May 6, 1987

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Docket Nos: 50-373
and 50-374

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.50 to Facility Operating License No. NPF-11 and Amendment No. 33 to Facility Operating License No. NPF-18 - La Salle County Station, Units 1 and 2

The U.S. Nuclear Regulatory Commission has issued the enclosed Amendment No. 50 to Facility Operating License No. NPF-11 and Amendment No. 33 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 October 23, 1986, and as supplemented by letters dated November 5, 1986, and March 6, 1987.

The amendments revise the La Salle County Station, Units 1 and 2 Technical Specifications to change the Group I Main Steam Isolation Valves' closure signal from Reactor Pressure Vessel Level 2 to Level 1.

A copy of the related safety evaluation supporting Amendment No. 50 to Facility Operating License No. NPF-11 and Amendment No.33 to Facility Operating License No. NPF-18 is enclosed.

Sincerely,

Original sign by/

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

Enclosures:
1. Amendment No. 50 to NPF-11
2. Amendment No. 33 to NPF-18
3. Safety Evaluation

cc w/enclosure:
See next page

*Previously Concurred

LA:BWD-3:DBL	PDIII-2 P.S	PDIII-2	OGC-Bethesda*
EHylton/vag*	PShemanski	DMuller	MYoung
04/01/87	05/5/87	05/6/87	4/8/87

8705110014 870506
PDR ADOCK 05000373
P PDR

Mr. Dennis L. Farrar
Commonwealth Edison Company

La Salle County Nuclear Power Station
Units 1 & 2

cc:
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Assistant Attorney General
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Chairman
La Salle County Board of Supervisors
La Salle County Courthouse
Ottawa, Illinois 61350

Attorney General
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Nuclear Facility Safety
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Regional Administrator, Region III
U. S. Nuclear Regulatory Commission
799 Rossevelt Road
Glen Ellyn, Illinois 60137

Local Public Document Room location: Public Library of Illinois Valley Community College, Rural Route No. 1, Oglesby, Illinois 61348.

Elinor G. Adensam, Director
BWR Project Directorate No. 3
Division of BWR Licensing

BWD-3:DBL
ABournia/vag
03/ 01/87

LA:BWD-3:DBL
EHylton
04/1 /87

OGC
MYoung
4/8/87

D:BWD-3:DBL
EAdensam
04/3 /87

Docket Nos: 50-373
and 50-374

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. to Facility Operating License No. NPF-11 and Amendment No. to Facility Operating License No. NPF-18 - La Salle County Station, Units 1 and 2

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 No. NPF-18 for the La Salle County Station, Units 1 and 2. These amendments are in response to your letter dated October 23, 1986, and as supplemented by letters dated November 5, 1986, and March 6, 1987.

The amendments revise the La Salle County Station, Units 1 and 2 Technical Specifications to change the Group I Main Steam Isolation Valves' closure signal from Reactor Pressure Vessel Level 2 to Level 1.

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

Sincerely,

Elinor G. Adensam, Director
BWR Project Directorate No. 3
Division of BWR Licensing

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

cc w/enclosure:
See next page

LA:BWD-3:DBL	BWD-3:DBL	D:BWD-3:DBL
EHylton/vag	ABournia	EAdensam
04/1/87	04/01/87	04/3/87

3. This amendment is effective upon date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Elinor G. Adensam, Director
BWR Project Directorate No. 3
Division of BWR Licensing

Enclosure:
Changes to the Technical
Specifications

Date of Issuance:

LA:BWD-3:DBL	aB BWD-3:DBL	MHY/ wanted revision to SE OGC	D:BWD-3:DBL
EHylton/vag	ABournia	MYoung	EAdensam
03/ 1 /87	03/ 01/87	04/ 8 /87	03/ /87

3. This amendment is effective upon date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Elinor G. Adensam, Director
BWR Project Directorate No. 3
Division of BWR Licensing

Enclosure:
Changes to the Technical
Specifications

Date of Issuance:

LA:BWD-3:DBL	BWD-3:DBL	OGC	D:BWD-3:DBL
EHylton/vag	ABournia	MYoung	EAdensam
03/1/87	03/01/87	04/8/87	03/ /87

approved revision to SE

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 No.50
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 October 23, 1986, as supplemented by letters dated November 5, 1986, and March 6, 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 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.50 , 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: May 6, 1987

