February 25, 1994

Docket Nos. 50-373
and 50-374

DISTRIBUTION:
Docket File
J. Roe
J. Dyer
A. Gody, Jr.
D. Hagan
C. Grimes
OPA
G. Hill (4)
I. Ahmed
NRC & Local PDRs
J. Zwolinski
T. Clark
OGC
PDIII-2 r/f
ACRS (10)
OC/LFDCB
B. Clayton RIII

Mr. D. L. Farrar, Manager
Nuclear Regulatory Services
Commonwealth Edison Company
Executive Towers West III, Suite 500
1400 OPUS Place
Downers Grove, Illinois 60515

Dear Mr. Farrar:

SUBJECT: ISSUANCE OF AMENDMENTS LASALLE COUNTY STATION, UNITS 1 AND 2 (TAC NOS. M88620 AND M88606)

The Commission has issued the enclosed Amendment No. 95 to Facility Operating License No. NPF-11 and Amendment No. 79 to Facility Operating License No. NPF-18 for the LaSalle County Station, Units 1 and 2, respectively. The amendments are in response to your application for exigent license amendment dated January 28, 1994.

The amendments minimize unnecessary testing for certain instruments in the Reactor Protection System and the End-of Cycle Recirculation Pump Trip system for LaSalle County Station, Units 1 and 2 Technical Specifications.

A copy of the Safety Evaluation is also enclosed. The Notice of Issuance will be included in the Commission's biweekly Federal Register notice.

Sincerely,

Original signed by:

Anthony T. Gody, Jr., Project Manager
Project Directorate III-2
Division of Reactor Projects - III/IV/V
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 95 to NPF-11
2. Amendment No. 79 to NPF-18
3. Safety Evaluation

cc w/enclosures:
See next page

OFC	LA:PDIII-2	PM:PDIII-2	D:PDIII-2	OGC		
NAME	TCLARK	AGODY	JDYER	R Bachmann		
DATE	2/22/94	2/22/94	2/22/94	2/23/94		
COPY	YES/NO	YES/NO	YES/NO	YES/NO	YES/NO	YES/NO

070082

9403080157 940225
PDR ADOCK 05000373
P PDR

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

February 25, 1994

Docket Nos. 50-373
and 50-374

Mr. D. L. Farrar, Manager
Nuclear Regulatory Services
Commonwealth Edison Company
Executive Towers West III, Suite 500
1400 OPUS Place
Downers Grove, Illinois 60515

Dear Mr. Farrar:

SUBJECT: ISSUANCE OF AMENDMENTS LASALLE COUNTY STATION, UNITS 1 AND 2 (TAC NOS. M88620 AND M88606)

The Commission has issued the enclosed Amendment No. 95 to Facility Operating License No. NPF-11 and Amendment No. 79 to Facility Operating License No. NPF-18 for the LaSalle County Station, Units 1 and 2, respectively. The amendments are in response to your application for exigent license amendment dated January 28, 1994.

The amendments minimize unnecessary testing for certain instruments in the Reactor Protection System and the End-of Cycle Recirculation Pump Trip system for LaSalle County Station, Units 1 and 2 Technical Specifications.

A copy of the Safety Evaluation is also enclosed. The Notice of Issuance will be included in the Commission's biweekly Federal Register notice.

Sincerely,

Anthony T. Gody, Jr.

Anthony T. Gody, Jr., Project Manager
Project Directorate III-2
Division of Reactor Projects - III/IV/V
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 95 to NPF-11
2. Amendment No. 79 to NPF-18
3. Safety Evaluation

cc w/enclosures:
See next page

Mr. D. L. Farrar
Commonwealth Edison Company

LaSalle County Station
Unit Nos. 1 and 2

cc:

Phillip P. Steptoe, Esquire
Sidley and Austin
One First National Plaza
Chicago, Illinois 60603

Assistant Attorney General
100 West Randolph Street
Suite 12
Chicago, Illinois 60601

Resident Inspector/LaSalle, NPS
U. S. Nuclear Regulatory Commission
Rural Route No. 1
P. O. Box 224
Marseilles, Illinois 61341

