact DC system loads. As such, the staff finds the licensee's evaluation of DC System acceptable.

EDGs

The standby emergency AC power source for each LGS unit consists of four EDG sets. Each EDG is provided as a standby source of power for one of the four Class 1E AC load groups in each unit. The EDG system automatically provides a safety-related source of AC power to sequentially energize and restart loads necessary to shutdown and maintain the plant in a safe shutdown condition to mitigate the consequences of postulated DBAs.

The licensee has reviewed the emergency onsite power system and has found no impact on EDG loading from the MUR power uprate. The duty cycle and duration for design basis EDG loads is based on analytical power levels of at least 102 percent of the current licensed thermal power. Hence, the EDG system is bounded by LGS current licensing basis.

Based on the above, the staff, after reviewing the LAR and UFSAR, concludes that the proposed power uprate does not impact EDG system loads and mission time. Therefore, the staff finds that the analyses for the EDG system bound MUR power uprate conditions, and the onsite power system conforms with the requirements of GDC-17.

Switchyard

The LGS UFSAR, Section 8.2, reflects 220 kV and a 500 kV switchyards supplying offsite power to the LGS, Units 1 and 2. The 220 kV substation is supplied from three transmission sources and the 500 kV switchyard is supplied from four transmission sources. Offsite power to the LGS units is supplied from two independent, physically separated sources from these switchyards, a 220-13 kV transformer (No. 10) located at the LGS 220 kV substation and a 13 kV tertiary winding on the No. 4A and 4B 500-220 kV bus tie autotransformers through the No. 20 13-13 kV regulating transformer. The primary function of the switchyard and distribution system is to connect the station electrical system to the transmission grid. In Attachments 6 and 8 of the submittal dated March 25, 2010, the licensee has concluded that the transformers and the associated switchyard components are rated for maximum transformer output and are adequate for the uprated condition. Based on the review of UFSAR and review of LAR, the staff agrees and concludes that the impact due to power uprate on the switchyard equipment remains within the existing limits. Therefore, the staff finds that the switchyard equipment and power supply systems will continue to conform with GDC-17.

Grid Stability

In Attachment 1 of the LAR, the licensee stated that two grid studies were completed to support the proposed power uprate. PJM [Pennsylvania-New Jersey-Maryland] Interconnection (PJM), the grid operator, completed a system stability analysis to assess the impact of the uprate on the rotor angle stability of generating plants in the PECO and neighboring control areas. The analysis bounds the expected output of each main generator under uprated conditions. The analysis conclusions are as follows: 1) All of the primary-clearing scenarios were found to be stable; 2) All of the primary-clearing scenarios with maintenance outages considered were found to be stable; and 3) All of the breaker failure scenarios considered in the study were found to be stable.

A second study was completed by PECO Transmission Planning to determine if the capacity and capability of the preferred power supply ensures the design and licensing basis for the Limerick Generating Station under uprated conditions. Adequacy of the preferred power supply is determined by verification of the transmission system's capability to maintain the post-trip voltage drops and voltages at the safety buses to remain above the reset value of the degraded voltage relay on a steady-state basis. The study assumed a 1240 MWe output for each LGS main generator, which is the expected maximum output of the main generators, as well as maximum mega volt ampere output, for both summer and winter conditions. Power flow simulations were performed using 2010 transmission grid models. Two independent offsite sources are required to be operable in accordance with Limerick TS 3.8.1.1, "AC Sources-Operating." The two primary offsite sources are No. 10 Start-up Transformer and No. 20 Start-up Transformer. The alternate third offsite source is the 8A/8B Transformer. The study demonstrates that the post-trip voltage drops are within the limits established in site procedures to maintain operability. The study also shows that the transmission system is capable of providing adequate voltage as required for operability of the offsite sources.

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

SBO

Section 50.63 of 10 CFR 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 a SBO.

In Attachments 6 and 8 of the application dated March 25, 2010, the licensee stated that the LGS SBO evaluation was performed assuming greater than or equal to 102 percent of the current licensed thermal power. Therefore, the postulated SBO scenarios for the thermal power optimization operation are bounded by the current evaluations. Based on this information, the staff finds that the MUR power uprate will have no impact on Limerick's SBO coping duration. Therefore, the staff finds that Limerick will continue to meet the requirements of 10 CFR 50.63 under MUR power uprate conditions.

EQ Program

In its application, the licensee stated that the proposed 1.65 percent power uprate does not increase the nominal vessel dome pressure; there is a very small effect on pressure and temperature conditions experienced by equipment during normal operation and accident conditions. The resulting environmental conditions are bounded by the existing environmental parameters specified for use in the environmental qualification program. In its response dated August 12, 2010, to an NRC staff RAI, the licensee stated that there will be a small increase in normal operational radiation levels. However, the radiation levels will be lower than the bounding parameter used by the LGS EQ program. As such, the licensee concluded that there will not be any impact on existing equipment qualification of the plant. Based on this information, the staff agrees that the current EQ parameters remain bounding for the proposed MUR power uprate. Therefore, the staff finds that the proposed MUR power uprate will have no impact on LGS EQ Program and will continue to meet the requirements of 10 CFR 50.49.

3.10.3 Electrical Systems Conclusion

Based on the regulatory and technical evaluations above, the staff finds that LGS, Units 1 and 2 will continue to meet GDC-17, 10 CFR 50.63, and 10 CFR 50.49. In the area of electrical systems, the application is generally consistent with RIS 2002-03. Therefore, based on these considerations, the staff finds the LGS MUR power uprate for the electrical systems acceptable.

3.11 MUR Power Uprate - Instrumentation and Controls (I&C)

The licensee's submittal references Cameron Topical Report ER-80P, and its supplement, Topical Report ER-157P. These topical reports, 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 measurements.

Topical Report ER-80P describes the LEFM technology, includes calculations of the power measurement uncertainty using a Cameron LEFM Check System in a typical two-loop pressurized-water reactor or two-feedwater-line boiling-water reactor, and provides guidelines and equations for determining the plant-specific power calorimetric uncertainties. Topical Report ER-157P describes the Cameron LEFM CheckPlus System and lists the nonproprietary results of an uncertainty calculation for a typical thermal power measurement of a pressurized-water reactor or boiling-water reactor using either the Cameron LEFM Check or the LEFM CheckPlus System. Together, these two reports provide a generic basis for an MUR power uprate.

Cameron Topical Report ER-739, "Bounding Uncertainty Analysis for Thermal Power Determination at Limerick Unit 1 Using the LEFM CheckPlus System," Revision 1, issued November 2009, and Cameron Topical Report ER-745, "Bounding Uncertainty Analysis for Thermal Power Determination at Limerick Unit 2 Using the LEFM CheckPlus System," Revision 1, issued December 2009, describe the plant-specific bases for the proposed uprate at LGS and were provided with the submittal dated March 25, 2010, as Attachment 9.

3.11.1 Regulatory Evaluation

Topical Report ER-80P and Topical Report ER-157P describe the Cameron LEFM CheckPlus System for the measurement of feedwater flow and provide a basis for the proposed uprate of approximately 1.65 percent of the licensed reactor thermal power. The staff also considered the guidance in RIS 2002-03, in its review of the licensee's submittals for the proposed MUR power uprate request.

