ATTACHMENT B
Proposed Change to Operating License and Technical Specifications for
LaSalle County Station, Units 1 and 2

MARKED-UP TS PAGES FOR PROPOSED CHANGES

REVISED PAGES

Am. 52
12/08/87

(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) <u>Maximum Power Level</u>

Am. 4
8/13/82

The licensee is authorized to operate the facility at core power levels not in excess of full power (~~3323~~ megawatts thermal).
3489

Am. 129
07/06/98

(2) <u>Technical Specifications and Environmental Protection Plan</u>

The Technical Specifications contained in Appendix A, as revised through **Amendment No. 129** 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) <u>Conduct of Work Activities During Fuel Load and Initial Startup</u>

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 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.

(4) <u>Resolution of Rebar Damage and Adequacy of Off-Gas Building Roof</u>

The licensee shall complete its assessment of the rebar damaged due to drilling and coring in concrete and the structural adequacy of the off-gas building roof. The results shall be reported to the NRC staff for review and approval, prior to power operation following initial criticality and zero power physics testing.

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 O-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 ~~3323~~ 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% ≤ ~~0.58W + 59%~~ with a maximum of	0.62W + 69.3% ≤ ~~0.58W + 62%~~ 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% ≤ ~~0.58W + 54.3%~~ with a maximum of	0.55W + 56.8% ≤ ~~0.58W + 57.3%~~ with a maximum of
b) High Flow Clamped	≤ 113.5% of RATED THERMAL POWER 108.1%	≤ 115.5% of RATED THERMAL POWER 112.3%
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	≤ ~~111 psid~~ 125 psid	≤ ~~116 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	0.62W + 52.3% ≤ ~~0.58 W + 47%~~	0.62W + 57.9% ≤ ~~0.58 W + 50%~~*
2) Single Recirculation Loop Operation	0.55W + 40.0% ≤ ~~0.58 W + 42.3%~~	0.55W + 45.4% ≤ ~~0.58 W + 45.3%~~*
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

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 ~~+2.0~~ 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.

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 39.6 psig.
39.9

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 ΔRT_{NDT} (°F)	MIN. UPPER SHELF (ft-lb)	MAX. EOL RT_{NDT}
Plate	SA-533,Gr.B,Cl.1	C5978-2	0.59	0.11	0.010	+23	~~30**~~ 18	~~110~~ 70	+59
Plate	SA-533,Gr.B,Cl.1	C6345-2	0.51	0.15	0.012	~~-35**~~ -20	~~49~~ 25	~~153~~ 92	~~+72~~ 30.5
Weld	3-308-A,B,C	IP3571/~~3978~~ 3958	0.15	0.37	0.017	-30	~~124~~ 58	~~***~~ 57.5	~~+94~~ 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,Cl.1	C6003-2	+12
Bottom Head Dollar Plate	SA-533,Gr.B,Cl.1	C6003-3	+58
Bottom Head Radial Plates	SA-533,Gr.B,Cl.1	C5328-1	+10
Top Head Dollar Plate	SA-533,Gr.B,Cl.1	C7343-1	-10
Top Head Side Plates	SA-533,Gr.B,Cl.1	C7376-2	-10
Top Head Flange	SA-508,Cl.2	ACT-USS-4P	+20
Vessel Flange	SA-508,Cl.2	2V-659ATF-112	+20
Feedwater Nozzle	SA-508,Cl.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)

~~* Combination of the higest starting RT_{NDT} plate and the highest ΔRT_{NDT} plate.~~ e

~~** These values are given only for the benefit of calculating the end-of-life (EOL) RTNDT.~~ e

~~***Not available.~~ e

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

39.9

The limitation on drywell and suppression chamber internal pressure ensure that the containment peak pressure of ~~39.6~~ 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 2.0~~ psig for initial positive primary containment pressure will limit the total pressure to ~~39.6~~ psig which is less than the design pressure and is consistent with the accident analysis. 39.9

0.75

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 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 peak bulk temperature of the suppression pool is maintained below 200°F during any period of relief valve operation with sonic conditions at the discharge exit for T-quencher devices. Specifications have been placed on the envelope of reactor operating conditions so that the reactor can be depressurized in a timely manner to avoid the regime of potentially high suppression chamber loadings.~~

REPLACE WITH INSERT A

INSERT A

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.

