Docket File
PDR

JUL 24 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. 3 to Facility Operating License No. NPF-18
La Salle County Station, Unit 2

The Nuclear Regulatory Commission has issued the enclosed Amendment No. 3 to Facility Operating License No. NPF-18 for La Salle County Station, Unit 2. This amendment is in response to Commonwealth Edison Company's letter dated May 24, 1984, which you submitted to revise the La Salle County Station, Unit 2 Technical Specifications to reflect a reactor scram on low control rod drive pump discharge pressure modification as required for completion by License Condition 2.C.(7). This amendment (1) deletes License Condition 2.C.(7), and (2) changes the Technical Specifications to incorporate the reactor scram on low control rod drive pump discharge pressure modification.

A copy of the related safety evaluation supporting Amendment No. 3 to Facility Operating License NPF-18 is enclosed.

Sincerely,

RC Martin
for A. Schwencer, Chief
Licensing Branch No. 2
Division of Licensing

Enclosures:
1. Amendment No. 3 to NPF-18
2. Staff Evaluation

cc w/ enclosures:
See next page

DL:LB#2/PM	DL:LB#2/LA	DL:LB#2/BC	OELD
ABournia:bdm	EHylton	ASchwencer	CWoodhead
7/20/84	7/20/84	7/23/84	7/24/84

JUL 24 1984

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

FOR THE NUCLEAR REGULATORY COMMISSION

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

Enclosure:
Changes to the Technical
Specifications

Date of Issuance: July 24, 1984

DL:LB#2/PM	DL:LB#2/LA	DL:LB#2/BC	OELD
ABournia:bdm	EHylton	ASchwencer	CWoodhead
7/20/84*	7/20/84*	7/23/84*	7/24/84*

* See previous concurrence.

JUL 24 1984

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FOR THE NUCLEAR REGULATORY COMMISSION

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

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with minor changes to SER

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

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R E Martin

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cc w/ enclosures:
See next page

La Salle

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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. 3
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 May 24, 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. 3, 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 deleted.

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

FOR THE NUCLEAR REGULATORY COMMISSION

R E Martin
A. Schwencer, Chief
Licensing Branch No. 2
Division of Licensing

Enclosure:
Changes to the Technical
Specifications

Date of Issuance: July 24, 1984

ENCLOSURE TO LICENSE AMENDMENT NO.3
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	$\leq$ 120 divisions of full scale	$\leq$ 122 divisions of full scale
2. Average Power Range Monitor:		
a. Neutron Flux-High, Setdown	$\leq$ 15% of RATED THERMAL POWER	$\leq$ 20% of RATED THERMAL POWER
b. Flow Biased Simulated Thermal Power - Upscale		
1) Two Recirculation Loop Operation		
a) Flow Biased	$\leq$ 0.66W + 51% with a maximum of	$\leq$ 0.66W + 54% with a maximum of
b) High Flow Clamped	$\leq$ 113.5% of RATED THERMAL POWER	$\leq$ 115.5% of RATED THERMAL POWER
2) Single Recirculation Loop Operation		
a) Flow Biased	$\leq$ 0.66W + 45.7% with a maximum of	$\leq$ 0.66W + 48.7% with a maximum of
b) High Flow Clamped	$\leq$ 113.5% of RATED THERMAL POWER	$\leq$ 115.5% of RATED THERMAL POWER
c. Fixed Neutron Flux-High	$\leq$ 118% of RATED THERMAL POWER	$\leq$ 120% of RATED THERMAL POWER
3. Reactor Vessel Steam Dome Pressure - High	$\leq$ 1043 psig	$\leq$ 1063 psig
4. Reactor Vessel Water Level - Low, Level 3	$\geq$ 12.5 inches above instrument zero*	$\geq$ 11 inches above instrument zero*
5. Main Steam Line Isolation Valve - Closure	$\leq$ 8% closed	$\leq$ 12% closed
6. Main Steam Line Radiation - High	$\leq$ 3 x full power background	$\leq$ 3.6 x full power background
7. Primary Containment Pressure - High	$\leq$ 1.69 psig	$\leq$ 1.89 psig
8. Scram Discharge Volume Water Level - High	$\leq$ 767' 5¼"	$\leq$ 767' 5¼"
9. Turbine Stop Valve - Closure	$\leq$ 5% closed	$\leq$ 7% closed
10. Turbine Control Valve Fast Closure, Trip Oil Pressure - Low	$\geq$ 500 psig	$\geq$ 414 psig
11. Reactor Mode Switch Shutdown Position	N.A.	N.A.
12. Manual Scram	N.A.	N.A.
13. Control Rod Drive		
a. Charging Water Header Pressure-Low	$\geq$ 1267 psig	$\geq$ 1185 psig
b. Delay Timer	$\leq$ 10 seconds	$\leq$ 10 seconds