3.11.2 Technical Evaluation

Neutron flux instrumentation is calibrated to the core thermal power, which is determined by automatically or manually calculating the energy balance around the plant 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. A more accurate measurement of this parameter will result in a more accurate determination of core thermal power.

The feedwater flow rate is typically measured using a venturi device that generates a differential pressure proportional to the feedwater velocity in the pipe. Because of the high cost of calibrating the venturi device 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 meter to be a viable alternative.

The LEFM CheckPlus system proposed for use at LGS uses a 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 they do in the opposite direction of 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.

The LEFM uses 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 an input to a table of thermodynamic properties of water. The velocity profile correction factor is derived from calibration testing of the LEFM in a plant-specific piping model at a calibration laboratory.

The NRC staff's review of the I&C area covers the proposed plant-specific implementation of the feedwater flow measurement technique and the power increase gained by implementing this technique in accordance with the guidelines (Items A–H) provided in Section I of Attachment 1 to RIS 2002-03. The staff conducted its review to confirm that the licensee's implementation of the proposed feedwater flow measurement device is consistent with the staff-approved Topical Report ER-80P and Topical Report ER-157P and that the licensee adequately addressed the four additional requirements detailed in the staff's topical report approvals as referenced in Section 2.0 of this SE. The NRC staff also reviewed the power measurement uncertainty calculations to ensure: (1) that the conservatively proposed uncertainty value of 0.31 percent correctly accounts for all uncertainties associated with power level instrumentation errors; and (2) that the uncertainty calculations meet the relevant requirements of Appendix K to 10 CFR Part 50.

In the application dated March 25, 2010, Attachment 1, the licensee provided the following information on the Cameron LEFM CheckPlus System feedwater flow measurement technique and its implementation at LGS, Units 1 and 2:

- The LEFM spool pieces will be installed in the feedwater piping of the three loops in each unit. The installations on Loop B and Loop C will be located in straight sections of pipe downstream of existing flow straighteners and upstream of existing feedwater flow nozzles. The installation on Loop A will be upstream of both the flow straightener and the feedwater flow nozzle.

- The transducers will be located in the turbine enclosure above the No. 6 feedwater heater rooms in an anticipated radiation field of 20 [millirem per hour] mR/h at full power. The electronics cabinet will be located in the corridor outside the turbine enclosure No. 6 feedwater heater rooms in an anticipated radiation field of less than 1 mR/h at full power. The licensee does not anticipate any radiation damage or degradation to the instruments (including electronics) due to such exposure.

- The licensee has developed modification packages that outline the steps necessary to install and test the LEFM system on each unit. After each unit has been shut down for its refueling outage, the licensee will install the LEFM spool pieces and the transducers, route the cables, and connect them to the plant process computer. Following installation, the licensee will perform an inservice leak test, compare feedwater flow and thermal power using various computational methods, and conduct the final testing required for commissioning.

Items A-C in Section I of Attachment 1 to RIS 2002-03

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 application, the licensee identified Revision 0 to Topical Report ER-80P and Revision 5 to Topical Report ER-157P as applicable to the Cameron LEFM CheckPlus System. The licensee also referenced NRC SEs for Topical Report ER-80P, dated March 8, 1999, and Topical Report ER-157P, dated December 20, 2001.

The licensee will install Cameron LEFM CheckPlus System in LGS, Units 1 and 2, in straight sections of piping. Installation locations are downstream of the flow straighteners and upstream of the existing feedwater flow nozzle.

Based on its review of the licensee's submittals as reflected in the above discussion, the staff finds that the licensee has sufficiently 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 its discussion of the implementation of the power uprate using this technique follow the guidance in Items A, B, and C of Section I of Attachment 1 to RIS 2002-03.

Item D in Section I of Attachment 1 to RIS 2002-03

Item D in Section I of Attachment 1 to RIS 2002-03 guides licensees to address four criteria when implementing the feedwater flow measurement uncertainty technique. The NRC staff SEs on Topical Reports ER-80P and ER-157P both include these four plant-specific criteria that a licensee must address when referencing these topical reports for a power uprate. The licensee's submittal[4] addresses each of the four criteria as follows:

(1) The licensee should discuss the maintenance and calibration procedures that will be implemented 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 the development of the necessary procedures and documents required for the maintenance and calibration of the new LEFM CheckPlus System. Exelon will revise plant maintenance and calibration procedures to incorporate Cameron's maintenance and calibration requirements before use of the LEFM to raise the power above the current 3458 MWt limit.

    The licensee will develop a preventive maintenance program for the LEFM system in accordance with vendor guidance for the Cameron LEFM CheckPlus System and its preventive maintenance template currently under development for the LEFM system. The licensee will perform power supply and pressure transmitter checks and verify clock settings during each refueling outage. The licensee is implementing these preventive maintenance activities with the associated plant modification package and in conjunction with the development of its preventive maintenance template for LEFMs.

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

    The licensee will continue to perform calibration and maintenance on equipment used for calorimetric computation in the event that the LEFM is nonfunctional in accordance with existing site procedures. Additional details on contingencies for an inoperable LEFM are evaluated as part of this SE in the following section covering Items G–H in Section I to Attachment 1 to RIS 2002-03.

    Based on the review of the licensee submittals, the NRC staff concludes that the licensee adequately addressed Criterion 1 of Item D.

(2) For plants that currently have LEFMs installed, provide an evaluation of the operational and maintenance history of the installed installation and confirmation that the installed

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4. The licensee provided a description of this topic in its application dated March 25, 2010, in Attachment 1 as well as in its supplement dated July 22, 2010, which was submitted in response to an NRC RAI.

instrumentation is representative of the LEFM system and bounds the analysis and assumptions set forth in Topical Report ER-80P.

Licensee Response

As of the date of the application, the licensee had not yet installed the Cameron LEFM CheckPlus System at LGS, Units 1 or 2. Thus, no operational or maintenance history exists for installation at the facility.

Because at the time of the original application the licensee had not yet installed the LEFM at either unit on the LGS site, the staff finds that Criterion 2 of Item D is not applicable.

(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 an alternative approach is used, the application should be justified and applied to both venturi and ultrasonic flow measurement instrumentation installations for comparison.

Licensee Response

The licensee stated that the LEFM uncertainty calculation is based on American Society of Mechanical Engineers Performance Test Code (PTC) 19.1 methodology and Instrument Society of America (ISA)-RP67.04.02-2000. The LEFM system uncertainty calculation methodology is based on a square root of the sum of the squares calculation, which is consistent with the licensee's current core thermal power uncertainty calculation for the existing feedwater instrumentation and with other Cameron engineering reports.

In accordance with Attachment 11 of the LAR submittal, the licensee used CC-MA-103-2001, "Setpoint Methodology for Peach Bottom Power Station and Limerick Generating Station," to perform its overall core thermal power calculation. The licensee stated that the methodology is consistent with NEDC-31336P-A, "General Electric Instrument Setpoint Methodology," which has been reviewed and approved by the NRC (ADAMS Accession No. ML072950103).

Based on the foregoing, the NRC staff concludes that the licensee adequately addressed Criterion 3 of Item D.

