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

APR 30 1985

Docket Nos. 50-373/374

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

Dear Mr. Farrar:

SUBJECT: ISSUANCE OF AMENDMENT NO. 22 TO FACILITY OPERATING LICENSE NO. NPF-11 AND AMENDMENT NO. 10 TO FACILITY OPERATING LICENSE NO. NPF-18 - LA SALLE COUNTY STATION, UNITS 1 AND 2

The Nuclear Regulatory Commission has issued the enclosed Amendment No. 22 to Facility Operating License No. NPF-11 and Amendment No. 10 to Facility Operating License No. NPF-18 for the La Salle County Station, Units 1 and 2. These amendments are in response to your letter dated February 21, 1985. The amendments would revise the La Salle, Units 1 and 2 Technical Specifications to delete the channel check requirements from certain instruments.

A copy of the related safety evaluation supporting Amendment No. 22 to Facility Operating License NPF-11 and Amendment No. 10 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. 22 to NPF-11
2. Amendment No. 10 to NPF-18
3. Safety Evaluation

cc w/enclosures:
See next page

8505080207 850430
PDR ADOCK 05000373
P PDR

La Salle

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

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Docket Nos. 50-373/374

APR 30 1985

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

Dear Mr. Farrar:

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The Nuclear Regulatory Commission has issued the enclosed Amendment No. 22 to Facility Operating License No. NPF-11 and Amendment No. 10 to Facility Operating License No. NPF-18 for the La Salle County Station, Units 1 and 2. These amendments are in response to your letter dated February 21, 1985. The amendments would revise the La Salle, Units 1 and 2 Technical Specifications to delete the channel check requirements from certain instruments.

A copy of the related safety evaluation supporting Amendment No. 22 to Facility Operating License NPF-11 and Amendment No. 10 to Facility Operating License NPF-18 is enclosed.

Sincerely,

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

Enclosures:
1. Amendment No. 22 to NPF-11
2. Amendment No. 10 to NPF-18
3. Safety Evaluation

cc w/enclosures:
See next page

DISTRIBUTION
See next page

LB#2/DL/LA	LB#2/DL/PM	OELD	LB#2/DL/BC
EHylton	ABournia:lb		ASchwencer
04/15/85	04/18/85	04/22/85	04/18/85

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

COMMONWEALTH EDISON COMPANY

DOCKET NO. 50-373

LA SALLE COUNTY STATION, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment 22
License No. NPF-11

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 February 21, 1985, 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 attachment 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. 22 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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PDR ADOCK 05000373
P PDR

3. This amendment is effective as of November 30, 1985.

FOR THE NUCLEAR REGULATORY COMMISSION

A. Schwencer

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

Enclosure:
Changes to the Technical
Specifications

Date of Issuance: April 30, 1985

3. This amendment is effective as of November 30, 1985.

FOR THE NUCLEAR REGULATORY COMMISSION

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

Enclosure:
Changes to the Technical
Specifications

Date of Issuance: April 30, 1985

LB#2/DL/LA	LB#2/DL/PM	OELD	LB#2/DL/BC	AD/L/DL
EHylton	ABournia:lb		ASchwencer	TMNovak
04/18/85	04/18/85	04/22/85	04/18/85	04/24/85

ENCLOSURE TO LICENSE AMENDMENT NO. 22
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
3/4 3-7	3/4 3-7
3/4 3-20	3/4 3-20
3/4 3-22	3/4 3-22
3/4 3-32	3/4 3-32
3/4 3-33	3/4 3-33
3/4 3-34	3/4 3-34
3/4 3-49	3/4 3-49

