August 31, 1988

Docket Nos. 50-373 and 50-374

DISTRIBUTION:

Docket file	ACRS (10)
PDIII-2 r/f	GPA/PA
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PDIII-2 Plant File	LLuther
PShemanski	OGC-Rockville
TBarnhart (8)	EJordan
DHagan	BGrimes
ARM/LFMB	EButcher

Mr. Henry Bliss
Nuclear Licensing Manager
Commonwealth Edison Company
P.O. Box 767
Chicago, IL 60690

Dear Mr. Bliss:

SUBJECT: ISSUANCE OF AMENDMENT NOS. AND TO FACILITY OPERATING LICENSES NPF-11 AND NO. NPF-18 - LASALLE COUNTY STATION, UNITS 1 AND 2 (TAC NOS. 66061 AND 66062)

The U.S. Nuclear Regulatory Commission has issued the enclosed Amendment No. to Facility Operating License No. NPF-11 and Amendment No. to Facility Operating License NPF-18 for the LaSalle County Station, Units 1 and 2. These amendments are in response to your letter dated April 29, 1987.

The amendments revise the LaSalle County Station, Units 1 and 2 Technical Specifications to correct an inconsistency between requirements regarding the suppression pool high level alarm.

A copy of the related Safety Evaluation supporting Amendment No. to Facility Operating License No. NPF-11 and Amendment No. to Facility Operating License No. NPF-18 is enclosed. The Notice of Issuance will be included in the Commission's biweekly Federal Register notice.

Sincerely,

Paul Shemanski

Paul Shemanski, Project Manager
Project Directorate III-2
Division of Reactor Projects - III
IV, V and Special Projects

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

cc w/enclosure:
See next page

PDII-2 P.S.	PDIII-2	OCG-Rockville	PDIII-2
PShemanski:km	LLuther	SH Lewis	WForney
8/16/88	8/16/88	8/24/88	8/16/88
		No legal objection, subject to modifications noted to SE.	

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P PNU

All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N.W., and at the Public Library of Illinois Valley Community College, Rural Route No. 1, Oglesby, Illinois 61348. A copy of items (2), and (3) may be obtained upon request addressed to the U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attention: Director, Division of Reactor Projects.

Dated at Rockville, Maryland this 31st day of August 1988.

FOR THE NUCLEAR REGULATORY COMMISSION

/S/

Daniel R. Muller, Director
Project Directorate III-2
Division of Reactor Projects - III,
IV, V and Special Projects

*See previous concurrence

PDIII-2	PDIII-2	*OGC-Rockville	PDIII-2
*PShemanski:km	*LLuther		*WForney
8/16/88	8/16/88	8/24/88	8/16/88

7590-01

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Dated at Rockville, Maryland this day of

FOR THE NUCLEAR REGULATORY COMMISSION

William L. Forney, Acting Director
Project Directorate III-2
Division of Reactor Projects - III,
IV, V and Special Projects

P.S.
PDIII-2
PShemanski:km
8/16/88

PDIII-2
LLuther
8/16/88

OGC-Rockville
SHLewis
8/24/88
No legal objection subject to modifications as noted.

PDIII-2
WForney
8/16/88

August 31, 1988

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Mr. Henry Bliss
Nuclear Licensing Manager
Commonwealth Edison Company
P.O. Box 767
Chicago, IL 60690

Dear Mr. Bliss:

SUBJECT: ISSUANCE OF AMENDMENT NOS. AND TO FACILITY OPERATING LICENSES NPF-11 AND NO. NPF-18 - LASALLE COUNTY STATION, UNITS 1 AND 2 (TAC NOS. 66061 AND 66062)

The U.S. Nuclear Regulatory Commission has issued the enclosed Amendment No. to Facility Operating License No. NPF-11 and Amendment No. to Facility Operating License NPF-18 for the LaSalle County Station, Units 1 and 2. These amendments are in response to your letter dated April 29, 1987.

The amendments revise the LaSalle County Station, Units 1 and 2 Technical Specifications to correct an inconsistency between requirements regarding the suppression pool high level alarm.

A copy of the related Safety Evaluation supporting Amendment No. to Facility Operating License No. NPF-11 and Amendment No. to Facility Operating License No. NPF-18 is enclosed. The Notice of Issuance will be included in the Commission's biweekly Federal Register notice.

Sincerely,

Paul Shemanski

Paul Shemanski, Project Manager
Project Directorate III-2
Division of Reactor Projects - III
IV, V and Special Projects

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

cc w/enclosure:
See next page

PDII-2 P.S.	PDIII-2	OCG-Rockville	PDIII-2
PShemanski:km	LLuther	SHLewis	WForney
8/16/88	8/16/88	8/24/88	8/16/88

No legal objection, subject to modifications noted to SE.

All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N.W., and at the Public Library of Illinois Valley Community College, Rural Route No. 1, Oglesby, Illinois 61348. A copy of items (2), and (3) may be obtained upon request addressed to the U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attention: Director, Division of Reactor Projects.

Dated at Rockville, Maryland this 31st day of August 1988.