(4) For plant installation where the ultrasonic meter (including LEFM) was not installed 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 Cameron LEFM CheckPlus System was calibrated using a site-specific piping configuration at Alden Research Laboratories. As part of the LAR, the licensee submitted Topical Report ER-789, "Meter Factor Calculation and Accuracy Assessment for Limerick Unit 1," Revision 0, issued November 2009, and Topical Report ER-797, "Meter Factor Calculation and Accuracy Assessment for Limerick Unit 2," Revision 0, issued December 2009. According to Topical Report ER-739 and Topical Report ER-745, the mass flow uncertainty for each unit is 0.28 percent. This flow uncertainty then becomes a primary input into the total power uncertainty calculation performed by the licensee.

For both LGS units, the licensee stated that acceptance of the final site-specific uncertainty analyses will occur after it completes the commissioning process and before it implements the proposed changes. The LEFM CheckPlus System was installed in LGS, Unit 1, during the spring 2010 refueling outage and is planned for installation in LGS, Unit 2, in spring 2011 refueling outage.

Based on the foregoing calibration, using a site-specific configuration, the NRC staff concludes that the licensee adequately addressed Criterion 4 of Item D. In addition, the licensee committed to confirming that the in situ test data are bounded by the calibration test data after the final commissioning of the Cameron LEFM CheckPlus System in the LGS units. The NRC staff agrees that this is an appropriate step and concurs that the licensee's commitment management program provides sufficient control over the performance of this activity.

Item E in Section I of Attachment 1 to RIS 2002-03

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

To address Item E, the licensee provided Revision 1 to Topical Report ER-739 and Revision 1 to Topical Report ER-745. By letter dated July 22, 2010, in response to an RAI, the licensee provided Revision 1 to LE-0113, "Reactor Core Thermal Power Uncertainty Calculation Unit 1," and has identified Revision 0 to CC-MA-103-2001 as the methodology used for the calculation.

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 licensee's calculations arithmetically summed uncertainties for parameters that are not statistically independent and that are statistically combined with other parameters. The licensee stated in LE-0113 that the level of confidence of each uncertainty used in the calculation is normalized to the 2σ confidence level and the final uncertainty (i.e., 0.31 percent) is identified as a 2σ uncertainty. The licensee combined random uncertainties using the square root of the sum of the 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 RG1.105, "Setpoints for Safety-Related Instrumentation."

Operation at 3515 MWt with a 0.31-percent uncertainty would ensure that the actual reactor power remains under 3527 MWt. Based on the current licensed thermal power (i.e., 3458 MWt) and the mandated two percent uncertainty in Appendix K to 10 CFR Part 50, LGS is analyzed for

operation at 3527.16 MWt. Thus, operation with the new uprated power level with the LEFM-supported thermal power uncertainty will be bounded by its existing safety analyses where those analyses are performed at 102 percent of the current licensed power level.
The NRC staff finds that the licensee has provided calculations of the total power measurement uncertainty at the plant and has explicitly identified all parameters and its individual contribution to the power uncertainty. Therefore, the licensee has adequately addressed the guidance in Item E of Section I of Attachment 1 to RIS 2002-03.

Item F in Section I of Attachment 1 to RIS 2002-03

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 related to all instruments that affect the power calorimetric as follows:

(1) Maintaining Calibration

The licensee stated that it will perform the calibration using procedures based on the appropriate LEFM CheckPlus System requirements and that it will revise plant procedures to incorporate Cameron's maintenance and calibration requirements before declaring the LEFM CheckPlus System operational. 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 in Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," to 10 CFR Part 50.

After installation, the licensee will maintain the LEFM CheckPlus System software configuration using existing LGS procedures and processes. The licensee maintains its plant computer software configuration in accordance with its change control process, which includes verification and validation of changes to 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 its configuration control processes.

(3) Performing Corrective Actions

The LGS corrective action program documents problems with plant instrumentation identified by LGS personnel. The licensee identifies and implements the necessary corrective actions as part of that program.

(4) Reporting Deficiencies to the Manufacturer

The licensee stated in the LAR that it reports any conditions associated with a vendor's processes or equipment to the vendor to support any corrective action as part of the LGS corrective action program.

(5) Receiving and Addressing Manufacturer Deficiency Reports

The licensee stated that it has existing processes in place to address the receipt of the manufacturer's deficiency reports. Any such deficiencies will be documented in the LGS 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 licensee meets the guidance in Item F of Section I of Attachment 1 to RIS 2002-03.

Items G and H in Section I of Attachment 1 to RIS 2002-03

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 application, the licensee proposed a 72-hour AOT for operating above 3458 MWt (i.e., the current licensed thermal power limit) if the LEFM becomes nonoperational. This AOT will be located in the Technical requirements Manual (TRM), a licensee-controlled document.

As described in response to an NRC RAI dated July 22, 2010, the LEFM system performs online self-diagnostics to verify that system operation is within the design-basis uncertainty limits. Any out-of-specification condition will result in a self-diagnostic alarm condition, either for "alert" status (e.g., increased flow measurement uncertainty) or "failure" status. In either of these cases, the licensee will consider the LEFM nonoperational and apply the actions outlined in the proposed LGS TRM. Additionally, if the communication link between the LEFM system and the plant computer fails, the licensee will consider the LEFM nonoperational and apply the proposed TRM actions.

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 the existing feedwater flow nozzle will be calibrated to the last validated data from the LEFM CheckPlus System measurements to ensure accurate power monitoring during the AOT.

If the core power level is below the current licensed core thermal power (i.e., 3458 MWt) at the time the LEFM is declared nonoperational or if the power level drops below the current licensed core thermal power during the AOT, LGS will not raise the power level above 3458 MWt before the LEFM returns to operational status. Section 3/4.0, "Applicability," of the LGS TRM states that the requirements of TS 3.0.4 are applicable to the TRM. TS 3.0.4 prohibits entering a mode or condition specified in Section 3/4.0 when a limiting condition for operation is not met, except when: (1) the associated actions permit operation in that condition for an unlimited period of time; (2) the licensee has performed a risk assessment; or (3) the TS specifically states otherwise. Exception 1 above cannot be used for the LEFM because the applicability for a proposed LEFM limiting condition for operation applies to core thermal power levels greater than 3458 MWt, and the TRM actions only permit operation above 3458 MWt for 72 hours with an inoperable LEFM. For Exception 2, the licensee will revise the LGS TRM to include the statement that TS 3.0.4.b does not apply to the LEFM. These revised TRM pages were submitted by the licensee by letter dated July 22, 2010. Exception 3 is not provided for in the TRM section for the LEFM. Thus, the application of TS 3.0.4 to the proposed TRM section for the LEFM would prohibit raising power above 3458 MWt without the LEFM being operational.

Both feedwater flow nozzle fouling and instrument measurement drift were considered as potential sources of error within the AOT window. LGS has not experienced any appreciable venturi fouling events in the past several years; however, the licensee has observed and has addressed fouling/defouling events since commencement of operations. The licensee stated that any fouling event would increase the pressure differential across the feedwater flow nozzle, which would tend to overestimate the flow (and thus conservatively force a reduction in power). Although unlikely, other plant parameters would detect a sudden defouling event.