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

Attorney General
500 South 2nd Street
Springfield, Illinois 62701

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

Illinois Department of Nuclear Safety
Office of Nuclear Facility Safety
1035 Outer Park Drive
Springfield, Illinois 62704

Regional Administrator
U. S. NRC, Region III
801 Warrenville Road
Lisle, Illinois 60532-4351

Robert Neuman
Office of Public Counsel
State of Illinois Center
100 W. Randolph
Suite 11-300
Chicago, Illinois 60601

Robert Cushing
Chief, Public Utilities Division
Illinois Attorney General's Office
100 West Randolph Street
Chicago, Illinois 60601

Michael I. Miller, Esquire
Sidley and Austin
One First National Plaza
Chicago, Illinois 60690

LaSalle Station Manager
LaSalle County Station
Rural Route 1
P. O. Box 220
Marseilles, Illinois 61341

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

COMMONWEALTH EDISON COMPANY

DOCKET NO. 50-373

LASALLE COUNTY STATION, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 95
License No. NPF-11

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

   A. The application for amendment filed by the Commonwealth Edison Company (the licensee), dated January 28, 1994, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's regulations set forth in 10 CFR Chapter I;

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

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

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

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

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the enclosure to this license amendment and paragraph 2.C.(2) of the Facility Operating License No. NPF-11 is hereby amended to read as follows:

(2) <u>Technical Specifications and Environmental Protection Plan</u>

The Technical Specifications contained in Appendix A, as revised through Amendment No. 95 , 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

James E. Dyer

James E. Dyer, Director
Project Directorate III-2
Division of Reactor Projects - III/IV/V
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: February 25, 1994

ATTACHMENT TO LICENSE AMENDMENT NO. 95

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 3-8	3/4 3-8
3/4 3-44	3/4 3-44
B 3/4 3-1	B 3/4 3-1
B 3/4 3-3	B 3/4 3-3
-	B 3/4 3-3a

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	Q	R	1
6. Main Steam Line Radiation - High	S	M	R	1, 2
7. Primary Containment Pressure - High	NA	M	Q	1, 2

TABLE 4.3.1.1-1 (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	Q	R	1
10. Turbine Control Valve Fast Closure Valve Trip System Oil Pressure - Low	NA	Q	R	1
11. Reactor Mode Switch Shutdown Position	NA	R	NA	1, 2, 3, 4, 5
12. Manual Scram	NA	W	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 levels calculated by a heat balance during OPERATIONAL CONDITION 1 when THERMAL POWER ≥ 25% of RATED THERMAL POWER. The APRM Gain Adjustment Factor (GAF) for any channel shall be equal to the power value determined by the heat balance divided by the APRM reading for that channel.

Within 2 hours, adjust any APRM channel with a GAF > 1.02. In addition, adjust any APRM channel within 12 hours, (1) if power is greater than or equal to 90% of RATED THERMAL POWER and the APRM channel GAF is < 0.98, or (2) if power is less than 90% of RATED THERMAL POWER and the APRM reading exceeds the power value determined by the heat balance by more than 10% of RATED THERMAL POWER. Until any required APRM adjustment has been accomplished, notification shall be posted on the reactor control panel.
(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).
(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.

TABLE 4.3.4.2.1-1

END-OF-CYCLE RECIRCULATION PUMP TRIP SYSTEM SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION
1. Turbine Stop Valve-Closure	Q	R
2. Turbine Control Valve-Fast Closure	Q	R

3/4.3.1 REACTOR PROTECTION SYSTEM INSTRUMENTATION

The reactor protection system automatically initiates a reactor scram to:

a. Preserve the integrity of the fuel cladding.

b. Preserve the integrity of the reactor coolant system.

c. Minimize the energy which must be adsorbed following a loss-of-coolant accident, and

d. Prevent inadvertent criticality.

This specification provides the limiting conditions for operation necessary to preserve the ability of the system to perform its intended function even during periods when instrument channels may be out of service because of maintenance. When necessary, one channel may be made inoperable for brief intervals to conduct required surveillance.