TABLE 4.3.1.1-1

REACTOR PROTECTION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION[a]	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
1. Intermediate Range Monitors				
a. Neutron Flux - High	S/U[b], S	S/U[c], W	R	2
	S	W	R	3, 4, 5
b. Inoperative	NA	W	NA	2, 3, 4, 5
2. Average Power Range Monitor:[f]				
a. Neutron Flux - High, Setdown	S/U[b], S	S/U[c], W	SA	1, 2
	S	W	SA	3, 5
b. Flow Biased Simulated Thermal Power-Upscale	S, D[g]	S/U[c], W	W[d][e], SA, R[h]	1
c. Fixed Neutron Flux - High	S	S/U[c], W	W[d], SA	1
d. Inoperative	NA	W	NA	1, 2, 3, 5
3. Reactor Vessel Steam Dome Pressure - High	NA	M	Q	1, 2
4. Reactor Vessel Water Level - Low, Level 3	NA	M	R	1, 2
5. Main Steam Line Isolation Valve - Closure	NA	M	R	1
6. Main Steam Line Radiation - High	S	M	R	1, 2
7. Primary Containment Pressure - High	NA	M	Q	1, 2

TABLE 4.3.2.1-1

ISOLATION ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
A. AUTOMATIC INITIATION				
1. PRIMARY CONTAINMENT ISOLATION				
a. Reactor Vessel Water Level				
1) Low, Level 3	NA	M	R	1, 2, 3
2) Low Low, Level 2	NA	M	R	1, 2, 3
b. Drywell Pressure - High	NA	M	Q	1, 2, 3
c. Main Steam Line				
1) Radiation - High	S	M	R	1, 2, 3
2) Pressure - Low	NA	M	Q	1
3) Flow - High	NA	M	R	1, 2, 3
d. Main Steam Line Tunnel Temperature - High	NA	M	R	1, 2, 3
e. Condenser Vacuum - Low	NA	M	Q	1, 2*, 3*
f. Main Steam Line Tunnel Δ Temperature - High	NA	M	R	1, 2, 3
2. SECONDARY CONTAINMENT ISOLATION				
a. Reactor Building Vent Exhaust Plenum Radiation - High	S	M	R	1, 2, 3 and **
b. Drywell Pressure - High	NA	M	Q	1, 2, 3
c. Reactor Vessel Water Level - Low Low, Level 2	NA	M	R	1, 2, 3, and #
d. Fuel Pool Vent Exhaust Radiation - High	S	M	R	1, 2, 3 and **
3. REACTOR WATER CLEANUP SYSTEM ISOLATION				
a. Δ Flow - High	S	M	R	1, 2, 3
b. Heat Exchanger Area Temperature - High	NA	M	Q	1, 2, 3
c. Heat Exchanger Area Ventilation ΔT - High	NA	M	Q	1, 2, 3
d. SLCS Initiation	NA	R	NA	1, 2, 3
e. Reactor Vessel Water Level - Low Low, Level 2	NA	M	R	1, 2, 3

TABLE 4.3.2.1-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
6. RHR SYSTEM SHUTDOWN COOLING MODE ISOLATION				
a. Reactor Vessel Water Level - Low, Level 3	NA	M	R	1, 2, 3
b. Reactor Vessel (RHR Cut-in Permissive) Pressure - High	NA	M	Q	1, 2, 3
c. RHR Pump Suction Flow - High	NA	M	Q	1, 2, 3
d. RHR Area Temperature - High	NA	M	Q	1, 2, 3
e. RHR Equipment Area ΔT - High	NA	M	Q	1, 2, 3
B. MANUAL INITIATION				
1. Inboard Valves	NA	R	NA	1, 2, 3
2. Outboard Valves	NA	R	NA	1, 2, 3
3. Inboard Valves	NA	R	NA	1, 2, 3 and **,#
4. Outboard Valves	NA	R	NA	1, 2, 3 and **,#
5. Inboard Valves	NA	R	NA	1, 2, 3
6. Outboard Valves	NA	R	NA	1, 2, 3
7. Outboard Valve	NR	R	NA	1, 2, 3

*When reactor steam pressure > 1043 psig and/or any turbine stop valve is open.

**When handling irradiated fuel in the secondary containment and during CORE ALTERATIONS and operations with a potential for draining the reactor vessel.