The licensee's July 22, 2010, RAI response made reference the existing feedwater flow measurement data, as compared to what is assumed in the Topical Report ER-80P analysis. The licensee stated that it uses Rosemount 1151 differential pressure transmitters for these measurements, and that based on its calibration records, the LGS instrument drift data are consistent with the measurement drift error values cited in the Topical Report ER-80P analysis. These errors, when extrapolated over a 72-hour period, would be negligible.

The NRC staff reviewed the licensee's submittals and found that the licensee provided sufficient justification for the proposed TRM AOT and the proposed actions to reduce the power level if the AOT is exceeded. Therefore, the licensee has followed the guidance in Items G and H of Section I of Attachment 1 to RIS 2002-03. The NRC staff agrees that the TRM, as described in LGS UFSAR 13.5.3, provides the appropriate location and control mechanism for the LEFM AOT.

Technical Specification Changes

The licensee proposed to modify several TSs related to the power uprate. The licensee performed the setpoint and allowable value changes to the TSs related to a limiting safety system setpoint (i.e., simulated thermal power upscale scram) using the LGS setpoint methodology contained in procedure CC-MA-103-2001. The licensee stated that this procedure is consistent with the methodology contained in NEDC-31336P-A, which the NRC had previously approved (ADAMS Accession No. ML072950103). The licensee's July 22, 2010, response to a RAI cited no differences between the methodologies that would make its calculations less conservative than those in NEDC-31336P-A.

Attachment 13 to the submittal dated March 25, 2010, contains a summary of the calculations used to derive the setpoints and allowable values for the simulated thermal power upscale scram. It also contains calculations related to setpoint and allowable value changes to the control rod block instrumentation setpoints. Attachment 13 identifies that individual uncertainties are normalized to the 2σ level in the setpoint calculations. Changes are also proposed for the TS for a simulated thermal power upscale scram to bring it into alignment with Revision 4 to TSTF-493 by adopting footnotes governing actions that the licensee must take in response to surveillance testing. The NRC staff reviewed the proposed setpoint and allowable value changes in conjunction with the submitted analysis. The revisions proposed are consistent with the change in rated thermal power and are therefore acceptable. The addition of footnotes to Table 4.3.1.1-1 is a conservative change, and is therefore also acceptable.

There are several other setpoint changes proposed in the LAR. The setpoints are related to turbine stop and control value closure, end-of-cycle recirculation pump trip system instrumentation, and recirculation loops and Oscillation Power Range Monitor (OPRM) upscale and are changes to the percent of full power, all of which are aimed at maintaining the TS values equivalent in terms of thermal reactor power. The staff finds that the proposed changes keep the values at the same thermal power levels, and are therefore acceptable.

3.11.3 I&C Systems 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 amendment is consistent with the staff's approved Topical Report ER-80P and its supplement, Topical Report ER-157P. The staff has also determined that the licensee adequately accounted for all instrumentation uncertainties in the uncertainty calculations for the reactor thermal power measurements and demonstrated that the calculations meet the relevant requirements of Appendix K to 10 CFR Part 50, as described in Section 2 of this SE. The staff reviewed the instrumentation setpoint changes proposed and concludes that they appropriately account for operation at the new licensed thermal power level.

Therefore, the NRC staff finds the I&C aspects of the proposed thermal power uprate acceptable.

3.12 MUR Power Uprate - Plant Systems

3.12.1 Regulatory Evaluation

The NRC staff's review in the area of plant systems covers the impact of the proposed MUR power uprate on reactor coolant and connected systems, safety-related cooling water systems, spent fuel pool (SFP) storage and cooling, radioactive waste systems, and power conversion systems. The staff's review is based on the guidance in SRP Chapters 3, 6, 9, 10, and 11, and RIS 2002-03, Attachment 1, Sections II, III, and VI. The licensee evaluated the effect of the MUR on the plant systems. This evaluation is reflected in Attachments 6 and 8 of the licensee's application dated, March 25, 2010.

3.12.2 Technical Evaluation

Reactor Coolant and Connected Systems

Reactor Coolant and connected systems include the RRS, MS line flow restrictors, MS isolation valves (MSIVs), reactor core isolation cooling (RCIC) system, residual heat removal (RHR) system, and RWCU system. The staff's review of the RWCU system is documented in Section 3.4.3 of this SE.

According to the licensee, the RRS is only minimally impacted by operation at the MUR power uprate conditions. The uprate does not require an increase in maximum core flow. The licensee also states that no significant reduction in the maximum flow capability occurs because of the less than 1 psi increase in core pressure drop. Net positive suction head margin is negligibly impacted. Cycle-specific reload analyses will consider the full range of the power and flow operating region. Based on the above considerations, the NRC staff concurs that the small increase in rated power level will have an insignificant impact on the RRS.

The licensee stated that requirements for the MS line flow restrictors remain unchanged for uprate conditions. Since the operating pressure remains the same, there is no change in steam line break flow rate. The generic evaluation provided in the NRC-approved Topical Report NEDC-32938P-A, is applicable to LGS. Based on these considerations, the NRC staff concurs that the safety and operational aspects of the MS line flow restrictors remain within the bounds of previous evaluations and are acceptable for MUR uprate conditions.

The licensee stated that the requirements for the MSIVs remain unchanged for uprate conditions. The licensee also stated that the generic evaluation provided in NEDC-32938P-A is applicable; therefore, the safety and operational aspects of the MSIVs remain within the bounds of previous evaluations. The NRC staff concurs with this evaluation.

The RCIC system provides inventory makeup to the reactor vessel when the vessel is isolated from the normal high-pressure makeup systems. The generic evaluation provided in NEDC-32938-P-A is applicable to LGS. The NRC staff concurs that the proposed MUR uprate does not affect the RCIC system operation initiation or capability requirements.

The RHR system is designed to restore and maintain the coolant inventory in the reactor vessel and to provide primary-system decay heat removal after reactor shutdown for both normal and post-accident conditions. The RHR system is designed to operate in the LPCI mode, the shutdown cooling mode, the suppression pool and the containment spray cooling mode, and fuel pool cooling assist mode. The licensee states, and the NRC agrees, that the generic discussion in NEDC-32938P-A is applicable to LGS. The slightly higher decay heat has a small effect on the operation of the RHR system in the shutdown cooling mode. The ability of the RHR system to perform required safety functions was demonstrated with analyses based on 102 percent of current licensed thermal power. Therefore, all safety aspects of the RHR system are within previous evaluations. The NRC staff, therefore, concludes that the requirements for the RHR system remain unchanged for MUR power uprate conditions.

In conclusion, for the reactor coolant and connected systems, the NRC staff reviewed the licensee's evaluation and concurs with the results. The licensee determined that there is no adverse impact on the reactor coolant and connected systems from the MUR power uprate. The staff does not anticipate that an MUR power uprate will challenge the reactor coolant and connected systems, and all systems have been shown to be operating within design limits. Therefore, the staff finds that the reactor coolant and connected systems are acceptable for the MUR uprate.