C. The license shall be deemed to contain and is subject to the conditions specified in the Commission's regulations set forth in 10 CFR Chapter I and is subject to all applicable provisions of the Act and to the rules, regulations, and others 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 (3323 megawatts thermal). Items in Attachment 1 shall be completed as specified. Attachment 1 is hereby incorporated into this license.

3489

Am. 116 (2) Technical Specifications and Environmental Protection Plan

03/16/99

The Technical Specifications contained in Appendix A, as revised through **Amendment No. 116** and the Environmental Protection Plan contained in Appendix B, are hereby incorporated into 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 actvities 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, knowledgable in the above area, and who meet the qualification 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 O-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 3323 MWT. 3489

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 [illegible].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	≤ ~~0.58W + 59%~~ 0.62W + 63.7% with a maximum of	≤ ~~0.58W + 62%~~ 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	≤ ~~0.58W + 54.3%~~ 0.55W + 51.5% with a maximum of	≤ ~~0.58W + 57.3%~~ 0.55W + 56.8% with a maximum of
b) High Flow Clamped	≤ ~~113.5%~~ 108.1% of RATED THERMAL POWER	≤ ~~115.5%~~ 112.5% 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	≤ ~~111 psid~~ 125 psid	≤ ~~116 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	$\geq$ 5% of RATED THERMAL POWER	$\geq$ 3% of RATED THERMAL POWER
2. APRM		
a. Flow Biased Simulated Thermal Power-Upscale		
1) Two Recirculation Loop Operation	0.62W + 52.3% $\leq$ ~~0.58 W + 47%~~*	0.62W + 57.9% $\leq$ ~~0.58 W + 50%~~*
2) Single Recirculation Loop Operation	0.55W + 40.0% $\leq$ ~~0.58 W + 42.3%~~	0.55W + 45.4% $\leq$ ~~0.58 W + 45.3%~~*
b. Inoperative	N.A.	N.A.
c. Downscale	$\geq$ 5% of RATED THERMAL POWER	$\geq$ 3% of RATED THERMAL POWER
d. Neutron Flux-High	$\leq$ 12% of RATED THERMAL POWER	$\leq$ 14% of RATED THERMAL POWER
3. SOURCE RANGE MONITORS		
a. Detector not full in	N.A.	N.A.
b. Upscale	$\leq 2 \times 10^5$ cps	$\leq 5 \times 10^5$ cps
c. Inoperative	N.A.	N.A.
d. Downscale	$\geq$ 0.7 cps	$\geq$ 0.5 cps
4. INTERMEDIATE RANGE MONITORS		
a. Detector not full in	N.A.	N.A.
b. Upscale	$\leq$ 108/125 of full scale	$\leq$ 110/125 of full scale
c. Inoperative	N.A.	N.A.
d. Downscale	$\geq$ 5/125 of full scale	$\geq$ 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 ~~+2.0~~ psig.
+0.75

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.

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 ~~39.6 psig~~. 39.9

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*	~~13*~~ 11.5	~~29~~ 69	~~+65~~ 75.5
Plate	SA-533,Gr.B,Cl.1	C9425-1	0.51	0.12	0.009	~~+30*~~ +32*	~~28*~~ 21.5	~~39~~ 59	~~+58~~ 75
Weld	1NMW/E8018-G	3P4966/1214	0.90	0.03	~~0.010~~ 0.011	- 6*	~~17*~~ 11	~~28~~ 75.5	~~+11~~ 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	~~+20~~ +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}.

~~** Not Available.~~

3/4.6.1 PRIMARY CONTAINMENT

3/4.6.1.5 DELETED

3/4.6.1.6 DRYWELL AND SUPPRESSION CHAMBER INTERNAL PRESSURE

39.9

The limitation on drywell and suppression chamber internal pressure ensure that the containment peak pressure of 39.6 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 2.0 psig for initial positive primary containment pressure will limit the total pressure to 39.6 psig which is less than the design pressure and is consistent with the accident analysis.

