Docket No. 50-374

SEP 21 1984

Mr. Dennis L. Farrar
Director of Nuclear Licensing
Commonwealth Edison Company
P.O. Box 767
Chicago, Illinois 60690

Dear Mr. Farrar:

Subject: Amendment No. 4 to Facility Operating License No. NPF-18
La Salle County Station, Unit 2

The Nuclear Regulatory Commission has issued the enclosed Amendment No. 4 to Facility Operating License No. NPF-18 for the La Salle County Station, Unit 2. This amendment is in response to your letter dated July 31, 1984, as supplemented by letters dated August 1 & 2, 1984. The amendment vacates Amendment No. 3 and reinstates License Condition 2.C.(7).

A copy of the related safety evaluation supporting Amendment No. 4 to Facility Operating License NPF-18 is enclosed.

Sincerely,

A. Schwencer, Chief
Licensing Branch No. 2
Division of Licensing

Enclosures:
1. Amendment No. 4 to NPF-18
2. Staff Evaluation

cc w/enclosures:
See next page

DISTRIBUTION
See Attached

AS

*DL:LB#2/PM	*DL:LB#2/LA	DL:LB#2/BC	OELD OK with minor changes to SER
ABournia:dh	EHylton	ASchwencer	CWoodhead
8/28/84	8/28/84	8/30/84	~~8~~/ /84 9-11

*See previous concurrence

8410170028 840921
PDR ADOCK 05000374
P PDR

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

SEP 21 1984

Docket No. 50-374

Mr. Dennis L. Farrar
Director of Nuclear Licensing
Commonwealth Edison Company
P.O. Box 767
Chicago, Illinois 60690

Dear Mr. Farrar:

Subject: Amendment No. 4 to Facility Operating License No. NPF-18
La Salle County Station, Unit 2

The Nuclear Regulatory Commission has issued the enclosed Amendment No. 4 to Facility Operating License No. NPF-18 for the La Salle County Station, Unit 2. This amendment is in response to your letter dated July 31, 1984, as supplemented by letters dated August 1 & 2, 1984. The amendment vacates Amendment No. 3 and reinstates License Condition 2.C.(7).

A copy of the related safety evaluation supporting Amendment No. 4 to Facility Operating License NPF-18 is enclosed.

Sincerely,

A. Schwencer

A. Schwencer, Chief
Licensing Branch No. 2
Division of Licensing

Enclosures:
1. Amendment No. 4 to NPF-18
2. Staff Evaluation

cc w/enclosures:
See next page

La Salle

Mr. Dennis L. Farrar
Director of Nuclear Licensing
Commonwealth Edison Company
P.O. Box 767
Chicago, Illinois 60690

cc: Philip P. Steptoe, Esquire
Suite 4200
One First National Plaza
Chicago, Illinois 60603

Assistant Attorney General
188 West Randolph Street
Suite 2315
Chicago, Illinois 60601

Michael J. Jordan, Resident Inspector
La Salle, NPS, U.S.N.R.C.
P.O. Box 224
Marseilles, Illinois 61364

Chairman
La Salle County Board of Supervisors
La Salle County Courthouse
Ottawa, Illinois 61350

Attorney General
500 South 2nd Street
Springfield, Illinois 62701

Department of Public Health
535 West Jefferson
Springfield, Illinois 62761
ATTN: Chief, Division of Nuclear Safety

The Honorable Tom Corcoran
United States House of Representatives
Washington, D.C. 20515

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

Mr. Gary N. Wright, Manager
Nuclear Facility Safety
Illinois Department of Nuclear Safety
1035 Outer Park Drive, 5th Floor
Springfield, Illinois 62704

B. Paragraph 2.C.(7) is added as follows:

(7) Low Pressure in Pump Discharge of the Control Rod Drive (Section 4.6.2, SSER #2, and Section 7.2.3.2, SSER #7)

Prior to completion of the startup test program, the licensee shall install instrumentation that would automatically shut down the reactor in the event of low control rod drive pump discharge pressure. This automatic scram shall be activated during startup and refueling modes only.