#During CORE ALTERATIONS and operations with a potential for draining the reactor vessel.

TABLE 4.3.3.1-1

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
A. DIVISION I TRIP SYSTEM				
1. RHR-A (LPCI MODE) AND LPCS SYSTEM				
a. Reactor Vessel Water Level - Low Low Low, Level 1	NA	M	R	1, 2, 3, 4*, 5*
b. Drywell Pressure - High	NA	M	Q	1, 2, 3
c. LPCS Pump Discharge Flow-Low	NA	M	Q	1, 2, 3, 4*, 5*
d. LPCS and LPCI A Injection Valve Injection Line Pressure Low Interlock	NA	M	R	1, 2, 3, 4*, 5*
e. LCPS and LPCI A Injection Valve Reactor Pressure Low Interlock	NA	M	R	1, 2, 3, 4*, 5*
f. LPCI Pump A Start Time Delay Relay	NA	M	Q	1, 2, 3, 4*, 5*
g. LPCI Pump A Flow-Low	NA	M	Q	1, 2, 3, 4*, 5*
h. Manual Initiation	NA	R	NA	1, 2, 3, 4*, 5*
2. AUTOMATIC DEPRESSURIZATION SYSTEM TRIP SYSTEM "A"#				
a. Reactor Vessel Water Level - Low Low Low, Level 1	NA	M	R	1, 2, 3
b. Drywell Pressure-High	NA	M	Q	1, 2, 3
c. ADS Timer	NA	M	Q	1, 2, 3
d. Reactor Vessel Water Level - Low, Level 3	NA	M	R	1, 2, 3
e. LPCS Pump Discharge Pressure-High	NA	M	Q	1, 2, 3
f. LPCI Pump A Discharge Pressure-High	NA	M	Q	1, 2, 3
g. Manual Initiation	NA	R	NA	1, 2, 3

TABLE 4.3.3.1-1 (Continued)

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
B. DIVISION 2 TRIP SYSTEM				
1. RHR B AND C (LPCI MODE)				
a. Reactor Vessel Water Level - Low Low Low, Level 1	NA	M	R	1, 2, 3, 4*, 5*
b. Drywell Pressure - High	NA	M	Q	1, 2, 3
c. LPCI B and C Injection Valve Injection Line Pressure Low Interlock	NA	M	R	1, 2, 3, 4*, 5*
d. LPCI Pump B Start Time Delay Relay	NA	M	Q	1, 2, 3, 4*, 5*
e. LPCI Pump Discharge Flow-Low	NA	M	Q	1, 2, 3, 4*, 5*
f. Manual Initiation	NA	R	NA	1, 2, 3, 4*, 5*
g. LPCI B and C Injection Valve Reactor Pressure Low Interlock	NA	M	R	1, 2, 3, 4*, 5*
2. AUTOMATIC DEPRESSURIZATION SYSTEM TRIP SYSTEM "B"#				
a. Reactor Vessel Water Level - Low Low Low, Level 1	NA	M	R	1, 2, 3
b. Drywell Pressure-High	NA	M	Q	1, 2, 3
c. ADS Timer	NA	M	Q	1, 2, 3
d. Reactor Vessel Water Level - Low, Level 3	NA	M	R	1, 2, 3
e. LPCI Pump B and C Discharge Pressure-High	NA	M	Q	1, 2, 3
f. Manual Initiation	NA	R	NA	1, 2, 3

TABLE 4.3.3.1-1 (Continued)