FOR THE NUCLEAR REGULATORY COMMISSION

/S/

Daniel R. Muller, Director
Project Directorate III-2
Division of Reactor Projects - III,
IV, V and Special Projects

*See previous concurrence

PDIII-2	PDIII-2	*OGC-Rockville	PDIII-2
*PShemanski:km	*LLuther		*WForney
8/16/88	8/16/88	8/24/88	8/16/88

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

August 31, 1988

Docket Nos. 50-373 and 50-374

Mr. Henry Bliss
Nuclear Licensing Manager
Commonwealth Edison Company
P.O. Box 767
Chicago, IL 60690

Dear Mr. Bliss:

SUBJECT: ISSUANCE OF AMENDMENT NOS. 59 AND39 TO FACILITY OPERATING LICENSES NPF-11 AND NO. NPF-18 - LASALLE COUNTY STATION, UNITS 1 AND 2 (TAC NOS. 66061 AND 66062)

The U.S. Nuclear Regulatory Commission has issued the enclosed Amendment No. 59 to Facility Operating License No. NPF-11 and Amendment No. 39 to Facility Operating License NPF-18 for the LaSalle County Station, Units 1 and 2. These amendments are in response to your letter dated April 29, 1987.

The amendments revise the LaSalle County Station, Units 1 and 2 Technical Specifications to correct an inconsistency between requirements regarding the suppression pool high level alarm.

A copy of the related Safety Evaluation supporting Amendment No. 59 to Facility Operating License No. NPF-11 and Amendment No. 39 to Facility Operating License No. NPF-18 is enclosed. The Notice of Issuance will be included in the Commission's biweekly Federal Register notice.

Sincerely,

Paul Shemanski

Paul Shemanski, Project Manager
Project Directorate III-2
Division of Reactor Projects - III
IV, V and Special Projects

Enclosures:
1. Amendment No. 59 to License No. NPF-11
2. Amendment No. 39 to License No. NPF-18
3. Safety Evaluation

cc w/enclosure:
See next page

Mr. Henry E. Bliss
Commonwealth Edison Company

LaSalle County Nuclear Power Station
Units 1 & 2

cc:

Philip P. Steptoe, Esquire
Sidley and Austin
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Chicago, Illinois 60603

Assistant Attorney General
100 West Randolph Street
Suite 12
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Resident Inspector/LaSalle, NPS
U.S. Nuclear Regulatory Commission
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P. O. Box 224
Marseilles, Illinois 61341

Chairman
LaSalle County Board of Supervisors
LaSalle County Courthouse
Ottawa, Illinois 61350

Attorney General
500 South 2nd Street
Springfield, Illinois 62701

Chairman
Illinois Commerce Commission
Leland Building
527 East Capitol Avenue
Springfield, Illinois 62706

Mr. Michael C. Parker, Chief
Division of Engineering
Illinois Department of Nuclear Safety
1035 Outer Park Drive, 5th Floor
Springfield, Illinois 62704

Regional Administrator, Region III
U. S. Nuclear Regulatory Commission
799 Roosevelt Road, Bldg. #4
Glen Ellyn, Illinois 60137

John W. McCaffrey
Chief, Public Utilities Division
SOIC
100 West Randolph Street
Chicago, Illinois 60601

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

COMMONWEALTH EDISON COMPANY

DOCKET NO. 50-373

LASALLE COUNTY STATION, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 59
License No. NPF-11

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

   A. The application for amendment filed by the Commonwealth Edison Company (the licensee), dated April 29, 1987 complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's regulations set forth in 10 CFR Chapter I;

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

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

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

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

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the enclosure to this license amendment and paragraph 2.C.(2) of the Facility Operating License No. NPF-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. 59, 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.

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3. This amendment is effective upon date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Daniel R. Muller

Daniel R. Muller, Director
Project Directorate III-2
Division of Reactor Projects - III,
IV, V and Special Projects

Enclosure:
Changes to the Technical
Specifications

Date of Issuance: August 31, 1988

ENCLOSURE TO LICENSE AMENDMENT NO. 59

FACILITY OPERATING LICENSE NO. NPF-11

DOCKET NO. 50-373

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
XIX	XIX
3/4 3-30	3/4 3-30
3/4 5-8	3/4 5-8
3/4 5-9	3/4 5-9
3/4 6-16	3/4 6-16
3/4 6-18	3/4 6-18
B 3/4 5-2	B 3/4 5-2
B 3/4 6-3	B 3/4 6-3
-------	b 3/4 6-3a (new page)

INDEX

LIST OF FIGURES

TABLE 3.3.3-2 (Continued)

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
C. DIVISION 3 TRIP SYSTEM		
1. HPCS SYSTEM		
a. Reactor Vessel Water Level - Low Low, Level 2	≥ -50 inches*	≥ -57 inches*
b. Drywell Pressure - High	≤ 1.69 psig	≤ 1.89 psig
c. Reactor Vessel Water Level - High, Level 8	≤ 55.5 inches*	≤ 56 inches*
d. Condensate Storage Tank Level - Low	≥ 715'7"	≥ 715'3"
e. Suppression Pool Water Level - High	≤ 2 inches**	≤ 3 inches**
f. Pump Discharge Pressure - High	≥ 120 psig	≥ 110 psig
g. HPCS System Flow Rate - Low	≥ 1000 gpm	≥ 900 gpm
h. Manual Intiation	NA	NA
D. LOSS OF POWER		
1. 4.16 kV Emergency Bus Undervoltage (Loss of Voltage)#		
a. 4.16 kV Basis		
1) Divisions 1 and 2	2625 ± 131 volts with ≤ 10 seconds time delay	2625 ± 262 volts with ≤ 11 seconds time delay
	2496 ± 125 volts with ≥ 4 seconds time delay	2496 ± 250 volts with ≥ 3 seconds time delay
2) Division 3	2870 ± 143 volts with ≤ 10 seconds time delay	2870 ± 287 volts with ≤ 11 seconds time delay

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

#These are inverse time delay voltage relays or instantaneous voltage relays with a time delay. The voltages shown are the maximum that will not result in a trip. Lower voltage conditions will result in decreased trip times.

