Mr. Oliver D. Kingsley, President
Nuclear Generation Group
Commonwealth Edison Company
Executive Towers West III
1400 Opus Place, Suite 500
Downers Grove, IL 60515

May 9, 2000

SUBJECT: LASALLE - ISSUANCE OF AMENDMENTS REGARDING POWER UPRATE (TAC NOS. MA6070 AND MA6071)

Dear Mr. Kingsley:

The U.S. Nuclear Regulatory Commission (Commission) has issued the enclosed Amendment No. 140 to Facility Operating License No. NPF-11 and Amendment No. 125 to Facility Operating License No. NPF-18 for the LaSalle County Station, Units 1 and 2, respectively. The amendments are in response to your application dated July 14, 1999, as supplemented by letters dated January 21, February 15, February 23, March 10, March 24, two letters on March 31, April 7 and April 14, 2000.

The amendments increase the maximum reactor core power level to 3489 megawatts thermal; an increase of 5 percent of rated core thermal power, for LaSalle County Station, Units 1 and 2. In addition, the proposed amendments correct a non-conservative value in the upper limit for drywell and suppression chamber internal pressure that was discovered during the course of the power uprate review.

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/

Donna M. Skay, Project Manager, Section 2
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-373, 50-374

Enclosures:
1. Amendment No. 140 to NPF-11
2. Amendment No. 125 to NPF-18
3. Safety Evaluation

cc w/encls: See next page

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*Input provided by memo dated April 5, 2000, incorporated with no significant changes
** Input provided by memo dated March 14, 2000, incorporated with no significant changes
***Input provided by memo dated March 14, 2000, incorporated with no significant changes
****Input provided by memo dated March 2, 2000, incorporated with no significant changes
Input provided by memo dated April 14, 2000, incorporated with no significant changes
Input provided by memo dated April 18, 2000, incorporated with no significant changes
Input provided by memo dated April 17, 2000, incorporated with no significant changes
Input provided by memo dated April 25, 2000, incorporated with no significant changes

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

May 9, 2000

Mr. Oliver D. Kingsley, President
Nuclear Generation Group
Commonwealth Edison Company
Executive Towers West III
1400 Opus Place, Suite 500
Downers Grove, IL 60515

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Sincerely,

Donna M. Skay

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Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-373, 50-374

Enclosures:
1. Amendment No. 140 to NPF-11
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3. Safety Evaluation

cc w/encls: See next page

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

COMMONWEALTH EDISON COMPANY

DOCKET NO. 50-373

LASALLE COUNTY STATION, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 140
License No. NPF-11

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

   A. The application for amendment filed by the Commonwealth Edison Company (the licensee), dated July 14, 1999, as supplemented on January 21, February 15, February 23, March 10, March 24, two letters on March 31, April 7 and April 14, 2000, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations set forth in 10 CFR Chapter I;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Operating License and the Technical Specifications as indicated in the attachment to this license amendment and paragraph 2.C.(2) of Facility Operating License No. NPF-11 is hereby amended to read as follows:

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A, as revised through Amendment No. 140, and the Environmental Protection Plan contained in Appendix B, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of the date of its issuance and shall be implemented within 60 days.

FOR THE NUCLEAR REGULATORY COMMISSION

Anthony J. Mendiola, Chief, Section 2
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Operating License
and Technical Specifications

Date of Issuance: May 9, 2000

ATTACHMENT TO LICENSE AMENDMENT NO. 140

FACILITY OPERATING LICENSE NO. NPF-11

DOCKET NO. 50-373

Replace the following page of Operating License NPF-11 with the enclosed page. The revised page is identified by amendment number and contains a vertical line indicating the area of change.

REMOVE	INSERT
3	3

Replace the following pages of the Appendix "A" Technical Specifications with the enclosed pages. The revised pages are identified by amendment number and contain a vertical line indicating the area of change

REMOVE	INSERT
1-6	1-6
2-4	2-4
3/4 3-15	3/4 3-15
3/4 3-53	3/4 3-53
3/4 6-13	3/4 6-13
3/4 6-20a	3/4 6-20a
B 3/4 4-6	B 3/4 4-6
B 3/4 6-2	B 3/4 6-2
B 3/4 6-3	B 3/4 6-3

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

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

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

(1) Maximum Power Level

The licensee is authorized to operate the facility at reactor core power levels not in excess of full power (3489 megawatts thermal).

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A, as revised through Amendment No. 140 , and the Environmental Protection Plan contained in Appendix B, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

(3) Conduct of Work Activities During Fuel Load and Initial Startup

The licensee shall review by committee all Unit 1 Preoperational Testing and System Demonstration activities performed concurrently with Unit 1 initial fuel loading or with the Unit 1 Startup Test Program to assure that the activity will not affect the safe performance of the Unit 1 fuel loading or the portion of the Unit 1 Startup Program being performed. The review shall address, as a minimum, system interaction, span of control, staffing, security and health physics, with respect to performance of the activity concurrently with the Unit 1 fuel loading or the portion of the Unit 1 Startup Program being performed. The committee for the review shall be composed of at least three members, knowledgeable in the above areas, and who meet the qualifications for professional-technical personnel specified by

e. The suppression chamber is OPERABLE pursuant to Specification 3.6.2.1.

f. The sealing mechanism associated with each primary containment penetration; e.g., welds, bellows or 0-rings, is OPERABLE.

g. Primary containment structural integrity has been verified in accordance with Surveillance Requirement 4.6.1.1.e.

PROCESS CONTROL PROGRAM

1.33 The PROCESS CONTROL PROGRAM (PCP) shall contain the current formulas, sampling, analyses, test, and determinations to be made to ensure that processing and packaging of solid radioactive wastes based on demonstrated processing of actual or simulated wet solid wastes will be accomplished in such a way as to assure compliance with 10 CFR 20, 61, and 71, State regulations, burial ground requirements, and other requirements governing the disposal of solid radioactive waste.

PURGE - PURGING

1.34 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

1.35 RATED THERMAL POWER shall be a total reactor core heat transfer rate to the reactor coolant of 3489 MWT.

REACTOR PROTECTION SYSTEM RESPONSE TIME

1.36 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.

REPORTABLE EVENT

1.37 A REPORTABLE EVENT shall be any of those conditions specified in Section 50.73 to 10 CFR Part 50.

ROD DENSITY

1.38 ROD DENSITY shall be the number of control rod notches inserted as a fraction of the total number of control rod notches. All rods fully inserted is equivalent to 100% ROD DENSITY.

TABLE 2.2.1-1
REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
1. Intermediate Range Monitor, Neutron Flux-High	≤ 120 divisions of full scale	≤ 122 divisions of full scale
2. Average Power Range Monitor:		
a. Neutron Flux-High, Setdown	≤ 15% of RATED THERMAL POWER	≤ 20% of RATED THERMAL POWER
b. Flow Biased Simulated Thermal Power - Upscale		
1) Two Recirculation Loop Operation		
a) Flow Biased	≤ 0.62W + 63.7% with a maximum of	≤ 0.62W + 69.3% with a maximum of
b) High Flow Clamped	≤ 113.5% of RATED THERMAL POWER	≤ 115.5% of RATED THERMAL POWER
2) Single Recirculation Loop Operation		
a) Flow Biased	≤ 0.55W + 51.5% with a maximum of	≤ 0.55W + 56.8% with a maximum of
b) High Flow Clamped	≤ 108.1% of RATED THERMAL POWER	≤ 112.3% of RATED THERMAL POWER
c. Fixed Neutron Flux-High	≤ 118% of RATED THERMAL POWER	≤ 120% of RATED THERMAL POWER
3. Reactor Vessel Steam Dome Pressure - High	≤ 1043 psig	≤ 1063 psig
4. Reactor Vessel Water Level - Low, Level 3	≥ 12.5 inches above instrument zero*	≥ 11.0 inches above instrument zero*

*See Bases Figure B 3/4 3-1.

TABLE 3.3.2-2

ISOLATION ACTUATION INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
A. AUTOMATIC INITIATION		
1. PRIMARY CONTAINMENT ISOLATION		
a. Reactor Vessel Water Level		
1) Low, Level 3	≥ 12.5 inches*	≥ 11.0 inches*
2) Low Low, Level 2	≥ -50 inches*	≥ -57 inches*
3) Low Low Low, Level 1	≥ -129 inches*	≥ -136 inches*
b. Drywell Pressure - High	≤ 1.69 psig	≤ 1.89 psig
c. Main Steam Line		
1) DELETED		
2) Pressure - Low	≥ 854 psig	≥ 834 psig
3) Flow - High	≤ 125 psid	≤ 128 psid
d. DELETED		
e. Main Steam Line Tunnel Δ Temperature - High	≤ 65°F	≤ 70°F
f. Condenser Vacuum - Low	> 7 inches Hg vacuum	> 5.5 inches Hg vacuum
2. SECONDARY CONTAINMENT ISOLATION		
a. Reactor Building Vent Exhaust Plenum Radiation - High	≤ 10 mr/hr	≤ 15 mr/hr
b. Drywell Pressure - High	≤ 1.69 psig	≤ 1.89 psig
c. Reactor Vessel Water Level - Low Low, Level 2	≥ -50 inches*	≥ -57 inches*
d. Fuel Pool Vent Exhaust Radiation - High	≤ 10 mr/hr	≤ 15 mr/hr
3. REACTOR WATER CLEANUP SYSTEM ISOLATION		
a. Δ Flow - High	≤ 70 gpm	≤ 87.5 gpm
b. Heat Exchanger Area Temperature - High	≤ 149°F	≤ 156.8°F
c. Heat Exchanger Area Ventilation ΔT - High	≤ 33°F	≤ 40.3°F
d. SLCS Initiation	NA	NA
e. Reactor Vessel Water Level - Low Low, Level 2	≥ -50 inches*	≥ -57 inches*

TABLE 3.3.6-2

CONTROL ROD WITHDRAWAL BLOCK INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
1. ROD BLOCK MONITOR		
a. Upscale The Rod Block Monitor Upscale Setpoints shall be established according to the relationships specified in the CORE OPERATING LIMITS REPORT.		
b. Inoperative	N.A.	N.A.
c. Downscale	≥ 5% of RATED THERMAL POWER	≥ 3% of RATED THERMAL POWER
2. APRM		
a. Flow Biased Simulated Thermal Power-Upscale		
1) Two Recirculation Loop Operation	≤ 0.62 W + 52.3%*	≤ 0.62 W + 57.9%*
2) Single Recirculation Loop Operation	≤ 0.55 W + 40.0%	≤ 0.55 W + 45.4%*
b. Inoperative	N.A.	N.A.
c. Downscale	≥ 5% of RATED THERMAL POWER	≥ 3% of RATED THERMAL POWER
d. Neutron Flux-High	≤ 12% of RATED THERMAL POWER	≤ 14% of RATED THERMAL POWER
3. SOURCE RANGE MONITORS		
a. Detector not full in	N.A.	N.A.
b. Upscale	≤ 2×10^5 cps	≤ 5×10^5 cps
c. Inoperative	N.A.	N.A.
d. Downscale	≥ 0.7 cps	≥ 0.5 cps
4. INTERMEDIATE RANGE MONITORS		
a. Detector not full in	N.A.	N.A.
b. Upscale	≤ 108/125 of full scale	≤ 110/125 of full scale
c. Inoperative	N.A.	N.A.
d. Downscale	≥ 5/125 of full scale	≥ 3/125 of full scale

CONTAINMENT SYSTEMS

DRYWELL AND SUPPRESSION CHAMBER INTERNAL PRESSURE

LIMITING CONDITION FOR OPERATION

3.6.1.6 Drywell and suppression chamber internal pressure shall be maintained between - 0.5 and 0.75 psig.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, AND 3

ACTION:

With the drywell and suppression chamber internal pressure outside of the specified limits, restore the internal pressure to within the limits within 1 hour or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.6 The drywell and suppression chamber internal pressure shall be determined to be within the limits at least once per 12 hours.

PLANT OPERATING PROCEDURES AND PROGRAMS (Continued)

7. Primary Containment Leakage Rate Testing Program

A program shall be established to implement the leakage rate testing of the primary containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Testing Program," dated September 1995.

The peak calculated primary containment internal pressure for the design basis loss of coolant accident, P_a, is 39.9 psig.

The maximum allowable primary containment leakage rate, L_a, at P_a, is 0.635% of primary containment air weight per day.

Leakage rate acceptance criteria are:

a. Primary containment overall leakage rate acceptance criterion is $\leq 1.0\ L_a$. During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are $\leq 0.60\ L_a$ for the combined Type B and Type C tests, and $\leq 0.75\ L_a$ for Type A tests.

b. Air lock testing acceptance criteria are:

1) Overall air lock leakage rate is $\leq 0.05\ L_a$ when tested at $\geq P_a$.

2) For each door, the seal leakage rate is $\leq$ 5 scf per hour when the gap between the door seals is pressurized to $\geq$ 10 psig.

The provisions of specification 4.0.2 do not apply to the test frequencies specified in the Primary Containment Leakage Rate Testing Program.

The provisions of specification 4.0.3 are applicable to the Primary Containment Leakage Rate Testing Program.

8. Ventilation Filter Testing Program (VFTP)

A program shall be established to implement the following required testing of Engineered Safety Feature (ESF) filter ventilation systems at the frequencies specified in Regulatory Guide 1.52, Revision 2, dated March 1978, and in accordance with ASME N510-1989.

The provisions of Specifications 4.0.2 and 4.0.3 are applicable to the VFTP test frequencies.

a. Demonstrate for each of the ESF systems that an inplace test of the high efficiency particulate air (HEPA) filters shows a penetration and system bypass <0.05% when tested in accordance with ASME N510-1989, at the system flowrate specified below:

ESF Ventilation System	Flowrate (cfm)
SBGT System	$\geq$ 3600 and $\leq$ 4400
CREF System	$\geq$ 3600 and $\leq$ 4400

BASES TABLE B 3/4.4.6-1

REACTOR VESSEL TOUGHNESS

BELTLINE

COMPONENT	MATERIAL TYPE OR WELD SEAM IDENTIFICATION	HEAT#/SLAB# OR HEAT#/LOT#	Ni(%)	CU(%)	P(%)	HIGHEST STARTING RT NDT (°F)	MAXIMUM Δ RT NDT (°F)	MIN. UPPER SHELF (ft-lb)	MAX. EOL RT NDT
Plate	SA-533,Gr.B,C1.1	C5978-2	0.59	0.11	0.010	+23	18	70	+59
Plate	SA-533,Gr.B,C1.1	C6345-2	0.51	0.15	0.012	-20	25	92	+30.5
Weld	3-308-A,B,C	IP3571/3958	0.15	0.37	0.017	-30	58	57.5	+84

NON-BELTLINE

COMPONENT	MATERIAL TYPE OR WELD SEAM IDENTIFICATION	HEAT#/SLAB# OR HEAT#/LOT#	HIGHEST STARTING RT NDT (°F)
Shell Ring	SA-533,Gr.B,C1.1	C6003-2	+12
Bottom Head Dollar Plate	SA-533,Gr.B,C1.1	C6003-3	+58
Bottom Head Radial Plates	SA-533,Gr.B,C1.1	C5328-1	+10
Top Head Dollar Plate	SA-533,Gr.B,C1.1	C7343-1	-10
Top Head Side Plates	SA-533,Gr.B,C1.1	C7376-2	-10
Top Head Flange	SA-508,C1.2	ACT-USS-4P	+20
Vessel Flange	SA-508,C1.2	2V-659ATF-112	+20
Feedwater Nozzle	SA-508,C1.2	#174W-3,Q2Q14VW	+40
Weld	15-308	NA/KOIB	0
Closure Stud	POH-16C,Gr.B and ATSM-A-540	14716	+70 (Lowest Service Temperature)

3/4.6.1 PRIMARY CONTAINMENT

3/4.6.1.4 DELETED

3/4.6.1.5 DELETED

3/4.6.1.6 DRYWELL AND SUPPRESSION CHAMBER INTERNAL PRESSURE

The limitation on drywell and suppression chamber internal pressure ensure that the containment peak pressure of 39.9 psig does not exceed the design pressure of 45 psig during LOCA conditions or that the external pressure differential does not exceed the design maximum external pressure differential of 5 psid. The limit of 0.75 psig for initial positive primary containment pressure will limit the total pressure to 39.9 psig which is less than the design pressure and is consistent with the accident analysis.

3/4.6.1.7 DRYWELL AVERAGE AIR TEMPERATURE

The limitation on drywell average air temperature ensures that the containment peak air temperature does not exceed the design temperature of 340°F during LOCA conditions and is consistent with the accident analysis.

3/4.6.1.8 DRYWELL AND SUPPRESSION CHAMBER PURGE SYSTEM

The drywell and suppression chamber purge supply and exhaust isolation valves are required to be closed during plant operation except as required for inerting, de-inerting and pressure control. During operations involving inerting, de-inerting and pressure control, only the drywell or suppression chamber purge supply and exhaust isolation valves may be open to prevent the creation of a bypass path between the drywell and suppression chamber. Creation of a bypass path between the drywell and the suppression chamber air space through the vent and purge lines would allow steam and gases from a LOCA to bypass the downcomers to the suppression pool in excess of design bypass leakage. These valves have been demonstrated capable of closing during a LOCA or steamline break accident from the full open position.

3/4.6.2. DEPRESSURIZATION SYSTEMS

The specifications of this section ensure that the primary containment pressure will not exceed the design pressure of 45 psig during primary system blowdown from full operating pressure.

The suppression chamber water provides the heat sink for the reactor coolant system energy release following a postulated rupture of the system. The suppression chamber water volume must absorb the associated decay and structural sensible heat released during the reactor coolant system blowdown from 1020 psig. Since all of the gases in the drywell are purged into the suppression chamber air space during a loss of coolant accident, the pressure of the liquid must not exceed 45 psig, the suppression chamber maximum pressure. The design volume of the suppression chamber, water and air, was obtained by considering that the total volume of reactor coolant and to be considered is discharged to the suppression chamber and that the drywell volume is purged to the suppression chamber. (See Figure B 3/4.6.2-1)

Using the minimum or maximum water volumes given in this specification, containment pressure during the design basis accident is approximately 39.6 psig which is below the design pressure of 45 psig. Maximum water volume of 131,900 ft^3 results in a downcomer submergence of 12.4 ft and the minimum volume of 128,800 ft^3 results in a submergence approximately 8 inches less. The majority of the Bogeda tests were run with a submerged length of four feet and with complete condensation. Thus, with respect to the downcomer submergence, this specification is adequate.