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
C. DIVISION 3 TRIP SYSTEM				
1. HPCS SYSTEM				
a. Reactor Vessel Water Level - Low Low, Level 2	NA	M	R	1, 2, 3, 4*, 5*
b. Drywell Pressure-High	NA	M	Q	1, 2, 3
c. Reactor Vessel Water Level-High Level 8	NA	M	R	1, 2, 3, 4*, 5*
d. Condensate Storage Tank Level - Low	NA	M	Q	1, 2, 3, 4*, 5*
e. Suppression Pool Water Level - High	NA	M	Q	1, 2, 3, 4*, 5*
f. Pump Discharge Pressure-High	NA	M	Q	1, 2, 3, 4*, 5*
g. HPCS System Flow Rate-Low	NA	M	Q	1, 2, 3, 4*, 5*
h. Manual Initiation	NA	R	NA	1, 2, 3, 4*, 5*
D. LOSS OF POWER				
1. 4.16 kv Emergency Bus Under-voltage (Loss of Voltage)	NA	NA	R	1, 2, 3, 4**, 5**

\# Not required to be OPERABLE when reactor steam dome pressure is less than or equal to 122 psig.

* When the system is required to be OPERABLE after being manually realigned, as applicable, per Specification 3.5.2.

** Required when ESF equipment is required to be OPERABLE.

TABLE 4.3.5.1-1

REACTOR CORE ISOLATION COOLING SYSTEM ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNITS	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION
a. Reactor Vessel Water Level - Low Low, Level 2	NA	M	R
b. Reactor Vessel Water Level - High, Level 8	NA	M	R
c. Manual Initiation	NA	R	NA

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 10
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 February 21, 1985, 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 attachment 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. 10 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 as of November 30, 1985.

FOR THE NUCLEAR REGULATORY COMMISSION

A. Schwencer

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

Enclosure:
Changes to the Technical
Specifications

Date of Issuance: April 30, 1985

3. This amendment is effective as of November 30, 1985.

FOR THE NUCLEAR REGULATORY COMMISSION

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

Enclosure:
Changes to the Technical
Specifications

Date of Issuance: April 30, 1985

LB#2/DL/LA	LB#2/DL/PM	OELD	LB#2/DL/BC	AD/L/DL
EHylton	ABournia:lb		ASchwencer	TMNovak
04/16/85	04/18/85	04/22/85	04/18/85	04/[illegible]/85

ENCLOSURE TO LICENSE AMENDMENT NO. 10
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
3/4 3-7	3/4 3-7
3/4 3-20	3/4 3-20
3/4 3-22	3/4 3-22
3/4 3-32	3/4 3-32
3/4 3-33	3/4 3-33
3/4 3-34	3/4 3-34
3/4 3-49	3/4 3-49

TABLE 4.3.1.1-1

REACTOR PROTECTION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION(a)	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
1. Intermediate Range Monitors				
a. Neutron Flux - High	S/U(b), S	S/U(c), W	R	2
	S	W	R	3, 4, 5
b. Inoperative	NA	W	NA	2, 3, 4, 5
2. Average Power Range Monitor:(f)				
a. Neutron Flux - High, Setdown	S/U(b), S	S/U(c), W	SA	1, 2
	S	W	SA	3, 5
b. Flow Biased Simulated Thermal Power-Upscale	S, D(g)	S/U(c), W	W(d)(e), SA, R(h)	1
c. Fixed Neutron Flux - High	S	S/U(c), W	W(d), SA	1
d. Inoperative	NA	W	NA	1, 2, 3, 5
3. Reactor Vessel Steam Dome Pressure - High	NA	M	Q	1, 2
4. Reactor Vessel Water Level - Low, Level 3	NA	M	R	1, 2
5. Main Steam Line Isolation Valve - Closure	NA	M	R	1
6. Main Steam Line Radiation - High	S	M	R	1, 2
7. Primary Containment Pressure - High	NA	M	Q	1, 2