3. This amendment is effective as of date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Darrell G. Eisenhut, Director
Division of Licensing

Enclosure:
Changes to the Technical
Specifications

Date of Issuance: SEP 21 1984

LB#2/DL	LB#2/DL	LB#2/DL	OELD	AD/L/DL	D/DL
ABournia:dh	PShuttleworth	ASchwencer	CWoodhead	TMNovak	DGEisenhut
8/30/84	8/30/84	8/30/84	8/ /84	08/10/84	09/21/84

COMMONWEALTH EDISON COMPANY

DOCKET NO. 50-374

LA SALLE COUNTY STATION, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 4
License No. NPF-18

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

   A. The application for amendment filed by the Commonwealth Edison Company, dated July 31,1984, as supplemented by letters dated August 1 & 2, 1984, 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 a 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 as follows:

   A. Page 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:

      The Technical Specifications contained in Appendix A, as revised through Amendment No. 4, 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.

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

COMMONWEALTH EDISON COMPANY

DOCKET NO. 50-374

LA SALLE COUNTY STATION, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 4
License No. NPF-18

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

   A. The application for amendment filed by the Commonwealth Edison Company, dated July 31,1984, as supplemented by letters dated August 1 & 2, 1984, 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 a 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 as follows:

   A. Page 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:

      The Technical Specifications contained in Appendix A, as revised through Amendment No. 4, 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.

B. Paragraph 2.C.(7) is added as follows:

(7) Low Pressure in Pump Discharge of the Control Rod Drive (Section 4.6.2, SSER #2, and Section 7.2.3.2, SSER #7)

Prior to completion of the startup test program, the licensee shall install instrumentation that would automatically shut down the reactor in the event of low control rod drive pump discharge pressure. This automatic scram shall be activated during startup and refueling modes only.

3. This amendment is effective as of date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Frank J. Miraglia

for Darrell G. Eisenhut, Director
Division of Licensing

Enclosure:
Changes to the Technical
Specifications

Date of Issuance: SEP 21 1984

ENCLOSURE TO LICENSE AMENDMENT NO.4
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 vertical lines indicating the area of change.

REMOVE	INSERT
2-4	2-4
B 2-13	-------
3/4 1-10	3/4 1-10
3/4 3-3	3/4 3-3
3/4 3-5	3/4 3-5
3/4 3-6	3/4 3-6
3/4 3-8	3/4 3-8
B 3/4 1-3	B 3/4 1-3

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	≤ 0.66W + 51% with a maximum of	≤ 0.66W + 54% 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	≤ 0.66W + 45.7% with a maximum of	≤ 0.66W + 48.7% with a maximum of
b) High Flow Clamped	≤ 113.5% of RATED THERMAL POWER	≤ 115.5% 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. Main Steam Line Radiation - High	≤ 3 x full power background	≤ 3.6 x full power background
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
10. Turbine Control Valve Fast Closure, Trip Oil Pressure - Low	≥ 500 psig	≥ 414 psig
11. Reactor Mode Switch Shutdown Position	N.A.	N.A.
12. Manual Scram	N.A.	N.A.

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

SURVEILLANCE REQUIREMENTS

4.1.3.5 Each control rod scram accumulator shall be determined OPERABLE:

a. At least once per 7 days by verifying that the indicated pressure is greater than or equal to 940 psig unless the control rod is inserted and disarmed or scrammed.

b. At least once per 18 months by:

1. Performance of a:

   a) CHANNEL FUNCTIONAL TEST of the leak detectors, and

   b) CHANNEL CALIBRATION of the pressure detectors, with the alarm setpoint 940 + 30, -0 psig on decreasing pressure.

2. Measuring and recording the time that each individual accumulator check valve maintains the associated accumulator pressure above the alarm setpoint with no control rod drive pump operating.