Should it be necessary to make the suppression chamber inoperable, this shall only be done as specified in Specification 3.5.3.

Under full power operating conditions, blowdown from an initial suppression chamber water temperature of 90°F results in a water temperature of approximately 135°F immediately following blowdown which is below the 200°F used for complete condensation via T-quencher devices. At this temperature and atmospheric pressure, the available NPSH exceeds that required by both the RHR and core spray pumps, thus there is no dependency on containment overpressure during the accident injection phase.

Experimental data indicates that excessive steam condensing loads can be avoided if the suppression pool peak bulk temperature can remain below saturation conditions. However, an additional concern raised related to the potential transfer of non-condensed SRV steam to the ECCS suction strainer, if local saturated conditions existed at the quencher and the ECCS suction is at a higher elevation than the SRV quencher. The LaSalle ECCS suction strainers are located above the elevation of the T-Quenchers. Further studies have shown that long steam plumes occur when subcooling levels are less than 9°F. However, the LaSalle T-Quenchers is at a submersion of 24 feet and provides 20°F subcooling with bulk temperature of 208°F with the wetwell at atmospheric pressure. This provides sufficient margin to ensure that exiting steam is condensed before posing a steam ingestion potential to the ECCS suction. Therefore, the peak bulk suppression pool limit for LaSalle will be 208°F.

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

COMMONWEALTH EDISON COMPANY

DOCKET NO. 50-374

LASALLE COUNTY STATION, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 125
License No. NPF-18

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

   A. The application for amendment filed by the Commonwealth Edison Company (the licensee), dated July 14, 1999, as supplemented on January 21, February 15, February 23, March 10, March 24, two letters on March 31, April 7 and April 14, 2000, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations set forth in 10 CFR Chapter I;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Operating License and the Technical Specifications as indicated in the attachment to this license amendment and paragraph 2.C.(2) of Facility Operating License No. NPF-18 is hereby amended to read as follows:

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A, as revised through Amendment No. 125 , and the Environmental Protection Plan contained in Appendix B, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of the date of its issuance and shall be implemented prior to start up of cycle 9.

FOR THE NUCLEAR REGULATORY COMMISSION

Anthony J. Mendiola, Chief, Section 2
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Operating License
and Technical Specifications

Date of Issuance: May 9, 2000

ATTACHMENT TO LICENSE AMENDMENT NO. 125

FACILITY OPERATING LICENSE NO. NPF-18

DOCKET NO. 50-374

Replace the following page of Operating License NPF-18 with the enclosed page. The revised page is identified by amendment number and contains a vertical line indicating the area of change.

REMOVE	INSERT
3	3

Replace the following pages of the Appendix "A" Technical Specifications with the enclosed pages. The revised pages are identified by amendment number and contain a vertical line indicating the area of change

REMOVE	INSERT
1-5a	1-5a
2-4	2-4
3/4 3-15	3/4 3-15
3/4 3-53	3/4 3-53
3/4 6-16	3/4 6-16
6-20a	6-20a
B 3/4 4-6	B 3/4 4-6
B 3/4 6-2a	B 3/4 6-2a
B 3/4 6-3	B 3/4 6-3

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

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

(1) Maximum Power Level

The licensee is authorized to operate the facility at reactor core power levels not in excess of full power (3489 megawatts thermal). Items in Attachment 1 shall be completed as specified. Attachment 1 is hereby incorporated into this license.

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A, as revised through Amendment No. 125 , and the Environmental Protection Plan contained in Appendix B, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

(3) Conduct of Work Activities During Fuel Load and Initial Startup

The licensee shall review by committee all Unit 2 Preoperational Testing and System Demonstration activities performed concurrently with Unit 2 initial fuel loading or with the Unit 2 Startup Test Program to assure that the activity will not affect the safe performance of the Unit 2 fuel loading or the portion of the Unit 2 Startup Program being performed. The review shall address, as a minimum, system interaction, span of control, staffing, security and health physics, with respect to performance of the activity concurrently with the Unit 2 fuel loading or the portion of the Unit 2 Startup Program being performed. The committee for the review shall be composed of at least three members, knowledgeable in the above areas, and who meet the qualifications for professional-technical personnel specified by section 4.4 of ANSI N18.7-1971. At least one of these three shall be a senior member of the Assistant Superintendent of Operation's staff.

PRIMARY CONTAINMENT INTEGRITY (Continued)

f. The sealing mechanism associated with each primary containment penetration; e.g., welds, bellows or 0-rings, is OPERABLE.

g. Primary containment structural integrity has been verified in accordance with Surveillance Requirement 4.6.1.1.e.

PROCESS CONTROL PROGRAM

1.33 The PROCESS CONTROL PROGRAM (PCP) shall contain the current formulas, sampling, analyses, test, and determinations to be made to ensure that processing and packaging of solid radioactive wastes based on demonstrated processing of actual or simulated wet solid wastes will be accomplished in such a way as to assure compliance with 10 CFR 20, 61, and 71, State regulations, burial ground requirements, and other requirements governing the disposal of solid radioactive waste.

PURGE - PURGING

1.34 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

1.35 RATED THERMAL POWER shall be a total reactor core heat transfer rate to the reactor coolant of 3489 MWT.

REACTOR PROTECTION SYSTEM RESPONSE TIME

1.36 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.

REPORTABLE EVENT

1.37 A REPORTABLE EVENT shall be any of those conditions specified in Section 50.73 to 10 CFR Part 50.

ROD DENSITY

1.38 ROD DENSITY shall be the number of control rod notches inserted as a fraction of the total number of control rod notches. All rods fully inserted is equivalent to 100% ROD DENSITY.

TABLE 2.2.1-1
REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
1. Intermediate Range Monitor, Neutron Flux-High	≤ 120 divisions of full scale	≤ 122 divisions of full scale
2. Average Power Range Monitor:		
a. Neutron Flux-High, Setdown	≤ 15% of RATED THERMAL POWER	≤ 20% of RATED THERMAL POWER
b. Flow Biased Simulated Thermal Power - Upscale		
1) Two Recirculation Loop Operation		
a) Flow Biased	≤ 0.62W + 63.7% with a maximum of	≤ 0.62W + 69.3% with a maximum of
b) High Flow Clamped	≤ 113.5% of RATED THERMAL POWER	≤ 115.5% of RATED THERMAL POWER
2) Single Recirculation Loop Operation		
a) Flow Biased	≤ 0.55W + 51.5% with a maximum of	≤ 0.55W + 56.8% with a maximum of
b) High Flow Clamped	≤ 108.1% of RATED THERMAL POWER	≤ 112.3% of RATED THERMAL POWER
c. Fixed Neutron Flux-High	≤ 118% of RATED THERMAL POWER	≤ 120% of RATED THERMAL POWER
3. Reactor Vessel Steam Dome Pressure - High	≤ 1043 psig	≤ 1063 psig
4. Reactor Vessel Water Level - Low, Level 3	≥ 12.5 inches above instrument zero*	≥ 11 inches above instrument zero*
5. Main Steam Line Isolation Valve - Closure	≤ 8% closed	≤ 12% closed
6. DELETED		
7. Primary Containment Pressure - High	≤ 1.69 psig	≤ 1.89 psig
8. Scram Discharge Volume Water Level - High	≤ 767' 5¼"	≤ 767' 5¼"
9. Turbine Stop Valve - Closure	≤ 5% closed	≤ 7% closed

*See Bases Figure B 3/4 3-1.

TABLE 3.3.2-2
ISOLATION ACTUATION INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
A. AUTOMATIC INITIATION		
1. PRIMARY CONTAINMENT ISOLATION		
a. Reactor Vessel Water Level		
1) Low, Level 3	≥ 12.5 inches*	≥ 11.0 inches*
2) Low Low, Level 2	≥ -50 inches*	≥ -57 inches*
3) Low Low Low, Level 1	≥ -129 inches*	≥ -136 inches*
b. Drywell Pressure - High	≤ 1.69 psig	≤ 1.89 psig
c. Main Steam Line		
1) DELETED		
2) Pressure - Low	≥ 854 psig	≥ 834 psig
3) Flow - High	≤ 125 psid	≤ 128 psid
d. DELETED		
e. Main Steam Line Tunnel Δ Temperature - High	≤ 65°F	≤ 70°F
f. Condenser Vacuum - Low	> 7 inches Hg vacuum	> 5.5 inches Hg vacuum
2. SECONDARY CONTAINMENT ISOLATION		
a. Reactor Building Vent Exhaust Plenum Radiation - High	≤ 10 mr/h	≤ 15 mr/h
b. Drywell Pressure - High	≤ 1.69 psig	≤ 1.89 psig
c. Reactor Vessel Water Level - Low Low, Level 2	≥ -50 inches*	≥ -57 inches*
d. Fuel Pool Vent Exhaust Radiation - High	≤ 10 mr/h	≤ 15 mr/h
3. REACTOR WATER CLEANUP SYSTEM ISOLATION		
a. ΔFlow - High	≤ 70 gpm	≤ 87.5 gpm
b. Heat Exchanger Area Temperature - High	≤ 149°F	≤ 156.8°F
c. Heat Exchanger Area Ventilation ΔT - High	≤ 33°F	≤ 40.3°F
d. SLCS Initiation	N.A.	N.A.
e. Reactor Vessel Water Level - Low Low, Level 2	≥ -50 inches*	≥ -57 inches*

TABLE 3.3.6-2
CONTROL ROD WITHDRAWAL BLOCK INSTRUMENTATION SETPOINTS

TRIP FUNCTION			TRIP SETPOINT	ALLOWABLE VALUE
1.	ROD BLOCK MONITOR			
	a.	Upscale The Rod Block Monitor Upscale Setpoints shall be established according to the relationships specified in the CORE OPERATING LIMITS REPORT.		
	b.	Inoperative	N.A.	N.A.
	c.	Downscale	≥ 5% of RATED THERMAL POWER	≥ 3% of RATED THERMAL POWER
2.	APRM			
	a.	Flow Biased Simulated Thermal Power-Upscale		
		1) Two Recirculation Loop Operation	≤ 0.62 W + 52.3%*	≤ 0.62 W + 57.9%*
		2) Single Recirculation Loop Operation	≤ 0.55 W + 40.0%	≤ 0.55 W + 45.4%*
	b.	Inoperative	N.A.	N.A.
	c.	Downscale	≥ 5% of RATED THERMAL POWER	≥ 3% of RATED THERMAL POWER
	d.	Neutron Flux-High	≤ 12% of RATED THERMAL POWER	≤ 14% of RATED THERMAL POWER
3.	SOURCE RANGE MONITORS			
	a.	Detector not full in	N.A.	N.A.
	b.	Upscale	≤ 2 x 10^5 cps	≤ 5 x 10^5 cps
	c.	Inoperative	N.A.	N.A.
	d.	Downscale	≥ 0.7 cps	≥ 0.5 cps
4.	INTERMEDIATE RANGE MONITORS			
	a.	Detector not full in	N.A.	N.A.
	b.	Upscale	≤ 108/125 of full scale	≤ 110/125 of full scale
	c.	Inoperative	N.A.	N.A.
	d.	Downscale	≥ 5/125 of full scale	≥ 3/125 of full scale

CONTAINMENT SYSTEMS

DRYWELL AND SUPPRESSION CHAMBER INTERNAL PRESSURE

LIMITING CONDITION FOR OPERATION

3.6.1.6 Drywell and suppression chamber internal pressure shall be maintained between - 0.5 and 0.75 psig.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, AND 3

ACTION:

With the drywell and suppression chamber internal pressure outside of the specified limits, restore the internal pressure to within the limits within 1 hour or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.6 The drywell and suppression chamber internal pressure shall be determined to be within the limits at least once per 12 hours.

PLANT OPERATING PROCEDURES AND PROGRAMS (Continued)

the Initial Structural Integrity Tests were not within 2 years of each other.

The Onsite Review and Investigative Function shall be responsible for reviewing and approving changes to the Inservice Inspection Program for Post Tensioning Tendons.

The provisions of 4.0.2 and 4.0.3 are applicable to the Tendon Surveillance Program inspection frequencies.

7. Primary Containment Leakage Rate Testing Program

A program shall be established to implement the leakage rate testing of the primary containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Testing Program," dated September 1995.

The peak calculated primary containment internal pressure for the design basis loss of coolant accident, P_a, is 39.9 psig.

The maximum allowable primary containment leakage rate, L_a, at P_a, is 0.635% of primary containment air weight per day.

Leakage rate acceptance criteria are:

a. Primary containment overall leakage rate acceptance criterion is $\leq 1.0\ L_a$. During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are $\leq 0.60\ L_a$ for the combined Type B and Type C tests, and $\leq 0.75\ L_a$ for Type A tests.

b. Air lock testing acceptance criteria are:

1) Overall air lock leakage rate is $\leq 0.05\ L_a$ when tested at $\geq P_a$.

2) For each door, the seal leakage rate is $\leq$ 5 scf per hour when the gap between the door seals is pressurized to $\geq$ 10 psig.

The provisions of specification 4.0.2 do not apply to the test frequencies specified in the Primary Containment Leakage Rate Testing Program.

The provisions of specification 4.0.3 are applicable to the Primary Containment Leakage Rate Testing Program.

8. Ventilation Filter Testing Program (VFTP)

A program shall be established to implement the following required testing of Engineered Safety Feature (ESF) filter ventilation systems at the frequencies specified in Regulatory Guide 1.52, Revision 2, dated March 1978, and in accordance with ASME N510-1989.

The provisions of Specifications 4.0.2 and 4.0.3 are applicable to the VFTP test frequencies.

BASES TABLE B 3/4.4.6-1

REACTOR VESSEL TOUGHNESS

BELTLINE COMPONENT	MATERIAL TYPE OR WELD SEAM IDENTIFICATION	HEAT#/SLAB# OR HEAT#/LOT#	Ni(%)	CU(%)	P(%)	HIGHEST STARTING RT_{NDT} (°F)	MAXIMUM Δ RT_{NDT} (°F)	MIN. UPPER SHELF (ft-lb)	MAX. EOL RT_{NDT}
Plate	SA-533,Gr.B,Cl.1	C9404-2	0.49	0.07	0.008	+52*	11.5	69	+75.5
Plate	SA-533,Gr.B,Cl.1	C9425-1	0.51	0.12	0.009	+32*	21.5	59	+75
Weld	1NMW/E8018-G	3P4966/1214	0.90	0.03	0.011	- 6*	11	75.5	+15.5

NON-BELTLINE COMPONENT	MATERIAL TYPE OR WELD SEAM IDENTIFICATION	HEAT#/SLAB# OR HEAT#/LOT#	HIGHEST STARTING RT_{NDT} (°F)
Shell Ring	SA-533,Gr.B,Cl.1	C9481-1	+10
Top Head Flange	SA-508,Cl.2	BWR-446	+10
Vessel Flange	SA-508,Cl.2	BRC-424	+26
Feedwater Nozzle	SA-508,Cl.2	Q2Q25W	-6
Weld	1NMW	3P4966/1214	-6
Closure Stud	SA-540, Grade B-24	82552	+70 (Lowest Service Temperature)

* These values are given only for the benefit of calculating the end-of-life (EOL) RT_{NDT}.

3/4.6.1 PRIMARY CONTAINMENT

3/4.6.1.5 DELETED

3/4.6.1.6 DRYWELL AND SUPPRESSION CHAMBER INTERNAL PRESSURE

The limitation on drywell and suppression chamber internal pressure ensure that the containment peak pressure of 39.9 psig does not exceed the design pressure of 45 psig during LOCA conditions or that the external pressure differential does not exceed the design maximum external pressure differential of 5 psid. The limit of 0.75 psig for initial positive primary containment pressure will limit the total pressure to 39.9 psig which is less than the design pressure and is consistent with the accident analysis.

3/4.6.1.7 DRYWELL AVERAGE AIR TEMPERATURE

The limitation on drywell average air temperature ensures that the containment peak air temperature does not exceed the design temperature of 340°F during LOCA conditions and is consistent with the accident analysis.

3/4.6.1.8 DRYWELL AND SUPPRESSION CHAMBER PURGE SYSTEM

The drywell and suppression chamber purge supply and exhaust isolation valves are required to be closed during plant operation except as required for inerting, de-inerting and pressure control. During operations involving inerting, de-inerting and pressure control, only the drywell or suppression chamber purge supply and exhaust isolation valves may be open to prevent the creation of a bypass path between the drywell and suppression chamber. Creation of a bypass path between the drywell and the suppression chamber air space through the vent and purge lines would allow steam and gases from a LOCA to bypass the downcomers to the suppression pool in excess of design bypass leakage. These valves have been demonstrated capable of closing during a LOCA or steamline break accident from the full open position.

3/4.6.2. DEPRESSURIZATION SYSTEMS

The specifications of this section ensure that the primary containment pressure will not exceed the design pressure of 45 psig during primary system blowdown from full operating pressure.

The suppression chamber water provides the heat sink for the reactor coolant system energy release following a postulated rupture of the system. The suppression chamber water volume must absorb the associated decay and structural sensible heat released during the reactor coolant system blowdown from 1020 psig. Since all of the gases in the drywell are purged into the suppression chamber air space during a loss of coolant accident, the pressure of the liquid must not exceed 45 psig, the suppression chamber maximum pressure. The design volume of the suppression chamber, water and air, was obtained by considering that the total volume of reactor coolant and to be considered is discharged to the suppression chamber and that the drywell volume is purged to the suppression chamber. (See Figure B 3/4.6.2-1)

Using the minimum or maximum water volumes given in this specification, containment pressure during the design basis accident is approximately 39.6 psig which is below the design pressure of 45 psig. Maximum water volume of 131,900 ft^3 results in a downcomer submergence of 12.4 ft and the minimum volume of 128,800 ft^3 results in a submergence approximately 8 inches less. The majority of the Bogeda tests were run with a submerged length of four feet and with complete condensation. Thus, with respect to the downcomer submergence, this specification is adequate.

Should it be necessary to make the suppression chamber inoperable, this shall only be done as specified in Specification 3.5.3.

Under full power operating conditions, blowdown from an initial suppression chamber water temperature of 90°F results in a water temperature of approximately 135°F immediately following blowdown which is below the 200°F used for complete condensation via T-quencher devices. At this temperature and atmospheric pressure, the available NPSH exceeds that required by both the RHR and core spray pumps, thus there is no dependency on containment overpressure during the accident injection phase.