**Level is referenced to a plant elevation of 699 feet 11 inches (See Figure B 3/4.6.2-1).

3/4.5.3 SUPPRESSION CHAMBER#

LIMITING CONDITION FOR OPERATION

3.5.3 The suppression chamber shall be OPERABLE:

a. In OPERATIONAL CONDITION 1, 2, or 3 with a contained water volume of at least 128,800 ft^3, equivalent to a level of -4 1/2 inches.**

b. In OPERATIONAL CONDITION 4 or 5* with a contained water volume of at least 70,000 ft^3, equivalent to a level of -12 feet 7 inches,** except that the suppression chamber level may be less than the limit or may be drained in OPERATIONAL CONDITION 4 or 5* provided that:

1. No operations are performed that have a potential for draining the reactor vessel,

2. The reactor mode switch is locked in the Shutdown or Refuel position,

3. The condensate storage tank contains at least 135,000 available gallons of water, equivalent to a level of 14.5 feet, and

4. The HPCS system is OPERABLE per Specification 3.5.2 with an OPERABLE flow path capable of taking suction from the condensate storage tank and transferring the water through the spray sparger to the reactor vessel.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, 3, 4, and 5*.

ACTION:

a. In OPERATIONAL CONDITION 1, 2, or 3 with the suppression chamber water level less than the above limit, restore the water level to within the limit within 1 hour or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. In OPERATIONAL CONDITION 4 or 5* with the suppression chamber water level less than the above limit or drained and the above required conditions not satisfied, suspend CORE ALTERATIONS and all operations that have a potential for draining the reactor vessel and lock the reactor mode switch in the Shutdown position. Establish SECONDARY CONTAINMENT INTEGRITY within 8 hours.

#See Specification 3.6.2.1 for pressure suppression requirements.

*The suppression chamber is not required to be OPERABLE provided that the reactor vessel head is removed, the cavity is flooded or being flooded from the suppression pool, the spent fuel pool gates are removed when the cavity is flooded, and the water level is maintained within the limits of Specifications 3.9.8 and 3.9.9.

**Level is referenced to a plant elevation of 699 feet 11 inches (See Figure B 3/4.6.2-1).

LIMITING CONDITION FOR OPERATION (Continued)

ACTION: (Continued)

c. With one suppression chamber water level instrumentation channel inoperable, restore the inoperable channel to OPERABLE status within 7 days or verify the suppression chamber water level to be greater than or equal to -4 1/2 inches** or -12 feet 7 inches**, as applicable, at least once per 12 hours by local indication.

d. With both suppression chamber water level instrumentation channels inoperable, restore at least one inoperable channel to OPERABLE status within 8 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours and verify the suppression chamber water level to be greater than or equal to -4 1/2 inches** or -12 feet 7 inches**, as applicable, at least once per 12 hours by local indication.

SURVEILLANCE REQUIREMENTS

4.5.3.1 The suppression chamber shall be determined OPERABLE by verifying:

a. The water level to be greater than or equal to, as applicable:

1. -4 1/2 inches** at least once per 24 hours.

2. -12 feet 7 inches** at least once per 12 hours.

b. Two suppression chamber water level instrumentation channels OPERABLE by performance of a:

1. CHANNEL CHECK at least once per 24 hours,

2. CHANNEL FUNCTIONAL TEST at least once per 31 days, and

3. CHANNEL CALIBRATION at least once per 18 months,

with the low water level alarm setpoint at greater than or equal to -3 inches.**

4.5.3.2 With the suppression chamber level less than the above limit or drained in OPERATIONAL CONDITION 4 or 5*, at least once per 12 hours:

a. Verify the required conditions of Specification 3.5.3.b. to be satisfied, or

b. Verify footnote conditions* to be satisfied.

*The suppression chamber is not required to be OPERABLE provided that the reactor vessel head is removed, the cavity is flooded or being flooded from the suppression pool, the spent fuel pool gates are removed when the cavity is flooded, and the water level is maintained within the limits of Specifications 3.9.8 and 3.9.9.

**Level is referenced to a plant elevation of 699 feet 11 inches (See Figure B 3/4.6.2-1).

SUPPRESSION CHAMBER#

LIMITING CONDITION FOR OPERATION

3.6.2.1 The suppression chamber shall be OPERABLE with:

a. The pool water:

1. Volume between 131,900 ft^3 and 128,800 ft^3, equivalent to a level between +3 inches** and -4 1/2 inches**, and a

2. Maximum average temperature of 100°F* during OPERATIONAL CONDITION 1 or 2, except that the maximum average temperature may be permitted to increase to:

a) 105°F,## during testing which adds heat to the suppression chamber.

b) 110°F with THERMAL POWER less than or equal to 1% of RATED THERMAL POWER.

c) 120°F with the main steam line isolation valves closed following a scram.