ENCLOSURE TO LICENSE AMENDMENT NO. 50

FACILITY OPERATING LICENSE NO. NPF-11

DOCKET NO. 50-373

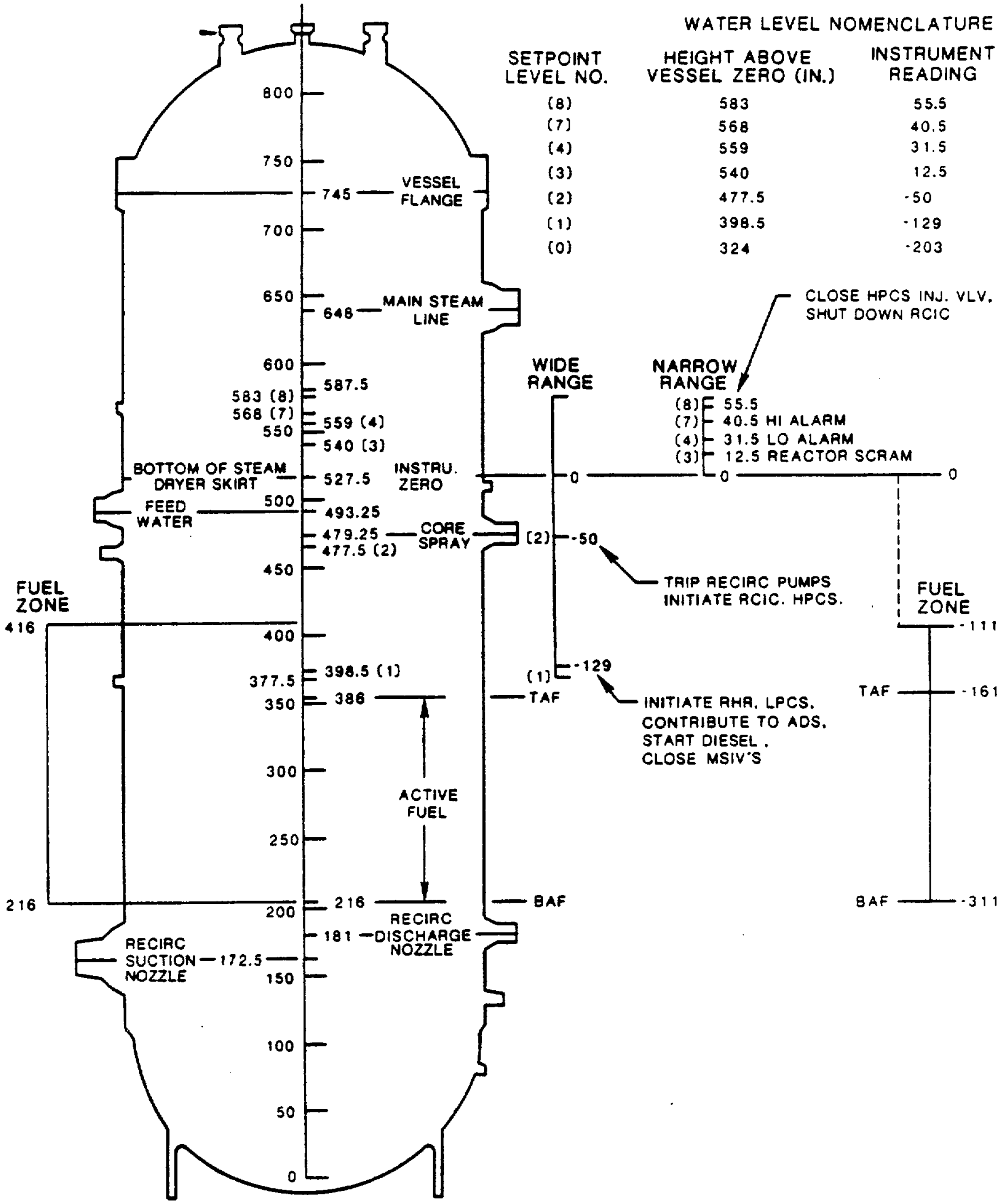
Replace the following pages of the Appendix A Technical Specifications with the enclosed pages. The revised pages are identified by Amendment number and contain a vertical line indicating the area of change.