TABLE 3.3.1-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	APPLICABLE OPERATIONAL CONDITIONS	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM (a)	ACTION
7. Primary Containment Pressure - High	1, 2[f]	2[g]	1
8. Scram Discharge Volume Water Level - High	1, 2,	2	1
	5[h]	2	3
9. Turbine Stop Valve - Closure	1[i]	4[j]	6
10. Turbine Control Valve Fast Closure, Valve Trip System Oil Pressure - Low	1[i]	2[j]	6
11. Reactor Mode Switch Shutdown Position	1, 2	1	1
	3, 4	1	7
	5	1	3
12. Manual Scram	1, 2	1	1
	3, 4	1	8
	5	1	9

TABLE 3.3.1-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION

TABLE NOTATIONS

(a) A channel may be placed in an inoperable status for up to 2 hours for required surveillance without placing the trip system in the tripped condition provided at least one OPERABLE channel in the same trip system is monitoring that parameter.

(b) The "shorting links" shall be removed from the RPS circuitry prior to and during the time any control rod is withdrawn* and during shutdown margin demonstrations performed per Specification 3.10.3.

(c) An APRM channel is inoperable if there are less than 2 LPRM inputs per level or less than 14 LPRM inputs to an APRM channel.

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

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

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

(g) Also actuates the standby gas treatment system.

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

(i) This function shall be automatically bypassed when turbine first stage pressure is $\leq$ 140 psig, equivalent to THERMAL POWER less than 30% of RATED THERMAL POWER.

(j) Also actuates the EOC-RPT system.

---

*Not required for control rods removed per Specification 3.9.10.1 or 3.9.10.2.

TABLE 3.3.1-2

REACTOR PROTECTION SYSTEM RESPONSE TIMES

FUNCTIONAL UNIT	RESPONSE TIME (Seconds)
1. Intermediate Range Monitors:	
a. Neutron Flux - High*	NA
b. Inoperative	NA
2. Average Power Range Monitor*	
a. Neutron Flux - High, Setdown	NA
b. Flow Biased Simulated Thermal Power-Upscale	$\leq$ 0.09**
c. Fixed Neutron Flux - High	$\leq$ 0.09
d. Inoperative	NA
3. Reactor Vessel Steam Dome Pressure - High	$\leq$ 0.55
4. Reactor Vessel Water Level - Low, Level 3	$\leq$ 1.05
5. Main Steam Line Isolation Valve - Closure	$\leq$ 0.06
6. Main Steam Line Radiation - High	NA
7. Primary Containment Pressure - High	NA
8. Scram Discharge Volume Water Level - High	NA
9. Turbine Stop Valve - Closure	$\leq$ 0.06
10. Turbine Control Valve Fast Closure, Trip Oil Pressure - Low	$\leq$ 0.08#
11. Reactor Mode Switch Shutdown Position	NA
12. Manual Scram	NA

*Neutron detectors are exempt from response time testing. Response time shall be measured from the detector output or from the input of the first electronic component in the channel.

**Not including simulated thermal power time constant.

#Measured from start of turbine control valve fast closure.

TABLE 4.3.1.1-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
8. Scram Discharge Volume Water Level - High	NA	M	R	1, 2, 5
9. Turbine Stop Valve - Closure	NA	M	R	1
10. Turbine Control Valve Fast Closure Valve Trip System Oil Pressure - Low	NA	M	R	1
11. Reactor Mode Switch Shutdown Position	NA	R	NA	1, 2, 3, 4, 5
12. Manual Scram	NA	M	NA	1, 2, 3, 4, 5