b. Drywell-to-suppression charmer bypass leakage less than or equal to 10% of the acceptable $A/\sqrt{k}$ design value of 0.03 ft^2.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

a. With the suppression chamber water level outside the above limits, restore the water level 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.

b. In OPERATIONAL CONDITION 1 or 2 with the suppression chamber average water temperature greater than or equal to 100°F, restore the average temperature to less than or equal to 100°F within 24 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours, except, as permitted above:

1. With the suppression chamber average water temperature greater than 105°F during testing which adds heat to the suppression chamber, stop all testing which adds heat to the suppression chamber and restore the average temperature to less than or equal to 100°F within 24 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.
2. With the suppression chamber average water temperature greater than 110°F, place the reactor mode switch in the Shutdown position and operate at least one residual heat removal loop in the suppression pool cooling mode.
3. With the suppression chamber average water temperature greater than 120°F, depressurize the reactor pressure vessel to less than 200 psig within 12 hours.

#See Specification 3.5.3 for ECCS requirements.
##See Special Test Exception 3.10.8.
**Level is referenced to a plant elevation of 699 feet 11 inches (See Figure B 3/4.6.2-1).

SURVEILLANCE REQUIREMENTS (Continued)

c. By verifying at least two suppression chamber water level instrumentation channels and at least 14 suppression pool water temperature instrumentation channels, 7 in each of two divisions, OPERABLE by performance of a:

   1. CHANNEL CHECK at least once per 24 hours,

   2. CHANNEL FUNCTIONAL TEST at least once per 31 days, and

   3. CHANNEL CALIBRATION at least once per 18 months.

   The suppression chamber water level and suppression pool temperature alarm setpoint shall be:

   a) High water level $\leq$ +2 inches*

   b) Low water level $\geq$ -3 inches*

   c) High temperature $\leq$ 100°F

d. By conducting drywell-to-suppression chamber bypass leak tests and verifying that the $A/\sqrt{k}$ calculated from the measured leakage is within the specified limit when drywell-to-suppression chamber bypass leak tests are conducted:

   1. At least once per 18 months at an initial differential pressure of 1.5 psi, and

   2. At the first refueling outage and then on the schedule required for Type A Overall Integrated Containment Leakage Rate tests by Specification 4.6.1.2.a; at an initial differential pressure of 5 psi,

   except that, if the first two 5 psi leak tests performed up to that time result in:

   1. A calculated $A/\sqrt{k}$ within the specified limit, and

   2. The $A/\sqrt{k}$ calculated from the leak tests at 1.5 psi is $\leq$ 20% of the specified limit,

   then the leak tests at 5 psi may be discontinued.

*Level is referenced to a plant elevation of 699 feet 11 inches (See Figure B 3/4.6.2-1).

ECCS-OPERATING and SHUTDOWN (Continued)

the suppression pool into the reactor, but no credit is taken in the hazards analyses for the condensate storage tank water.

With the HPCS system inoperable, adequate core cooling is assured by the OPERABILITY of the redundant and diversified automatic depressurization system and both the LPCS and LPCI systems. In addition, the reactor core isolation cooling (RCIC) system, a system for which no credit is taken in the hazards analysis, will automatically provide makeup at reactor operating pressures on a reactor low water level condition. The HPCS out-of-service period of 14 days is based on the demonstrated OPERABILITY of redundant and diversified low pressure core cooling systems.

The surveillance requirements provide adequate assurance that the HPCS system will be OPERABLE when required. Although all active components are testable and full flow can be demonstrated by recirculation through a test loop during reactor operation, a complete functional test with reactor vessel injection requires reactor shutdown. The pump discharge piping is maintained full to prevent water hammer damage and to provide cooling at the earliest moment.

Upon failure of the HPCS system to function properly, if required, the automatic depressurization system (ADS) automatically causes selected safety-relief valves to open, depressurizing the reactor so that flow from the low pressure core cooling systems can enter the core in time to limit fuel cladding temperature to less than 2200°F. ADS is conservatively required to be OPERABLE whenever reactor vessel pressure exceeds 122 psig even though low pressure core cooling systems provide adequate core cooling up to 350 psig.

ADS automatically controls seven selected safety-relief valves. Six valves are required to be OPERABLE since the LOCA analysis assumes 6 ADS valves in addition to a single failure. It is therefore appropriate to permit one of the required valves to be out-of-service for up to 14 days without materially reducing system reliability.

3/4.5.3 SUPPRESSION CHAMBER

The suppression chamber is also required to be OPERABLE as part of the ECCS to ensure that a sufficient supply of water is available to the HPCS, LPCS and LPCI systems in the event of a LOCA. This limit on suppression chamber minimum water volume ensures that sufficient water is available to permit recirculation cooling flow to the core. The OPERABILITY of the suppression chamber in OPERATIONAL CONDITIONS 1, 2 or 3 is required by Specification 3.6.2.1.

Repair work might require making the suppression chamber inoperable. This specification will permit those repairs to be made and at the same time give assurance that the irradiated fuel has an adequate cooling water supply when the suppression chamber must be made inoperable, including draining, in OPERATIONAL CONDITION 4 or 5.

In OPERATIONAL CONDITION 4 and 5 the suppression chamber minimum required water volume is reduced because the reactor coolant is maintained at or below 200°F. Since pressure suppression is not required below 212°F, the minimum water volume is based on NPSH, recirculation volume, vortex prevention plus a 2'-4" safety margin for conservatism.

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.