39.9 0.75

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 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 peak bulk temperature of the suppression pool is maintained below 200°F during any period of relief valve operation with sonic conditions at the discharge exit for T-quencher devices. Specifications have been placed on the envelope of reactor operating conditions so that the reactor can be depressurized in a timely manner to avoid the regime of potentially high suppression chamber loadings.~~

REPLACE WITH INSERT A

INSERT A

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.

INFORMATION SUPPORTING A FINDING OF NO SIGNIFICANT HAZARDS CONSIDERATION

ComEd has evaluated the proposed change and determined that it does not involve a significant hazards consideration. According to 10 CFR 50.92(c), a proposed amendment to an operating license involves no significant hazards consideration if operation of the facility in accordance with the proposed amendment would not:

> Involve a significant increase in the probability of occurrence or consequences of an accident previously evaluated;
>
> Create the possibility of a new or different kind of accident from any previously analyzed; or
>
> Involve a significant reduction in a margin of safety.

ComEd proposes to increase rated core thermal power from 3323 MWt to 3489 MWt. The method for achieving higher reactor power is to increase core thermal power with a more uniform and flattened power distribution to create an increase in steam flow. A corresponding increase in feedwater flow will be required. In addition, the power/flow map will be extended to the Maximum Extended Load Line Limit (MELLL) domain to increase operational flexibility and minimize need for frequent reactor control rod pattern adjustments. The maximum allowable core flow rate does not change as a result of power uprate. Uprated operation will not involve increasing reactor pressure vessel (RPV) dome pressure because the plants have sufficient pressure control and turbine flow capabilities to control the inlet pressure conditions at the turbine.

The determination that the criteria set fourth in 10 CFR 50.92 are met for this amendment request is indicated below.

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

A. Evaluation of the Probability of Previously Evaluated Accidents.

The proposed power uprate imposes only minor increases in plant operating conditions. No change is made to the reactor operating pressure. Operation at uprated conditions will result in moderate flow increases in those systems associated with the turbine cycle in that steam flow increases by approximately six (6)% and feed flow increases by approximately six (6)%. The increase in flow in the carbon steel piping systems was evaluated for the effect on flow induced erosion and corrosion rates and it was confirmed that power uprate has no significant effect on flow induced erosion or corrosion. The affected systems are currently monitored by the Flow Accelerated Corrosion (FAC) program that addresses erosion and corrosion concerns. Continued monitoring of the systems provides a high level of confidence in the integrity of potentially susceptible high energy piping systems.

Plant systems and components have been verified to be capable of performing their intended design functions at uprated power conditions. Where necessary, some components will be modified prior to implementation of uprated power conditions to accommodate the revised operating conditions. The review has concluded that operation at power uprate conditions will not affect the reliability of plant equipment, and that current Technical Specifications (TS) surveillance requirements ensure adequate monitoring of system operability. Systems continue to be operated in accordance with current design requirements under uprated conditions, therefore no new components or system interactions were identified that could lead to an increase in accident probability. Changes to reactor scram setpoints are such that no significant increase in scram frequency due to operation at uprated conditions will occur.

B. Evaluation of the Consequences of Previously Evaluated Accidents.

The radiological consequences due to the Loss of Coolant Accident (LOCA) were calculated and are found to be below the applicable regulatory limits. The results are presented in Table 9-3 of Attachment E.

The LOCA radiological consequences have not significantly increased due to power uprate, and radiological consequences continue to meet established regulatory limits.

The radiological evaluations for other non-LOCA Design Basis Accidents (DBAs) were also performed and the dose consequences for these events did not significantly increase. These changes are outlined in Section 9.2 of Attachment E and they demonstrate that LaSalle County Station (LCS), Units 1 and 2 still meets the applicable regulatory limits.

Non-DBA Radiological Doses

All of the other radiological releases discussed in Updated Final Safety Analysis Report (UFSAR) are either unchanged because they are not power-dependent, or increase approximately in linear proportion to the amount of the uprate. The dose consequences for all of the non-LOCA radiological release accident events did not significantly increase, and are bounded by the "LOCA Radiological Consequences" events discussed above and were shown to meet the current dose acceptance limits. These events are discussed in Section 9.2 of Attachment E.