REMOVE	INSERT
3/4 3-11	3/4 3-11
3/4 3-15	3/4 3-15
3/4 3-18	3/4 3-18
3/4 3-20	3/4 3-20
3/4 6-26	3/4 6-26
B3/4 3-7	B3/4 3-7

NOTE: SCALE IN INCHES ABOVE VESSEL ZERO

WATER LEVEL NOMENCLATURE

SETPOINT LEVEL NO.	HEIGHT ABOVE VESSEL ZERO (IN.)	INSTRUMENT READING
(8)	583	55.5
(7)	568	40.5
(4)	559	31.5
(3)	540	12.5
(2)	477.5	-50
(1)	398.5	-129
(0)	324	-203



REACTOR VESSEL WATER LEVEL

BASES FIGURE B 3/4 3-1

LA SALLE - UNIT 1

B 3/4 3-7

TABLE 3.3.2-1

ISOLATION ACTUATION INSTRUMENTATION

TRIP FUNCTION	VALVE GROUPS OPERATED BY SIGNAL (a)	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM (b)	APPLICABLE OPERATIONAL CONDITION	ACTION
A. AUTOMATIC INITIATION				
1. PRIMARY CONTAINMENT ISOLATION				
a. Reactor Vessel Water Level				
(1) Low, Level 3	7	2	1, 2, 3	20
(2) Low Low, Level 2	2, 3	2	1, 2, 3	20
(3) Low Low Low, Level 1	1, 10	2	1, 2, 3	20
b. Drywell Pressure - High	2, 7, 10	2	1, 2, 3	20
c. Main Steam Line				
1) Radiation - High	1	2	1, 2, 3	21
	3	2	1, 2, 3	22
2) Pressure - Low	1	2	1	23
3) Flow - High	1	2/line(d)	1, 2, 3	21
d. Main Steam Line Tunnel Temperature - High	1	2	1, 2, 3	21
e. Main Steam Line Tunnel ΔTemperature - High	1	2	1(i), 2(i), 3(i)	21
f. Condenser Vacuum - Low	1	2	1, 2*, 3*	21
2. SECONDARY CONTAINMENT ISOLATION				
a. Reactor Building Vent Exhaust Plenum Radiation - High	4(c)(e)	2	1, 2, 3 and **	24
b. Drywell Pressure - High	4(c)(e)	2	1, 2, 3	24
c. Reactor Vessel Water Level - Low Low, Level 2	4(c)(e)	2	1, 2, 3, and #	24
d. Fuel Pool Vent Exhaust Radiation - High	4(c)(e)	2	1, 2, 3, and **	24

TABLE 3.3.2-2

ISOLATION ACTUATION INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
A. AUTOMATIC INITIATION		
1. PRIMARY CONTAINMENT ISOLATION		
a. Reactor Vessel Water Level		
1) Low, Level 3	≥ 12.5 inches*	≥ 11.0 inches*
2) Low Low, Level 2	≥ -50 inches*	≥ -57 inches*
3) Low Low Low, Level 1	≥ -129 inches*	≥ -136 inches*
b. Drywell Pressure - High	≤ 1.69 psig	≤ 1.89 psig
c. Main Steam Line		
1) Radiation - High	≤ 3.0 x full power background	≤ 3.6 x full background
2) Pressure - Low	≥ 854 psig	≥ 834 psig
3) Flow - High	≤ 111 psid	≤ 116 psid
d. Main Steam Line Tunnel Temperature - High	≤ 140°F	≤ 146°F
e. Main Steam Line Tunnel Δ Temperature - High	≤ 36°F	≤ 42°F
f. Condenser Vacuum - Low	> 7 inches Hg vacuum	> 5.5 inches Hg vacuum
2. SECONDARY CONTAINMENT ISOLATION		
a. Reactor Building Vent Exhaust Plenum Radiation - High	≤ 10 mr/hr	≤ 15 mr/hr
b. Drywell Pressure - High	≤ 1.69 psig	≤ 1.89 psig
c. Reactor Vessel Water Level - Low Low, Level 2	≥ -50 inches*	≥ -57 inches*
d. Fuel Pool Vent Exhaust Radiation - High	≤ 10 mr/hr	≤ 15 mr/hr
3. REACTOR WATER CLEANUP SYSTEM ISOLATION		
a. Δ Flow - High	≤ 70 gpm	≤ 87.5 gpm
b. Heat Exchanger Area Temperature - High	≤ 181°F	≤ 187°F
c. Heat Exchanger Area Ventilation ΔT - High	≤ 85°F	≤ 91°F
d. SLCS Initiation	NA	NA
e. Reactor Vessel Water Level - Low Low, Level 2	≥ -50 inches*	≥ -57 inches*