Safety-Related Cooling Water Systems

The safety-related Emergency Service Water (ESW) and Residual Heat Removal Service Water (RHRSW) systems provide cooling water to essential equipment and the RHR system, respectively. The systems provide cooling water during and following a design-basis-accident. The safety-related performance of these two systems during and following a LOCA, the most demanding design-basis event for the systems, does not change because the current LOCA analysis was based on 102 percent of the current licensed thermal power. Therefore, the ESW and RHRSW systems are acceptable for the MUR uprate.

SFP Storage and Cooling

The principal function of the SFP storage and cooling systems is to provide storage and cooling of spent fuel. The primary impact of a power uprate would be to the decay heat of the fuel recently discharged from the core. The licensee concluded that the fuel pool cooling and cleanup system, as supplemented by the RHR system, is not impacted by the MUR power uprate because the anticipated heat load remains within the existing design-basis limits. Based on the licensee's compliance with the assumptions in the existing design-basis analysis, the

NRC staff finds that the SFP storage and cooling systems are acceptable for the proposed uprate.

Radioactive Waste (Radwaste) Systems

The liquid radwaste system collects, monitors, processes, stores, and returns processed radwaste to the plant for reuse, discharge, or shipment. The activated corrosion products in the radwaste stream are expected to increase proportionally to the uprate. However, the licensee states that the total volume of processed waste is not expected to increase appreciably because the only significant increase in processed waste is due to more frequent backwashes of the condensate demineralizers and RWCU filter demineralizers. Therefore, the radiological limits of 10 CFR 20 and 10 CFR 50, Appendix I, continue to be met, and the uprate does not adversely affect the processing of liquid radwaste.

The gaseous waste systems, including the offgas system and the various building ventilation systems, collect, control, process, and dispose of gaseous radwaste. The activity of airborne effluents does not increase significantly due to the uprate, and the release limit is administratively controlled and is not a function of core power. The expected flow through the offgas system will increase slightly due to the uprate, but it remains well within the capacity of the system. Therefore, the uprate does not affect the gaseous waste systems design or operation.

The NRC staff reviewed the licensee's assessment. The staff does not expect the MUR uprate to result in a significant change to the operation of the radwaste systems; therefore, based on the licensee's assessment, the staff finds that the radwaste systems will function adequately for the proposed change.

Power Conversion Systems

Power conversion systems include the turbine generator, turbine steam bypass system, and the FW and condensate systems. These systems are not safety-related, but the operation of these systems can affect safety-related systems.

The turbine generator was evaluated for the potential to generate missiles with the potential to affect safety-related components. The licensee evaluated the existing missile analysis and found its assumptions bound the operation at the MUR power uprate conditions. Similarly, the assumptions in the turbine overspeed evaluation bound operation at the uprated condition and no changes in overspeed trip settings are required. Therefore, the MUR power uprate does not change the potential for turbine missile generation and/or overspeed.

The turbine steam bypass system was originally designed for a steam flow capacity of approximately 25 percent of the rated steam flow at the current licensed thermal power. While the bypass capacity as a percent of rated steam flow is reduced to 24.3 percent of rated steam flow at MUR power uprate conditions, the actual steam bypass capacity is unchanged. The transient analyses that credit the turbine bypass system use a bypass capacity that is less than the actual capacity. Therefore, the turbine bypass capacity remains adequate because the actual capacity remains within the value used in the analyses.

The FW and condensate systems are designed to provide FW at the temperature, pressure, quality, and flow rate required by the reactor. These systems are not safety-related; however, their performance may have an effect on plant availability and the capability to operate reliably at the MUR power uprate condition. The licensee reviewed the FW heaters, heater drains, condensate demineralizers, and the pumps (FW and condensate) and determined that the components are capable of performing in the proper design range to provide the slightly higher FW flow rate at the necessary temperature and pressure.

The staff reviewed the licensee's evaluation of the power conversion systems and concurs with the results. The licensee determined that there is no adverse impact on the power conversion systems from the MUR power uprate. The staff does not anticipate that an MUR power uprate will challenge the power conversion systems, and the systems most affected by the MUR power uprate have been shown to be operating within design limits. Therefore, the staff finds that the power conversion systems are acceptable for the MUR uprate.

3.12.3 Plant Systems Conclusion

The NRC staff has reviewed the licensee's analyses of the impact of the proposed MUR power uprate on Reactor Coolant and connected systems, safety-related cooling water systems, SFP storage and cooling, radioactive waste systems, and power conversion systems. The NRC staff has determined that the results of the licensee's analyses related to these areas will continue to meet the applicable acceptance criteria following implementation of the MUR power uprate. Therefore, the NRC staff finds the proposed MUR power uprate to be acceptable with respect to the plant systems review.

3.13 MUR Power Uprate – Containment and Heating, Ventilating and Air Conditioning (HVAC) Systems

3.13.1 Regulatory Evaluation

Section 1.2.2.1 of the LGS UFSAR states that the Unit 1 and Unit 2 designs conform to the requirements given in 10 CFR 50, Appendix A. The GDC applicable to the review of the MUR with respect to the containment are GDC-16, "Containment design," and GDC-50, "Containment design basis." The GDC applicable to the review of the MUR with respect to HVAC systems are GDC-19, "Control room," GDC-60, "Control of releases of radioactive materials to the environment," and GDC-64, "Monitoring radioactivity releases."

Regulatory guidance for the containment systems (primary and secondary) is found in the SRP, Sections 6.2.1.1.C, 6.2.1.3 and 6.2.2. Regulatory guidance for the habitability, filtration and ventilation systems is found in the SRP, Sections 6.4, 6.5.1, 9.4.1, 9.4.2, 9.4.3, 9.4.4 and 9.4.5 and in RG 1.52 and RG 1.78.

3.13.2 Technical Evaluation

Containment

Section 4.1 of Attachments 6 and 8 of the licensee's application dated March 25, 2010, discusses changes to the containment system performance due to the MUR. The table in this section states that the short term and the long term containment responses to the LOCA are bounded by the analysis performed for the current power level which includes an allowance for operation at 2 percent higher power than the current licensed power level. Additionally, the containment dynamic loads at MUR conditions are similarly bounded. Evaluations of the ability of containment isolation valves and operators to perform their required functions at MUR conditions remain bounded by the analysis for current power level. Finally, according to the licensee's application, the net positive suction head (NPSH) margin of the ECCS pumps is bounded by the values at the current power level since the existing analyses have already been performed at 102 percent of the current licensed power level.

Therefore, the licensee asserts, and the NRC staff agrees, that the relevant containment analyses have been performed at a power level that bounds operation at the uprated conditions.

HVAC Systems

Control room habitability following a postulated accident is bounded by the current evaluation of the Main Control Room Atmosphere Control System at two percent above the current power level. The licensee evaluated the Standby Gas Treatment System at two percent above the current licensed power level. The results were acceptable. This bounds operation at MUR conditions. The licensee also evaluated the MS isolation valve leakage alternate drain pathway at two percent above the current licensed power level and found to be acceptable.

Therefore, the licensee asserts, and the NRC staff agrees, that the HVAC analyses described above have been performed at a power level that bounds operation at the uprated conditions.

3.13.3 Containment and HVAC Systems Conclusion

Therefore, since the containment and HVAC analyses have been performed at a power level that bounds operation at the uprated power level, the NRC staff finds the proposed uprate acceptable with respect to the containment and HVAC systems.