Suppression Chamber Level	Plant Elevation	Control Room/Local Indication
26' 10"	700' 2"	+3" High Level LCO (Volume 131,900 ft^3)
26' 9"	700' 1"	+2" High Level Alarm HPCS Suction Valve Transfer
26' 7"	699' 11"	0" Instrument Zero
26' 4"	699' 8"	-3" Low Level Alarm
26' 2 1/2"	699' 6 1/2"	-4 1/2" Low Level LCO Operational Condition 1, 2, or 3 (Volume 128,800 ft^3)
14'	687' 4"	-12' 7" Low Level LCO Operational Condition 4 or 5 (Volume 70,000 ft^3)

SUPPRESSION POOL LEVEL SETPOINTS

BASES FIGURE B 3/4.6.2-1

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

COMMONWEALTH EDISON COMPANY

DOCKET NO. 50-374

LASALLE COUNTY STATION, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 39
License No. NPF-18

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

   A. The application for amendment filed by the Commonwealth Edison Company (the licensee), dated April 29, 1987 complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's regulations set forth in 10 CFR Chapter I;

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

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

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

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

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

   (2) Technical Specifications and Environmental Protection Plan

   The Technical Specifications contained in Appendix A, as revised through Amendment No.39, 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 amendment is effective upon date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Daniel R. Muller

Daniel R. Muller, Director
Project Directorate III-2
Division of Reactor Projects - III,
IV, V and Special Projects

Enclosure:
Changes to the Technical
Specifications

Date of Issuance: August 31, 1988

ENCLOSURE TO LICENSE AMENDMENT NO. 39

FACILITY OPERATING LICENSE NO. NPF-18

DOCKET NO. 50-374

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
XIX	XIX
3/4 3-30	3/4 3-30
3/4 5-8	3/4 5-8
3/4 5-9	3/4 5-9
3/4 6-19	3/4 6-19
3/4 6-21	3/4 6-21
B 3/4 5-2	B 3/4 5-2
B 3/4 6-3	B 3/4 6-3
-------	B 3/4 6-3a (new page)

LIST OF FIGURES

TABLE 3.3.3-2 (Continued)
EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
C. DIVISION 3 TRIP SYSTEM		
1. HPCS SYSTEM		
a. Reactor Vessel Water Level - Low Low, Level 2	≥ 50 inches*	≥ 57 inches*
b. Drywell Pressure - High	≤ 1.69 psig	≤ 1.89 psig
c. Reactor Vessel Water Level - High, Level 8	≤ 55.5 inches*	≤ 56 inches*
d. Condensate Storage Tank Level - Low	≥ 715'7"	≥ 715'3"
e. Suppression Pool Water Level - High	≤ 2 inches**	≤ 3 inches**
f. Pump Discharge Pressure - High	≥ 120 psig	≥ 110 psig
g. HPCS System Flow Rate - Low	≥ 1000 gpm	≥ 900 gpm
h. Manual Intiation	N.A.	N.A.
D. LOSS OF POWER		
1. 4.16 kV Emergency Bus Undervoltage (Loss of Voltage)#		
a. 4.16 kV Basis		
1) Divisions 1 and 2	2625 ± 131 volts with ≤ 10 second time delay	2625 ± 262 volts with ≤ 11 second time delay
	2496 ± 125 volts with ≥ 4 second time delay	2496 ± 250 volts with ≥ 3 second time delay
2) Division 3	2870 ± 143 volts with ≤ 10 second time delay	2870 ± 287 volts with ≤ 11 second time delay

TABLE NOTATIONS

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

#These are inverse time delay voltage relays or instantaneous voltage relays with a time delay. The voltages shown are the maximum that will not result in a trip. Lower voltage conditions will result in decreased trip times.

N.A. Not Applicable

**Level is referenced to a plant elevation of 699 feet 11 inches (See Figure B 3/4.6.2-1).

3/4.5.3 SUPPRESSION CHAMBER#

LIMITING CONDITION FOR OPERATION

3.5.3 The suppression chamber shall be OPERABLE:

a. In OPERATIONAL CONDITION 1, 2, or 3 with a contained water volume of at least 128,800 ft^3, equivalent to a level of -4 1/2 inches.**

b. In OPERATIONAL CONDITION 4 or 5* with a contained water volume of at least 70,000 ft^3, equivalent to a level of -12 feet 7 inches**, except that the suppression chamber level may be less than the limit or may be drained in OPERATIONAL CONDITION 4 or 5* provided that:

1. No operations are performed that have a potential for draining the reactor vessel,

2. The reactor mode switch is locked in the Shutdown or Refuel position,

3. The condensate storage tank contains at least 135,000 available gallons of water, equivalent to a level of 14.5 feet, and

4. The HPCS system is OPERABLE per Specification 3.5.2 with an OPERABLE flow path capable of taking suction from the condensate storage tank and transferring the water through the spray sparger to the reactor vessel.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, 3, 4, and 5*.

ACTION:

a. In OPERATIONAL CONDITION 1, 2, or 3 with the suppression chamber water level less than the above limit, restore the water level to within the limit within 1 hour or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. In OPERATIONAL CONDITION 4 or 5* with the suppression chamber water level less than the above limit or drained and the above required conditions not satisfied, suspend CORE ALTERATIONS and all operations that have a potential for draining the reactor vessel and lock the reactor mode switch in the Shutdown position. Establish SECONDARY CONTAINMENT INTEGRITY within 8 hours.