Experimental data indicates that excessive steam condensing loads can be avoided if the suppression pool peak bulk temperature can remain below saturation conditions. However, an additional concern raised related to the potential transfer of non-condensed SRV steam to the ECCS suction strainer, if local saturated conditions existed at the quencher and the ECCS suction is at a higher elevation than the SRV quencher. The LaSalle ECCS suction strainers are located above the elevation of the T-Quenchers. Further studies have shown that long steam plumes occur when subcooling levels are less than 9°F. However, the LaSalle T-Quenchers is at a submersion of 24 feet and provides 20°F subcooling with bulk temperature of 208°F with the wetwell at atmospheric pressure. This provides sufficient margin to ensure that exiting steam is condensed before posing a steam ingestion potential to the ECCS suction. Therefore, the peak bulk suppression pool limit for LaSalle will be 208°F.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 140 TO FACILITY OPERATING LICENSE NO. NPF-11

AND AMENDMENT NO. 125 TO FACILITY OPERATING LICENSE NO. NPF-18

COMMONWEALTH EDISON COMPANY

LASALLE COUNTY STATION, UNITS 1 AND 2

DOCKET NOS. 50-373 AND 50-374

I. INTRODUCTION

By letter dated July 14, 1999, as supplemented on January 21, February 15, February 23, March 10, March 24, two letters on March 31, April 7, and April 14, 2000, Commonwealth Edison Company (ComEd, the licensee) proposed changes to the licenses for LaSalle County Station, Units 1 and 2. The letters dated January 21, February 15, February 23, March 10, March 24, two letters on March 31, April 7, and April 14, 2000, contained supplemental, clarifying information that did not change the staff's initial proposed no significant hazards consideration determination. The proposed amendments would increase the maximum reactor core power level to 3489 megawatts thermal, an increase of 5 percent of rated core thermal power, for LaSalle County Station, Units 1 and 2. In addition, the proposed amendments correct a non-conservative value in the upper limit for drywell and suppression chamber internal pressure that was discovered during the course of the power uprate review.

II. EVALUATION

1.0 Background

LaSalle Units 1 and 2, are currently licensed to operate at a maximum reactor power level of 3323 megawatts thermal (MWt). The licensee, in conjunction with General Electric Company (GE), undertook a program to uprate the maximum reactor power by 5 percent to 3489 MWt.

The planned approach to achieving the higher power level consists of: (1) an increase in the core thermal power (with a more uniform and flattened power distribution) to create increased steam flow, (2) a corresponding increase in the feedwater system flow, (3) no increase in maximum core flow, and (4) reactor operation primarily along standard maximum extended load line limit (MELLL) rod/flow control lines. This approach is based on, and is technically consistent with, the NRC-approved boiling water reactor (BWR) generic power uprate guidelines that are presented in GE Topical Report NEDC-31897P, "Generic Guidelines For General Electric Boiling Water Reactor Power Uprate" (Reference 1). The plant-unique evaluations are based on a review of plant design and operating data to confirm excess design capabilities and, if necessary, identify any areas that may require modifications associated with power uprate. For some items, bounding analyses and evaluations in NEDC-31984P, "Generic Evaluations of General Electric Boiling Water Reactor Power Uprate" (Reference 2) demonstrate plant operability and safety.

2.0 Reactor Core and Fuel Performance

2.1 Fuel Design and Operation

Both LaSalle units currently contain a mixed core of GE and Siemens Power Corporation (SPC) fuel. Unit 1 is now operating in cycle 9 and the majority of the fuel bundles are GE fuel. Unit 2 is now operating in cycle 8 and the majority of the fuel bundles are GE fuel.

All fuel and core design limits will continue to be met by control rod pattern and/or core flow adjustments. Current design methods will not be changed for the power uprate. The power uprate will increase the core power density, and will have some effects on operating flexibility, reactivity characteristics, and energy requirements. These issues are discussed in the following sections.

2.2 Thermal Limits Assessment

Operating limits are established to ensure that regulatory requirements and safety limits are not exceeded for a range of postulated events as is currently the practice. The operating limit and safety limit Minimum Critical Power Ratio (MCPR) as well as the Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) and Linear Heat Generation Rate (LHGR) limits are cycle dependent and as such will be established or confirmed at each reload as described in reference 2.

2.3 Power/Flow Operating Map

The uprated power/flow operating map includes the operating domain changes for uprated power. The map includes the increased core flow (ICF) range and an uprated Maximum Extended Load Line Limit (MELLL). The maximum thermal operating power and maximum core flow correspond to the uprated power and the maximum core flow for ICF. Power has been rescaled so that uprated power is equal to 100 percent rated power. The changes to the power/flow operating map are consistent with the previously approved generic descriptions given in NEDC-31897P-A (Reference 1) and are acceptable to the staff.

2.4 Stability

LaSalle County Station, Units 1 and 2, is currently operating under the guidelines of reactor stability Interim Corrective Actions in response to NRC Generic Letter (GL) 94-02, "Long-Term Solutions and Upgrade of Interim Operating Recommendations for Thermal-Hydraulic Instabilities in Boiling Water Reactors." The units will implement the long-term stability Option III. The Option III solution monitors Oscillation Power Range Monitor (OPRM) signals to determine when a reactor scram is required. Until the implementation of Option III, the plant will continue to rely on the revised Interim Corrective Actions for both units using uprated stability region boundaries on the power/flow operating map. This is acceptable to the staff.

2.5 Reactivity Control - Control Rod Drives (CRD) and CRD Hydraulic System

The control rod drive (CRD) system controls gross changes in core reactivity by positioning neutron absorbing control rods within the reactor. It is also required to scram the reactor by rapidly inserting withdrawn rods into the core. The CRD system was evaluated at the uprated steam flow and dome pressure. Since there is no increase in the reactor operating pressure, CRD scram performance is not affected.

For CRD insertion and withdrawal, the required minimum differential pressure between the hydraulic control unit (HCU) and the vessel bottom head is 250 psid. The CRD pumps were evaluated against this requirement and were found to have sufficient capacity. The flows required for CRD cooling and driving are assured by automatic opening of the system control valve. The CRD system will continue to perform all its functions at uprated power and will function adequately during insert and withdraw modes.

The licensee indicated that the control rod drive mechanisms (CRDMs) have been designed in accordance with the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (Code) Section III, 1971 Edition. The components of the CRDMs that form part of the primary pressure boundary, have been designed for a bottom head pressure of 1250 psig, which is higher than the reactor bottom head pressure of 1040 psig for the uprated operating condition. The licensee concluded that the existing stress and fatigue analyses of CRDM components remain valid. The staff agrees with the licensee's conclusion.

3.0 Reactor Coolant System and Connected Systems

3.1 Nuclear System Pressure Relief

The nuclear boiler pressure relief system prevents overpressurization of the nuclear system during abnormal operating transients. The plant safety/relief valves (SRVs), in conjunction with the reactor protection system, provide this protection.

The licensee stated that safety-related SRV operability is not affected by the proposed changes. The licensee stated that the plant-specific analyses for the power uprate condition conservatively assume one SRV out-of-service. This additional margin in the plant-specific analyses provides reasonable assurance that the postulated SRV setpoint drift would not result in the maximum allowable system pressure being exceeded. Furthermore, the maximum operation reactor dome pressure remains unchanged for the power uprate. Consequently, the licensee concluded that the SRV setpoints and analytical limits are not affected by the proposed power uprate, and that the SRV loads for the SRV discharge line piping will remain unchanged. The staff agrees with the licensee's conclusion that the SRVs and the SRV discharge piping will continue to maintain their structural integrity and to provide sufficient overpressure protection to accommodate the proposed power uprate.

3.2 Reactor Overpressure Protection

The results of the overpressure protection analysis are contained in each cycle-specific reload amendment submittal. The design pressure of the reactor pressure vessel (RPV) remains at

1250 psig. The ASME code allowable peak pressure for the reactor vessel is 1375 psig (110 percent of the design value), which is the acceptance limit for pressurization events. The limiting pressurization event is a main steam isolation valve (MSIV) closure with a failure of the valve position scram. The MSIV closure was analyzed by the licensee using NRC-approved methods (ODYN), with the following assumptions: (1) 102 percent of the uprated core power and 105 percent of core flow; (2) the maximum initial reactor dome pressure was assumed to be 1020 psig, which is higher than the nominal uprated pressure; (3) one SRV was assumed out-of-service; and (4) the analysis did not take credit for externally actuated mode, via electro-pneumatic mode. The analysis took credit only for 12 SRVs and assumed that the SRV opening pressures were +3 percent above the nominal setpoint for the available valves. The calculated peak reactor pressure increases to 1332 psig, but remains below the 1375 psig code limit. This overpressure analysis is acceptable to the staff.

3.3 Reactor Vessel and Internals

The staff reviewed the licensee's submittal to determine whether the proposed licensing action would reduce the margins of safety that have been established in the licensing basis to ensure the structural integrity of the LaSalle, Units 1 and 2, reactor coolant pressure boundary and, in particular, to ensure the integrity of the RPV.

The plant parameters that could be affected by a power uprate include: pressure-temperature (P-T) limits and adjusted reference temperature (ART) calculations, upper shelf energy (USE) drop for the RPV materials, and the surveillance capsule withdrawal schedule.

The staff evaluated the P-T limits based on the following NRC regulations and guidance: 10 CFR Part 50, Appendix G; GL 88-11, "NRC Position on Intergranular Stress Corrosion Cracking (IGSCC) in BWR Austenitic Stainless Steel Piping"; GL 92-01, Revision 1, Supplement 1, "Reactor Vessel Structural Integrity"; Regulatory Guide (RG) 1.99, Revision 2; and the Standard Review Plan (SRP) 5.3.2. GL 88-11 advised licensees that the staff would use RG 1.99, Revision 2, to review P-T limit curves. RG 1.99, Revision 2, contains methodologies for determining the increase in transition temperature and the decrease in USE from neutron radiation. GL 92-01, Revision 1, Supplement 1, requested that licensees provide and assess data from other licensees that could affect their RPV integrity evaluations. These data sets are used as the basis for the staff's review of the P-T limit curves. Appendix G to 10 CFR Part 50 requires that P-T limit curves for the RPV be at least as conservative as those obtained by applying the methodology of Appendix G to Section XI of the ASME Code.

The staff evaluated the surveillance program based upon Appendix H to 10 CFR Part 50, "Reactor Vessel Material Surveillance Program Requirements." Appendix H to 10 CFR Part 50 includes criteria to monitor changes in the fracture toughness properties of ferritic materials in the RPV beltline region of light water nuclear power reactors which result from exposure of these materials to neutron irradiation and the thermal environment. Appendix H to 10 CFR Part 50 endorses the American Society for Testing and Materials (ASTM) E185, "Surveillance Tests for Nuclear Reactor Vessels." Appendix H states that "the design of the surveillance program and the withdrawal schedule must meet the requirements of the edition of ASTM E185 that is current on the issue date of the ASME Code to which the reactor vessel was purchased."

3.3.1 Reactor Vessel Integrity/Neutron Irradiation

Several analyses are performed to determine the impact that neutron irradiation has on the integrity of the reactor vessel. The most critical area is the beltline region of the reactor vessel, since it is predicted to be most susceptible to neutron damage. In regard to the power uprate and the reactor vessel integrity, the analyses should include an evaluation of the: (1) ART calculations, (2) heat-up and cooldown P-T limit curves, (3) upper shelf energy, and (4) surveillance capsule withdrawal schedule. It should be noted that these evaluations could be affected by changes in the neutron fluence and operating temperatures and pressures that result from a power uprate.

Licensee's Evaluation

ComEd assessed the effects of the LaSalle power uprate on the RPVs. The licensee evaluated the integrity of the reactor vessel at the revised design conditions in terms of impact due to the neutron fluence. More specifically, ComEd provided an assessment on the impact of the power uprate to: (1) the ART of the limiting RPV material, (2) the need to revise the ComEd P-T limit curves, (3) the change in the predicted USE drop for the RPV materials, and (4) determine whether changes in the RPV surveillance program (as required by 10 CFR Part 50, Appendix H) are necessary. Details of the licensee's assessment are provided below.

LaSalle, Unit 1

The licensee stated that the highest current ART end-of-license (EOL) value for LaSalle, Unit 1, reactor vessel (for the vertical weld 1P3571) remains as 83.8° F. With a nominal 5 percent increase in fluence, the licensee determined that the change in the ART value would not be significant, and therefore, revised P-T curves were not required.

LaSalle, Unit 2

In regard to the ART calculation, the licensee stated that the highest current ART EOL value for LaSalle, Unit 2, reactor vessel (for the lower intermediate shell plate C9404-2) remains as 75.3° F. With a nominal 5 percent increase in fluence, the licensee determined that the change in the ART value would not be significant and, therefore, revised P-T curves were not required.

In addition, ComEd found that the revised design conditions showed continued compliance with the existing design and licensing criteria for the LaSalle, Units 1 and 2, reactor vessels. ComEd went on to explain that with regard to the application of the requirements of 10 CFR Part 50, Appendices G and H, to the LaSalle, Units 1 and 2, RPV materials:

(1) The USE values would still remain above the 50 ft-lb value throughout the life of the vessel.

(2) There is no significant change in the 32 effective full power year (EFPY) shift in adjusted reference temperature and, therefore, the existing P-T curves remain bounding for power operation up to 3489 MWt.

(3) No changes in the Appendix H program (the RPV surveillance program) are required.

Staff Evaluation

LaSalle, Unit 1

The staff independently calculated the ARTs at the 1/4 thickness position in the vessel wall for the LaSalle, Unit 1, RPV beltline materials, considering the 5 percent power uprate. The staff verified that the licensee used a fluence increase that was greater than 5 percent, which is conservative when compared to the power increase. In calculating the ARTs for the beltline materials of LaSalle, Unit 1, the staff used the higher fluence of 5.04×10^{17} n/cm^2, as proposed by the licensee for the 5 percent power uprate. The staff independently verified that the limiting material for LaSalle, Unit 1, was the vertical weld 1P3571. In addition, the staff independently calculated the shift in the adjusted reference temperature (delta RT_{NDT} value) as a result of the power uprate, and determined that the uprate had a negligible effect on the value. The shift in the ART remained at 57.8° F , which resulted with an ART at EOL remaining at 83.8° F.

The staff also independently evaluated the USE, based upon the revised fluence value of 5.04×10^{17} n/cm^2, as a result of the power uprate. The staff determined that the minimum USE at EOL for the beltline materials of LaSalle, Unit 1, is 57.3 ft-lb. Therefore, the staff verified that the USE remains greater than 50 ft-lb for the design life of the LaSalle, Unit 1, vessel and maintains the margin requirements of 10 CFR Part 50, Appendix G.

In evaluating the surveillance program for LaSalle, Unit 1, the staff determined that the predicted transition temperature shift at vessel inside surface, as a result of the 5 percent power uprate, remained below 100° F ; therefore, the staff determined that the minimum number of capsules to be withdrawn and the capsule withdrawal schedule for LaSalle, Unit 1, still meet the ASTM E185-82 Standard. Since the minimum number of capsules and withdrawal schedule for LaSalle, Unit 1, meet the ASTM E185-82 Standard, the surveillance program is in compliance with Appendix H, 10 CFR Part 50, and is acceptable.

LaSalle, Unit 2

Regarding the RPV assessment, the staff independently calculated the ARTs at the 1/4 thickness position in the vessel wall for the LaSalle, Unit 2, RPV beltline materials, considering the 5 percent power uprate. The staff verified that the licensee used a fluence increase that was greater than 5 percent, which is conservative when compared to the power increase. In calculating the ARTs for the beltline materials of LaSalle, Unit 2, the staff used the higher fluence of $6.03 \text{ X } 10^{17}$ n/cm^2, as proposed by the licensee for the 5 percent power uprate. The staff independently verified that the limiting material for LaSalle, Unit 2, was the lower-intermediate shell plate C9404-2. In addition, the staff independently calculated the shift in the ART as a result of the power uprate, and determined that the uprate had a negligible effect on the value. The shift in the ART remained at 11.6° F , which resulted with an ART at EOL remaining at 75.3° F.

The staff also independently evaluated the USE, based upon the revised fluence value of 6.03×10^{17} n/cm^2, as a result of the power uprate. The staff determined that the minimum USE at EOL for the beltline materials of LaSalle, Unit 2, is 53.6 ft-lb. Therefore, the staff verified that the USE remains greater than 50 ft-lb for the design life of the LaSalle, Unit 2, vessel and maintains the margin requirements of 10 CFR Part 50, Appendix G.

In evaluating the surveillance program for LaSalle, Unit 2, the staff determined that the predicted transition temperature shift at vessel inside surface, as a result of the 5 percent power uprate, remained below 100° F; therefore, the staff determined that the minimum number of capsules to be withdrawn and the capsule withdrawal schedule for LaSalle, Unit 2, still meet the ASTM E185-82 Standard. Since the minimum number of capsules and withdrawal schedule for LaSalle, Unit 2, meet the ASTM E185-82 Standard, the surveillance program is in compliance with Appendix H, 10 CFR Part 50, and is acceptable.

Conclusion

Based on the staff's review of the ComEd submittal, the staff found that the issues regarding the integrity and operation of the LaSalle, Units 1 and 2, RPVs is adequately addressed in the ComEd submittal. The staff also determined that as a result of the power uprate, the LaSalle, Units 1 and 2, RPVs still meet the requirements of Appendices G and H of 10 CFR Part 50.

3.3.2 Reactor Internals Evaluation

The licensee evaluated the reactor vessel and internal components in accordance with the current licensing basis. Load combinations include reactor internal pressure difference (RIPD), loss-of-coolant accident (LOCA), safety/relief valves (SRV) discharge, seismic, and fuel lift loads. The seismic loads are unaffected by the power uprate. There is no increase in LOCA and SRV dynamic loads because the existing design loads are bounding for the power uprate. In its response dated February 15, 2000, to the staff's request for additional information, the licensee indicated that the asymmetric pressurization (AP) and line break thrust loads were also considered in appropriate load combinations for the evaluation of the reactor vessel and internal components. However, the existing AP and line break thrust loads are bounding for the power uprate. In its evaluation of the reactor internals, the licensee recalculated the reactor internal pressure differences for the proposed power uprate for normal, upset and faulted conditions, respectively.

The stresses and cumulative fatigue usage factors (CUFs) for the reactor internal and vessel components were evaluated by the licensee in accordance with the code of record at LaSalle County Station; the ASME Code, Section III, 1968 Edition with addenda to and including Winter 1969 for Unit 1, and the 1968 Edition with addenda to and including 1970 addenda for Unit 2, with certain exceptions and modifications as specified in LaSalle's Updated Final Safety Analysis Report (UFSAR). The load combinations for normal, upset and faulted conditions were considered in the evaluation. The maximum stresses for critical components of the reactor internals were calculated for the power uprate conditions. The calculated stresses are less than the allowable code limits. The licensee indicated that the stresses were determined by scaling the existing (pre-uprate) stresses based on bounding uprated conditions (pressure, temperature, and flow). The staff finds this to be conservative and consistent with the methodology approved by the NRC in a letter to GE, "Staff Position Concerning Generic Boiling

Water Reactor Power Uprate Program," dated September 30, 1991, and is, therefore, acceptable.