3.14 SLCS Changes

3.14.1 Introduction

By application dated March 25, 2010, the licensee requested NRC approval of a change to the operation of the SLCS system pursuant to 10 CFR 50.59, "Changes, tests, and experiments," paragraph (c)(2)(ii), in that the proposed modification involves a change to the facility that may result in more than a minimal increase in the likelihood of occurrence of a malfunction of a structure, system, or component important to safety previously evaluated in the UFSAR. In addition, by application dated March 25, 2010, the licensee proposed a wording change to TS Limiting Condition for Operation (LCO) 3.1.5 regarding the SLCS system. This proposed change is a separate but related licensing action from the proposed MUR power uprate. Specifically, the proposed SLCS change involves the installation of a plant modification involving a modified hand switch that limits the auto start function of SLCS to two pumps on an ATWS signal. The third SLCS pump would be available for manual start, if required. Previously, all three SLCS pumps would auto-start on an ATWS signal. This change is necessary because a revised ATWS analysis performed for the MUR power uprate indicated that all three SLCS pumps operating together would challenge the discharge relief valve setpoints under certain conditions. This could divert flow back to the pump(s) suction, potentially impacting the boron injection capability of the system. The lower pressure in the combined discharge header with only two pumps running ensures sufficient margin to the relief valve(s) setpoint so that valves do not lift under the postulated conditions. The licensee proposes to modify TS 3.1.5 to remove the phrase "a minimum of" from the LCO, as it applies to LCO 3.1.5.a. This change is consistent with the revised ATWS analysis which assumes two pumps are aligned for automatic operation in operational conditions 1 and 2.

3.14.2 Regulatory Evaluation

The SLCS is subject to requirements in 10 CFR 50.36, "Technical Specifications," and 10 CFR 50.62, "Requirements for Reduction of Risk from Anticipated Transients Without Scram for Light-Water-Cooled Nuclear Power Plants." Because LGS was licensed using the GDC, the NRC staff also considered 10 CFR 50, Appendix A, "General Design Criteria for Nuclear Power Plants," in its review.

The requirement for a TS LCO for the SLCS is promulgated by 10 CFR 50.36(c)(2)(ii)(D). A TS LCO of a nuclear reactor must be established for the SLCS, because it is a structure, system or component which operating experience or probabilistic risk assessment has shown to be significant to public health and safety.

The SLCS is required by, among other things, 10 CFR 50.62(c)(4), which states that each BWR must have a standby liquid control system with the capability of injecting into the reactor pressure vessel a borated water solution at such a flow rate, level of boron concentration and boron-10 isotope enrichment, and accounting for reactor pressure vessel volume, that the resulting reactivity control is at least equivalent to that resulting from injection of 86 gpm of 13 weight percent sodium pentaborate decahydrate solution at the natural boron-10 isotope abundance into a 251-inch diameter reactor pressure vessel for a given core design. The SLCS and its injection location must be designed to perform its function in a reliable manner. Its initiation must be automatic.

GDC applicable to the SLCS are provided in Chapter 9, Section 3.5 of the SRP. Among others listed in SRP Section 9.3.5, the following GDC are applicable to the requested change, and were hence considered by the NRC staff[5].

(1) GDC-26, insofar as it states that two independent reactivity control systems of different design principles be provided, and that one of the systems be capable of holding the reactor in a subcritical condition, and

(2) GDC-27, insofar as it states that the reactivity control systems have a combined capability, in conjunction with poison addition by the emergency core cooling system (ECCS), to reliably control reactivity changes under postulated accident conditions.

The proposed SLCS changes were also evaluated in the area of human factors. This evaluation deals with programs, procedures, training, and plant design features related to operator performance during normal and accident conditions. The NRC's acceptance criteria for human factors are based on the following regulatory requirements and guidance:

- 10 CFR 50.62, "Requirements for reduction of risk from anticipated transients without scram (ATWS) events for light-water-cooled nuclear power plants."

- NUREG-1764, "Guidance for the Review of Changes to Human Actions."

- NUREG-0800, "Standard Review Plan," Revision 1, Section 18.0.

Finally, the NRC staff also reviewed the impact of the proposed changes to the LGS SLCS to ensure the capability of the LGS SLCS to maintain suppression pool pH at a level of 7.0 or greater following a LOCA so that the assumptions of the AST analysis performed pursuant to 10 CFR 50.67 and issued to LGS by the NRC on August 23, 2006, (ADAMS Accession No. ML062210214) are preserved.

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5. According to the LGS UFSAR, Section 3.1, LGS meets these two GDC.

3.14.3 Technical Evaluation

Section 9.3.5 of the Limerick UFSAR describes the SLCS. It is designed to be manually initiated from the control room to cause standby liquid control fluid, a sodium pentaborate solution, to be pumped into the reactor if the operator determines that the reactor cannot be shut down or kept shut down with the control rods. The system operates with positive displacement pumps; each pump is sized to inject solution into the reactor at 43 gpm.

The SLCS is also automatically initiated upon receipt of a signal from the RRCS logic. Low vessel water level, high vessel pressure, or manual initiation of the RRCS starts a timer. After a predetermined time delay, the SLCS injection is initiated. The sodium pentaborate solution is injected through the core spray line and sparger. With two pumps operating, the SLCS can begin to deliver the control liquid to the reactor pressure vessel within about 53 seconds after actuation. To further ensure the system performance, the actuation signal currently starts all three pumps and actuates the three squib valves that are normally closed to isolate the standby liquid control fluid from the reactor coolant.

The licensee stated that only two of the three SLCS pumps are required to meet the requirements of 10 CFR 50.62. Correspondingly, TS 3.1.5 currently requires that a minimum of two SLCS pumps and corresponding flow paths be operable in operational conditions 1 and 2.

The licensee also explained that the SLCS manual initiation feature is controlled by a control room operator, with each of the three SLCS loops having its own key-locked switch. The RRCS logic, as stated above, overrides the manual initiation signal and starts all three trains after a time delay.

The proposed modification would install a modified hand switch for the 'C' SLCS pump, which would allow the operators to inhibit the auto-start ATWS signal to the 'C' SLCS pump. On all three trains, the current switch has a stop position, but it spring-returns to the "norm" position, which does not defeat the auto-start logic. The modification will allow the operator to inhibit selectively the auto-start signal to the 'C' pump, keeping the 'A' and 'B' pumps aligned for automatic initiation, and leaving the 'C' pump available for manual start if required.

The licensee performed a plant-specific ATWS analysis to determine the predicted increase in reactor pressure associated with the requested power uprate. The results of that analysis indicated that with all three pumps running, and at the uprated power level, the SLCS discharge pressure could exceed the SLCS relief valve setpoints, thus causing the standby liquid control solution recirculation described in NRC Information Notice 2001-13, "Inadequate Standby Liquid Control System Relief Valve Margin."

The results of the licensee's analysis indicated that, when considering pulsations from the positive displacement standby liquid control pumps, piping losses, and the predicted vessel pressure at the time of SLCS initiation, three-pump operation would cause system pressures in excess of 1400 psi, owing mainly to line losses that are approximately 40 psi greater with three-pump operation than with two. As a result, three-pump operation would lift the SLCS relief valves, whereas two-pump operation would leave approximately 20 psi of margin to the SLCS relief valve setpoint.