#See Specification 3.6.2.1 for pressure suppression requirements.
*The suppression chamber is not required to be OPERABLE provided that the reactor vessel head is removed, the cavity is flooded or being flooded from the suppression pool, the spent fuel pool gates are removed when the cavity is flooded, and the water level is maintained within the limits of Specifications 3.9.8 and 3.9.9.
**Level is referenced to a plant elevation of 699 feet 11 inches (see Figure B 3/4 6.2-1).

LIMITING CONDITION FOR OPERATION (Continued)

ACTION: (Continued)

c. With one suppression chamber water level instrumentation channel inoperable, restore the inoperable channel to OPERABLE status within 7 days or verify the suppression chamber water level to be greater than or equal to -4 1/2 inches** or -12 feet 7 inches**, as applicable, at least once per 12 hours by local indication.

d. With both suppression chamber water level instrumentation channels inoperable, restore at least one inoperable channel to OPERABLE status within 8 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours and verify the suppression chamber water level to be greater than or equal to -4 1/2 inches** or -12 feet 7 inches**, as applicable, at least once per 12 hours by local indication.

SURVEILLANCE REQUIREMENTS

4.5.3.1 The suppression chamber shall be determined OPERABLE by verifying:

a. The water level to be greater than or equal to, as applicable:

 1. -4 1/2 inches** at least once per 24 hours.

 2. -12 feet 7 inches** at least once per 12 hours.

b. Two suppression chamber water level instrumentation channels OPERABLE by performance of a:

 1. CHANNEL CHECK at least once per 24 hours,

 2. CHANNEL FUNCTIONAL TEST at least once per 31 days, and

 3. CHANNEL CALIBRATION at least once per 18 months,

 with the low water level alarm setpoint at greater than or equal to -3 inches.**

4.5.3.2 With the suppression chamber level less than the above limit or drained in OPERATIONAL CONDITION 4 or 5*, at least once per 12 hours:

a. Verify the required conditions of Specification 3.5.3.b. to be satisfied, or

b. Verify footnote conditions* to be satisfied.

*The suppression chamber is not required to be OPERABLE provided that the reactor vessel head is removed, the cavity is flooded or being flooded from the suppression pool, the spent fuel pool gates are removed when the cavity is flooded, and the water level is maintained within the limits of Specifications 3.9.8 and 3.9.9.

**Level is referenced to a plant elevation of 699 feet 11 inches (See Figure B 3/4 6.2-1).

CONTAINMENT SYSTEMS

3/4.6.2 DEPRESSURIZATION SYSTEMS

SUPPRESSION CHAMBER#

LIMITING CONDITION FOR OPERATION

3.6.2.1 The suppression chamber shall be OPERABLE with:

a. The pool water:

1. Volume between 131,900 ft^3 and 128,800 ft^3, equivalent to a level between +3 inches** and -4 1/2 inches**, and a

2. Maximum average temperature of 100°F during OPERATIONAL CONDITION 1 or 2, except that the maximum average temperature may be permitted to increase to:

a) 105°F, during testing which adds heat to the suppression chamber.

b) 110°F with THERMAL POWER less than or equal to 1% of RATED THERMAL POWER.

c) 120°F with the main steam line isolation valves closed following a scram.

b. Drywell-to-suppression chamber bypass leakage less than or equal to 10% of the acceptable $A/\sqrt{k}$ design value of 0.03 ft^2.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

a. With the suppression chamber water level outside the above limits, restore the water level 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.

b. In OPERATIONAL CONDITION 1 or 2 with the suppression chamber average water temperature greater than or equal to 100°F, restore the average temperature to less than or equal to 100°F within 24 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours, except, as permitted above:

1. With the suppression chamber average water temperature greater than 105°F during testing which adds heat to the suppression chamber, stop all testing which adds heat to the suppression chamber and restore the average temperature to less than or equal to 100°F within 24 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

2. With the suppression chamber average water temperature greater than 110°F, place the reactor mode switch in the Shutdown position and operate at least one residual heat removal loop in the suppression pool cooling mode.

3. With the suppression chamber average water temperature greater than 120°F, depressurize the reactor pressure vessel to less than 200 psig within 12 hours.

#See Specification 3.5.3 for ECCS requirements.

**Level is referenced to a plant elevation of 699 feet 11 inches (See Figure B 3/4.6.2-1).

SURVEILLANCE REQUIREMENTS (Continued)

c. By verifying at least 2 suppression chamber water level instrumentation channels and at least 14 suppression pool water temperature instrumentation channels, 7 in each of two divisions, OPERABLE by performance of a:

1. CHANNEL CHECK at least once per 24 hours,

2. CHANNEL FUNCTIONAL TEST at least once per 31 days, and

3. CHANNEL CALIBRATION at least once per 18 months.

The suppression chamber water level and suppression pool temperature alarm setpoint shall be:

a) High water level $\leq$ +2 inches*

b) Low water level $\geq$ -3 inches*

c) High temperature $\leq$ 100°F

d. By conducting drywell-to-suppression chamber bypass leak tests and verifying that the $A/\sqrt{k}$ calculated from the measured leakage is within the specified limit when drywell-to-suppression chamber bypass leak tests are conducted:

1. At least once per 18 months at an initial differential pressure of 1.5 psi, and

2. At the first refueling outage and then on the schedule required for Type A Overall Integrated Containment Leakage Rate tests by Specification 4.6.1.2.a., at an initial differential pressure of 5 psi,

except that, if the first two 5 psi leak tests performed up to that time result in:

1. A calculated $A/\sqrt{k}$ within the specified limit, and

2. The $A/\sqrt{k}$ calculated from the leak tests at 1.5 psi is $\leq$ 20% of the specified limit,

then the leak tests at 5 psi may be discontinued.