TABLE 3.3.2-3

ISOLATION SYSTEM INSTRUMENTATION RESPONSE TIME

TRIP FUNCTION	RESPONSE TIME (Seconds)#
A. AUTOMATIC INITIATION	
1. PRIMARY CONTAINMENT ISOLATION	
a. Reactor Vessel Water Level	
1) Low, Level 3	NA
2) Low Low, Level 2	$\leq$ 13[(a)]
3) Low Low Low, Level 1	$\leq$ 1.0*/$\leq$ 13[(a)]**
b. Drywell Pressure - High	$\leq$ 13[(a)]
c. Main Steam Line	
1) Radiation - High[(b)]	$\leq$ 1.0*/$\leq$ 13[(a)]**
2) Pressure - Low	$\leq$ 2.0*/$\leq$ 13[(a)]**
3) Flow - High	$\leq$ 0.5*/$\leq$ 13[(a)]**
d. Main Steam Line Tunnel Temperature - High	NA
e. Condenser Vacuum - Low	NA
f. Main Steam Line Tunnel Δ Temperature - High	NA
2. SECONDARY CONTAINMENT ISOLATION	
a. Reactor Building Vent Exhaust Plenum Radiation - High[(b)]	$\leq$ 13[(a)]
b. Drywell Pressure - High	$\leq$ 13[(a)]
c. Reactor Vessel Water Level - Low, Level 2	$\leq$ 13[(a)]
d. Fuel Pool Vent Exhaust Radiation - High[(b)]	$\leq$ 13[(a)]
3. REACTOR WATER CLEANUP SYSTEM ISOLATION	
a. Δ Flow - High	$\leq$ 13[(a)]##
b. Heat Exchanger Area Temperature - High	NA
c. Heat Exchanger Area Ventilation ΔT-High	NA
d. SLCS Initiation	NA
e. Reactor Vessel Water Level - Low Low, Level 2	$\leq$ 13[(a)]
4. REACTOR CORE ISOLATION COOLING SYSTEM ISOLATION	
a. RCIC Steam Line Flow - High	$\leq$ 13[(a)]###
b. RCIC Steam Supply Pressure - Low	$\leq$ 13[(a)]
c. RCIC Turbine Exhaust Diaphragm Pressure - High	NA
d. RCIC Equipment Room Temperature - High	NA
e. RCIC Steam Line Tunnel Temperature - High	NA
f. RCIC Steam Line Tunnel Δ Temperature - High	NA
g. Drywell Pressure - High	NA
h. RCIC Equipment Room Δ Temperature - High	NA
5. RHR SYSTEM STEAM CONDENSING MODE ISOLATION	
a. RHR Equipment Area Δ Temperature - High	NA
b. RHR Area Cooler Temperature - High	NA
c. RHR Heat Exchanger Steam Supply Flow High	NA

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
3) Low Low Low, Level 1	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 3.6.3-1 (Continued)

PRIMARY CONTAINMENT ISOLATION VALVES

VALVE FUNCTION AND NUMBER	VALVE GROUP[(a)]	MAXIMUM ISOLATION TIME (Seconds)
Automatic Isolation Valves (Continued)		
11. Containment Monitoring Valves	2	$\leq$ 5
1CM017A,B		
1CM018A,B		
1CM019A,B		
1CM020A,B		
1CM021B[(h)]		
1CM022A[(h)]		
1CM025A[(h)]		
1CM026B[(h)]		
1CM027		
1CM028		
1CM029		
1CM030		
1CM031		
1CM032		
1CM033		
1CM034		
12. Drywell Pneumatic Valves		
1IN001A and B	10	$\leq$ 30
1IN017	10	$\leq$ 22
1IN074	10	$\leq$ 22
1IN075	10	$\leq$ 22
1IN031	2	$\leq$ 5
13. RHR Shutdown Cooling Mode Valves	6	
1E12-F008		$\leq$ 40
1E12-F009		$\leq$ 40
1E12-F023		$\leq$ 90
1E12-F053 A and B		$\leq$ 29
1E12-F099A and B[(g)(i)]		$\leq$ 30

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. 33
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 October 23, 1986, as supplemented by letters dated November 6, 1986, and March 6, 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 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. 33, 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: May 6, 1987