The licensee indicated that the evaluations of structural integrity of the reactor vessel were performed considering operating conditions such as feedwater flow and temperature and steam flow that are affected by the power uprate. The licensee provided the calculated CUFs in Table 3-3 of its July 14, 1999, submittal. The reactor vessel components not listed in Table 3-3 have maximum stress and CUF that are either not affected by the power uprate or bounded by those listed in Table 3-3. The primary plus secondary stresses shown in the table are within the allowable limit and the CUF is less than the code limit of unity. The staff finds that the methodology used by the licensee is consistent with the NRC-approved methodology in Appendix I of NEDC-31897P-1 and is, therefore, acceptable.

The licensee assessed the potential for flow-induced vibration based on the GE prototype plant vibration data for the reactor internal components recorded during startup testing and on operating experience from similar BWR/5 plants with a 251-inch diameter vessel. The vibration levels were calculated by extrapolating the recorded vibration data to power uprate conditions and compared to the plant allowable limits for acceptance. The licensee confirmed that vibration levels of all safety-related reactor internal components are within the acceptance limit of 10 Ksi peak stress intensity, specified by the GE design criteria. The staff finds this acceptable in comparison to the ASME criteria of 13.6 Ksi.

Based on its review of the information provided by the licensee, the staff finds that the maximum stresses and fatigue usage factors are within the code-allowable limit, and concludes that the reactor vessel and internal components will continue to maintain their structural integrity for the power uprate condition.

3.4 Reactor Recirculation System

The power uprate will be accomplished by operating along extensions of rod lines on the power/flow map with no increase in maximum core flow. The cycle-specific core reload analyses will be performed with the most conservative core flow. The evaluation by the licensee of reactor recirculation system performance at uprated power determined that the system drive flow increases by approximately 0.6 percent, which is within the capability of the recirculation system. An evaluation of the net positive suction head (NPSH) for the recirculation pumps, jet pumps, and flow control valves found that power uprate does not significantly change the NPSH required or the NPSH margin. The cavitation protection interlocks for the recirculation pumps, jet pumps, and flow control valves remain the same in terms of absolute thermal power values, and there is no change in the downshift or runback set points.

3.5 Reactor Coolant Pressure Boundary Piping

The licensee evaluated the effects of the power uprate condition, including higher flow rate, temperature, pressure, fluid transients and vibration effects on the reactor coolant pressure boundary (RCPB) and components. The components evaluated included equipment nozzles, anchors, guides, penetrations, pumps, valves, flange connections, and pipe supports. The evaluation was performed using the original code of record specified in the LaSalle County

Station UFSAR, the code allowables, and analytical techniques similar to those used in the original and existing design-basis analysis. The licensee indicated that no new assumptions were introduced that were not in the original analyses.

The RCPB piping systems evaluated include the main steam piping, reactor recirculation piping, feedwater piping, RPV bottom head drain line, reactor water cleanup (RWCU), reactor vessel head vent line, reactor core isolation cooling (RCIC), core spray piping, high pressure core spray piping (HPCS), residual heat removal (RHR), safety/relief valve (SRV) discharge piping and CRD piping. The licensee evaluated the RCPB piping systems by reviewing the increase in pressure, temperature, and flow rate against the same parameters in the original design-basis analyses. The bounding percentage increases for affected limiting piping systems were applied to obtain the maximum calculated stresses, displacements, and the CUF for the power uprate. The staff finds that the approach is consistent with the methodology provided in Appendix K to Reference 1.

In its February 15, 2000, letter, the licensee provided the calculated maximum stresses and fatigue usage factors at critical locations of the piping systems evaluated for the power uprate. Based on the information provided by the licensee, all calculated stresses are within ASME allowable limits and the calculated fatigue usage factors are less than the allowable limit of 1.0. The licensee also concluded that the evaluation showed compliance with all appropriate code requirements for the piping systems evaluated and that power uprate will not have an adverse effect on the RCP system design. The staff reviewed the licensee's evaluation and finds that the licensee's conclusions are acceptable.

Based on the above review, the staff concludes that the design of piping, components and their supports will be adequate to maintain the structural and pressure boundary integrity of the RCPB components and supports in the proposed power uprate.

3.6 Main Steamline Flow Restrictors

The licensee stated that there is no impact on the structural integrity of the main steamline flow restrictors for the power uprate. The licensee indicated that a higher peak RPV transient pressure of 1332 psig results from plant operation at 3489 MWt conditions, but this value remains below the ASME Code limit of 1375 psig. Therefore, the main steam line flow restrictor will maintain its structural integrity following the power uprate since the restrictor was designed for a differential pressure of 1375 psig, which exceeds that for uprated power conditions.

3.7 Main Steam Isolation Valves (MSIVs)

The MSIVs are part of the reactor coolant pressure boundary and perform the safety function of steamline isolation during certain design basis abnormal events. The MSIVs must be able to close within a specified time range at all design and operating conditions, upon receipt of a closure signal. The generic evaluation for MSIVs (as discussed in Reference 2), is based on an operating pressure increase of less than or equal to 40 psi, a temperature increase of less than or equal to 5° F, and a system flow increase of less than or equal to 5 percent. The LaSalle power uprate is bounded by the conclusions of the generic evaluation and, therefore, the MSIVs are acceptable for power uprate operation.

3.8 Reactor Core Isolation Cooling System (RCIC)

The RCIC provides core cooling when the RPV is isolated from the main condenser, and the RPV pressure is greater than the maximum allowable for initiation of a low pressure core cooling system. The licensee evaluated the RCIC system for loss of feedwater transient events and has determined that its operation will continue to be consistent with the bases and conclusions of the generic evaluation in Reference 2.

The occurrence of transient speed peaks during the turbine start up is effectively minimized through the use of a special contoured plug for the steam admission valve, which limits steam flow to the turbine during initial valve opening. This is an alternate approach to the control system modifications described in GE SIL No. 377 and is acceptable to the staff.

3.9 Residual Heat Removal System (RHR)

The residual heat removal system (RHR) is designed to restore and maintain the coolant inventory in the reactor vessel and to provide primary system decay heat removal following reactor shutdown for both normal and post-accident conditions. The RHR system is designed to operate in the low pressure coolant injection (LPCI) mode, shutdown cooling mode, suppression pool cooling mode, and containment spray cooling mode. The effects of the power uprate on these operating modes are discussed in the following paragraphs with the exception of LPCI which is discussed in Section 4.2.2.

3.9.1 Shutdown Cooling Mode

The operational objective for normal shutdown is to reduce the bulk reactor temperature to 125°F in approximately 20 hours, using two RHR loops. At the uprated power level, the decay heat is increased proportionally, thus slightly increasing the time required to reach the shutdown temperature. This increased time is judged to have an insignificant impact on plant safety.

3.9.2 Suppression Pool Cooling Mode

The Suppression Pool Cooling (SPC) mode is designed to remove heat discharged into the suppression pool to maintain pool temperature below the Technical Specification limit during normal plant operation and below the suppression pool design temperature limit of 212° F following a LOCA. The power uprate increases the reactor decay heat which increases the heat input to the suppression pool during a LOCA, resulting in a slightly higher peak suppression pool temperature. The objective of the SPC mode of RHR is met with the power uprate, because the peak suppression pool temperature analysis by the licensee confirms that the pool temperature has not increased and will stay below its design limit at uprated conditions.

3.9.3 Fuel Pool Cooling Assist Mode

The Containment Spray Cooling (CSC) mode is designed to spray water from the suppression pool via spray headers into the containment airspace, to reduce containment pressure and temperature during post-accident conditions. The licensee indicated that the power uprate slightly increases the containment spray water temperature by about 3° F. This increase has a negligible effect on the calculated values of drywell pressure, drywell temperature, and suppression chamber pressure because these parameters reach peak values prior to actuation of the containment spray.

3.9.4 Conclusion

Based on its review of the licensee's rationale and evaluation, and review of power uprate applications for similar BWR plants, the staff has concluded that plant operations at the proposed uprated power level will have an insignificant or no impact on the shutdown cooling mode, fuel pool cooling assist mode, or the suppression pool cooling mode of RHR.

3.10 Reactor Water Cleanup System

The operating conditions of the Reactor Water Cleanup System (RWCU) (flow, temperature, and pressure) are not changed as a result of power uprate. Due to the increased feedwater flow rate, a slight reduction in the proportion of RWCU system flow to feedwater flow results in a slightly higher reactor water conductivity. The conductivity will remain within the chemistry limits bounding plant operation which are unchanged for power uprate conditions. Therefore, the staff concludes that operations at the proposed uprated power level will have an insignificant impact on the RWCU system.

3.11 Balance of Plant Piping

The licensee evaluated the stress levels for balance of plant (BOP) piping and appropriate components, connections and supports in a manner similar to the evaluation of the reactor coolant pressure boundary piping and supports based on increases in temperature and pressure from the design-basis analysis input. The evaluated BOP systems include lines which are affected by the power uprate such as the feedwater heater piping, main steam bypass lines, and portions of the main steam, recirculation, feedwater, RCIC, HPCI, and RHR systems outside the primary containment. The existing design analyses of the affected BOP piping systems were reviewed against the uprated power conditions. The licensee concluded that in all cases there is sufficient margin between the calculated stresses and the code-allowable limits to accommodate the increase in stresses due to the increase in pressure, temperature and flow as a result of the power uprate. The staff finds that the stress ratios provided by the licensee are within the code-allowable limits and are, therefore, acceptable.

The licensee evaluated pipe supports by evaluating the piping interface loads due to the increases in pressure, temperature, and flow for affected limiting piping systems. The licensee indicated that there is an adequate margin between the original design stresses and code limits for the supports to accommodate the load increase and as such, all evaluated pipe supports were within the code-allowable limits. The licensee reviewed the original postulated pipe break analysis and concluded that the existing pipe break locations were not affected by the power uprate, and no new pipe break locations were identified. The staff finds the licensee's evaluation to be acceptable.

4.0 Engineered Safety Features

4.1 Containment System Performance

The LaSalle County Station UFSAR provides the results of analyses of the containment response to various postulated accidents that constitute the design basis for the containment. Operation with a 5 percent power uprate from 3323 MWt to 3489 MWt would change some of the conditions and assumptions of the containment analyses. Topical Report NEDC-31897, "Generic Guidelines For General Electric Boiling Water Reactor Power Uprate," Section 5.10.2 requires the power uprate applicant to show the acceptability of the effect of the uprated power on containment capability. These evaluations will include containment pressures and temperatures, LOCA containment dynamic loads, and safety-relief valve containment dynamic loads. Appendix G of NEDC-31897 prescribes the generic approach for this evaluation and outlines the methods and scope of plant-specific containment analyses to be done in support of power uprate. Appendix G states that the applicant will analyze short-term containment pressure and temperature response using the GE M3CPT code (current analyses). These analyses will cover the response through the time of peak drywell pressure throughout the range of power/flow operating conditions with the power uprate. A more detailed computer model of the nuclear steam supply steam (LAMB) may be used to determine more realistic RPV break flow rates for input to the M3CPT code. The use of the LAMB code has been reviewed by the NRC for application to LOCA analysis in accordance with 10 CFR Part 50, Appendix K. The results from these analyses will also be used for input to the LOCA dynamic loads evaluation.

Appendix G of NEDC-31897 also requires the applicant to perform long-term containment heatup (suppression pool temperature) analyses for the limiting UFSAR events to show that pool temperatures will remain within limits for suppression pool design temperature, ECCS NPSH and equipment qualification temperatures. These analyses can be performed using the GE computer code SHEX. SHEX is partially based on M3CPT and is used to analyze the period from when the break begins until after peak pool heatup (i.e., the long-term response). The SHEX computer code has been used by GE on all BWR power uprates and has been shown to be acceptable based on confirmatory calculations for validation of the results.

4.1.1 Containment Pressure and Temperature Response

Short-term and long-term analyses of the containment pressure and temperature response following a large break inside the drywell (DBA LOCA) are documented in the LaSalle UFSAR. The short-term analysis was performed to determine the peak drywell and wetwell pressure response during the initial blowdown of the reactor vessel inventory into the containment following a large break inside the drywell, while the long-term analysis was performed to determine the peak pool temperature response considering decay heat addition.

The licensee indicated that the containment analyses were performed in accordance with Regulatory Guide 1.49 and NEDC-31897 using GE codes and models. The M3CPT code was used to model the short-term containment pressure and temperature response. The more detailed RPV model (LAMB) was used for determining the vessel break flow for input to the M3CPT code in the containment analyses for power uprate. The use of the LAMB model is justified in "General Electric Company Analytical Model for Loss-of-Coolant Accident Analysis in Accordance with 10CFR50 Appendix K," NEDE-20566-P-A, September 1986. The staff finds the use of the LAMB model detailed RPV break flow input to the M3CPT code in the containment analysis for power uprate acceptable.

The licensee also indicated that the SHEX code was used to model the long-term containment pressure and temperature response. Based on the experience gained from our review of power uprates for similar BWR plants, the staff finds the use of this code acceptable for the LaSalle power uprate.

4.1.1.1 Long-Term Suppression Pool Temperature Response

(1) Bulk Pool Temperature

The licensee indicated that the long-term bulk suppression pool temperature response was evaluated for the limiting DBA LOCA in Section 6.2 of the UFSAR. The bounding analysis was performed at 102 percent of uprate power (3489 MWt.) using the SHEX code and the more realistic decay heat model (ANS/ANSI 5.1+two sigma). The staff has determined that the use of ANS/ANS 5.1-1979 decay heat model with an uncertainty adder of two sigma is acceptable.

The revised long-term containment response analyses were performed at 102 percent of the uprated power level and at 102 percent of the original power level using updated methods and decay heat model to show the difference in containment pressure and temperature due to uprated power. These analyses calculated the peak suppression pool temperature of 193° F at the uprated power level and 190° F at the current power level. The present UFSAR value for the above case was 200° F with the previous methods and decay heat model. The peak calculated suppression pool temperature of 193° F at uprated power remains below the suppression pool temperature limit of 212° F required for ECCS pumps to have adequate NPSH and the structural design temperature of 275° F.

The licensee also analyzed the highest bulk pool temperature response for the alternate shutdown cooling transient from a non-LOCA event at 102° F of the uprated power level. The limiting alternate shutdown activity assumes reactor isolation with availability of one RHR heat exchanger. This analysis calculated the peak bulk pool temperature to be 207° F at 102 percent of uprated power, which is also within the ECCS NPSH pump limit of 212° F.

Based on the results of these analyses, the staff concludes that the peak bulk suppression pool temperature response remains acceptable from both NPSH and structural design standpoints for the power uprate.

(2) Local Pool Temperature with SRV Discharge

NUREG-0783, "Suppression Pool Temperature Limits for BWR Containments," dated November 1981, specified local suppression pool temperature limits to ensure smooth steam condensation without the imposition of significant loads on the containment. Specifically, for all plant transients involving SRV operations during which the steam flux through the quencher perforations is less than 42 lb_m/ft^2-sec, the suppression pool local temperature shall be at least

20° F subcooled, and in no case shall it exceed 210° F. However, the Boiling Water Reactor Owners Group (BWROG) requested the elimination of the local suppression pool temperature limits which was approved by the staff in its safety evaluation report dated August 29, 1994. That safety evaluation report recognized the potential for the extended plume when the local pool temperature limit is removed. This is particularly important for plants with ECCS strainer suction inlets above or at the same elevation of the SRV quenchers. The LaSalle T-quenchers and ECCS suction strainers are located essentially at the same elevation.

The licensee evaluated all design and licensing basis events at uprated power conditions. During the review, a potential steam ingestion concern with the reactor core isolation cooling (RCIC) suction strainer was identified. The RCIC suction strainer is in close proximity to the SRV "K" T-quencher. As such, maintaining the local suppression pool temperature to 20° F subcooled is not possible at a suppression pool temperature greater than 200° F. Therefore, there is a potential for the bubble produced by the discharge of the SRV "K" T-quencher to be ingested by the RCIC suction strainer at a suppression pool temperature greater than 200° F. According to the licensee's analysis, this scenario is only applicable to the station blackout (SBO) event. The licensee's proposed solution for this potential problem is to impose procedural controls within the emergency operating procedures (EOPs) to limit the RPV cooldown rate. By limiting the RPV cooldown rate, the licensee's analysis indicates that a 20° F subcooling margin can be maintained, thus, eliminating extended plume ingestion concerns. This will maintain the maximum suppression pool temperature to 196° F for an SBO event at uprated power conditions. With the maximum suppression pool temperature under 200° F, the required subcooling will be maintained which will preclude steam ingestion into the RCIC suction strainer. The licensee indicates that this is a temporary solution "pending evaluation of the permanent resolution," of extended steam plumes within the suppression pool. Therefore, the staff concludes that the local suppression pool temperature limits will be maintained in the uprated power condition pending the implementation of the permanent resolution.

The licensee also evaluated the heat capacity temperature limit curve (HCTL) which is contained in the emergency operating procedures. The HCTL curve evaluates the reactor pressure vessel pressure versus the suppression pool temperature as a function of suppression pool level. For uprated power conditions, the HCTL curve would be increased to account for the effects of the higher power level. The licensee noted that for certain low suppression pool level cases, the 20° F subcooling margin could not be maintained with the increased HCTL curve. However, the licensee has concluded that there is adequate orientation and separation of the ECCS suction strainers to prevent steam ingestion during SRV operation. Therefore, the staff concludes that the HCTL curve can be modified to address the effects of the higher power level.

(3) <u>Steam Bypass Case</u>

A concern during a LOCA event is steam bypass of the suppression pool due to a leakage between the drywell and the wetwell airspace in the highly unlikely event of a reactor depressurization to the drywell accompanied by a simultaneous open bypass path between the drywell and suppression pool. The licensee indicated that a steam bypass analysis in UFSAR Section 6.2 was performed to bound 102 percent of uprated power for the limiting event of an intermediate steamline break with the maximum allowable leakage path to ensure that there is sufficient time for corrective operator action. The evaluation showed that the operator will have 59 minutes after being alerted to steam bypass leakage by wetwell pressure to effectively take corrective action, which is more than the 15 minutes postulated in the UFSAR.