*Level is referenced to a plant elevation of 699 feet 11 inches (See Figure B 3/4.6.2-1).

ECCS-OPERATING and SHUTDOWN (Continued)

the suppression pool into the reactor, but no credit is taken in the hazards analyses for the condensate storage tank water.

With the HPCS system inoperable, adequate core cooling is assured by the OPERABILITY of the redundant and diversified automatic depressurization system and both the LPCS and LPCI systems. In addition, the reactor core isolation cooling (RCIC) system, a system for which no credit is taken in the hazards analysis, will automatically provide makeup at reactor operating pressures on a reactor low water level condition. The HPCS out-of-service period of 14 days is based on the demonstrated OPERABILITY of redundant and diversified low pressure core cooling systems.

The surveillance requirements provide adequate assurance that the HPCS system will be OPERABLE when required. Although all active components are testable and full flow can be demonstrated by recirculation through a test loop during reactor operation, a complete functional test with reactor vessel injection requires reactor shutdown. The pump discharge piping is maintained full to prevent water hammer damage and to provide cooling at the earliest moment.

Upon failure of the HPCS system to function properly, if required, the automatic depressurization system (ADS) automatically causes selected safety-relief valves to open, depressurizing the reactor so that flow from the low pressure core cooling systems can enter the core in time to limit fuel cladding temperature to less than 2200°F. ADS is conservatively required to be OPERABLE whenever reactor vessel pressure exceeds 122 psig even though low pressure core cooling systems provide adequate core cooling up to 350 psig.

ADS automatically controls seven selected safety-relief valves. Six valves are required to be OPERABLE since the LOCA analysis assumes 6 ADS valves in addition to a single failure. It is therefore appropriate to permit one of the required valves to be out-of-service for up to 14 days without materially reducing system reliability.

3/4.5.3 SUPPRESSION CHAMBER

The suppression chamber is also required to be OPERABLE as part of the ECCS to ensure that a sufficient supply of water is available to the HPCS, LPCS and LPCI systems in the event of a LOCA. This limit on suppression chamber minimum water volume ensures that sufficient water is available to permit recirculation cooling flow to the core (See Figure B 3/4.6.2-1). The OPERABILITY of the suppression chamber in OPERATIONAL CONDITIONS 1, 2 or 3 is required by Specification 3.6.2.1.

Repair work might require making the suppression chamber inoperable. This specification will permit those repairs to be made and at the same time give assurance that the irradiated fuel has an adequate cooling water supply when the suppression chamber must be made inoperable, including draining, in OPERATIONAL CONDITION 4 or 5.

In OPERATIONAL CONDITION 4 and 5 the suppression chamber minimum required water volume is reduced because the reactor coolant is maintained at or below 200°F. Since pressure suppression is not required below 212°F, the minimum water volume is based on NPSH, recirculation volume, vortex prevention plus a 2'-4" safety margin for conservatism.

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.

Suppression Chamber Level	Plant Elevation	Control Room/Local Indication
26' 10"	700' 2"	+3" High Level LCO (Volume 131,900 ft^3)
26' 9"	700' 1"	+2" High Level Alarm HPCS Suction Valve Transfer
26' 7"	699' 11"	0" Instrument Zero
26' 4"	699' 8"	-3" Low Level Alarm
26' 2 1/2"	699' 6 1/2"	-4 1/2" Low Level LCO Operational Condition 1, 2, or 3 (Volume 128,800 ft^3)
14'	687' 4"	-12' 7" Low Level LCO Operational Condition 4 or 5 (Volume 70,000 ft^3)

SUPPRESSION POOL LEVEL SETPOINTS

BASES FIGURE B 3/4.6.2-1

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 59 TO FACILITY OPERATING LICENSE NO. NPF-11 AND

AMENDMENT NO. 39 TO FACILITY OPERATING LICENSE NO. NPF-18

COMMONWEALTH EDISON COMPANY

LASALLE COUNTY STATION, UNITS 1 AND 2

DOCKET NOS. 50-373 AND 50-374

1.0 INTRODUCTION

By letter dated April 29, 1987, Commonwealth Edison proposed to amend Facility Operating License NPF-11 and NPF-18 pursuant to 10 CFR 50.90. The proposed amendment corrects an inconsistency between Technical Specification requirements regarding the suppression pool high level alarm.

2.0 EVALUATION

Commonwealth Edison discovered that an inconsistency exists between the requirements in Technical Specification 4.6.2.1.c and Technical Specification Table 3.3.3-2. Technical Specification 4.6.2.1.c.1 requires a setpoint of less than or equal to 26 feet 8 inches (equivalent to a plant elevation of 700 feet 0 inches) for the suppression pool high level alarm. Technical Specification 3.3.3-2 and the Updated Final Safety Analysis Report (UFSAR) Table 7.3-1 require trip setpoint of less than or equal to 700 feet 1 inch and an allowable value of less than or equal 700 feet 2 inches for Suppression Pool Water Level - High. Both of these alarm setpoints are below the maximum allowable Suppression Pool level of 26 feet 10 inches indicated in Technical Specification 3.6.2.1.a.1.