TABLE 4.3.2.1-1

ISOLATION ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
A. AUTOMATIC INITIATION				
1. PRIMARY CONTAINMENT ISOLATION				
a. Reactor Vessel Water Level				
1) Low, Level 3	NA	M	R	1, 2, 3
2) Low Low, Level 2	NA	M	R	1, 2, 3
b. Drywell Pressure - High	NA	M	Q	1, 2, 3
c. Main Steam Line				
1) Radiation - High	S	M	R	1, 2, 3
2) Pressure - Low	NA	M	Q	1
3) Flow - High	NA	M	R	1, 2, 3
d. Main Steam Line Tunnel Temperature - High	NA	M	R	1, 2, 3
e. Condenser Vacuum - Low	NA	M	Q	1, 2*, 3*
f. Main Steam Line Tunnel Δ Temperature - High	NA	M	R	1, 2, 3
2. SECONDARY CONTAINMENT ISOLATION				
a. Reactor Building Vent Exhaust Plenum Radiation - High	S	M	R	1, 2, 3 and **
b. Drywell Pressure - High	NA	M	Q	1, 2, 3
c. Reactor Vessel Water Level - Low Low, Level 2	NA	M	R	1, 2, 3, and #
d. Fuel Pool Vent Exhaust Radiation - High	S	M	R	1, 2, 3 and **
3. REACTOR WATER CLEANUP SYSTEM ISOLATION				
a. Δ Flow - High	S	M	R	1, 2, 3
b. Heat Exchanger Area Temperature - High	NA	M	Q	1, 2, 3
c. Heat Exchanger Area Ventilation ΔT - High	NA	M	Q	1, 2, 3
d. SLCS Initiation	NA	R	NA	1, 2, 3
e. Reactor Vessel Water Level - Low Low, Level 2	NA	M	R	1, 2, 3

TABLE 4.3.2.1-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
6. RHR SYSTEM SHUTDOWN COOLING MODE ISOLATION				
a. Reactor Vessel Water Level - Low, Level 3	NA	M	R	1, 2, 3
b. Reactor Vessel (RHR Cut-in Permissive) Pressure - High	NA	M	Q	1, 2, 3
c. RHR Pump Suction Flow - High	NA	M	Q	1, 2, 3
d. RHR Area Temperature - High	NA	M	Q	1, 2, 3
e. RHR Equipment Area ΔT - High	NA	M	Q	1, 2, 3
B. MANUAL INITIATION				
1. Inboard Valves	NA	R	NA	1, 2, 3
2. Outboard Valves	NA	R	NA	1, 2, 3
3. Inboard Valves	NA	R	NA	1, 2, 3 and **,#
4. Outboard Valves	NA	R	NA	1, 2, 3 and **,#
5. Inboard Valves	NA	R	NA	1, 2, 3
6. Outboard Valves	NA	R	NA	1, 2, 3
7. Outboard Valve	NA	R	NA	1, 2, 3

*When reactor steam pressure > 1043 psig and/or any turbine stop valve is open.

**When handling irradiated fuel in the secondary containment and during CORE ALTERATIONS and operations with a potential for draining the reactor vessel.

#During CORE ALTERATIONS and operations with a potential for draining the reactor vessel.

TABLE 4.3.3.1-1

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
A. DIVISION I TRIP SYSTEM				
1. RHR-A (LPCI MODE) AND LPCS SYSTEM				
a. Reactor Vessel Water Level - Low Low Low, Level 1	NA	M	R	1, 2, 3, 4*, 5*
b. Drywell Pressure - High	NA	M	Q	1, 2, 3
c. LPCS Pump Discharge Flow-Low	NA	M	Q	1, 2, 3, 4*, 5*
d. LPCS and LPCI A Injection Valve Injection Line Pressure Low Interlock	NA	M	R	1, 2, 3, 4*, 5*
e. LPCS and LCPI A Injection Valve Reactor Pressure Low Interlock	NA	M	R	1, 2, 3, 4*, 5*
f. LPCI Pump A Start Time Delay Relay	NA	M	Q	1, 2, 3, 4*, 5*
g. LPCI Pump A Flow-Low	NA	M	Q	1, 2, 3, 4*, 5*
h. Manual Initiation	NA	R	NA	1, 2, 3, 4*, 5*
2. AUTOMATIC DEPRESSURIZATION SYSTEM TRIP SYSTEM "A"#				
a. Reactor Vessel Water Level - Low Low Low, Level 1	NA	M	R	1, 2, 3
b. Drywell Pressure-High	NA	M	Q	1, 2, 3
c. ADS Timer	NA	M	Q	1, 2, 3
d. Reactor Vessel Water Level - Low, Level 3	NA	M	R	1, 2, 3
e. LPCS Pump Discharge Pressure-High	NA	M	Q	1, 2, 3
f. LPCI Pump A Discharge Pressure-High	NA	M	Q	1, 2, 3
g. Manual Initiation	NA	R	NA	1, 2, 3