Based on its review of the licensee's evaluation, and review of power uprate applications for similar BWR plants, the staff concludes that the suppression pool steam bypass case will remain acceptable after the power uprate.

4.1.1.2 Short-Term Containment Gas Temperature Response

The licensee indicated that the drywell design temperature of 340° F was determined based on a bounding analysis of the superheated gas temperature which can be reached with blowdown of steam to the drywell during a LOCA as documented in UFSAR Section 6.2.1. Because the reactor vessel dome pressure assumed for the maximum drywell temperature analysis still bounds the power uprate condition, the power uprate has no impact on the short-term peak drywell temperature.

The wetwell gas space peak temperature is calculated assuming thermal equilibrium between the pool and wetwell gas space. The analyses calculated the maximum pool temperature of 193° F during a LOCA event and 207° F during an Alternate Shutdown Cooling event at uprated power. These values remain below the suppression chamber design value of 275° F; therefore, the containment gas temperature response for the power uprate has no adverse effect on the containment structure.

Based on our review of the licensee's evaluation, and review of power uprate applications for similar plants, we agree with the licensee's conclusion that the drywell and wetwell gas temperature response will remain acceptable after the power uprate.

4.1.1.3 Short-Term Containment Pressure Response

The licensee indicated that the short-term containment response analyses were performed for the limiting DBA LOCA, which assumes a double ended guillotine break of a recirculation suction line to demonstrate that operation at the proposed uprated power level does not result in exceeding the containment design pressure limits. The short-term analysis covers the blowdown period during which the maximum drywell pressure, maximum wetwell pressure and maximum differential pressure between the drywell and containment occur. These analyses were performed at 102 percent of the uprated power level and 100 percent of the current power level, using updated methods reviewed and accepted by NRC to show the difference due to uprated power. The revised analyses calculated a maximum containment pressure of 39.9 psig at the uprated power level and 39.3 psig at the current power level. The calculated maximum pressure of 39.9 psig at the uprated power level remains below the containment design pressure of 45 psig.

The peak calculated drywell-to wetwell pressure difference remains less than or equal to the design value of 25 psid at uprated power using the M3CPT code. The licensee indicated that the M3CPT code is overly conservative in that it does not consider compressibility of the wetwell air space in response to pool swell following a LOCA event. The drywell-to-wetwell pressure difference was also evaluated using the PICSM computer code. The PICSM code uses the GE Pool Swell Analytical Model and was accepted in NUREG-0487 and NUREG-0808. The peak drywell-to-wetwell pressure difference of 22.4 psid was determined for the limiting condition with power uprate, which remains less than the design value of 25 psid.

Based on our review of the licensee's evaluation, and review of power uprate applications for similar BWR plants, the staff concludes that the containment pressure response following a postulated LOCA will remain acceptable after the power uprate.

4.1.2 Containment Dynamic Loads

4.1.2.1 LOCA Containment Dynamic Loads

The licensee indicated that the LOCA containment dynamic loads for the power uprate are based primarily on the short-term LOCA analyses, which provide calculated values for the controlling parameters for the dynamic loads throughout the blowdown. The key parameters are the drywell and wetwell pressures, vent flow rate and suppression pool temperature. The LOCA dynamic loads which are considered in the power uprate evaluations include pool swell, condensation oscillation, and chugging.

The licensee stated that the short-term containment response conditions with power uprate are within the range of test conditions used to define the pool swell and condensation oscillation loads. The long-term response conditions with power uprate in which chugging would occur, are within the range of test conditions used to define the chugging loads. Therefore, the LOCA dynamic loads are not impacted by the power uprate.

Based on our review of the licensee's evaluation, and review of power uprate applications for similar BWR plants, the staff concludes that the LOCA containment dynamic loads will remain acceptable after the power uprate.

4.1.2.2 Safety Relief Valve Loads

The safety-relief valve (SRV) air-clearing loads include SRV discharge line (SRVDL) loads, suppression pool boundary pressure loads, and drag loads on submerged structures. These loads are influenced by the SRV opening setpoint pressure, SRVDL geometry, and suppression pool geometry and operating parameters. The licensee indicated that the only parameter change which can affect the SRV loads is an increase in SRV opening setpoint pressure. The SRV setpoint, which was the basis for the SRVDL loads and the SRV loads on the suppression pool boundary and submerged structures, is the highest setpoint from among the SRV groups. The highest setpoint with power uprate does not increase from the pre-power uprate value, and therefore, the power uprate will not impact the SRV load definitions.

Based on its review of the licensee's evaluation, and review of power uprate applications for similar BWR plants, the staff concludes that, because there is no change in SRV setpoint pressure, the plant operation at uprated power will not impact the SRV containment loads.

4.1.2.3 Subcompartment (Annulus and Drywell Head) Pressurization

The loads due to subcompartment pressurization are controlled by the initial energy release rates from the break, which in turn are governed by the initial reactor thermal-hydraulic conditions. The licensee indicated that the results of its evaluation show that the mass and energy release rates at uprated conditions are no greater than the design basis release rates. Therefore, the design basis subcompartment pressurization loads of record would bound the loads applicable to the uprated conditions. Because the energy release rates from the break remain within the design basis, the staff concludes that operation at uprated power will not impact pressurization of the annulus and drywell.

4.1.3 Containment Isolation

The licensee indicated that the system designs for containment isolation are not affected by the power uprate. No new containment isolation device is required because of the power uprate. The capability of the actuation devices to perform at power uprate conditions has been evaluated and determined to be acceptable. No new containment isolation devices are required because of power uprate. All motor-operated valves (MOVs) used as containment isolation valves will comply with Generic Letter 89-10, "Safety Related Motor Operated Valve Testing and Surveillance" at uprated conditions as discussed below. Based on the licensee's evaluation, the staff concludes that plant operations at the proposed uprated power level will have an insignificant or no impact on the containment isolation system.

4.1.4 Generic Letter 89-10 Program

In its letter dated March 31, 2000, the licensee stated that the safety-related MOV's were evaluated for changes in fluid flow, pressure, temperature and differential pressures for the power uprate. The licensee identified valves whose operating conditions have changed due to power uprated plant conditions. For these affected valves, the existing design basis parameters were evaluated to be bounding for the power uprate. In its letter dated April 9, 2000, the licensee further confirmed that all safety-related MOVs and air operated valves (AOVs) will perform their intended function(s) following the power uprate. All MOVs in the LaSalle County Station MOV program will continue to comply with Generic Letter 89-10. Because the operating conditions of the safety-related MOVs will continue to be within the design basis parameters, the staff concludes that operation at the uprated power level will have no effect on the ability of the valves to perform their intended function.

4.1.5 Generic Letters 95-07 and 96-06

The licensee reviewed the plant-specific information on systems and components for the power uprate to determine its potential effect on the performance of mechanical components. The licensee concluded that there will be no significant effect on pumps and valves at LaSalle from the power uprate. The licensee evaluated changes in environmental temperatures and determined that the long term peak suppression pool post-accident temperature is lower than the previously analyzed value and the peak drywell temperature remains unchanged. Therefore, the proposed power uprate has no impact on the licensee's GL 95-07 evaluation regarding valve pressure locking or thermal binding.

The licensee also indicated that the proposed power uprate condition is bounded by the current containment analysis, and thus, has no impact on the evaluation in response to GL 96-06, "Assurance of Equipment Operability and Containment Integrity During Design Basis Accident Conditions" on potential over-pressurization of isolated piping segments for LaSalle Units 1 and 2.

Based on the licensee's evaluation, the staff concludes that operation at the uprated power level will not impact the results of the analyses performed in response to Generic Letters 95-07 and 96-06 and the operability of the mechanical components is not affected.

4.2 Emergency Core Cooling Systems (ECCS)

The effect of the power uprate on each ECCS system is addressed below. The licensee has concluded that the current ECCS flow rates are sufficient for the power uprate conditions and the LaSalle ECCS operating conditions bound the generic evaluation approved by the staff.

4.2.1 High Pressure Core Spray System (HPCS)

The HPCS system was evaluated by the licensee and the evaluation is consistent with the bases and conclusions contained in the generic evaluation for the power uprate. The maximum injection pressure for the HPCS system is conservatively based on the upper analytical setpoint for the lowest available group of SRVs, which does not change for power uprate. HPCS system operation at the power uprate conditions does not change, and thus does not have any effect on the availability or reliability of the system, and does not change any of the original design pressures or temperatures for the system components. This is acceptable to the staff.

4.2.2 Low Pressure Core Injection System (LPCI mode of RHR)

The hardware for the low pressure portions of the RHR are not affected by the power uprate. The upper limit of the low pressure ECCS injection setpoints will not be changed for the power uprate; therefore, the low pressure portions of these systems will not experience any higher pressures. The licensing and design flow rates of the low pressure ECCS will not be increased. Since the system does not experience different operating conditions due to the power uprate, there is no impact. The licensee stated that LaSalle, Units 1 and 2, are bounded by the generic analyses presented in Section 4.1 of Reference 2. This is acceptable to the staff.

4.2.3 Low Pressure Core Spray System (LPCS)

The hardware for the low pressure core spray system are not affected by the power uprate. The upper limit of the low pressure ECCS injection set points will not be changed for the power uprate, therefore the low pressure portions of these systems will not experience any higher pressures. The licensing and design flow rates of the low pressure ECCS will not be increased. Therefore, since these systems do not experience different operating conditions due to the power uprate, there is no impact. The licensee stated that LaSalle, Units 1 and 2, are bounded by the generic analyses presented in Section 4.1 of Reference 2. This is acceptable to the staff.

4.2.4 Automatic Depressurization Systems (ADS)

The ADS uses safety/relief valves to reduce reactor pressure following a small break LOCA with HPCS failure. This function allows low pressure coolant injection (LPCI) and low pressure core spray (LPCS) to flow to the vessel. The ADS initiation logic and ADS valve control are adequate for the power uprate, and the power uprate does not require any change in logic and control of ADS. Plant design requires a minimum flow capacity for the SRVs, and requires that ADS initiate after a time delay on either low water level plus high drywell pressure, or on low water level alone. The ability to perform either of these functions is not affected by the power uprate.

4.2.5 Net Positive Suction Head (NPSH)

The effect of changes to the calculated suppression pool temperature on the NPSH of the RHR pumps during the suppression pool cooling and containment spray cooling modes is discussed in the licensee's submittal. The results show that there is adequate NPSH for the RHR and core spray (CS) pumps. Since the peak suppression pool temperature is not increased from the pre-uprate conditions, the NPSH available at peak temperature conditions is not adversely affected and the power uprate will not affect compliance with the ECCS pump NPSH requirements.

4.3 ECCS Performance Evaluation

The emergency core cooling systems (ECCS) are designed to provide protection against hypothetical loss-of-coolant accidents (LOCAs) caused by ruptures in the primary systems piping. The ECCS performance under all LOCA conditions and their analysis models must satisfy the requirements of 10 CFR 50.46 and 10 CFR Part 50, Appendix K. The fuel used in LaSalle Units 1 and 2 was analyzed by the licensee with the NRC-approved methods (SAFER/GESTR). The results of the base ECCS-LOCA analysis using NRC-approved methods is presented in NEDC-32258P (Reference 4), the plant specific ECCS-LOCA results for the LaSalle units.

The licensee used the staff-approved SAFER/GESTR (S/G) methodology to assess the ECCS capability for meeting the 10 CFR 50.46 criteria. The S/G-LOCA analysis for LaSalle, Units 1 and 2, was performed by the licensee with the appropriate reload fuel in accordance with NRC requirements to demonstrate conformance with the ECCS acceptance criteria of 10 CFR 50.46 and Appendix K. The base S/G-LOCA analyses were performed at a nominal power level of 3323 MWt (100 percent of the current rated power) and an Appendix K power level of 3454 MWt (104 percent of the current rated power).

The LOCA analysis performed for the 102 percent of uprated power conditions indicates a peak calculated temperature of 1276° F. This is less than the acceptance criteria of 2200° F and, therefore, is acceptable. The analyses described in Reference 4 for the uprated conditions were performed in accordance with the NRC requirements, and demonstrates conformance with the ECCS acceptance criteria of 10 CFR 50.46 and, therefore, are acceptable.

4.4 Main Control Room Atmosphere Control System

The control room atmosphere control system (CRACS) containing an emergency filtration system is designed to maintain the control room envelope at a slightly positive pressure relative to the outside atmosphere and thus minimizes unfiltered in-leakage of contaminated outside air into the control room following a LOCA. The licensee stated that the changes in core inventory and the resulting post-accident radiation levels resulting from the power uprate are within regulatory limits, and that operation of the plant at the uprated power level will have no significant impact on the CRACS.

Based on our review of the licensee's evaluation and review of power uprate applications for similar BWR plants, we find that plant operations at the proposed uprated power level do not change the design and operational aspects of the CRACS, and will have no significant impact on the CRACS.

4.5 Standby Gas Treatment System

The standby gas treatment system (SGTS) is designed to ensure controlled and filtered release of particulates and halogens from primary and secondary containment to the environment during abnormal and accident situations in order to maintain off-site thyroid doses within the 10 CFR Part 100 limits. The SGTS is sized to maintain the secondary containment at a slight negative pressure of 0.25 inch water gauge with respect to the outside atmosphere. Maintaining this negative pressure serves to prevent unfiltered release of radioactive material from the secondary containment to the environment. The licensee stated that the capability of the SGTS to maintain this negative pressure is not affected by the proposed power uprate.

The licensee also stated that the charcoal filter beds are not affected by uprated power level operation. The post-LOCA iodine loading increases slightly at the uprated conditions and remains below the original design capability of the filter.

Based on our review of the licensee's evaluation, and review of power uprate applications for similar BWR plants, we conclude that the uprated power level operation will have no significant impact on the ability of the SGTS to meet its design objectives.

4.6 Main Steam Isolation Valve Leakage Control System

The main stream isolation valve leakage control system is designed to control any leakage of contaminated steam through redundant isolation valves on each main stream line from the reactor to the turbine in the event of a LOCA. The licensee performed an evaluation and stated that the existing analyses in the Updated Final Safety Analysis Report (UFSAR) have sufficient margin to ensure that the increases in LOCA radiological consequences resulting from the 5 percent power uprate remain within the guidance of 10 CFR Part 100.

Based on our review of the licensee's evaluation and review of power uprate applications for similar BWR plants, we find that plant operations at the proposed uprated power level do not change the design and operational aspects of the main stream isolation valve leakage control system, and will have no significant impact on the main stream isolation valve leakage control system.

4.7 Post-LOCA Combustible Gas Control

The combustible gas control system is designed to control the hydrogen concentrations of the drywell and containment atmospheres below the lower flammability limit of 4.0 volume percent (v/o) following a LOCA. Design of the system is based on evolution of hydrogen from three sources including (1) metal-water reaction of active fuel cladding, (2) corrosion of zinc and aluminum exposed to water during the LOCA, and (3) radiolysis of water. As a result of the power uprate, only the post-LOCA production of hydrogen by radiolysis will increase in proportion to the power. The licensee indicated that the increase in hydrogen generation due to the power uprate has a minor impact on the time available to start the system before reaching procedurally controlled limits, but does not impact the ability of the system to maintain hydrogen below the lower flammability limit. The recombiner start time decreases from 6 hours at rated power to 5 hours for the power uprate to maintain the hydrogen concentration below the lower flammability limit of 4.0 v/o following the LOCA. Without recombiner operation, the hydrogen concentration is expected to reach the 4.0 v/o limit 15 hours after the LOCA for the uprated condition, compared to 22 hours for the current rated power. The power uprate has no impact on recombiner maximum operating temperature which is dependent only on the containment hydrogen concentration when the recombiners are started.

Based on our review of the licensee's evaluation, and review of power uprate applications for similar BWR plants, we agree with the licensee's conclusion that plant operations at the proposed uprated power level will have no significant impact on the post-LOCA combustible gas control system and the system will remain acceptable.

5.0 Instrumentation and Control

The control and instrumentation signal ranges and analytical limits for setpoints were evaluated to establish the effects of the changes in various process parameters, such as power, neutron flux, steam flow, and feedwater flow. As required, analyses were performed to determine the need for setpoint changes for various functions (e.g., main steamline high-flow isolation setpoints). On the basis of the analytical evaluation, the following changes are proposed to maintain an adequate difference between plant operating parameters and trip setpoints:

(1) The trip setpoint and allowable value for the Average Power Range Monitor (APRM) Flow Biased Simulated Thermal Power - High Scram for two-loop operation contained in TS Section 2.2, "Limiting Safety System Settings - Reactor Protection System Instrumentation Setpoints," are proposed for revision in TS Table 2.2.1-1, Function 2.b.(1)(a).

(2) The trip setpoints and allowable value for the APRM Flow Biased Simulated Thermal Power - High Scram and High Flow Clamped for single-loop operation contained in TS Section 2.2, "Limiting Safety System Settings - Reactor Protection System Instrumentation Setpoints," are proposed for revision in TS Table 2.2.1-1, Functions 2.b.(2)(a) and 2.b.(2)(b).

(3) The trip setpoint and allowable value for the Automatic Initiation - Primary Containment Isolation - Main Steam Line - Flow- High contained in TS Section 3.3.2, "Isolation Actuation Instrumentation, Main Steam Line - Flow - High," are proposed for revision in TS Table 3.3.2-2, Function A.1.c.(3).

(4) The trip setpoint and allowable value for the APRM Flow Biased Simulated Thermal Power - Upscale Control Rod Withdrawal Block for two-loop operation contained in TS Section 3.3.6, "Control Rod Withdrawal Block Instrumentation," are proposed for revision in TS Table 3.3.6-2, Function 2.a.(1).

(5) The trip setpoint and allowable value for the APRM Flow Biased Simulated Thermal Power - Upscale Control Rod Withdrawal Block for single-loop operation contained in TS Section 3.3.6, "Control Rod Withdrawal Block Instrumentation," are proposed for revision in TS Table 3.3.6-2, Function 2.a.(2).

The revised setpoints proposed by the licensee have been established using ComEd setpoint methodology. Each setpoint was selected with a sufficient difference between the system setting and the actual value in the safety analysis (analytical limit) to preclude inadvertent initiation of the protective action while assuring adequate allowances for instrument accuracy, calibration, drift, and applicable design basis events relative to the analytical limit.

An increase in the core thermal power and the steam flow affects some instrument setpoints. These setpoints are adjusted to maintain comparable differences between system settings and actual limits and reviewed to ensure that adequate operational flexibility and the necessary safety functions are maintained at the uprated power level. As part of power uprate implementation, the ComEd setpoint methodology was used to generate the allowable values and the nominal trip setpoints related to the analytical limit changes.