Therefore the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The configuration, operation and event response of the LCS, Units 1 and 2 systems, structures or components are uncharged by operation at uprated power conditions. Analysis of transient events has confirmed that the same transients remain limiting and that no transient event results in a new sequence of events that could lead to a new accident scenario.

An increase in power level will not create a new fission product release path, or result in a new fission product barrier failure mode. The current fission product barriers consisting of the reactor fuel rod cladding, the reactor coolant pressure boundary, and the containment structure remain in place. Fuel rod cladding integrity is ensured by operating within thermal, mechanical, and exposure design limits, and was confirmed for a representative core by performance of transient and accident analysis. Cycle specific analysis will continue to be performed for each fuel reload to demonstrate the compliance with the applicable transient analysis criteria and to establish the cycle specific Minimum Critical Power Ratio (MCPR) safety limit and fuel operating limits. The integrity of the reactor coolant pressure boundary was confirmed by evaluation of the bounding overpressurization event and ensuring that the corresponding pressure remained below the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code, Section III, "Rules for Construction of Nuclear Power Plant Components," overpressure protection requirements. Similarly, analysis of the primary containment structure has demonstrated under worst case design basis accident conditions that the containment structure remains below the containment design pressure.

The effect of operation at uprated conditions on plant equipment has been evaluated. No new operating mode, safety-related equipment lineup, accident scenario, or equipment failure mode was identified as a result of operating at uprated conditions. In addition, operation at power uprated conditions does not create any new sequence of events or failure modes that lead to a new type of accident. Plant modifications required to support implementation of power uprated conditions will be made to existing systems rather than by adding new systems of a different design, which might introduce new failure modes or accident sequences.

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

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

The power uprate analysis for LCS, Units 1 and 2 assures that the power dependent safety margin will be maintained by meeting the appropriate regulatory criteria as prescribed by the applicable regulations. Similarly, factors of safety specified by application of the regulatory required design rules have been maintained, as have other acceptance criteria used to judge the acceptability of current plant operation.

No change is required in the basic fuel design to achieve the uprated power levels, or to maintain current operating and safety margins. No increase in the allowable peak bundle power is requested as a result of operation at uprated conditions. The abnormal transients have been evaluated for a representative core configuration and confirmed that operation at uprated conditions does not have an adverse effect on the operating limit MCPR. No change to the Safety Limit MCPR results, thus the margin of safety as assured by the safety limit MCPR is maintained. The fuel operating limits related to heat generation rate would still be met at uprated conditions. Cycle specific analysis will continue to be performed for each fuel reload to demonstrate the compliance with the applicable transient analysis criteria and to establish the cycle specific safety limit and fuel operating limits.

The Emergency Core Cooling System (ECCS)-LOCA performance has been evaluated at power uprated conditions using methodologies that have been approved by the NRC for 10CFR50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Nuclear Power Reactors," analysis. The current ECCS performance requirements were used in the power uprate analysis. The ECCS-LOCA analysis was conducted at 102 % of the proposed uprated thermal power in accordance with regulatory guidance. The necessary analysis for operation of General Electric (GE) fuel under uprated conditions and the determination that the peak cladding temperature (PCT) remains below the 10CFR50.46 limit of 2200°F have been performed. However, LCS Unit 2 currently contains a mixed core of GE and Siemens Power Corporation (SPC) fuel. LCS obtained an TS amendment that allows operation with SPC fuel, and approved the use of the SPC analytical methodology. The ECCS-LOCA analysis performed to support use of the SPC fuel was conducted at a power level that bounds 102 % of the proposed uprated power level and determined that the PCT, for SPC fuel, remains below the 10CFR50.46 limit of 2200°F. The analysis for both GE and SPC fuel types demonstrate all 10CFR 50.46 criteria are met. Therefore, there is no reduction in margin with respect to maintaining ECCS performance.