TABLE 4.3.3.1-1 (Continued)

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
B. DIVISION 2 TRIP SYSTEM				
1. RHR B AND C (LPCI MODE)				
a. Reactor Vessel Water Level - Low Low Low, Level 1	NA	M	R	1, 2, 3, 4*, 5*
b. Drywell Pressure - High	NA	M	Q	1, 2, 3
c. LPCI B and C Injection Valve Injection Line Pressure Low Interlock	NA	M	R	1, 2, 3, 4*, 5*
d. LPCI Pump B Start Time Delay Relay	NA	M	Q	1, 2, 3, 4*, 5*
e. LPCI Pump Discharge Flow-Low	NA	M	Q	1, 2, 3, 4*, 5*
f. Manual Initiation	NA	R	NA	1, 2, 3, 4*, 5*
g. LPCI B and C Injection Valve Reactor Pressure Low Interlock	NA	M	R	1, 2, 3, 4*, 5*
2. AUTOMATIC DEPRESSURIZATION SYSTEM TRIP SYSTEM "B"#				
a. Reactor Vessel Water Level - Low Low Low, Level 1	NA	M	R	1, 2, 3
b. Drywell Pressure-High	NA	M	Q	1, 2, 3
c. ADS Timer	NA	M	Q	1, 2, 3
d. Reactor Vessel Water Level - Low, Level 3	NA	M	R	1, 2, 3
e. LPCI Pump B and C Discharge Pressure-High	NA	M	Q	1, 2, 3
f. Manual Initiation	NA	R	NA	1, 2, 3

TABLE 4.3.3.1-1 (Continued)

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
C. DIVISION 3 TRIP SYSTEM				
1. HPCS SYSTEM				
a. Reactor Vessel Water Level - Low Low, Level 2	NA	M	R	1, 2, 3, 4*, 5*
b. Drywell Pressure-High	NA	M	Q	1, 2, 3
c. Reactor Vessel Water Level-High Level 8	NA	M	R	1, 2, 3, 4*, 5*
d. Condensate Storage Tank Level - Low	NA	M	Q	1, 2, 3, 4*, 5*
e. Suppression Pool Water Level - High	NA	M	Q	1, 2, 3, 4*, 5*
f. Pump Discharge Pressure-High	NA	M	Q	1, 2, 3, 4*, 5*
g. HPCS System Flow Rate-Low	NA	M	Q	1, 2, 3, 4*, 5*
h. Manual Initiation	NA	R	NA	1, 2, 3, 4*, 5*
D. LOSS OF POWER				
1. 4.16 kV Emergency Bus Under-voltage (Loss of Voltage)	NA	NA	R	1, 2, 3, 4**, 5**
2. 4.16 kV Emergency Bus Under-voltage (Degraded Voltage) (Division 3)	NA	NA	R	1, 2, 3, 4**, 5**

TABLE NOTATIONS

#Not required to be OPERABLE when reactor steam dome pressure is less than or equal to 122 psig.

*When the system is required to be OPERABLE after being manually realigned, as applicable, per Specification 3.5.2.