The ComEd setpoint methodology is based on American National Standards Institute/ Instrument Society of America (ANSI/ISA) S67.04, Parts 1 and 2 of the 1994 version. This methodology does not deviate from, nor does it require ComEd to make any exceptions to, Regulatory Guide 1.105, "Setpoints for Safety-Related Instrumentation." This methodology was inspected and approved by the NRC as part of an instrumentation and control inspection at Dresden Nuclear Power Station in 1994. An evaluation of this methodology was also included in Amendment No. 129 to Facility Operating License No. NFP-11 for LaSalle Unit 1. In this amendment, the NRC approved the licensee's setpoint methodology with the statement that "the staff compared the methodology used in Calculation No. L-001420 to the methodology shown in licensee document NES-EIC-20.04 and determined that the methodology as shown in NES-EIC-20.04 and Appendix A, B, and C of that document was suitable for use in Calculation No. L-001420 because it contained the proper terms for establishing setpoints."

As a result of the increase in reactor power, all potentially affected analytical limits for setpoints were assessed. The proposed setpoints and allowable values will provide adequate allowances between the operational settings and the analytical limits to ensure the necessary safety functions and are, therefore, acceptable.

6.0 Electrical Power and Auxiliary Systems

6.1 AC Power

6.1.1 Offsite Power

The staff has reviewed information provided by the licensee to determine the impact of the power uprate on offsite power. Areas included in the review were grid stability and electrical systems associated with the main turbine auxiliary systems.

6.1.1.1 Grid Stability and Reliability Analysis

The licensee performed a grid stability uprate review to determine the adequacy of grid stability for the LaSalle power uprate. The grid stability studies assume not only the effect of the LaSalle uprate on power output but also changes in electrical output from other stations. The increases reduce system voltage support (reduced volt amperes reactive (MVARs) with increased power factor (PF)) and affect the generator critical clearing time. The LaSalle 345-KV switchyard circuit breaker 1-2 local breaker backup (LBB) timer settings require reduction. The licensee stated that grid stability remains adequate with the uprate LBB timer settings.

The staff requested the licensee to discuss how it compensates for the loss of MVARs with the power uprate to maintain the grid stability and whether it needs to add or adjust the capacitor banks at the switchyard. In response to the staff request, the licensee stated that there are no physical changes to the 345-KV switchyard equipment. The voltage rating and operating ranges remain unchanged. ComEd's Bulk Power Operations (BPO) requests the LaSalle Station to increase or decrease the MVARs as required by the grid needs. The generator rating (MVA and PF), as described by the generator's capacity curves that are part of the LaSalle Station Operating Procedures, determine MVAR production. Megawatt electric output limitations may be necessary for uprate conditions to meet the MVAR requirements.

Studies performed in support of the LaSalle power uprate indicate the decreased MVAR capacity at the higher MW output will have a slight impact on voltage stability. Using a 2001-year base case, the licensee's studies indicate that the reduced MVAR output will decrease the amount of load at which voltage collapse may occur. However, this reduced collapse point is still greater than 105 percent of the predicted peak load for year 2001.

In addition to the studies mentioned above, the licensee performs ongoing analyses of the power system to insure that adequate reliability can be maintained under many different contingency situations. All of these studies follow established criteria defined by ComEd's System Planning Department. The criteria define the various contingencies which are simulated to ensure adequate reliability of the bulk power transmission system during steady state and transient conditions. These criteria are maintained by System Planning and are reviewed annually. The criteria are documented and reported to Federal Energy Regulatory Commission (FERC) on an annual basis as part of ComEd's FERC 715 filing. Additional capacitors may be required at key locations to compensate for system load growth. The LaSalle power uprate is considered during these periodic power system analyses.

The protective relay settings for the generator, main power transformer, unit auxiliary transformer, and system auxiliary transformer correlate with the respective equipment MVA, voltage, impedance, and current parameters. Except for one of the LaSalle switchyard circuit breaker (1-2 LBB) timer settings which requires reduction, the existing protective relay settings remain adequate for power uprate. The required change to the LBB settings will be implemented by the licensee prior to uprate power ascension. The 345-KV switchyard equipment and stability remain adequate with the uprate LBB timer settings. The licensee has reviewed and determined that the connections to the switchyard remain adequate under uprate conditions.

On the basis of this information, the staff concludes that the proposed power uprate at LaSalle in conjunction with the periodic system analyses will ensure that the grid stability and reliability will remain adequate.

6.1.1.2 Electrical Systems Associated With the Main Turbine Auxiliary Systems

The licensee performed a review to determine the adequacy of electrical systems associated with the main turbine auxiliary systems. The review determined that the electrical system's configuration and operating voltage ranges are unchanged and remain adequate for operation at the higher output. The review determined that the isophase bus rating, the main power transformer ratings, the unit auxiliary power transformer ratings, the system auxiliary power transformer ratings, the 345-KV switchyard equipment ratings and operating voltage ranges, the generator voltage and current ratings, and operating voltage ranges bound the uprate operating conditions. The operation of the isophase bus, main power transformers, unit auxiliary power transformers, system auxiliary power transformers, and 345-KV switchyard equipment remain adequate with incorporation of the uprate LBB settings. Based on the licensee's review, the staff concludes that the turbine/generator and major electrical components remain adequate for operation at the higher output.

6.1.2 Onsite Power

6.1.2.1 Onsite Power Distribution System

The onsite power distribution system consists of transformers, buses, switchgear, and distribution panels. The transmission system, motor-generator power supplies, and diesel generators power the alternate current (ac) distribution system. Station batteries provide direct current (dc) power to the dc distribution system. The licensee noted that operation at the uprated level is achieved by utilizing existing equipment operating at or below the nameplate and service factor ratings. Equipment running loads or nameplate data loads establish the station load calculation basis. The heater drain pumps and condensate/condensate booster pumps experience some increased flow because of uprated conditions and are the only motor loads affected by power uprate operation. Because flows are only slightly increased, the motor demand for each of these loads remains bounded by the electrical system's capacity and capability. Therefore, under normal conditions, the electrical supply and distribution components (switchgear, motor control centers, cables, etc.) remain adequate.

The licensee stated that there is no increase in flow or pressure required for any of the ac powered emergency core cooling system equipment because of power uprate, and the operation of engineered safety feature loads are unaffected by power uprate conditions. Therefore, the existing diesel generator load calculations are unchanged by the uprated conditions, and the current emergency power system design remains adequate. The system has sufficient capacity to support the required loads for safe shutdown, to maintain a safe-shutdown condition, and to operate the required engineered safeguards equipment following a postulated accident. Based on the discussion above, the staff concludes that operation at the uprated power level has no impact on the current emergency power system.

6.1.3 Station Blackout Analysis (SBO)

In a request for additional information, the staff requested the licensee to verify that the assumptions for the existing SBO analysis are still valid for the power uprate conditions, particularly as they relate to issues such as heatup analysis, equipment operability, and battery capacity. In response to the staff's request, the licensee stated that the primary effect of power uprate on the SBO coping analysis is the influence that the increase in decay heat has on the final suppression pool and drywell temperatures. The licensee has determined that an optimal reactor pressure vessel (RPV) depressurization cooldown rate ($\leq$ 20°F/hr) can be selected, which will accommodate the increase in decay heat to enable Units 1 and 2 to withstand and recover from an SBO. The licensee stated that the optimal RPV depressurization cooldown rate is being incorporated into station procedures and the updated final safety analysis report (UFSAR).

The existing loss of heating, ventilation, and air conditioning analyses for a SBO at LaSalle is for a coping duration of four hours. The heat loads associated with lighting and equipment are applied over the entire 4-hour SBO. The licensee evaluated equipment operability and concluded that the small temperature changes resulting from the power uprate conditions have no effect on the equipment as previously evaluated for the SBO event. This evaluation included equipment located in the auxiliary electrical equipment room, the control room, and the reactor core isolation cooling room.

The required dc electrical loads for an SBO are assumed to be energized for the entire 4-hour SBO event plus recovery. Loads that are not required for SBO coping are shed through station procedures. The licensee determined that the power uprate will not affect the equipment required to support SBO nor will it affect the loads that are required to be shed. Since the required SBO dc loads are conservatively assumed to be energized over the entire duration of the SBO, the power uprate will not result in any changes to parameters associated with the SBO Class 1E battery coping analysis.

On the basis of this information, the staff concludes that operation at the uprated power level does not significantly affect the SBO coping analysis.

6.2 DC Power

There are no load changes as a result of the power uprate that affect the existing DC power system design. Operation at uprated power levels does not change the existing design basis loading or revise any control logic. Therefore, the power uprate does not result in any changes to the DC power system.

6.3 Spent Fuel Pool Cooling System

As a result of plant operations at the proposed uprated power level, the decay heat load for any specific fuel discharge scenario will increase slightly and will be determined for each reload. The licensee stated that there are administrative controls and procedures in place to ensure that adequate cooling capability is provided for planned refuelings and for unplanned offloads. Also, the UFSAR will be revised to require refueling-specific analyses under the current plant conditions to be conducted prior to core offload. These analyses will ensure that the SFP water temperature will be maintained below the temperature limit of 140° F in accordance with the acceptance criteria as described in Section 9.1.3 of the Standard Review Plan (SRP), NUREG-0800.

LaSalle has a separate SFP for each unit. The SFPs are separated by gates (transfer canal gates/cask well gates). The SFP cooling system for each SFP consists of two 100 percent capacity coolant pumps and two 100 percent capacity heat exchangers. The piping configuration is such that each SFP pump is able to operate with either heat exchanger. Also, the opposite Unit's SFP cooling system can be used to provide cooling to the SFP by removing the transfer canal gates/cask well gates. The licensee stated that LaSalle's Shutdown Safety Management Program and procedures describe the requirement for removing these transfer canal gates/cask well gates.

In addition, as discussed in Section 3.9.4 above, supplemental fuel pool cooling is provided by the RHR system in the event that the SFP heat load exceeds the heat removal capability of the SFP cooling system. The licensee has an existing procedure governing the use of the B loop of RHR in the event that the calculated heat load in the SFP would cause the SFP water temperature to exceed 150° F.

Maintaining the SFP temperature limit at LaSalle is based on two primary parameters. The first is the ultimate heat sink (UHS) temperature, since the heat removal capability of the SFP cooling systems is a function of ultimate heat sink temperature. The second is the core in-vessel decay time following reactor shutdown, since this determines the heat load in the SFP. The licensee stated that prior to planned refueling outages, administrative controls and procedures are in place to require analyses to be performed for determining the required core in-vessel decay time following reactor shutdown and for determining which equipment must be placed in service to maintain the SFP water below the temperature of 140° F.

Based on our review of the licensee's evaluation, our review of power uprate applications for similar BWR plants, and the fact that the plant has administrative controls and operating procedures in place to ensure that backup cooling capability is provided for all SFP cooling scenarios, we find that the design and operation of the SFP cooling systems (SFP cooling system and RHR system in the SFP cooling assist mode) for the power uprate conditions at LaSalle meet the intent of the guidance described in the SRP for SFPs. Therefore, we conclude that operation of the SFP at power uprate conditions is acceptable.

6.4 Water Systems

6.4.1 Service Water Systems

The service water systems are designed to provide cooling water to various systems (both safety-related and nonsafety-related).

6.4.1.1 Safety-Related Loads

These safety-related loads include the loads from the following components/systems: RHR heat exchangers, RHR pump seal coolers, low pressure core spray pump motor coolers, emergency diesel generator heat exchangers, core standby cooling system area coolers, high pressure core spray diesel generator heat exchangers, and spent fuel pool emergency makeup. The safety-related performance of the service water system to provide cooling for these components/systems during and following the design basis accident is not significantly dependent on the reactor rated power.

The diesel generator loads and the RHR system flows remain unchanged for LOCA conditions following uprated operation. The cooling loads from core spray and RHR pump room coolers, RHR pump seal coolers, and miscellaneous room coolers remain virtually the same as that for the current rated power level operation because the equipment performance in the areas serviced by these coolers does not change for power uprate post-LOCA conditions. In addition, the ability to supply emergency makeup to the spent fuel pool is not impacted by the uprate condition. Therefore, the licensee concluded that plant operation at the uprated power level does not require the modification of the service water system for the safety-related loads.

Based on our review of the licensee's evaluation and review of power uprate applications for similar BWR plants, we find that plant operations at the proposed uprated power level will have an insignificant or no impact on the service water system regarding the safety-related loads.

6.4.2 Main Condenser, Circulating Water, and Normal Heat Sink Performance

The main condenser, circulating, and normal heat sink systems are designed to provide the main condenser with a continuous supply of cooling water for removing heat rejected to the condenser by turbine exhaust, turbine bypass steam, and other exhausts over the full range of operating loads thereby maintaining low condenser pressure as recommended by the turbine vendor. The licensee stated that the performance of the main condenser, circulating water, and normal heat sink systems was evaluated and found adequate for plant operations at the proposed uprated power level. The staff finds the licensee's analysis acceptable.

6.4.3 Reactor Building Closed Cooling Water System

The reactor building closed cooling water (RBCCW) system is designed to remove heat from various auxiliary plant equipment housed in the reactor building. The licensee performed evaluations and stated that the increase in heat loads on this system due to uprated power operations is insignificant.

Based on our review of the licensee's evaluation and review of power uprate applications for similar BWR plants, we find that plant operations at the proposed uprated power level do not change the design aspects and operations of the RBCCW system. Therefore, we agree with the licensee's conclusion that the impact of plant operations at the proposed uprated power level on the RBCCW system is insignificant.

6.4.4 Turbine Building Closed Cooling Water System

The turbine building closed cooling water (TBCCW) system supplies cooling water to auxiliary plant equipment in the turbine building. The licensee stated that the TBCCW system heat-load increases due to power uprate are those related to the operation of the turbine-generator. The licensee stated that the TBCCW system has adequate heat removal capability for plant operations at the proposed uprated power level. Therefore, the staff finds the TBCCW system acceptable for power uprate operations

6.4.5 Ultimate Heat Sink

The ultimate heat sink (UHS) for LaSalle County Station is a cooling pond that remains after the main dike of the cooling lake is breached. As a result of operation at the uprated power level, the post-LOCA UHS temperature increases due to higher reactor decay heat. The licensee performed an evaluation and concluded that the UHS will provide a sufficient quantity of water at a temperature of less than 100° F (design temperature) following a design basis LOCA.

Based on its review of the licensee's evaluation, the staff concludes that plant operations at the proposed uprated power level will have an insignificant impact on the UHS.

6.5 Standby Liquid Control System

The Standby Liquid Control System (SLCS) is designed to assure reactor shutdown, from full power operation to cold subcritical by mixing boron with the primary reactor coolant, in an event when no control rods can be inserted. The ability of the SLCS boron solution to achieve and maintain safe shutdown is not a direct function of core thermal power, and therefore, is not affected by the power uprate. SLCS shutdown capability is re-evaluated for each reload core. Since there is no increase in the reactor operating pressure at uprate conditions, SLCS pump relief valve margin is not affected. The system was found to have the capability to deliver its design rated flow rate and is therefore acceptable.

6.6 Power-Dependent Heating, Ventilation, and Air Conditioning (HVAC) Systems

The HVAC systems consist mainly of heating, cooling supply, exhaust and recirculation units in the reactor building, turbine building and the drywell. Power uprate results in a small increase in these system heat loads due to slightly higher process temperature and higher electrical currents in some motors and cables. The licensee stated that some areas in the reactor building, turbine building and main steam tunnel will experience slightly higher heat loads as a result of power uprate. The licensee performed evaluations and stated that based on the small increase in overall heat load and excess design capacity, plant operations at the proposed uprated power level will have no impact on the HVAC systems for the above cited areas.

Based on its review of the licensee's evaluation and review of power uprate applications for similar BWR plants, the staff concludes that plant operations at the proposed uprated power level has no significant impact on the HVAC systems for the above cited areas.

6.7 Fire Protection

Fire suppression or detection is not expected to be impacted due to plant operations at the proposed uprated power level since there are no physical plant configurations or combustible load changes resulting from the uprated power operations. In addition, the safe shutdown systems and equipment used to achieve and maintain cold shutdown conditions do not change for the uprated conditions, and the operator actions required to mitigate the consequences of a fire are not affected. The licensee performed an evaluation to demonstrate post-fire safe shutdown capability in compliance with the requirements of 10 CFR Part 50, Appendix R, assuming power uprate conditions. The licensee concluded that plant operation at the proposed uprated power level does not affect the ability of the Appendix R systems to perform their safe shutdown function.

Based on our review of the licensee's evaluation and review of power uprate applications for similar BWR plants, we agree with the licensee that the post-fire safe shutdown capability will not be affected by plant operations at the proposed uprated power level.

6.8 Other Systems Not Impacted or Insignificantly Impacted by the Power Uprate

In its application, the licensee stated that the following systems and equipment are either not impacted or are potentially impacted in a minor way by operation at uprated power.

Auto Generation Control	Aux Building Floor Drains	Aux Equip Room Panel
Auxiliary Steam	Caustic Handling	Clean Condensate Storage
Communication	Containment Monitoring	Main Control Room Atmos. Control
Misc.Drains and Vents	Miscellaneous Drains	Oil Drain Disposal
Penetration Pressurization	Primary Containment Purge	Pump House Ventilation
QA Records Vault HVAC	Control Rod Grapples	Control Rod Position Ind
Diesel Room Floor Drains	Domestic Water	Drains-Station Heating Cond.
Elect. Welder outlets	Emergency Breathing Air	Fuel Service Equipment
Fuel Storage	Grounding & Cathodic Prot.	Heat Tracing
Hoists & Cranes	Laundry Equip Radwaste	Lighting
Industrial Security	Machine Shop Ventilation	Radwaste Drains
Instrument & Control	Radwaste Facility HVAC	Reactor Drains & Vents
Laboratory HVAC	Reactor Vessel Service Equip.	
Laundry & Floor Drains	Refueling Equipment	Rod Sequence Control System
Warehouse	Screen Wash	Service Air
Sewage Treatment	Station Heating	Suppression Pool Cleanup
Test Instruments	Treated Water	Turbine Bldg Equip Drains
Turbine Bldg	Turbine Test	Turbine/Generator Vent & Drain
Under Vessel Service Equip	Vendor Supplied/Serviced Equipment	
Acid Feed & Handling	Alternate Rod Insertion	Annunciator

Area Radiation Monitor	Auxiliary Building	Auxiliary Power
Carbon Dioxide	Chem Feed and Handling	Well Water
Chemical Radwaste Disp.	Chlorination	Control Room Panels
Cycled Condensate Storage	Diesel Fuel Oil	Diesel Gen Room Vent
Environmental Monitor	Fire Seals	Fuel Storage
Gland Water	Hydrogen Water Chemistry	Loose Parts Monitoring
Makeup Demineralizer	Misc. Ventilation	Nitrogen Inerting
Off-Gas Sampling	Process Rad Monitor	Process Sampling
Radwaste Equip Drains	Radwaste Floor Drains	Reactor Building
Control Room Panels	Remote Shutdown	Service Bldg HVAC
Refrigerator Piping	Switchyard Relay House HVAC	
Turbine and Aux Drains	Turbine Bldg Floor Drains	Vibration Monitoring
Instrument Air	Instrument Nitrogen	Zinc Injection

These systems and equipment are not impacted by the power uprate because their operation is not affected by changes to the parameters resulting from the power uprate. The staff reviewed this list and agrees with the licensee that plant operations at the proposed uprated power level has no impact or insignificant impact on these systems.