ENCLOSURE TO LICENSE AMENDMENT NO.33

FACILITY OPERATING LICENSE NO. NPF-18

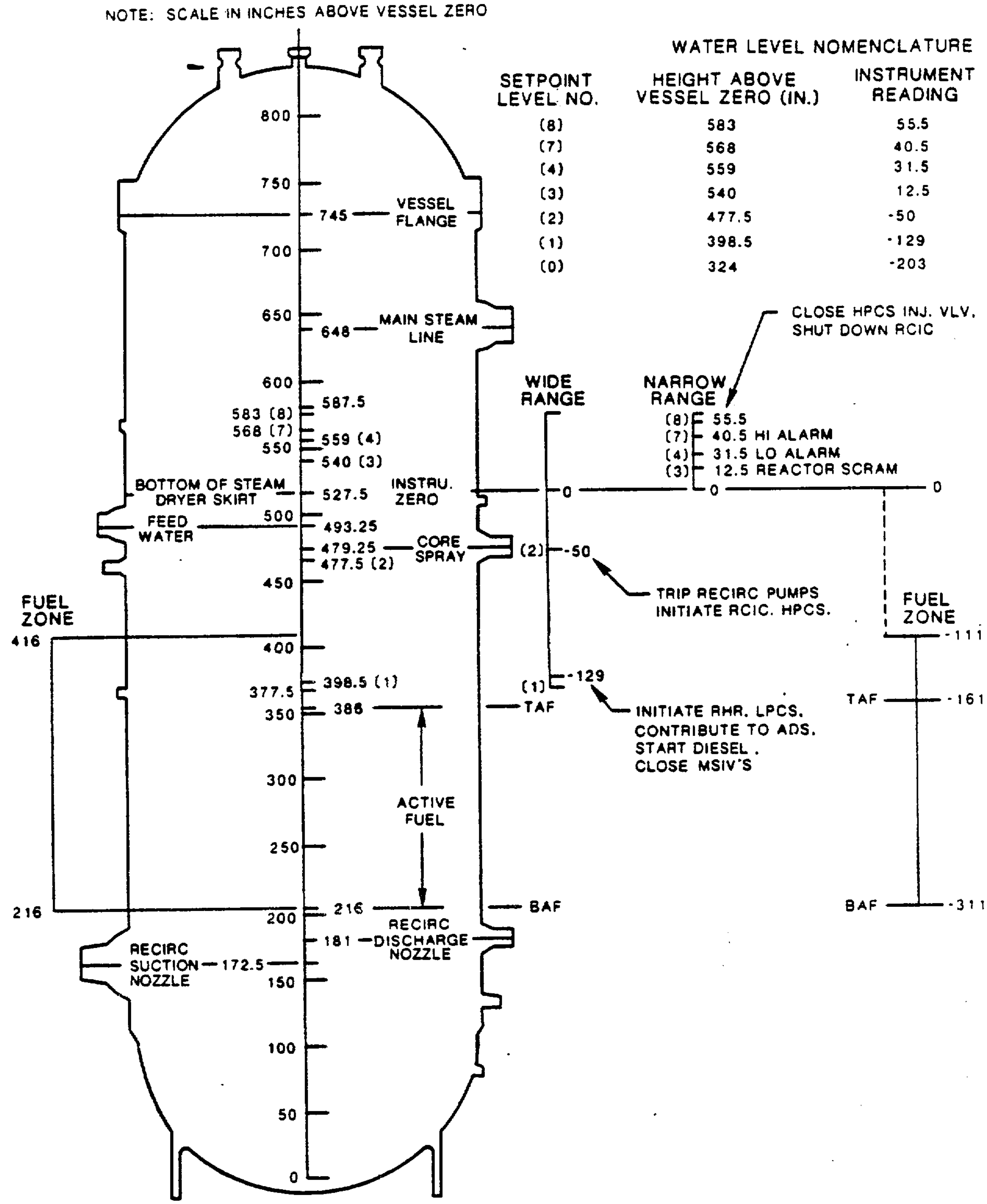
DOCKET NO. 50-374

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 3-11	3/4 3-11
3/4 3-15	3/4 3-15
3/4 3-18	3/4 3-18
3/4 3-20	3/4 3-20
3/4 6-29	3/4 6-29
B3/4 3-7	B3/4 3-7

WATER LEVEL NOMENCLATURE

SETPOINT LEVEL NO.	HEIGHT ABOVE VESSEL ZERO (IN.)	INSTRUMENT READING
(8)	583	55.5
(7)	568	40.5
(4)	559	31.5
(3)	540	12.5
(2)	477.5	-50
(1)	398.5	-129
(0)	324	-203