**Required when ESF equipment is required to be OPERABLE.

TABLE 4.3.5.1-1

REACTOR CORE ISOLATION COOLING SYSTEM ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNITS	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION
a. Reactor Vessel Water Level - Low Low, Level 2	NA	M	R
b. Reactor Vessel Water Level - High, Level 8	NA	M	R
c. Manual Initiation	NA	R	NA

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

SAFETY EVALUATION

AMENDMENT NO. 22 TO NPF-11 AND

AMENDMENT NO. 10 TO NPF-18

LA SALLE COUNTY STATION, UNITS 1 & 2

DOCKET NOS. 50-373 AND 50-374

Introduction

By letter dated February 21, 1985, Commonwealth Edison Company (the licensee) proposed amendments requesting changes to the La Salle Units 1 and 2 Technical Specifications to delete channel check surveillance requirements for reactor vessel level and main steam line flow sensors (differential pressure switches) installed in the reactor protection system, primary containment isolation system, emergency core cooling system, and the reactor core isolation cooling system. These changes are the result of replacement of existing Barton differential pressure (dp) indicating switches that have local readout capability, with "blind" dp switches (i.e., local readout capability is not provided) manufactured by Static-O-Ring. Thus, the capability to perform channel checks no longer exists. The reason for the changeout of the dp switches is that the Barton switches were not environmentally qualified in accordance with the requirements of 10 CFR 50.49, "Environmental Qualification of Electrical Equipment Important to Safety for Nuclear Power Plants." A list of the specific instruments and the associated systems affected by the changeout is provided in Attachment 1.

Evaluation

The following definition is provided in the BWR Standard Technical Specifications (STS) for a channel check:

> A CHANNEL CHECK shall be the qualitative assessment of channel behavior during operation by observation. This determination shall include, where possible, comparison of the channel indication and/or status with other indications and/or status derived from independent instrument channels measuring the same parameter.

That is, a channel check is a comparison of a group of instrument channel readouts/displays, typically control room indicators, for a given monitored parameter (e.g., reactor vessel water level). An instrument channel readout that differs significantly from the readouts of the remaining instrument channels is indicative of a channel malfunction. The STS typically require that channel checks be performed once each operating shift (8 hrs.) for those protection system instruments having readout capability. The performance of a channel check provides a quick and easy method for detecting gross instrument failures in the non-conservative direction (i.e., failures away from the trip setpoint) between the surveillance intervals of other more extensive tests (e.g., monthly channel functional tests) which would detect the failure.

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A gross failure in the conservative direction would typically be detected in the form of a channel trip (i.e., the setpoint value would be exceeded). For those instruments for which readout/display capability is not provided, periodic surveillance in the form of channel checks is not required. Channel checks are not relied on to ensure the operability of protection system equipment. This is accomplished by more extensive testing; channel functional tests and channel calibrations. A channel functional test involves the injection of a simulated signal into the sensor to verify operability including alarm and/or trip setpoint functions. A channel calibration is the adjustment, as necessary, of the instrument channel output such that it responds with the necessary range and accuracy to known values of the parameter which the channel monitors. It is the combination of these periodic surveillance activities which ensures that protection system instrument channels remain operable, and thus, that this portion of the protection system remains in compliance with the single failure criterions.

The performance of channel checks does not increase protection system reliability, although they may result in increased availability of individual instrument channels. Protection system reliability is achieved and maintained by the design and installation of equipment that satisfies safety criteria set forth in the Commission's regulations (e.g., diversity of parameters, single failure, equipment qualification, channel independence, etc.), and has the capability of being tested and calibrated while retaining the capability to accomplish protective functions. The staff does not require protection system sensors to be provided with readout/display capability in order to perform instrument channel checks. If readout capability is provided, then channel check surveillance is typically required by plant technical specifications because of the potential benefit gained (i.e., early detection and repair of a failed instrument channel) from a surveillance activity that is simple to perform and does not require significant time or manpower on behalf of the utility. It is noted that requirements exist for the display of information in the control room to allow the operator(s) to assess the status of the plant and the protection system, and to perform manual actions required for safety. However, these generic requirements are independent of plant specific capability for performing channel checks.