7.0 Power Conversion Systems

7.1 Turbine-Generator

The licensee performed evaluations for turbine operations with respect to design acceptance criteria to verify the mechanical integrity under the conditions imposed by the power uprate. Results of the evaluations showed that there would be insignificant increase in the probability of turbine overspeed and its associated turbine missile production due to plant operations at the proposed uprated power level. The licensee stated that there is sufficient design margin in the current turbine overspeed protection trip settings. Therefore, the turbine could continue to be operated safely at the proposed uprated power levels.

Based on our review of the licensee's evaluation and review of power uprate applications for similar BWR plants, we find that operation of the turbine at the proposed uprated power level is acceptable.

7.2 Turbine Steam Bypass

The turbine bypass valves were initially rated for total steam flow capacity of 25 percent of the original rated steam flow. The transient analysis for uprated power assumes the same bypass capacity. Therefore, the bypass capacity is adequate for the power uprate.

7.3 Feedwater and Condensate Systems

The feedwater and condensate systems do not perform a safety-related function. However, their performance has an effect on the capability of the plant to operate at uprated conditions. The condensate and feedwater systems were originally designed for 105 percent warranted steam flows. Operation at the uprated power level does not significantly affect operating conditions of these systems. During steady state conditions, the systems have adequate net positive suction head for all of the pumps to operate without cavitation in the uprated condition. In addition, the condensate and feedwater systems have adequate flow capacity margin during flow transients to support the uprated condition.

8.0 Radwaste Systems and Radiation Sources

8.1 Liquid and Solid Waste Management

The single largest source of liquid and wet solid waste is from the backwash of the condensate demineralizers. The licensee stated that the power uprate results in an increase of flow rate through the condensate demineralizers, therefore, the average time between backwashes will be reduced slightly. This reduction does not affect plant safety. Similarly, the reactor water cleanup (RWCU) filter demineralizer may require more frequent backwashes due to slightly higher levels of activation and fission products. The licensee further stated that the activated corrosion products in liquid wastes are expected to increase proportionally to the power uprate. However, the total volume of processed waste is not expected to increase appreciably, since the only significant increase in processed waste is due to the more frequent backwash of the condensate demineralizers and RWCU filter demineralizer. The licensee performed evaluations of plant operations and effluent reports, and concluded that the requirements of 10 CFR Part 20 and 10 CFR Part 50, Appendix I, will continue to be satisfied.

Based on our review of the licensee's evaluation and review of power uprate applications for similar BWR plants, we agree with the above licensee's conclusion and find the liquid and wet solid radwaste system acceptable for power uprate conditions.

8.2 Gaseous Waste Management

Gaseous wastes generated during normal and abnormal operation are collected, controlled, processed, stored, and disposed utilizing the gaseous waste processing treatment systems. These systems which are designed to meet the requirements of 10 CFR Part 20 and 10 CFR Part 50, Appendix I, include the offgas system and standby gas treatment system, as well as other building ventilation systems. Various devices and processes, such as radiation monitors, filters, isolation dampers, and fans, are used to control airborne radioactive gases. Results of licensee analyses demonstrate that airborne effluent activity released through building vents is not expected to increase significantly due to plant operations at the proposed uprated power level. The release limit is an administratively controlled variable, and is not a function of core power.

Based on our review of the licensee's evaluation and review of power uprate applications for similar BWR plants, we conclude that plant operations at the proposed uprated power level will have an insignificant impact on the above systems.

Offgas System

Core radiolysis (i.e., formation of H_2 and O_2) increases linearly with core power, thus increasing the heat load on the offgas recombiner and related components. The licensee evaluated the impact of the increases of these offgases resulting from plant operation at the uprated power level on the offgas system. The licensee stated the operational increase in offgas due to the power uprate remains well within the design capacity of the system. The system radiological release rate is administratively controlled, and is not changed with operating power. Therefore, power uprate does not affect the offgas system design or operation.

Based on our review of the licensee's evaluation and review of power uprate applications for similar BWR plants, we conclude that plant operations at the proposed uprated power level will have an insignificant impact on the offgas system.

8.3 Radiation Sources in the Core and the Coolant

Radioactive materials in the reactor core are produced in direct proportion to the reactor power and the duration of irradiation. Most of the nuclides having the greatest significance with regard to dose calculations have short half lives and reach equilibrium during the operation cycle. Nuclides with longer half lives continue to accumulate with irradiation time.

During reactor operation, the coolant passing through the core region becomes radioactive because of activation of impurities in the reactor water, activation of corrosion products suspended in the coolant, and release of fission products from fuel rods. Coolant and corrosion activation products in the reactor water would increase in approximate proportion to the increase in reactor power. However, the steam concentration would remain nearly constant because the increase in activity is effectively balanced by the increase in steam flow. ComEd stated that while the increased feedwater flow could reduce the filter efficiency of the condensate demineralizers resulting in an increase in corrosion product production, the resulting concentrations are expected to remain within the existing design basis concentrations.

With regard to the release of fission products, ComEd stated that although the design basis offgas rate is 0.1 Curies/second after thirty minutes decay, the observed offgas rates are well below the design basis. ComEd stated that the power uprate does not change the design basis noble gas release rates from the fuel.

Based on our review of the licensee's evaluation and experience with other similar power uprates, the staff agrees with the licensee that the power uprate at LaSalle Units 1 and 2 will not have an adverse effect on radiation sources in the reactor core or reactor coolant.

9.0 Reactor Safety Performance Evaluations

9.1 Reactor Transients

The Unit 1 Reload 7 cycle 8 was used as the representative fuel. Reload licensing analyses evaluate the limiting plant transients. Disturbances of the plant caused by a malfunction, a single failure of equipment, or personnel error are investigated according to the type of initiating event. The licensee's methodology to calculate the effects of the limiting reactor transients has been approved by the NRC. The limiting events to be analyzed are identified in Reference 1. The relatively small changes in rated power and maximum allowed core flow are not expected to affect the selection of limiting events. The events explicitly evaluated for the power uprate analysis are:

Loss of Feedwater Heating (LOFWH)
Feedwater Controller Failure (FWCF)
Generator Load Rejection without Bypass (GLRWOB)
Turbine Trip without Bypass (TTWOB)
Rod Withdrawal Error (RWE)
Slow Recirculation Flow Increase

The limiting events which establish the minimum critical power ratio (MCPR) operating limits for the power uprate conditions are GLRWOB, TTWOB and FWCF. The analyses for the limiting transients were performed at 100 percent of uprated power. These events are analyzed with the staff approved ODYN code and the GEMINI methodology which include an allowance for core thermal power uncertainty. The input parameters for the transient analyses are presented in Table 9-1, and the results of the transient analyses are presented in Table 9-2 of NEDC-32701P. The licensee, in its letter dated January 21, 2000, stated that LaSalle, Units 1 and 2, have virtually identical system geometries, reactor protection system configurations and mitigation functions. Additionally, both units have similar thermal-hydraulic and transient behavior characteristics. Therefore, system behavior after the power uprate is expected to be the same for both units. Direct or statistical allowance for a 2 percent power uncertainty is included in the analysis. Transient events identified in the generic evaluation are analyzed at the full uprated power and maximum allowed core flow operating point which bounds the power/flow map shown in Figure 2-1 of NEDC-32701P.

Cycle specific analyses will be done at each reload and the results will be part of the Core Operating Limits Report (COLR) developed by the licensee.

The safety limit minimum critical power ratio (SLMCPR) is calculated by the licensee as part of the reload licensing analyses using the NRC-approved methodology for the appropriate reload fuel. No change will be made to this methodology due to the power uprate. The analysis plan proposed by the licensee is acceptable. The licensee will submit the results of the cycle specific analysis with each reload document.

9.2 Design Basis Accidents

ComEd considered the effects of the power uprate at LaSalle Units 1 and 2 on the postulated consequences of DBAs. The magnitude of the consequences of a DBA is dependent upon the quantity of fission products released to the environment, atmospheric dispersion factors, and dose exposure pathways. The dose exposure pathways and atmospheric dispersion factors are unaffected by the power uprate. The power uprate results in an increased inventory of radioactivity available in the core for release. The quantity of fission products released to the environment is dependent on the inventory available for release and the transport mechanisms between the core and the environment.

ComEd performed plant specific accident reviews at uprated conditions for selected postulated events. These postulated design basis events include the loss-of-coolant accident (LOCA), main steam line break (MSLB), fuel handling accident (FHA), control rod drop accident (CRDA), inadvertent main steam isolation valve (MSIV) closure, instrument line break, feedwater line break, radioactive gas waste system leak or failure, and liquid radwaste tank failure. The refueling cask drop accident was also evaluated.

9.2.1 Loss-of-Coolant Accident

For the LOCA, the radiological consequences have increased due to the power uprate although the current analysis assumed a power level that bounds the uprated power level. The current LOCA analysis assumes a power level of 3910 MWt with a core burnup of 2034 effective full power days. All doses, except for the control room and auxiliary electric equipment room whole body doses, are the same as those previously approved by the staff in amendments 126 and 111 to NPF-11 and NPF-18, respectively.[1] These values for the proposed change have increased from those previously reviewed by the staff, due to the addition of a dose contributor previously not accounted for in the LOCA calculation. This contributor is direct radiation shine from gamma radiation sources external to the facility. With this contributor included the whole body dose is 1.6 Rem to both the control room and auxiliary electric equipment room. This value is less than the acceptance criterion of 5 Rem. This increase is, therefore, acceptable.

9.2.2 Main Steam Line Break Accident

The main steam line break (MSLB) accident is based on primary coolant technical specification limits which are unchanged by the power uprate. For the 5 percent power uprate, the licensee determined that the MSLB accident will release amounts of steam and reactor coolant that are bounded by the assumptions in the current updated final safety analysis report (UFSAR) (the amount of steam released with the power uprate is the same as in the UFSAR and the amount of coolant is 98.8 percent of the value assumed in the UFSAR). The closure times of the MSIVs are not changed by power uprate. Therefore, the present UFSAR dose for the MSLB remains limiting.

9.2.3 Fuel Handling Accident and Control Rod Drop Accident

Two design-basis accidents that involve fuel rod failure were reassessed for the power uprate by power level scaling. The two design-basis accidents involved are the fuel handling accident (FHA) and control rod drop accident (CRDA). The whole body and thyroid doses for these accidents were scaled based on power level comparisons for the exclusion area boundary. The doses resulting from the accidents analyzed are compared with applicable dose limits. The plant specific results for power uprate remain below established regulatory limits.

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[1] See Calculation L-001166 included as Attachment C of Letter from F. Dacimo (ComEd) to U.S. NRC, "LaSalle County Station Units 1 and 2, Supplement to Application for Amendment of Facility Operating Licenses NPF-11 and NPF-18, Appendix A, Technical Specifications, Addition of a Ventilation Filter Testing Program," dated May 1, 1998, Tables 13 and 14.

The staff notes that the CRDA UFSAR results were calculated for three cases which vary the time to trip the main condenser vacuum pump after a CRDA, but the staff review of the analyses performed is limited to case 2 (15 minute operator action). Therefore, in the review of the proposed power uprate the staff based its conclusions with regards to the CRDA only on case 2.

9.2.4 Inadvertent MSIV Closure

The model described in the current UFSAR Section 15.2.4 applies to the present evaluation of offsite dose impact for the inadvertent MSIV closure. There is no additional impact on the reactor vessel system, the steam line relief system, the primary containment structures, suppression pool structures, and offsite doses (which are controlled by station administration and procedures) due to a 5 percent power increase. An analysis performed by ComEd demonstrates that fuel thermal margin requirements are met. Therefore, inadvertent MSIV closure at 105 percent power does not result in fuel damage. Additionally, the reactor dome pressure and the design basis normal reactor coolant fission product inventories do not change. Therefore, the mass blowdown to the suppression pool and design fission product inventory in the suppression pool do not change for 105 percent power. The radioactive material releases to the environment (containment purging) and primary containment access are administratively controlled, and therefore do not change as a result of the uprate. The actual airborne activity can be analyzed to determine processing (filtering) and release rates to assure that 10 CFR Part 50, Appendix I, and 10 CFR Part 20 requirements are met.

9.2.5 Instrument Line Break (Inside Secondary Containment)

The reactor coolant instrument line break model described in the current UFSAR Section 15.6.2 applies to the present evaluation of offsite dose impact. The off-site dose is a function of the mass release and the normal reactor coolant fission product source term. Since the reactor pressure does not increase and there has been no physical change to the instrument tubing, the mass and activity release rate does not change. Thus, power uprate does not change the dose values reported in the UFSAR Section 15.6.2.

9.2.6 Feedwater Line Break (FLB)

The 5 percent power uprate is conservatively expected to increase the total feedwater mass release and the amount of feedwater that flashes by 6 percent. The mass that flashes will go from 165,000 pounds to 175,000 pounds. There is no fuel damage and the normal reactor coolant source term is not impacted by the power uprate. The offsite doses would increase by 6 percent based on the current model. The current values are less than 8.1E-09 rem whole body dose and 7.9E-07 rem thyroid dose. This small offsite dose increase will remain below one tenth of the 10 CFR Part 100 limits.

9.2.7 Radioactive Gas Waste System Leak or Failure

A power uprate to 105 percent of the current licensed core thermal power has no impact on the radioactive gas waste system releases. The existing design basis concentrations of fission products in the reactor coolant are conservative and remain applicable. Therefore, the radioactive gas source term is not dependent on the reactor power level. The waste gas activity accumulation will be controlled in the same manner following a power uprate to 105 percent of the current licensed core thermal power to 3489 MWt. In summary, a power uprate to 105 percent of the current licensed core thermal power will have no impact on the offsite doses due to a radioactive gas waste system leak or failure as described in the current UFSAR Section 15.7.1.

9.2.8 <u>Postulated Radioactive Releases Due to Liquid Radwaste Tank Failure</u>

The radwaste equipment's radioactive inventory is a combination of the collection volume, the source stream(s) supply rates, and the activity in the source streams. For liquid radwaste following the 5 percent power increase, all these parameters will remain unchanged. The increase in the volume of processed condensate (which actually decreases the concentrations in the condensate) will increase the frequency of the resin processing, but it will not affect the physical operation (pumping rates, tank volume, and batch processing size). Equipment leakage, process sampling, decontamination operation, and housekeeping water requirements are not expected to change. Airborne radioactivity would still be processed by an HVAC system and would be mixed in the main stack before being released to the environment. Therefore, the environmental impact of a full waste concentrate tank rupture will not change. In summary, the 5 percent power uprate will not have any impact on the postulated radioactive releases due to liquid radwaste tank failure described in the current UFSAR Section 15.7.3.

9.2.9 <u>Refueling Cask Accident (RCA)</u>

The refueling cask accident, which involves fuel rod failure, was reassessed for the power uprate by power level scaling. ComEd performed this review at an actual uprated power level (3489 MWt) which did not account for the 2 percent uncertainty described in Regulatory Guide 1.49, "Power Levels for Nuclear Power Plants." A review of the UFSAR shows that no design basis analysis was performed for this accident. An accident analysis entitled "Conservative Engineering Assumptions" was performed. Given the low consequences of this accident and the assumed decay of 360 days, use of the actual uprated power is acceptable. The whole body and thyroid doses for this accident were scaled based on power level comparisons for the exclusion area boundary. The doses resulting from the accidents analyzed were compared with applicable dose limits. The plant specific results for power uprate remain below established regulatory limits.

9.2.10 <u>Summary</u>

Based on the information above, and information provided by the licensee, the staff finds reasonable assurance that the consequences of postulated DBA accidents are not adversely affected by the power uprate. The accidents affected by this change are the LOCA, FHA, CRDA, FLB and the RCA. The postulated radiological consequences of the impacted accidents at LaSalle will be less than the dose guidelines of 10 CFR Part 100 and the criteria of 10 CFR Part 50, Appendix A, General Design Criterion 19, and Sections 6.4 (LOCA, FHA, CRDA, FLB, RCA), 15.6.5 (LOCA), 15.7.4 (FHA), 15.4.9 (CRDA), and 15.7.5 (RCA) of NUREG-0800 and are, therefore, acceptable.

9.3 Special Events

9.3.1 Anticipated Transients Without Scram (ATWS)

A generic evaluation of the ATWS event is presented in Section 3.7 of Supplement 1 to Reference 2. This evaluation concludes that the ATWS acceptance criteria for fuel, reactor pressure vessel (RPV), and containment integrity will not be violated for the power uprate if the following conditions are met: reactor power increase is equal to or less than 5 percent; dome pressure increase is equal to or less than 40 psi; SRV opening setpoint increase is equal to or less than 80 psi; and ATWS high pressure recirculation pump trip (RPT) value increases equal to or less than 20 psi. The results of the ATWS analyses for LaSalle meet all of the four criteria. The licensee evaluated the limiting ATWS event, the MSIV closure. RPV integrity was reanalyzed at the uprated 3489 MW core thermal power. The results were submitted in the licensee's letter dated January 21, 2000 which showed the peak RPV pressure to be 1477 psig, which is below the ASME code service level C limit of 1500 psig. The peak suppression pool temperature is 204° F, which is less than the 212° F limit. These results are acceptable.

9.3.2 Station Blackout

The licensee stated that the plant response and coping capabilities for a station blackout (SBO) event are impacted slightly by plant operations at the proposed uprated power level due to the increase in the operating temperature of the primary coolant system, decay heat, and main steam safety/relief valve set points. The licensee analyzed the impact of these increases on the condensate water requirement and the temperature heat-up in the areas which contain equipment necessary to mitigate the SBO event, and concluded that no changes to the systems and equipment used to cope with an SBO event are required. However, as discussed in Section 4.1.1.1(2), the licensee will impose procedural controls within the EOPs to limit the RPV cooldown rate during an SBO. The staff has found this approach to be acceptable.