REACTOR VESSEL WATER LEVEL

BASES FIGURE B 3/4 3-1

LA SALLE - UNIT 2

B 3/4 3-7

TABLE 3.3.2-1

ISOLATION ACTUATION INSTRUMENTATION

TRIP FUNCTION	VALVE GROUPS OPERATED BY SIGNAL (a)	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM (b)	APPLICABLE OPERATIONAL CONDITION	ACTION
A. AUTOMATIC INITIATION				
1. PRIMARY CONTAINMENT ISOLATION				
a. Reactor Vessel Water Level				
(1) Low, Level 3	7	2	1, 2, 3	20
(2) Low Low, Level 2	2, 3	2	1, 2, 3	20
(3) Low Low Low, Level 1	1, 10	2	1, 2, 3	20
b. Drywell Pressure - High	2, 7, 10	2	1, 2, 3	20
c. Main Steam Line				
1) Radiation - High	1	2	1, 2, 3	21
	3	2	1, 2, 3	22
2) Pressure - Low	1	2	1	23
3) Flow - High	1	2/line(d)	1, 2, 3	21
d. Main Steam Line Tunnel Temperature - High	1	2	1, 2, 3	21
e. Main Steam Line Tunnel Δ Temperature - High	1	2	1(i), 2(i), 3(i)	21
f. Condenser Vacuum - Low	1	2	1, 2*, 3*	21
2. SECONDARY CONTAINMENT ISOLATION				
a. Reactor Building Vent Exhaust Plenum Radiation - High	4(c)(e)	2	1, 2, 3 and **	24
b. Drywell Pressure - High	4(c)(e)	2	1, 2, 3	24
c. Reactor Vessel Water Level - Low Low, Level 2	4(c)(e)	2	1, 2, 3, and #	24
d. Fuel Pool Vent Exhaust Radiation - High	4(c)(e)	2	1, 2, 3, and **	24

TABLE 3.3.2-2

ISOLATION ACTUATION INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
A. AUTOMATIC INITIATION		
1. PRIMARY CONTAINMENT ISOLATION		
a. Reactor Vessel Water Level		
1) Low, Level 3	≥ 12.5 inches*	≥ 11.0 inches*
2) Low Low, Level 2	≥ -50 inches*	≥ -57 inches*
3) Low Low Low, Level 1	≥ -129 inches*	≥ -136 inches*
b. Drywell Pressure - High	≤ 1.69 psig	≤ 1.89 psig
c. Main Steam Line		
1) Radiation - High	≤ 3.0 x full power background	≤ 3.6 x full background
2) Pressure - Low	≥ 854 psig	≥ 834 psig
3) Flow - High	≤ 111 psid	≤ 116 psid
d. Main Steam Line Tunnel Temperature - High	≤ 140°F	< 146°F
e. Main Steam Line Tunnel Δ Temperature - High	≤ 36°F	≤ 42°F
f. Condenser Vacuum - Low	≥ 7 inches Hg vacuum	≥ 5.5 inches Hg vacuum
2. SECONDARY CONTAINMENT ISOLATION		
a. Reactor Building Vent Exhaust Plenum Radiation - High	≤ 10 mr/h	≤ 15 mr/h
b. Drywell Pressure - High	≤ 1.69 psig	≤ 1.89 psig
c. Reactor Vessel Water Level - Low Low, Level 2	≥ -50 inches*	≥ -57 inches*
d. Fuel Pool Vent Exhaust Radiation - High	≤ 10 mr/h	≤ 15 mr/h
3. REACTOR WATER CLEANUP SYSTEM ISOLATION		
a. ΔFlow - High	≤ 70 gpm	≤ 87.5 gpm
b. Heat Exchanger Area Temperature - High	≤ 181°F	≤ 187°F
c. Heat Exchanger Area Ventilation ΔT - High	≤ 85°	≤ 91°F
d. SLCS Initiation	N.A.	N.A.
e. Reactor Vessel Water Level - Low Low, Level 2	≥ -50 inches*	≥ -57 inches*