The NRC staff review establishes the following:

1. The proposed TS LCO change meets the requirements of 10 CFR 50.36(c)(2)(ii)(D), because the appropriate operability requirements and system surveillances apply to all three pumps, such that any two pumps may be declared operable at a given time.

2. The SLCS meets the requirements of 10 CFR 50.62(c)(4) because it has adequate injection capability, given that only two pumps operate.

3. The SLCS provides adequate, redundant reactivity control under postulated accident conditions, consistent with the intent of GDC-26 and GDC-27.

The review is discussed in further detail in the following sub-sections.

Adequate Assurance of System Operability

The evaluation provided by the licensee demonstrates that two-pump SLCS operation provides appropriate mitigation of ATWS events, since three-pump operation would result in excessive pressure losses in the injection line and cause SLC solution recirculation. Based on this evaluation, the NRC staff finds that LCO 3.1.5.a specifying that two pumps shall be operable in operational conditions 1 and 2 is appropriate. By deleting the phrase "a minimum of" as it applies to LCO 3.1.5.a, having more than two pumps aligned for automatic operation does not meet the LCO. The phrase "a minimum of" is retained for LCO 3.1.5.b in the revised specification. It is retained so that it is clear that having one or two pumps and corresponding flowpaths operable meets the LCO requirements in operational condition 3. Consistent with the revised LCO, surveillance requirement (SR) 4.1.5.b.4 ensures that no more than two pumps are aligned for automatic operation. This applies in operational conditions 1, 2, and 3. Thus, in operational condition 3, the maximum number of pumps aligned for automatic operation is ensured by the SR and the provisions of SR 4.0.1.

Because the proposed TS revision will allow for any combination of two pumps to remain operable, the NRC staff requested information from the licensee to determine whether system surveillance, testing, inspection, and maintenance requirements would remain the same for all three pumps. In a supplemental letter dated April 26, 2010, the licensee stated that these requirements would remain the same for the 'C' SLCS pump as those for the 'A' and 'B' pumps. In light of this information, the NRC staff finds that it is acceptable to allow any two SLCS pumps to fulfill the operability requirement specified in LCO 3.1.5 because there is the same degree of assurance that any of the three pumps can perform its safety function when needed.

Adequate Injection Capability

The licensee stated that only two pumps are required to be operating in order to meet the prescriptive requirements of 10 CFR 50.62. The NRC staff confirmed this statement by using the following equivalency relationship:

$$\frac{Q}{86} \times \frac{M}{M_{251}} \times \frac{C}{13} \times \frac{E}{19.8} > 1,$$

where

Q	=	expected SLCS flow rate (gallons per minute),
M	=	mass of water in the reactor vessel and recirculation system in hot rated

		condition (pounds),
C	=	sodium pentaborate solution concentration (weight percent),
E	=	boron-10 isotopic enrichment (percent), and
M_{251}	=	mass of water in a 251-inch reactor vessel (pounds).

The NRC staff requested that the licensee provide the inside diameter of the LGS reactor vessels, and the licensee responded by letter dated April 26, 2010, stating that the inner diameter of both reactor vessels is 251 inches. This causes the mass term to drop out of the equivalency expression given above.

The NRC staff obtained the minimum allowable SLCS operating parameters to examine this relationship from the LGS TS. The minimum flow requirement for each SLCS pump is 41.2 gallons per minute. With the required two pumps operable, this will total 82.4 gallons per minute. The isotopic boron-10 enrichment must be at least 29 percent. At a nominal 65°F, the minimum boron concentration is approximately 13 percent[6].

Using these parameters in the expression above, the NRC staff confirmed that the prescriptive requirements of 10 CFR 50.62 are met by a factor of 1.4. Because the licensee will meet the SLCS injection requirements specified in 10 CFR 50.62, the NRC staff finds two-pump SLCS operation acceptable for Limerick.

Redundant Reactivity Control Under Postulated Accident Conditions

The SLCS is provided for mitigation of transient sequences that fail to include a reactor trip when needed. As such, SLCS performance is demonstrated in part by analysis of limiting ATWS scenarios. The limiting scenario analyzed to support this LAR postulated automatic initiation of the standby liquid control system. However, the system is designed, and EOPs have provisions for, manual system initiation. The UFSAR describes the current initiation logic, stating that an automatic initiation signal will override a manual initiation signal, but that a manual shutoff signal will override the automatic initiation signal. Based on this logic description, the NRC staff requested additional information from the licensee to establish whether various manual initiation sequences could occur sooner than the expiry of the RRCS timer, which is the basis for the boron injection timing assumed in the ATWS analysis.

The licensee's April 26, 2010, supplement stated that operating procedures will allow for manual initiation of only two SLCS pumps. Based on this statement, it is clear that operating procedures preclude operator initiation of either a single or all three pumps.

The NRC staff also reviewed the ATWS analysis to determine whether two-pump manual initiation earlier than assumed in the analysis would have any significant, detrimental impact on the analytic results. Based on the facts that: (1) the reactor vessel pressure is predicted to peak and subside to a level that is stable on the safety/relief valve setpoints after a period of about 30 seconds; and (2) the two-pump injection capacity is approximately 86 gpm, it is expected that additional SLC solution added to the vessel would not exceed 200 gallons, should the operator initiate the SLCS earlier than assumed in the ATWS analysis. It is the NRC staff's judgment that this is a trivial amount compared to either the total inventory contained in the vessel, or the amount of steam released through the safety/relief valves during this postulated transient. Therefore, the NRC staff finds that operator initiation of the SLCS earlier than the 2-minute

6. The minimum boron concentration allowed by the LGS TS is a function of the SLC solution temperature. The LGS, Units 1 and 2, TS also specify a minimum boron concentration that follows a similar, but more conservative, expression to that given above.

automatic SLCS initiation assumed in the ATWS analysis would have minimal impact on the analysis. Analyses indicate, therefore, that either manual or automatic initiation from the proposed, two-pump configuration will enable the SLCS to perform its reactivity control function under postulated accident conditions.

In the area of human factors, by letter dated July 28, 2010 (ADAMS Accession No. ML102110047), the NRC staff requested information in the following areas:

- Operator task requirements
- Alarms, displays and controls needed for the operator to operate the SLCS system with the proposed switch alignment
- Operator training
- Normal, abnormal, and emergency operating procedure changes needed to support this change.

Each of these is discussed below:

Operator Task Requirements

By letter dated July 28, 2010, Exelon described the operator task requirements for the proposed change:

> Operator tasks are minimally affected by the modification. For anticipated transient without scram (ATWS) events, the SLCS is an automatic system but can also be initiated manually. Operator actions in response to an ATWS are unaffected. If a pump fails to start, the current operator response as directed by procedures is to manually start SLCS pumps. The action to manually start SLCS pumps is not changed, with the exception that the operator may need to obtain a key for the C SLCS pump control switch if the A and B pumps fail to manually start. The key for this switch will be available locally in the control room. The task of manipulating a key lock switch is familiar to operators. The removal of the key from the switch for the C SLCS pump is intended to reduce the potential for the operator to manually start three SLCS pumps in an ATWS situation, since the operator would need to obtain and insert the key before manually actuating the C SLCS pump.