Channel checks surveillance requirements are typically associated with analog instrument channels that provide 4-20mA signals to the control room. These signals or corresponding voltage signals, typically 1-5V, are usually displayed on meters/indicators at the main control boards or back row instrument cabinets. The comparison of indicators monitoring the same parameter by observation constitutes the channel check. The indicating dp switches used to sense reactor vessel water level and main steam line flow at La Salle Units 1 & 2 only provide a digital/bistable signal (i.e., the value of the monitored parameter is either above or below the trip setpoint) to the protection system cabinets in the control room. There is no associated display capability in the control room which allow the operator(s) to determine how close the monitored parameter value is to the trip setpoint. The dp switches are located at instrument racks in different areas of the reactor building. At these racks, the dp switches have an associated mechanical indication consisting of a linkage assembly connected to a torque tube (connected to the monitored process) at one end, and a pointer and dial assembly at the other end. Channel checks are currently performed by comparison of these local mechanical indications. Although not as accurate or convenient as channel checks of analog channel readouts from the control room, channel check surveillance was still required because of the potential benefits discussed above. Following replacement of the Barton indicating dp switches with the environmentally qualified Static-O-Ring dp switches, the capability to perform instrument channel checks will no longer exist for those instruments identified in Attachment 1. The "blind" Static-O-Ring dp switches are completely sealed; no mechanical indication is provided. The licensee has thus requested to delete channel check surveillance requirements for these instruments from the La Salle Units 1 & 2 Technical Specifications.

Based on the NRC staff's review of information provided by the licensee, and a review of the bases for channel check surveillance requirements in the STS, the staff concludes that the proposed revisions to the La Salle Units 1 & 2 Technical Specifications to delete channel check surveillance requirements for those instruments identified in Attachment 1, are acceptable. The licensee has committed to continue to perform channel checks each shift on all instruments to be replaced up to the time of replacement. This approach is acceptable to the staff.

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 and changes a surveillance requirement. 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 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

The Commission made a proposed determination that the amendment involves no significant hazards consideration which was published in the Federal Register (50 FR 12141) on March 27, 1985. No public comments were received.

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: April 30, 1985

ATTACHMENT 1

LIST OF AFFECTED INSTRUMENTS

INSTRUMENT NUMBER	MONITORED PARAMETER	AFFECTED SYSTEM(S) & FUNCTIONS
1(2)B21-N026A,B,C,&D	Reactor Vessel Level	Primary and Secondary Containment Isolation, and RWCU system isolation on low level (level 2)
1(2)B21-N024A,B,C,&D	Reactor Vessel Level	Reactor Scram, and Primary Containment and RHR system Shutdown Cooling Mode Isolation on low level (level 3)
1(2)E31-N008A,B,C,&D	Main Steam Line Flow	Primary Containment Isolation on High Flow
1(2)E31-N009A,B,C,&D	Main Steam Line Flow	Primary Containment Isolation on High Flow
1(2)E31-N010A,B,C,&D	Main Steam Line Flow	Primary Containment Isolation on High Flow
1(2)E31-N011A,B,C,&D	Main Steam Line Flow	Primary Containment Isolation on High Flow
1(2)B21-N037A&C	Reactor Vessel Level	Initiation of ADS (Div. 1), LPCS, and LPCI A on low level (level 1); and RCIC initiation (level 2).
1(2)B21-N037B&D	Reactor Vessel Level	Initiation of ADS (Div. 2) and LPCI B&C on low level (level 1); and RCIC initiation (level 2)
1(2)B21-N101A&B	Reactor Vessel Level	RCIC termination on high level (level 8)
1(2)B21-N038A	Reactor Vessel Level	ADS (Div. 1) low level permissive (level 3)

INSTRUMENT NUMBER	MONITORED PARAMETER	AFFECTED SYSTEM(S) & FUNCTIONS
1(2)B21-N038B	Reactor Vessel Level	ADS (Div. 2) low level permissive (level 3)
1(2)B21-N100A&B	Reactor Vessel Level	HPCS termination on high level (level 8)
1(2)B21-N031A,B,C,&D	Reactor Vessel Level	HPCS initiation on low level (level 2)