(a) Neutron detectors may be excluded from CHANNEL CALIBRATION.
(b) The IRM, and SRM channels shall be determined to overlap for at least 1/2 decades during each startup and the IRM and APRM channels shall be determined to overlap for at least 1/2 decades during each controlled shutdown, if not performed within the previous 7 days.
(c) Within 24 hours prior to startup, if not performed within the previous 7 days.
(d) This calibration shall consist of the adjustment of the APRM channel to conform to the power values calculated by a heat balance during OPERATIONAL CONDITION 1 when THERMAL POWER $\geq$ 25% of RATED THERMAL POWER. Adjust the APRM channel if the absolute difference is greater than 2%. Any APRM channel gain adjustment made in compliance with Specification 3.2.2 shall not be included in determining the absolute difference.
(e) This calibration shall consist of the adjustment of the APRM flow biased channel to conform to a calibrated flow signal.
(f) The LPRMs shall be calibrated at least once per 1000 effective full power hours (EFPH) using the TIP system.
(g) Measure and compare core flow to rated core flow.
(h) This calibration shall consist of verifying the 6 ± 1 second simulated thermal power time constant.

BASES

CONTROL RODS (Continued)

Control rod coupling integrity is required to ensure compliance with the analysis of the rod drop accident in the FSAR. The overtravel position feature provides the only positive means of determining that a rod is properly coupled and therefore this check must be performed prior to achieving criticality after completing CORE ALTERATIONS that could have affected the control rod drive coupling integrity. The subsequent check is performed as a backup to the initial demonstration.

In order to ensure that the control rod patterns can be followed and therefore that other parameters are within their limits, the control rod position indication system must be OPERABLE.

The control rod housing support restricts the outward movement of a control rod to less than 3 inches in the event of a housing failure. The amount of rod reactivity which could be added by this small amount of rod withdrawal is less than a normal withdrawal increment and will not contribute to any damage to the primary coolant system. The support is not required when there is no pressure to act as a driving force to rapidly eject a drive housing.

The required surveillance intervals are adequate to determine that the rods are OPERABLE and not so frequent as to cause excessive wear on the system components.

3/4.1.4 CONTROL ROD PROGRAM CONTROLS

Control rod withdrawal and insertion sequences are established to assure that the maximum insequence individual control rod or control rod segments which are withdrawn at any time during the fuel cycle could not be worth enough to result in a peak fuel enthalpy greater than 280 cal/gm in the event of a control rod drop accident. The specified sequences are characterized by homogeneous, scattered patterns of control rod withdrawal. When THERMAL POWER is greater than 20% of RATED THERMAL POWER, there is no possible rod worth which, if dropped at the design rate of the velocity limiter, could result in a peak enthalpy of 280 cal/gm. Thus requiring the RSCS and RWM to be OPERABLE when THERMAL POWER is less than or equal to 20% of RATED THERMAL POWER provides adequate control.

The RSCS and RWM provide automatic supervision to assure that out-of-sequence rods will not be withdrawn or inserted.

The analysis of the rod drop accident is presented in Section 15.4.9 of the FSAR and the techniques of the analysis are presented in a topical report, Reference 1, and two supplements, References 2 and 3.

The RBM is designed to automatically prevent fuel damage in the event of erroneous rod withdrawal from locations of high power density during high power operation. Two channels are provided. Tripping one of the channels will block erroneous rod withdrawal soon enough to prevent fuel damage. This system backs up the written sequence used by the operator for withdrawal of control rods.

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

SAFETY EVALUATION

AMENDMENT NO. 4 TO NPF-18

LA SALLE COUNTY STATION, UNIT 2

DOCKET NO. 50-374

Introduction

License Condition 2.C.(7) of License NPF-18 required installation of instrumentation that would automatically, shut down the reactor (in startup and refueling modes only) in the event of low control rod drive pump discharge pressure. This condition was to have been satisfied prior to completion of the startup test program.

By letter dated May 24, 1984, Commonwealth Edison Company (licensee) requested a license amendment which was issued as Amendment No. 3 effective July 24, 1984, and was intended to show compliance with License Condition 2.C.(7) and provide the necessary Technical Specifications to assure proper operation of the new scram capability.