The reactor protection system is made up of two independent trip systems. There are usually four channels to monitor each parameter with two channels in each trip system. The outputs of the channels in a trip system are combined in a logic so that either channel will trip that trip system. The tripping of both trip systems will produce a reactor scram. The system meets the intent of IEEE-279, 1971, for nuclear power plant protection systems. Specified surveillance intervals for MSIV-Closure, TSV-Closure, TCV-Closure, and the Manual Scram have been determined in accordance with NEDC-30851P-A, "Technical Specification Improvement Analyses for BWR Reactor Protection System", March 1988. The bases for the trip settings of the RPS are discussed in the bases for Specification 2.2.1.

The measurement of response time at the specified frequencies provides assurance that the protective functions associated with each channel are completed within the time limit assumed in the accident analysis. No credit was taken for those channels with response times indicated as not applicable. Response time may be demonstrated by any series of sequential, overlapping or total channel test measurement, provided such tests demonstrate the total channel response time as defined. Sensor response time verification may be demonstrated by either (1) inplace, onsite or offsite test measurements, or (2) utilizing replacement sensors with certified response times.

3/4.3.4 RECIRCULATION PUMP TRIP ACTUATION INSTRUMENTATION

The anticipated transient without scram (ATWS) recirculation pump trip system provides a means of limiting the consequences of the unlikely occurrence of a failure to scram during an anticipated transient. The response of the plant to this postulated event falls within the envelope of study events in General Electric Company Topical Report NEDO-10349, dated March 1971 and NEDO-24222, dated December, 1979, and Appendix G of the FSAR.

The end-of-cycle recirculation pump trip (EOC-RPT) system is a part of the Reactor Protection System and is an essential safety supplement to the reactor trip. The purpose of the EOC-RPT is to recover the loss of thermal margin which occurs at the end-of-cycle. The physical phenomenon involved is that the void reactivity feedback due to a pressurization transient can add positive reactivity to the reactor system at a faster rate than the control rods add negative scram reactivity. Each EOC-RPT system trips both recirculation pumps, reducing coolant flow in order to reduce the void collapse in the core during two of the most limiting pressurization events. The two events for which the EOC-RPT protective feature will function are closure of the turbine stop valves and fast closure of the turbine control valves.

A generic analysis, which provides for continued operation with one or both trip systems of the EOC-RPT system inoperable, has been performed. The analysis determined bounding cycle independent MINIMUM CRITICAL POWER RATIO (MCPR) Limiting Condition for Operation (LCO) values which must be used if the EOC-RPT system is inoperable. These values ensure that adequate reactivity margin to the MCPR safety limit exists in the event of the analyzed transient with the RPT function inoperable. The analysis results are further discussed in the bases for Specification 3.2.3.

A fast closure sensor from each of two turbine control valves provides input to the EOC-RPT system; a fast closure sensor from each of the other two turbine control valves provides input to the second EOC-RPT system. Similarly, a position switch for each of two turbine stop valves provides input to one EOC-RPT system; a position switch from each of the other two stop valves provides input to the other EOC-RPT system. For each EOC-RPT system, the sensor relay contacts are arranged to form a 2-out-of-2 logic for the fast closure of turbine control valves and a 2-out-of-2 logic for the turbine stop valves. The operation of either logic will actuate the EOC-RPT system and trip both recirculation pumps.

Each EOC-RPT system may be manually bypassed by use of a keyswitch which is administratively controlled. The manual bypasses and the automatic Operating Bypass at less than 30% of RATED THERMAL POWER are annunciated in the control room.

Specified surveillance intervals have been determined in accordance with the following:

1. NEDC-30851P-A, "Technical Specification Improvement Analyses for BWR Reactor Protection System", March 1988.

3/4.3.4 RECIRCULATION PUMP TRIP ACTUATION INSTRUMENTATION

2. GENE-770-06-1-A, "Bases for Changes to Surveillance Test Intervals and Allowed Out-of-Service Times for Selected Instrumentation Technical Specifications", December 1992.