The margin of safety of the reactor coolant pressure boundary is maintained under power uprated conditions. The design pressure of the RPV and reactor pressure coolant pressure boundary remains at 1250 psig. The ASME B&PV Code allowable peak pressure is 1375 psig (i.e., 110% of 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 valve position scram and this event results in a calculated peak RPV pressure of 1332 psig at the bottom of the RPV. The peak pressure remains below the 1375 psig ASME limit. Therefore, there is no decrease in margin of safety in the reactor coolant pressure boundary.

The margin of safety of the containment structure is maintained under power uprated conditions. The analyses were conducted using a newer NRC-reviewed methodology. The pre-uprated cases were run using the new methodology and the re-baselined cases were compared to the uprated cases. The short-term containment peak pressure analysis re-baseline result was 39.3 psig compared to the original analysis of 39.6 psig. At uprated conditions the peak containment drywell pressure would be 39.9 psig, and is below the design value of 45 psig. The long-term containment suppression pool temperature analysis re-baseline result was 190°F compared to the original analysis result of 200°F. At uprated conditions the analysis concluded that in the event of a LOCA, the calculated peak bulk suppression pool temperature would be 193°F. This is less than the design temperature of the suppression pool of 275°F, and the criteria used to ensure adequate Net Positive Suction Head (NPSH) to the ECCS pumps which is 212°F. Therefore, power uprate does not challenge the structural integrity of the containment structure and ECCS NPSH is assured.

Therefore, operation at power uprated conditions does not involve a significant reduction in the margin of safety.

Conclusion

Therefore, based upon the above evaluation, ComEd has concluded that these changes involve no significant hazards consideration.

INFORMATION SUPPORTING AN ENVIRONMENTAL ASSESSMENT

Identification of Proposed Action

LaSalle County Stations (LCS), Units 1 and 2 are currently licensed to operate at a core thermal power level of 3323 MWt. This license amendment request proposes to increase the licensed core thermal power to 3489 MWt, or 105 % of the current licensed maximum steady state thermal power limit.

Need for the Proposed Action

The proposed change allows an increase in licensed core thermal power from 3323 MWt to 3489 MWt and allows the flexibility to increase the potential electrical output of the LCS, Unit 1 and Unit 2.

Environmental Assessment of the Proposed Actions

This proposed change allowing operation at uprated conditions will not cause a significant impact on the environment and does not constitute an unreviewed environmental question. The radiological assessment of power uprate is summarized below. Details of the non-radiological assessment of the impact of power uprate are also provided.

A. Radiological Environmental Assessment

The impact to the radwaste systems due to operation at power uprated conditions was evaluated. The evaluation concluded that the operation of the radwaste systems at LCS would not be impacted by operation at uprated power conditions and the slight increase in effluents discharged would continue to meet the requirements of 10CFR 20, "Standards for Protection Against Radiation," and 10CFR50, Appendix I, "Numerical Guides for Design Objectives and Limiting Conditions for Operation to Meet the Criterion 'As Low as is Reasonably Achievable' for Radioactive Material in Light-Water-Cooled Nuclear Power Reactor Effluents." Therefore, power uprate does not have an adverse effect on the processing of radioactive effluents, and there are no significant environmental effects from radiological releases.

The potential effects of power uprate conditions on the radiation sources within the plant and the radiation levels during normal and post-accident conditions were evaluated. For normal operations it was determined that conservatism in the analyses to determine operational doses and radiation shielding requirements and the margins added to calculated doses and specific shield thickness are sufficient to accommodate any increases attributed to the five (5)% increase in rated thermal power. For post-accident conditions the resulting radiation levels were determined to be within current regulatory limits, and that there would be no effect on the plant or habitability of the control room envelope, or the Technical Support Center.

The calculated whole body and thyroid doses at the exclusion area boundary that might result from the postulated design basis Loss of Coolant Accident (LOCA) were evaluated. All offsite doses evaluated at uprated conditions remain below established regulatory limits.

B. Non-Radiological Environmental Assessment

The non-radiological environmental impacts of power uprate were reviewed based on information submitted in the Environmental Report – Operating License Stage to support original licensing of LCS, Units 1 and 2, the Final Environmental Impact Statement (i.e., NUREG-0486), the requirements of the Environmental Protection Plan and the National Pollutant Discharge Elimination System (NPDES) Permit.