The staff has determined that the licensee has adequately analyzed and described the changes to operator task requirements resulting from the proposed LAR. Based upon the information provided, the staff is satisfied that, upon appropriate implementation of the modification, the proposed operator actions are within the operators' capability to complete SLCS task requirements.

Alarms, Displays, and Controls

By letter dated July 28, 2010, Exelon provided the following information regarding Alarms, Displays and Controls:

> The proposed change modifies the C SLCS pump control switch such that it can be aligned to maintain the "stop" position to inhibit automatic start of the C SLCS

pump in response to an ATWS signal.

Additionally, the modification adds a main control room (MCR) alarm and associated alarm switch. The new MCR alarm is labeled "C SLCS Pump Auto-Start Status Trouble." The new alarm switch is a two-position hand switch. The alarm will actuate when the 'C' SLCS pump control switch is misaligned in the following two conditions.

- With the alarm switch in the "inhibit" position, the annunciator alarms when the C SLCS pump control switch is aligned to the "norm" position, which would allow C SLCS pump auto-start. Procedures will direct that the alarm switch be placed in the "inhibit" position when the A and B SLCS pumps are aligned for automatic operation. This will alert the operators if the C SLCS should inadvertently become aligned for automatic operation with the A and B pumps already so aligned.

- With the alarm switch in the "enable" position, the annunciator alarms when the C SLCS pump control switch is aligned to the "stop" position, which would inhibit C SLCS pump auto-start. Procedures will direct that the alarm switch be placed in the "enable" position when either the A or B SLCS pump is not aligned for automatic operation, for example, during maintenance of the A or B pump. This will alert the operators if the auto-start capability of the C SLCS pump is unavailable when needed.

The alarm switch affects only the functioning of the alarm and does not affect the operation of the 'C' SLCS pump.

The staff has determined that the licensee has adequately analyzed and described the changes to control room alarms and controls resulting from the proposed LAR. The staff is satisfied that, upon appropriate implementation of the modification, the proposed control room changes will support reliable operator action to perform SLCS task requirements.

Operator Training and Procedure Changes

As noted in the application dated March 25, 2010, and/or in the RAI response dated July 25, 2010, operator training will be completed prior to implementation of the proposed changes. Also, changes to plant procedures and the plant simulator will be made in accordance with plant configuration control processes. Procedure changes include normal system operating procedures, which will be revised to ensure that only two pumps are aligned for automatic operation, and to direct positioning of the modified key lock switch for the C SLCS pump. Additionally, the normal system operating procedures will be revised to direct positioning of the alarm switch to ensure that 'C' SLCS Pump Auto-Start Status Trouble alarm functions as described above. An annunciator response procedure has been developed to direct operator response to the new alarm. No changes to the AOPs or EOPs are necessary regarding the SLCS modification.

The staff has determined that the licensee has adequately analyzed and described the changes to operator training requirements resulting from the proposed LAR. Operator training updates will be performed in accordance with the plant configuration control process that is used for other plant changes and hence is familiar to the operators. The staff is satisfied that upon appropriate implementation of the modification, the proposed operator training and procedure changes will be sufficient to allow operators to complete SLCS task requirements.

Finally, the NRC staff reviewed the regulatory and technical analyses performed by the licensee in support of its proposed modification to the SLCS, as they relate to the radiological consequences of DBA analyses. In accordance with the current radiological consequence analyses of record, issued to LGS by the NRC on August 23, 2006, the suppression pool pH at LGS must be controlled at values of 7.0 or greater. LGS accomplishes this by injecting sodium pentaborate from the SLCS during a LOCA.

According to the licensee, the post-LOCA SLCS injection is manually initiated and hence, the changes proposed in this application to the automatic start logic do not impact the post-LOCA pH control function.

The NRC staff agrees that the proposed changes do not impact the ability of the operator to manually control post-LOCA suppression pool pH greater than 7 because no changes in the delivery of sodium pentaborate will occur under those postulated conditions. According to the LGS UFSAR, Section 9.3.5.1, only one SLCS pump is required to meet the design basis requirement for post-LOCA suppression pool pH control. Thus, the revised TS 3.1.5.a, does not impact the ability to perform this function. Also, the NRC staff agrees with the licensee that the revised TS and revised switch configuration will not adversely impact the ability of the operator to manually control pump operation. Therefore, LGS DBA radiological dose consequence analyses will continue to meet the dose acceptance criteria given in 10 CFR 50.67. The staff concludes that the proposed SLCS change is acceptable with respect to the radiological dose consequences of the DBAs.

3.14.4 SLCS Change Conclusion

Based on the systems analysis, human factors, and dose consequence considerations discussed above, the NRC staff finds the proposed modification to the SLCS acceptable.

4.0 STATE CONSULTATION

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

5.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes a surveillance requirement. The NRC staff has determined that the amendment involves 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 amendment involves no significant hazards consideration, and there has been no public comment on such finding (75 FR 32512). Accordingly, the amendment meets 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 amendment.

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 amendment will not be inimical to the common defense and security or to the health and safety of the public.

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Date: April 8, 2011

concludes that all materials from the LGS, Units 1 and 2, Reactor Pressure Vessels which require evaluation in accordance with 10 CFR Part 50 Appendix G for upper-shelf energy, have been demonstrated to meet the requirements of 10 CFR Part 50, Appendix G, either by having their upper shelf energy values comply with criteria provided in the regulation, or by having an acceptable equivalent margins analysis (EMA) performed for them, as permitted by the regulation. Specifically, the NRC staff approves the EMAs submitted for the low-pressure coolant injection nozzles and their associated welds, at the uprated (3515 MWt) power level, up to 32 effective full-power years, for LGS, Units 1 and 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,
/ra/
Peter Bamford, Project Manager
Plant Licensing Branch I-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-352 and 50-353
Enclosures:
1. Amendment No. 201 to License No. NPF-39
2. Amendment No. 163 to License No. NPF-85
3. Safety Evaluation

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Amendment: ML110691095 * Concurrence via memo ** concurrence via email

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NAME	PBamford	ABaxter **	WKemper*	RElliott	MMitchell*
DATE	03/ 10/2011	03 /17/2011	08/19/2010	03/23 /2011	3/02/2011

DE/EEEB	DCI/CSGB	DE/EMCB	DCI/CPTB	DSS/SBPB	DRA/AFPB
RMathew*	RTaylor*	MKhanna*	AMcMurtray*	GCasto, GPurciarello for	AKlein*
10/04/2010	07/07/2010	08/04/2010	07/21/2010	03 / 28 /2011	08/06/2010

DSS/SCVB	DRA/AADB	DSS/SRXB	DIRS/IHPB	OGC
RDennig*	TTate*	AUlses*	UShoop*	LSubin, NLO w/comments
09/02/2010	10/18/2010	08/09/2010	08/12/2010	03 /29/2011

LPLI-2/BC	DORL/DD	DORL/D
HChernoff	RNelson	JGiitter
04 / 06 /2011	04 / 07 /2011	04 / 08 /2011

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