TABLE 3.3.2-3

ISOLATION SYSTEM INSTRUMENTATION RESPONSE TIME

TRIP FUNCTION	RESPONSE TIME (Seconds)#
A. AUTOMATIC INITIATION	
1. PRIMARY CONTAINMENT ISOLATION	
a. Reactor Vessel Water Level	
1) Low, Level 3	N.A
2) Low Low, Level 2	≤13(a)
3) Low Low Low, Level 1	≤1.0*/≤13.0(a)**
b. Drywell Pressure - High	≤13(a)
c. Main Steam Line	
1) Radiation - High(b)	≤1.0*/≤13(a)**
2) Pressure - Low	≤2.0*/≤13(a)**
3) Flow - High	≤0.5*/≤13(a)**
d. Main Steam Line Tunnel Temperature - High	N.A.
e. Condenser Vacuum - Low	N.A.
f. Main Steam Line Tunnel Δ Temperature - High	N.A.
2. SECONDARY CONTAINMENT ISOLATION	
a. Reactor Building Vent Exhaust Plenum Radiation - High(b)	≤13(a)
b. Drywell Pressure - High	≤13(a)
c. Reactor Vessel Water Level - Low, Level 2	≤13(a)
d. Fuel Pool Vent Exhaust Radiation - High(b)	≤13(a)
3. REACTOR WATER CLEANUP SYSTEM ISOLATION	
a. Δ Flow - High	≤13(a)##
b. Heat Exchanger Area Temperature - High	N.A.
c. Heat Exchanger Area Ventilation ΔT-High	N.A.
d. SLCS Initiation	N.A
e. Reactor Vessel Water Level - Low Low, Level 2	≤13(a)
4. REACTOR CORE ISOLATION COOLING SYSTEM ISOLATION	
a. RCIC Steam Line Flow - High	≤13(a)###
b. RCIC Steam Supply Pressure - Low	≤13(a)
c. RCIC Turbine Exhaust Diaphragm Pressure - High	N.A.
d. RCIC Equipment Room Temperature - High	N.A.
e. RCIC Steam Line Tunnel Temperature - High	N.A.
f. RCIC Steam Line Tunnel ΔTemperature - High	N.A.
g. Drywell Pressure - High	N.A.
h. RCIC Equipment Room Δ Temperature - High	N.A.
5. RHR SYSTEM STEAM CONDENSING MODE ISOLATION	
a. RHR Equipment Area ΔTemperature - High	N.A.
b. RHR Area Cooler Temperature - High	N.A.
c. RHR Heat Exchanger Steam Supply Flow High	N.A.

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
3) Low Low Low, Level 1	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 3.6.3-1 (Continued)

PRIMARY CONTAINMENT ISOLATION VALVES

VALVE FUNCTION AND NUMBER	VALVE GROUP[a]	MAXIMUM ISOLATION TIME (Seconds)
Automatic Isolation Valves (Continued)		
11. Containment Monitoring Valves	2	≤ 5
2CM017A,B		
2CM018A,B		
2CM019A,B		
2CM020A,B		
2CM021B[h]		
2CM022A[h]		
2CM025A[h]		
2CM026B[h]		
2CM027		
2CM028		
2CM029		
2CM030		
2CM031		
2CM032		
2CM033		
2CM034		
12. Drywell Pneumatic Valves		
2IN001A and B	10	≤ 30
2IN017	10	≤ 22
2IN074	10	≤ 22
2IN075	10	≤ 22
2IN031	2	≤ 5
13. RHR Shutdown Cooling Mode Valves	6	
2E12-F008		≤ 40
2E12-F009		≤ 40
2E12-F023		≤ 90
2E12-F053 A and B		≤ 29
2E12-F099A and B[g][i]		≤ 30

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 50 TO FACILITY OPERATING LICENSE NO. NPF-11 AND

AMENDMENT NO.33 TO FACILITY OPERATING LICENSE NO. NPF-18

COMMONWEALTH EDISON COMPANY

LA SALLE COUNTY STATION, UNITS 1 AND 2

DOCKET NOS. 50-373 AND 50-374

1.0 INTRODUCTION

By letter dated October 23, 1986, (Ref. 1), as supplemented November 5, 1986 (Ref. 2) and March 6, 1987 (Ref. 3), Commonwealth Edison Company (licensee) proposed a change to the La Salle County Station, Units 1 & 2 Technical Specifications (TS). The main purpose of this change is to reduce challenges to the safety relief valves (SRVs). This would be accomplished by changing the water level setpoint for closure of the main steam isolation valves (MSIVs) and main steam line drain valves (MSLDVs) from Level 2 to the lower Level 1. The probability of closing an MSIV due to variation of water level following a scram would thus be reduced. With the MSIVs open and the main condenser available, prime mover steam for the reactor feed pump turbines will also remain available to allow continued operation of the condensate and main feedwater system. This will aid in water level recovery before the low level isolation setpoint is reached.

Should Level 1 be reached, MSIV closure will cause reactor pressure to rise causing the SRVs to open and discharge to the suppression pool. The new heat load for the suppression pool will be reduced by the amount of heat picked up by the main condenser.

The MSIVs fail closed upon loss of pneumatic (nitrogen) supply. The nitrogen supply is presently isolated from the MSIVs at Level 2; and, therefore, its setpoint will also have to be lowered to isolate at Level 1.

The November 5, 1986 and the March 6, 1987 submittals provided analyses to support the proposed changes. Thus the action noticed in the Federal Register on November 19, 1986 and the staff's proposed no significant hazards consideration determination were not affected by these submittals.

2.0 EVALUATION

The licensee request to lower the water level trip setpoint for MSIV closure is consistent with staff solutions for NUREG-0737, item II.K.3.16, "Reduction of Challenges and Failures of Relief Valves". In addition, the General Electric Information Letter, SIL No. 367, also recommends the lower trip setpoint. The staff evaluation of the change in setpoint is discussed below.