The subsequent Commonwealth Edison investigation into the cause of the inconsistency concluded that one of the contributing factors was the use of different reference points for the suppression pool high level alarm setpoints identified in the Technical Specifications. That is, the levels were referenced to plant elevation in one case and to the bottom of the suppression chamber in the other. The investigation also found that the reference points used in the Technical Specifications for the Limiting Conditions for Operations and trip setpoints are not consistent with the instrument references used for the control room and local suppression pool level indications. The instrument zero for all plant suppression pool level indications is set at a plant elevation of 699 feet 11 inches.

These inconsistencies within the Technical Specifications and between the Technical Specifications and plant indications have the potential to cause future personnel errors. The licensee proposed that the following amendments be made to the Technical Specifications:

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1. The suppression pool high water level alarm setpoint in Technical Specification 4.6.2.1.c.1 be raised 1 inch to be consistent with Technical Specification Table 3.3.3-2 and the UFSAR.

2. All references to suppression pool level in the Technical Specifications be amended to be consistent with plant indications.

3. A figure be added to the Technical Specification bases which will correlate plant evaluation, suppression chamber levels and suppression pool level indications.

The change to the suppression pool high level alarm setpoint does not effect the LCO for suppression pool level. By making the Technical Specification limits consistent with plant indications, the potential for future personnel errors occurring, due to misinterpretation of the technical specifications, will be reduced.

The staff has reviewed the proposed Technical Specification changes and concludes that they are consistent with the analyses described above and are acceptable. We conclude that the proposed Unit 1 and Unit 2 license amendments are acceptable.

3.0 ENVIRONMENTAL CONSIDERATION

Pursuant to 10 CFR 51.32 the Commission has previously determined in an environmental assessment of the proposed action published in the Federal Register that granting this amendment will have no significant impact on the quality of the human environment (53 FR 8520).

4.0 CONCLUSION

The NRC staff has reviewed the licensee's submittal to correct inconsistencies between Technical Specification requirements regarding the suppression pool high level alarm. Based on this review, the staff concludes that the proposed Technical Specification changes are acceptable.

The staff 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, and (2) such activities will be conducted in compliance with the Commission's regulations and the issuance of these amendments will not be inimical to the common defense and security or to the health and safety of the public.

REFERENCE

Letter from C. Allen, Commonwealth Edison to USNRC dated April 29, 1987.

Principal Contributor: Paul Shemanski, NRR/PDIII-2

Dated: August 31, 1988

UNITED STATES NUCLEAR REGULATORY COMMISSION

COMMONWEALTH EDISON COMPANY

DOCKET NOS. 50-373 AND 50-374

NOTICE OF ISSUANCE OF AMENDMENT TO

FACILITY OPERATING LICENSE

The U.S. Nuclear Regulatory Commission (Commission) has issued Amendment No. 59 to Facility Operating License No. NPF-11 and Amendment No. 39 to Facility Operating License No. NPF-18, issued to Commonwealth Edison Company, (the licensee), which revised the Technical Specifications for operation of The LaSalle County Station, (the facility), Units 1 and 2, located in LaSalle County, Illinois. The amendments are effective 30 days after the date of issuance.

The amendments correct an inconsistency between Technical Specification requirements regarding the suppression pool high level alarm. The following changes to the Technical Specification have been made:

1. The suppression pool high water level alarm setpoint in Technical Specification 4.6.2.1.c.1 has been raised 1 inch to be consistent with Technical Specification Table 3.3.3-2 and the UFSAR.

2. All references to suppression pool level in the Technical Specifications have been amended to be consistent with plant indications.

3. A figure has been added to the Technical Specification bases which correlates plant elevation, suppression chamber levels and suppression pool level indications.

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These revisions to the licenses of LaSalle County Station, Units 1 and 2 are in response to the licensee's application for amendment dated April 29, 1987.

The application for the amendment complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendment.

Notice of Consideration of Issuance of Amendment and Opportunity for Hearing in connection with this action was published in the FEDERAL REGISTER on December 8, 1987 (52 FR 46541). No request for a hearing or petition for leave to intervene was filed following this notice.

The Commission has prepared an Environmental Assessment and Finding of No Significant Impact related to the action and has concluded that an environmental impact statement is not warranted because the action will not have a significant adverse effect on the quality of the human environment and there will be no environmental impact attributable to the action beyond that which has been predicted and described in the Commission's Final Environmental Statement for the facility dated November 1978.

For further details with respect to the actions see (1) the application for amendment April 29, 1987, and Amendment No. 39 to License No. NPF-18, (2) Amendment No. 59 to License No. NPF-11 and (3) the Commission's related Safety Evaluation and Environmental Assessment and Finding of No Significant Impact.

All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N.W., and at the Public Library of Illinois Valley Community College, Rural Route No. 1, Oglesby, Illinois 61348. A copy of items (2), and (3) may be obtained upon request addressed to the U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attention: Director, Division of Reactor Projects.

Dated at Rockville, Maryland this 31st day of August 1988.

FOR THE NUCLEAR REGULATORY COMMISSION

Daniel R Muller

Daniel R. Muller, Director
Project Directorate III-2
Division of Reactor Projects - III,
IV, V and Special Projects