The main non-radiological issues within the environmental licenses that are impacted by power uprate are cooling lake makeup, blowdown, and evaporation. A 2,058 acre perched lake provides normal cooling for the plant. Three baffle dikes are constructed within the lake to channel the flow of water and to increase the flow path for efficient heat dissipation. Blowdown, which originates in the cooler portion of the lake, is discharged into the Illinois River. A constant volume in the lake must be maintained; this volume corresponds to a lake elevation of 700 ft. Makeup water for the lake is supplied from the Illinois River, which is approximately 2 miles north of the lake at its closest point. The plant systems that require lake water are the condenser cooling system, the service water systems, and the fire protection system.

As a result of power uprate to 105% of current licensed core power, normal heat loads to the perched lake will increase primarily from an increase in heat load from the condenser and from other increased heat loads rejected by the plant service water system. Circulating water flow rate remains the same at uprated conditions because the pumps are constant speed centrifugal pumps. An increase in steam and condensate flow to the condenser will result in a corresponding increase in the net heat rejection to the cooling lake. Based on a condenser backpressure of 3.5 in. Hga, a 1°F rise in circulating water temperature is expected relative to the current temperature rise value of approximately 24 °F. This, in turn, will raise cooling lake temperature, thus increasing circulating water inlet temperature to the condenser. The lake is expected to experience a 0.4 °F increase in temperature on a long-term basis. Based on this minimal temperature rise, thermal shock to the fish population of the lake is not expected. The effect on lake evaporation, makeup, and blowdown was evaluated and found to be acceptable. The effect on cooling lake total dissolved solids was determined to remain within the ComEd administrative limit of 750 ppm.

An evaluation of the thermal discharge standards in place related to the temperature of the water in the river in the vicinity of the cooling lake blowdown was conducted. This evaluation concluded that at uprated conditions, significant margin exists between the maximum expected edge of mixing zone temperature and imposed regulatory limits. The expected increase in peak lake temperature will not approach the blowdown thermal limits.

The noise effects due to operation of the LCS, Units 1 and 2 at uprated power conditions were evaluated. Since the turbine and the reactor building supply and exhaust fans will continue to operate at current speeds and noise levels at uprated conditions, the overall noise level will not increase due to power uprate.

Conclusions

The limits and licensing basis have sufficient margin to accommodate any environmental changes due to the proposed power uprate. Based on this review, it is concluded that the uprate will have insignificant impacts on the non-radiological elements of concern. Existing Federal, State and local regulatory permits presently in effect will accommodate power uprate without modification. Impacts to air, water, and land resources will be essentially non-existent.

ComEd has evaluated this proposed operating license amendment request against the criteria for identification of licensing and regulatory actions requiring environmental assessment in accordance with 10 CFR 51.21. ComEd has determined that this proposed license amendment request meets the criteria for a categorical exclusion set forth in 10 CFR 51.22(c)(9) and as such, has determined that no irreversible consequences exist in accordance with 10 CFR 50.92(b). This determination is based on the fact that this change is being proposed as an amendment to a license issued pursuant to 10 CFR 50 that changes a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, and the amendment meets the following specific criteria.

(i) The amendment involves no significant hazards consideration.

This proposed amendment does not involve any significant hazards consideration.

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

The normal offsite doses are not significantly affected by operation at the uprated power level and will remain below the limits in 10 CFR 20 and 10 CFR 50, Appendix I, Technical Specifications, and the Off-site Dose Calculation Manual (ODCM).

(iii) There is no significant increase in individual or cumulative occupational radiation exposure.

Although expected radiation levels will increase a small amount, individual worker exposures will be maintained within acceptable limits by the site ALARA program, which is in accordance with 10CFR50 Appendix I. Therefore, there will be no significant increase in individual or cumulative occupational radiation exposure.