Based on our review of the licensee's evaluation and review of power uprate applications for similar BWR plants, we find that there is no significant impact on coping with an SBO event due to plant operations at the proposed uprated power level.

10.0 Additional Aspects of Power Uprate

10.1 High Energy Line Breaks

Temperature, Pressure and Humidity Profiles

Plant operations at the proposed uprated power level will cause a small increase in the mass and energy release rates following a high energy line break (HELB) outside the primary containment. This results in a small increase in the subcompartment pressure and temperature profiles. The licensee performed a HELB analysis for all systems (e.g., main steam system, feedwater system, reactor core isolation cooling system, etc.) evaluated in the Updated Safety Analysis Report (USAR) and stated that the resulting pressure and temperature profiles resulting from plant operations at the proposed uprate power level are bounded by the existing profiles due to the conservatism in the original HELB analysis.

Based on our review of the licensee's evaluation and review of power uprate applications for similar BWR plants, we agree with the licensee that the existing analysis for HELB remains bounding and is acceptable for plant operations at the proposed uprated power level.

10.2 Moderate Energy Line Break (MELB)

The licensee determined that uprated power level operation has no impact on the moderate energy line break. Based on a review of the emergency core cooling system, the reactor core isolation cooling system, the service water system, the RHR system, and the fire protection system, the licensee concluded that the original moderate energy line break analysis is not affected by plant operation at the uprated power level.

Based on our review of the licensee's evaluation and review of power uprate applications for similar BWR plants, we agree with the licensee that the existing analyses for MELB are not impacted by plant operations at the proposed uprated power level.

10.3 Equipment Qualification

10.3.1 Electrical Equipment

The licensee evaluated the safety-related electrical equipment to ensure qualification for the normal and accident conditions expected in the areas in which the devices are located.

10.3.1.1 Inside Primary Containment

Environmental qualification (EQ) for safety-related electrical equipment located inside the primary containment is based on a steam line break and/or design basis accident loss-of-coolant accident (LOCA) conditions and their resultant temperature, pressure, humidity, and radiation consequences and includes the environments expected to exist during normal plant operation. The licensee evaluated the EQ for safety-related electrical equipment located inside the primary containment and determined that the current accident and normal plant conditions for temperature, pressure, and humidity inside containment are nearly unchanged for the power uprate conditions.

The staff requested the licensee to describe the changes to accident and normal temperature, pressure, and humidity profiles inside the containment and why these changes have no impact on the EQ of safety-related electrical equipment. In response to the staff's request, the licensee stated that the normal operating pressure condition in the primary containment does not change for the power uprate. The drywell operating pressure is controlled between -0.5 and 0.75 psig during power operation. The drywell heat load is increased by approximately 0.1 percent, thus, the area temperature increase in the drywell is negligible. The two factors that affect relative humidity under normal operating conditions are temperature and leakage. Since the normal operating temperature increase in the drywell is negligible with the power uprate, and leakage into the drywell is not affected, it is concluded that drywell humidity remains within the band of 40-55 percent. Following an accident, relative humidity increases to 100 percent for the pre-uprate condition. Since this is the maximum value for relative humidity, there is no change with the power uprate.

The licensee also stated that the peak post-LOCA primary containment pressure and temperature are expected to increase following a power uprate. The licensee indicated that the uprate temperature and pressure (post-LOCA) remain less than the electrical equipment qualification temperatures and pressures for safety-related equipment located in the primary containment.

The licensee determined that current EQ radiation levels under normal plant conditions are nearly unchanged for the uprated conditions, and the accident conditions are conservatively evaluated to increase 16 percent. A 16 percent increase represents the largest increase possible regardless of plant configuration and barriers to radiological transport from the containment. A 10 percent increase is applied to zones H4A though H4H on the basis of a more realistic evaluation of the standby gas treatment system and the surrounding area. The increases do not adversely affect the bounding environmental conditions currently in the UFSAR. The existing normal operation design basis source term bound the power uprate conditions.

In summary, the power uprate has a negligible effect on normal plant operating environmental conditions and has no significant effect on the environmental conditions currently used for the safety-related electrical equipment EQ program inside the primary containment.

10.3.1.2 <u>Outside Primary Containment</u>

Accident temperature, pressure, and humidity environments used for qualification of equipment outside primary containment result from a main steamline break in the pipe tunnel, or other high-energy line breaks (HELBs). The accident temperature, pressure, and humidity conditions resulting from a LOCA do not change with power level, but some of the HELB profiles increase by a small amount.

LaSalle is designed to withstand the effects of postulated pipe breaks and leakage cracks, including pipe whip, jet impingement, and reaction forces, for a power uprate to 105 percent. The design bases for pipe whip restraints, equipment shields, interior flood control, HELB pressurization, and environmental analyses have sufficient margin to accommodate changes to system parameters as a result of power uprate. The increase in the blowdown rate is insignificant and the resulting profiles are bounded by the existing profiles because of conservatism in the original analyses. The licensee performed an evaluation which shows that the system and components required to mitigate the postulated HELB events are designed to withstand the resulting pressure and thermal loading following an HELB. On the basis of the analysis performed at uprated power, the mass and energy releases resulting from pipe breaks outside primary containment are bounded by the original analysis. Because the mass and energy releases are bounded, there is no increase in the environmental parameters that would effect equipment operability. Thus, all equipment remains qualified.

The current EQ radiation levels under normal plant conditions are unchanged, and accident conditions are conservatively evaluated to increase 16 percent for all zones, except zones H4A through H4H, the accident conditions are conservatively evaluated to increase 10 percent. These increases do not effect the bounding equipment environmental conditions.

In summary, the power uprate has a negligible effect on the environmental conditions currently used for the safety-related electrical equipment EQ program outside the primary containment.

10.3.2 Mechanical Equipment

The licensee evaluated equipment qualification for the power uprate condition. The dynamic loads such as SRV discharge and LOCA loads (including pool swell, condensation oscillation, and chugging loads) that were used in the equipment design will remain unchanged. This is because the plant-specific hydrodynamic loads that are based on the range of test conditions for the design-basis analysis at LaSalle are bounding for the power uprate.

Based on its review of the proposed power uprate amendment, the staff finds that the original seismic and dynamic qualification of the safety-related mechanical equipment is not affected by the power uprate conditions for the following reasons:

1. Seismic loads are unchanged for the power uprate;

2. No new pipe break locations or pipe whip and jet impingement targets are postulated as a result of the uprated conditions;

3. Pipe whip and jet impingement loads do not increase for the power uprate; and

4. SRV and LOCA dynamic loads used in the original design-basis analyses are bounding for the power uprate.

In response to the staff's Request for Additional Information (RAI), the licensee performed an evaluation of the effects of plant operations at the proposed uprated power level on the non-metallic components of safety-related mechanical equipment. The licensee stated that the changes for the normal and accident environmental conditions inside and outside the containment are negligible. In addition, the process temperatures and radiation effects from power uprate are within the pre-uprate design limits. Therefore, the licensee concluded that the environmental qualification of the non-metallic materials in the mechanical equipment exposed to the power uprate process conditions is not adversely impacted.

Based on our review of the licensee's evaluation and review of power uprate applications for similar BWR plants, we agree with the licensee that the existing EQ of mechanical equipment with non-metallic components remain bounding and is acceptable for plant operations at the proposed uprated power level.

10.4 Required Testing

10 CFR Part 50, Appendix J

The licensee proposed to change the value of the calculated peak containment pressure (Pa) from 39.6 psig to 39.9 psig used in the 10 CFR Part 50, Appendix J, test program to reflect the new calculated peak containment pressure resulting from plant operation at the uprated power level.

Based on our review of the licensee's evaluation, we find the licensee's proposal to change the value of Pa from 39.6 psig to 39.9 psig used in the 10 CFR Part 50, Appendix J, test program, to reflect the new calculated peak containment pressure resulting from plant operation at the uprated power level, is acceptable.

10.5 Operator Training and Human Factors

The staff reviewed information provided by the licensee relative to the effects of the power uprate on operator licensing and human performance. Five topics were reviewed; the staff's evaluation of each is provided below.

10.5.1 Effect of Power Uprate on the Type and Scope of Plant Emergency and Abnormal Operating Procedures

The licensee stated in its letter dated July 14, 1999 that, "The plant Emergency Operating Procedures (EOPs) will be reviewed for any effects of power uprate, and the EOPs will be updated, as necessary." The review will be based on Section 2.3 of NEDC-31984P (Reference 2).

The licensee further states that,

> For uprated power conditions, operator and equipment interfaces to transient, accident and special events are not affected. Most abnormal events result in automatic plant shutdown (scram). Some abnormal events result in automatic RCPB [reactor coolant pressure boundary] pressure relief, ADS [automatic depressurization system] actuation and/or automatic ECCS [emergency core cooling system] actuation (for low water level events). Power uprate does not change any of the automatic safety functions. After the applicable automatic responses have initiated, the follow on operator actions (e.g., maintaining safe shutdown, core cooling, containment cooling, etc.) for plant safety do not change for power uprate. The emergency operating procedures are symptom-based and the slight changes in accident and transient behavior will have minimal effect.

The staff finds that the licensee's response is satisfactory.

10.5.2 Effect of Power Uprate on Operator Reliability or Performance or Operator Response Times

The licensee stated, in its letter dated April 7, 2000, that the EOPs for LaSalle are symptom based and that as symptoms exceed the established limits, mitigative actions are taken as prescribed by the procedures. The licensee states:

> Power Uprate effects on the LaSalle Probabilistic Safety Analysis (PSA) indicate slight reductions in the available times for several operator actions. Those operator actions that have a measurable impact are Standby Liquid Control (SBLC) initiation during and Anticipated Transient without Scram (ATWS) event, reduction of power by Reactor Pressure Vessel (RPV) level control during an ATWS event, and alignment of the Fire Protection System as an alternate injection system during an internal flooding event.

> However, the reductions in time for these three operator actions are small and do not affect the results of the current PSA calculation.
>
> In addition, no new operator actions will be required as a result of the proposed power uprate.

The staff finds that the licensee's response is satisfactory.

10.5.3 Effect of Power Uprate on Control Room Alarms, Controls, and Displays

The licensee indicated, in a teleconference with the staff on February 16, 2000, that changes to control room instrumentation that are a result of the proposed power uprate are limited to scales or banding and instrument setpoint changes (e.g., main steam line high flow isolation setpoints). One alarm response procedure also will be affected by the power uprate. The proposed power uprate will not change any of the automatic safety functions.

The staff finds that the licensee's response is satisfactory.

10.5.4 Effect of Power Uprate on the Safety Parameter Display System (SPDS)

The licensee stated in its letter dated July 14, 1999, that plant EOPs will be reviewed for any effects of power uprate and will be revised accordingly. The licensee indicated that the review would be based on Section 2.3 of NEDC-31984P (reference 2). In accordance with NEDO-31984,

> Changing some of the variables and limit curves [as a result of changing the rated reactor power] will require modifying the values in the EOPs and updating utility supporting documentation. EOP curves and limits may also be included in the safety parameter display system. It must also be updated accordingly.

This commitment is acceptable to the staff.

10.5.5 Effect of Power Uprate on the Operator Training Program and Plant Simulator

The licensee stated in its letter dated July 14, 1999 that,

> Additional training required to operate the plant in an uprated condition is expected to be minimal. The changes to the plant have been identified and the operator training program is being evaluated to determine the specific changes required for operator training. This evaluation includes the effect on the plant simulator.

The licensee indicated that, "when applicable, the results from the uprate test program will be used to revise the operator training program to reflect the effects of the uprated conditions."

In a teleconference with the staff on February 16, 2000, the licensee further indicated that the LaSalle simulator currently models the L1C09 core design, evaluated in the cycle specific analysis for uprate conditions, and complies with the requirements of ANSI/ANS 3.5, "Nuclear

Power Plant Simulators for Use in the Operator Training and Examination." The licensee stated that turbine performance will be updated to predicted heat balance data prior to implementation of the power uprate. Upon completion of the uprate power ascension, the simulator will be compared to actual plant performance to confirm compliance with ANSI/ANS 3.5 modeling requirements.

The staff finds that the licensee's response is satisfactory and consistent with the existing simulation facility certification.

10.5.6 Conclusion

For the reasons stated above, the staff concludes that the previously discussed review topics associated with the proposed LaSalle County Station, Units 1 and 2, uprate have been satisfactorily addressed. The staff further concludes that the power uprate should not adversely affect simulation facility fidelity, operator performance, or operator reliability.

11.0 Maine Yankee Lessons Learned

The LaSalle County Station power uprate amendments were reviewed with regard to the recommendations from the Report of the Maine Yankee Lessons Learned Task Group, dated December 5, 1996. This report is documented in SECY-97-042, "Response to OIG Event Inquiry 96-04S Regarding Maine Yankee," dated February 18, 1997.

The staff requested that the licensee identify all codes/methodologies used to obtain safety limits and operating limits and how they verified that these limits were correct for the appropriate uprate core. The licensee was also requested to identify and discuss any limitations associated with these codes/methodologies that may have been imposed by the staff. In the letter dated January 21, 2000, the licensee responded to the staff request and identified the codes/methodologies used for the power uprate analyses and confirmed that the models/methodologies are used appropriately for the power uprate evaluation.

The letter dated January 21, 2000 also confirmed that the licensee audited GE to assure that the codes are used correctly by GE for power uprate conditions and the limitations and restrictions were followed appropriately by GE.

The main findings centered around the use and applicability of the code methodologies used to support the uprated power. The licensee has verified that the codes are appropriate and applicable to the plant given the uprated conditions. The licensee indicated that the LOCA and transients analyses conform with the generic analyses approved by the staff for power uprate. This is acceptable.

III. CHANGES TO THE TECHNICAL SPECIFICATIONS

Based on the considerations discussed above, the increase in rated thermal power from 3323 MWt to 3489 MWt is acceptable. This change will be reflected in condition C.1 of the Unit 1 and Unit 2 operating licenses. In addition, Section 1.1, "Definitions" will be revised to reflect the change in rated thermal power.

In addition, the increase in rated thermal power requires that the following setpoints be revised:

1. TS Table 2.2.1-1, Functions 2.b(1)(a), "APRM Flow Biased Simulated Thermal Power - High Scram" for two loop operation, Function 2.b.(2)(a), "APRM Flow Biased Simulated Thermal Power - High Scram for Single Loop Operation, and Function 2.b.(2)(b)," "APRM Flow Biased Simulated Thermal Power - High Flow Clamped" will be revised.

2. TS Table 3.3.6-2

   Function 2.a.(1), APRM Flow Biased Simulated Thermal Power - Upscale Control Rod Withdrawal Block for two loop operation. The trip setpoint will be revised from 0.58W + 47 percent to 0.62W + 52.3 percent and the allowable value will be changed from 0.58W + 50 percent to 0.62W + 57.9 percent.

   Function 2.a.(2), APRM Flow Biased Simulated Thermal Power - Upscale Control Rod Withdrawal Block for single loop operation. The trip setpoint will be revised rom 0.58W + 42.3 percent to 0.55W + 40.0 percent and the allowable value will be changed from 0.58W + 45.3 percent to 0.55W + 45.4 percent.

3. TS Table 3.3.2-2, Function A.1.c.(3), Automatic Initiation - Primary Containment Isolation - Main Steam Line - Flow - High. The trip setpoint will be changed from 111 psid to 125 psid and the allowable value will be changed from 116 psid to 128 psid.

4. TS Section 3.6.1.6, Drywell and Suppression Chamber Internal Pressure will be changed from +2.0 psig to +0.75 psig.

   During the course of the power uprate review, a non-conservative value was noted in the TS Section 3.6.1.6, "Drywell and Suppression Chamber Internal Pressure." The licensee proposed to change the upper limit for drywell and suppression chamber internal pressure to reflect the input assumptions of the accident analysis.

   The current short-term containment pressure response analysis assumed a drywell pressure, as an initial condition, of 0.75 psig. The corresponding TS limit is 2.0 psig. The licensee's practice has been to take action when containment pressure reaches 0.3 psig, and the units have not operated at a pressure greater than 0.75 psig in the containment. The analysis for power uprate assumed an initial pressure of 0.75 for determining containment peak pressure. Therefore, the licensee proposed revising TS 3.6.1.6 to change the upper limit from 2.0 to 0.75. This change is consistent with the assumptions used in the analysis and is acceptable.

5. TS Section 6.2.F.7, Primary Containment Leakage Rate Testing Program. The peak post accident containment pressure will be changed from 39.6 psig to 39.9 psig. This value was changed as a result of the short-term containment response analysis conducted at power uprate conditions.

6. Bases changes

The licensee proposed changes to TS Bases Sections B 3/4.6.1.6, 3/4.6.2, and Table B3/4.4.6-1 to reflect power uprated conditions.

IV. STATE CONSULTATION

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

V. ENVIRONMENTAL CONSIDERATION

Pursuant to 10 CFR 51.21, 51.32 and 51.35 an Environmental Assessment and Finding of No Significant Impact was published in the Federal Register on April 21, 2000 (65 FR 21491). Accordingly, based on the Environmental Assessment, the Commission has determined that issuance of the amendments will not have a significant effect on the quality or the human environment.

VI. CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributors:
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VII. REFERENCES

(1) GE Nuclear Energy, "Generic Guidelines For General Electric Boiling Water Reactor Power Uprate," Licensing Topical Report NEDO-31897, Class I (non-proprietary), February 1992; and NEDC-31897P-A, Class III (Proprietary), May 1992.

(2) GE Nuclear Energy, "Generic Evaluations of General Electric Boiling Water Reactor Power Uprate," Licensing Topical Report NEDC-31984P, Class III (Proprietary), July 1991; NEDO-31984, Class I (Non-proprietary), March 1992; and Supplements 1 and 2.

(3) GE Nuclear Energy, "Power Uprate Safety Analysis for the LaSalle County Station Units 1 & 2," Licensing Topical Report NEDC-32701P, Class III (Proprietary), July 1999.

(4) GE Nuclear Energy, LaSalle County Station Units 1, and 2 , SAFER/GESTR-LOCA Loss-of-Coolant Accident Analysis (Revision 1), NEDC-32258P, October 1998.

(5) GE Nuclear Energy, "Generic Evaluations of General Electric Boiling Water Reactor Extended Power Uprate," Licensing Topical Report NEDC-32523P, Supplement 1, Volume II, June 1996; and NRC Safety Evaluation, July 31, 1992.

(6) SECY-91-401, "Generic Boiling Water Reactor Power Uprate Program," December 12, 1991.