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

REACTOR PROTECTON SYSTEM INSTRUMENTATION SETPOINTS (Continued)

13. Control Rod Drive (CRD) Charging Water Header Pressure - Low

The Hydraulic Control Unit (HCU) scram accumulator is precharged with high pressure nitrogen (N_2). When the Control Rod Drive (CRD) pump is activated, the pressurized charging water forces the accumulator piston down to mechanical stops. The piston is maintained seated against this mechanical stop with normal charging water pressure, typically above 1400 psig. If the charging water header pressure decreases below the N_2 pressure, such as would be the case with high leakage through the check valves of the CRD charging water lines, the accumulator piston would eventually rise off its stops. This results in a reduction of the accumulator energy and thereby degrades normal scram performance of the CRD's in the absence of sufficient reactor pressure.

The CRD low charging water header pressure trip setpoint initiates a scram at the charging water header pressure which assures the seating of the accumulator piston. With this trip setpoint, full accumulator capability, and therefore, normal scram performance, is assured at all reactor pressures. An adjustable time-delay relay is provided for each pressure transmitter/trip channel to protect against inadvertant scram due to pressure fluctuations in the charging line.

Four channels of pressure transmitter/pressure indicating switch combinations measure the charging water header pressure using one-out-of-two twice logic. The trip function is automatically bypassed in RUN mode because reactor pressure is available there to assist the CRD scram action. A keylock switch bypass is available in the SHUTDOWN and REFUEL modes to allow the scram reset of the RPS and to establish nominal/CRD valve line up.

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.

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, 5[h]	2 2	1 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 3, 4 5	1 1 1	1 7 3
12. Manual Scram	1, 2 3, 4 5	1 1 1	1 8 9
13. Control Rod Drive			
a. Charging Water Header Pressure - Low	2[k] 5[h]	2 2	1 3
b. Delay Timer	2[k] 5[h]	2 2	1 3

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.

(k) With reactor pressure < 950 psig.

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*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
13. Control Rod Drive	
a. Charging Water Header Pressure - Low	NA
b. Delay Timer	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
13. Control Rod Drive				
a. Charging Water Header Pressure - Low	NA	M	R	2, 5
b. Delay Timer	NA	M	R	2, 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.

CONTROL RODS (Continued)

In addition, the automatic CRD charging water header low pressure scram (see Table 2.2.1-1) initiates well before any accumulator loses its full capability to insert the control rod. With the added automatic scram feature, the surveillance of each individual accumulator check valve is no longer necessary to demonstrate adequate stored energy is available for normal scram action.

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.

SAFETY EVALUATION
AMENDMENT NO. 3 TO NPF-18
LA SALLE COUNTY STATION, UNIT 2
DOCKET NO. 50-374

Introduction

By letter dated May 24, 1984, Commonwealth Edison Company (licensee) proposed an amendment that would change the La Salle County Station, Unit 2 Technical Specifications to include a previously approved reactor trip setting on low control rod drive pump discharge water header pressure and delete an associated surveillance requirement. The proposed Technical Specification change is in accordance with the setpoints and conditions as discussed in Section 7.2.3.2 of Supplement No. 7 to the Safety Evaluation Report.

Evaluation

In response to a concern that the control rod drive accumulators do not maintain adequate pressure for a period of time compatible with operator action when the reactor is at less than operating pressure, the licensee proposed installation of an automatic reactor trip that would scram the control rods in the event of low control rod drive pump discharge pressure. This scram would be operational by the mode switch. This modification to the La Salle design was evaluated and approved by the NRC staff in Section 4.6.2 of Supplement No. 2 and Section 7.2.3.2 of Supplement No. 7 to the Safety Evaluation Report, and accordingly, a license condition was included in the license for its completion.

We have reviewed the Technical Specification limiting condition for operations and surveillance requirements for the proposed reactor trip instrumentation (Tables 3.3.1-1 and 4.3.1.1-1) and conclude that they are acceptable. The reactor trip on low control rod drive pump discharge water header pressure has been designed and is being installed in La Salle, Unit 2. The licensee also requested the deletion of surveillance requirement 4.1.3.5.b.2, which verifies that each accumulator maintains its pressure above the alarm setpoint for at least 10 minutes. This surveillance requirement will no longer be needed to insure that the reactor can be scrammed when the new trip is installed. Therefore, based upon our evaluations, we have determined that the Technical Specification changes, for addition of the reactor trip on low control rod drive pump discharge pressure and deletion of the surveillance requirement on accumulator pressure, are acceptable. Accordingly, we find that the licensee has complied with the condition set forth in license condition 2.C.(7) and that license condition 2.C.(7) be deleted.

Environmental Consideration

This amendment involves a change in the 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 increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that this amendment involves no significant hazards consideration and there has been no public comment on such finding. Accordingly, this amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR Sec 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of this amendment.

Conclusion

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

Dated: July 24, 1984