2.1 LOSS OF COOLANT ACCIDENT (LOCA) AND MAIN STEAM LINE BREAK (MSLB)

2.1.1 LOCA

To justify the setpoint change, LOCA analyses for the Design Basis Accident (DBA), small break, and feedwater line break were performed. At the request of the staff, the licensee submitted a report (1-83-008) which included the LOCA analyses (Ref. 2). Our review of the report revealed that the emergency core cooling system (ECCS) evaluation used the RETRAN-02, MOD2 computer code. This code is not staff approved as an Appendix K-ECCS evaluation model. In addition, the analysis was performed by Quadrex; it also is not an approved user of the code. The staff stated that the portion of the Quadrex report relating to the above breaks would be accepted, if, on a LOCA comparison basis, the peak cladding temperature (PCT) with the lower MSIV level trip point did not increase. The remaining alternative was to use a staff approved ECCS evaluation model. The licensee chose the latter alternative.

By letter dated March 3, 1987 (Ref. 3), the licensee submitted a new LOCA safety evaluation to justify changing the MSIV water level isolation set point. Presently, the most limiting LOCA, the one that results in the highest peak cladding temperature and determines the maximum average planar linear heat generation rate (MAPLHGR) limit, is the recirculation suction line break DBA. ECCS calculations were performed using the staff approved codes of SAFE, REFLOOD and CHASTE. The effects of the proposed lower setpoint for large, intermediate and small break LOCAs were considered.

The licensee stated that large and intermediate LOCA events would not be affected by the setpoint change. For these events, there would be a rapid depressurization and inventory loss within the reactor vessel resulting in a fast actuation of the MSIVs. It was concluded that the lower MSIV setpoint would not significantly increase the reactor core inventory loss, the total core uncovery time or subsequent fuel heatup, or the radiation release to the environment. Thus, the setpoint change would not affect the consequences of design basis accidents. The staff accepts these findings.

For a small break LOCA there is a potential of initiation of MSIV closure at the proposed lower level setpoint which results in raising the peak cladding temperature (PCT). This event was analyzed. The results show that increase in PCT is less than 30° F. The highest small break LOCA PCT would be substantially less than the 2200° F limit. The results of the LOCA analyses show that the DBA remains unchanged. Therefore, the MAPLHGR will not be changed. We find this acceptable.

2.1.2 MSLB

MSLB was not analyzed. The Quadrex report stated that the MSIV closure is initiated by high steam flow rather than by water level. We find this acceptable.

2.2 ABNORMAL OPERATIONAL TRANSIENTS

Abnormal transient events were not evaluated by the licensee. Based on previous reviews with a lowered MSIV setpoint level, the staff does not consider these events to have any adverse effect with regard to the reactor safety performance.

2.3 NITROGEN SUPPLY

To allow the MSIVs to isolate at Level 1, the nitrogen supply to MSIVs must also actuate Level 1. This actuation is also changed from Level 2 to Level 1. We find this to be acceptable.

2.4 TECHNICAL SPECIFICATION (TS) CHANGES

We have reviewed the following TS changes:

TABLE 3.3.2-1	for isolation actuation instrumentation
TABLE 3.3.2-2	for isolation actuation instrumentation setpoints
TABLE 3.3.2-3	for isolation system instrumentation response time
TABLE 4.3.2.1-1	for isolation actuation instrumentation surveillance requirements
TABLE 3.6.3-1	for primary containment isolation valves
Bases for figure B 3/4 3-1	for reactor vessel water levels

We conclude that the proposed changes are acceptable.

3.0 ENVIRONMENTAL CONSIDERATION

These amendments involve changes in the installation and use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes in surveillance requirements. The staff has determined that these amendments involve 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 these amendments involve no significant hazards consideration and there has been no public comment on such finding. Accordingly, these amendments meet 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 these amendments.

4.0 CONCLUSION

The Commission made a proposed determination that the amendments involve no significant hazards consideration which was published in the FEDERAL REGISTER (51 FR 41847) on November 19, 1986, and consulted with the state of Illinois. No public comments were received, and the state of Illinois did not have any comments.

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.

Principal Contributor: D. Katze, RSB

Dated: May 6, 1987

REFERENCES

1. Letter from C. Allen Commonwealth Edison to H. R. Denton, USNRC, dated October 23, 1986.

2. Letter from C. Allen, Commonwealth Edison to H. R. Denton, USNRC, dated November 5, 1986.

3. Letter from C. Allen, Commonwealth Edison to H. R. Denton, USNRC, dated March 6, 1987.