ATTACHMENT E
Proposed Change to Operating License and Technical Specifications for
LaSalle County Station, Units 1 and 2

GE REPORT NEDC-32701P,

POWER UPRATE SAFETY ANALYSIS REPORT
FOR
LASALLE COUNTY STATION
UNITS 1 AND 2,

JULY 1999 (PROPRIETARY)

ATTACHMENT F
Proposed Change to Operating License and Technical Specifications for
LaSalle County Station, Units 1 and 2

GE AFFIDAVIT FOR NEDC-32701P

General Electric Company

AFFIDAVIT

I, George B. Stramback, being duly sworn, depose and state as follows:

(1) I am Project Manager, Regulatory Services, General Electric Company ("GE") and have been delegated the function of reviewing the information described in paragraph (2) which is sought to be withheld, and have been authorized to apply for its withholding.

(2) The information sought to be withheld is contained in the GE proprietary report NEDC-32701P, *Power Uprate Safety Analysis Report for LaSalle County Station Units 1 and 2*, Revision 2, Class III (GE Proprietary Information), dated July 1999. This document, taken as a whole, constitutes a proprietary compilation of information, some of it also independently proprietary, prepared by the General Electric Company. The independently proprietary elements are identified by light gray shading of the text and tables or by bars marked in the margin adjacent to the specific material.

(3) In making this application for withholding of proprietary information of which it is the owner, GE relies upon the exemption from disclosure set forth in the Freedom of Information Act ("FOIA"), 5 USC Sec. 552(b)(4), and the Trade Secrets Act, 18 USC Sec. 1905, and NRC regulations 10 CFR 9.17(a)(4), 2.790(a)(4), and 2.790(d)(1) for "trade secrets and commercial or financial information obtained from a person and privileged or confidential" (Exemption 4). The material for which exemption from disclosure is here sought is all "confidential commercial information", and some portions also qualify under the narrower definition of "trade secret", within the meanings assigned to those terms for purposes of FOIA Exemption 4 in, respectively, Critical Mass Energy Project v. Nuclear Regulatory Commission, 975F2d871 (DC Cir. 1992), and Public Citizen Health Research Group v. FDA, 704F2d1280 (DC Cir. 1983).

(4) Some examples of categories of information which fit into the definition of proprietary information are:

a. Information that discloses a process, method, or apparatus, including supporting data and analyses, where prevention of its use by General Electric's competitors without license from General Electric constitutes a competitive economic advantage over other companies;

b. Information which, if used by a competitor, would reduce his expenditure of resources or improve his competitive position in the design, manufacture, shipment, installation, assurance of quality, or licensing of a similar product;

c. Information which reveals cost or price information, production capacities, budget levels, or commercial strategies of General Electric, its customers, or its suppliers;

d. Information which reveals aspects of past, present, or future General Electric customer-funded development plans and programs, of potential commercial value to General Electric;

e. Information which discloses patentable subject matter for which it may be desirable to obtain patent protection.

Both the compilation as a whole and the marked independently proprietary elements incorporated in that compilation are considered proprietary for the reason described in items (4)a. and (4)b., above.

(5) The information sought to be withheld is being submitted to NRC in confidence. That information (both the entire body of information in the form compiled in this document, and the marked individual proprietary elements) is of a sort customarily held in confidence by GE, and has, to the best of my knowledge, consistently been held in confidence by GE, has not been publicly disclosed, and is not available in public sources. All disclosures to third parties including any required transmittals to NRC, have been made, or must be made, pursuant to regulatory provisions or proprietary agreements which provide for maintenance of the information in confidence. Its initial designation as proprietary information, and the subsequent steps taken to prevent its unauthorized disclosure, are as set forth in paragraphs (6) and (7) following.

(6) Initial approval of proprietary treatment of a document is made by the manager of the originating component, the person most likely to be acquainted with the value and sensitivity of the information in relation to industry knowledge. Access to such documents within GE is limited on a "need to know" basis.

(7) The procedure for approval of external release of such a document typically requires review by the staff manager, project manager, principal scientist or other equivalent authority, by the manager of the cognizant marketing function (or his delegate), and by the Legal Operation, for technical content, competitive effect, and determination of the accuracy of the proprietary designation. Disclosures outside GE are limited to regulatory bodies, customers, and potential customers, and their agents, suppliers, and licensees, and others with a legitimate need for the information, and then only in accordance with appropriate regulatory provisions or proprietary agreements.