By letter dated July 31, 1984, as supplemented by letters dated August 1 & 2, 1984, the licensee requested an exigent amendment to vacate Amendment No. 3 to the license, citing the fact that, on July 30, 1984 while testing the modification, spurious scrams occurred, indicating that with the existing trip setpoints, the modification cannot yet be declared fully operable, pending identification and correction of the cause of these scrams. Accordingly, the licensee has requested that the changes to the license and the Technical Specification authorized by Amendment No. 3 to NPF-18 be delayed until completion of the startup test program, as was originally provided for in License Condition 2.C.(7).

Evaluation

Prior to issuance of License Amendment No. 3, License condition 2.C.(7) allowed completion of the startup test program before installation of this additional scram capability was required. The startup test program is expected to be completed in October 1984 at which time the instrumentation must be installed. Therefore, vacating License Amendment No. 3, which reflected installation of the instrumentation, does not constitute a significant hazards consideration.

Since this plant modification has caused spurious reactor trips, we agree that it is prudent not to require the instrumentation to be operable until this problem is resolved and appropriate corrective action is taken. In the interim, to avoid spurious scrams while resolving the problem, the licensee intends to simply jumper out the low pressure scram logic, thus isolating this installed modification from the plant's reactor protection system. Therefore, the system and Technical Specifications will be the same as it was prescribed in the original license. We find this acceptable. The NRC will inspect these jumpers to assure that they have been properly installed.

8410170035 840921
PDR ADOCK 05000374
P PDR

Also, since appropriate corrective action to resolve the spurious scrams may involve numeric changes in the instrument set points, we concur that the prudent course of action is to vacate Amendment No. 3 and thus reinstate License Condition 2.C.(7).

No Significant Hazards Determination

The basis for the previous determination of no significant hazards concerning the proposed amendment remains unchanged. The plant will be the same as it was prior to issuance of Amendment No. 3. Amendment No. 3 was intended to both satisfy the License Condition 2.C.(7) and provide the necessary Technical Specifications to assure proper operation of a scram capability required which the licensee discovered to be premature. License Condition 2.C.(7) granted until the completion of the startup test program for completion of this additional scram capability. The startup test program is expected to be completed in October 1984 at which time the licensee must demonstrate operability of the instrumentation. The NRC staff concludes the proposed amendment will not:

(1) Involve a significant increase in the probability or consequence of an accident previously evaluated because the plant is as originally licensed, and the amendment only provides additional time for operability of scram instrumentation required by license condition.

(2) Create the possibility of a new or different kind of accident from any previously evaluated because the accident analysis is unchanged by a delay in operability of the instrumentation to completion of startup testing.

(3) Involve a significant reduction in the margin of safety since the plant remain the same as originally licensed.

The staff has determined that a timely application for the proposed change was made and that exigent circumstances do exist and were not any fault of the licensee. The exigent circumstances result from the fact that the time is needed to confirm the cause of the problem and to establish an adequate basis for the proper scram setpoint. By requesting an exigent review, the licensee is avoiding a possible emergency amendment request in the event the facility were to incur a shutdown in the immediate future, and this new scram capibility would allow the Unit to restart due to additional spurious scrams.

Environmental Consideration

This amendment which vacates Amendment No. 3 to NPF-18 and reinstates License Condition 2.C.(7) involves a change in the time of installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite and that there is no significant

Issuance of Amendment No. 4 to Facility Operating License No. NPF-18
La Salle County Station, Unit 2

DISTRIBUTION
Docket File
NRC PDR
Local PDR
PRC System
NSIC
LB#2 Reading
PShuttleworth
ABournia
TNovak
JSaltzman, SAB
CWoodhead, OELD
OMiles
HDenton
JRutberg
AToalston
WMiller, LFMB
NGrace
ÉJordan
LHarman
DBrinkman, SSPB
TBarnhart(4)
EHylton