(8) The information identified by light gray shading of the text and tables or by bars in the margin is classified as proprietary because it contains detailed results and conclusions from these evaluations, utilizing analytical models and methods, including computer codes, which GE has developed, obtained NRC approval of, and applied to perform evaluations of transient and accident events in the GE Boiling Water Reactor ("BWR"). The development and approval of these system, component, and thermal hydraulic models and computer codes was achieved at a significant cost to GE, on the order of several million dollars.

The remainder of the information identified in paragraph (2), above, is classified as proprietary because it constitutes a confidential compilation of information, including detailed results of analytical models, methods, and processes, including computer codes, and conclusions from these applications, which represent, as a whole, an integrated process or approach which GE has developed, obtained NRC approval of, and applied to perform evaluations of the safety-significant changes necessary to demonstrate the regulatory acceptability of a given increase in licensed power output for a GE BWR. The development and approval of this overall approach was achieved at a significant additional cost to GE, in excess of a million dollars, over and above the very large cost of developing the underlying individual proprietary analyses.

To effect a change to the licensing basis of a plant requires a thorough evaluation of the impact of the change on all postulated accident and transient events, and all other regulatory requirements and commitments included in the plant's FSAR. The analytical process to perform and document these evaluations for a proposed power uprate was developed at a substantial investment in GE resources and expertise. The results from these evaluations identify those BWR systems and components, and those postulated events, which are impacted by the changes required to accommodate operation at increased power levels, and, just as importantly, those which are not so impacted, and the technical justification for not considering the latter in changing the licensing basis. The scope thus determined forms the basis for GE's offerings to support utilities in both performing analyses and providing licensing consulting services. Clearly, the scope and magnitude of effort of any attempt by a competitor to effect a similar licensing change can be narrowed considerably based upon these results. Having invested in the initial evaluations and developed the solution strategy and process described in the subject document GE derives an important competitive advantage in selling and performing these services. However, the mere knowledge of the impact on each system and component reveals the process, and provides a guide to the solution strategy.

(9) Public disclosure of the information sought to be withheld is likely to cause substantial harm to GE's competitive position and foreclose or reduce the availability of profit-making opportunities. The information is part of GE's comprehensive BWR technology base, and its commercial value extends beyond the original

development cost. The value of the technology base goes beyond the extensive physical database and analytical methodology and includes development of the expertise to determine and apply the appropriate evaluation process. In addition, the technology base includes the value derived from providing analyses done with NRC-approved methods, including justifications for not including certain analyses in applications to change the licensing basis.

GE's competitive advantage will be lost if its competitors are able to use the results of the GE experience to avoid fruitless avenues, or to normalize or verify their own process, or to claim an equivalent understanding by demonstrating that they can arrive at the same or similar conclusions. In particular, the specific areas addressed by any document and submittal to support a change in the safety or licensing bases of the plant will clearly reveal those areas where detailed evaluations must be performed and specific analyses revised, and also, by omission, reveal those areas not so affected.

While some of the underlying analyses, and some of the gross structure of the process, may at various times have been publicly revealed, enough of both the analyses and the detailed structural framework of the process have been held in confidence that this information, in this compiled form, continues to have great competitive value to GE. This value would be lost if the information as a whole, in the context and level of detail provided in the subject GE document, were to be disclosed to the public. Making such information available to competitors without their having been required to undertake a similar expenditure of resources, including that required to determine the areas that are not affected by a power uprate and are therefore blind alleys, would unfairly provide competitors with a windfall, and deprive GE of the opportunity to exercise its competitive advantage to seek an adequate return on its large investment in developing its analytical process.

STATE OF CALIFORNIA)
) ss:
COUNTY OF SANTA CLARA)

George B. Stramback, being duly sworn, deposes and says:

That he has read the foregoing affidavit and the matters stated therein are true and correct to the best of his knowledge, information, and belief.

Executed at San Jose, California, this 12th day of July 1999.

George B. Stramback

George B. Stramback
General Electric Company

Subscribed and sworn before me this 12 day of July 1999.

Vicky D. Schroer

Notary Public, State of California

VICKY D. SCHROER
Commission # 1224251
Notary Public - California
Santa Clara County
My Comm. Expires Jun 12, 2003