The EOC-RPT system response time is the time assumed in the analysis between initiation of valve motion and complete suppression of the electric arc, i.e., 190 ms, less the time allotted for sensor response, i.e., 10 ms, and less the time allotted for breaker arc suppression determined by test, as correlated to manufacturer's test results, i.e., 83 ms, and plant pre-operational test results.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

COMMONWEALTH EDISON COMPANY

DOCKET NO. 50-374

LASALLE COUNTY STATION, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 79
License No. NPF-18

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

   A. The application for amendment filed by the Commonwealth Edison Company (the licensee), dated January 28, 1994, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's regulations set forth in 10 CFR Chapter I;

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

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

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

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

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

(2) <u>Technical Specifications and Environmental Protection Plan</u>

The Technical Specifications contained in Appendix A, as revised through Amendment No. 79 , 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

James E. Dyer

James E. Dyer, Director
Project Directorate III-2
Division of Reactor Projects - III/IV/V
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: February 25, 1994

ATTACHMENT TO LICENSE AMENDMENT NO. 79

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 3-8	3/4 3-8
3/4 3-44	3/4 3-44
B 3/4 3-1	B 3/4 3-1
B 3/4 3-3	B 3/4 3-3
-	B 3/4 3-3a

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	Q	R	1
6. Main Steam Line Radiation - High	S	M	R	1, 2
7. Primary Containment Pressure - High	NA	M	Q	1, 2

TABLE 4.3.1.1-1 (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	Q	R	1
10. Turbine Control Valve Fast Closure Valve Trip System Oil Pressure - Low	NA	Q	R	1
11. Reactor Mode Switch Shutdown Position	NA	R	NA	1, 2, 3, 4, 5
12. Manual Scram	NA	W	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 levels calculated by a heat balance during OPERATIONAL CONDITION 1 when THERMAL POWER $\geq$ 25% of RATED THERMAL POWER. The APRM Gain Adjustment Factor (GAF) for any channel shall be equal to the power value determined by the heat balance divided by the APRM reading for that channel.

Within 2 hours, adjust any APRM channel with a GAF > 1.02. In addition, adjust any APRM channel within 12 hours, (1) if power is greater than or equal to 90% of RATED THERMAL POWER and the APRM channel GAF is < 0.98, or (2) if power is less than 90% of RATED THERMAL POWER and the APRM reading exceeds the power value determined by the heat balance by more than 10% of RATED THERMAL POWER. Until any required APRM adjustment has been accomplished, notification shall be posted on the reactor control panel.

(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).
(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.

TABLE 4.3.4.2.1-1

END-OF-CYCLE RECIRCULATION PUMP TRIP SYSTEM SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION
1. Turbine Stop Valve Closure	Q	R
2. Turbine Control Valve-Fast Closure	Q	R

3/4.3.1 REACTOR PROTECTION SYSTEM INSTRUMENTATION

The reactor protection system automatically initiates a reactor scram to:

a. Preserve the integrity of the fuel cladding.

b. Preserve the integrity of the reactor coolant system.

c. Minimize the energy which must be adsorbed following a loss-of-coolant accident, and

d. Prevent inadvertent criticality.

This specification provides the limiting conditions for operation necessary to preserve the ability of the system to perform its intended function even during periods when instrument channels may be out of service because of maintenance. When necessary, one channel may be made inoperable for brief intervals to conduct required surveillance.

The reactor protection system is made up of two independent trip systems. There are usually four channels to monitor each parameter with two channels in each trip system. The outputs of the channels in a trip system are combined in a logic so that either channel will trip that trip system. The tripping of both trip systems will produce a reactor scram. The system meets the intent of IEEE-279, 1971, for nuclear power plant protection systems. Specified surveillance intervals for MSIV-Closure, TSV-Closure, TCV-Closure, and the Manual Scram have been determined in accordance with NEDC-30851P-A, "Technical Specification Improvement Analyses for BWR Reactor Protection System", March 1988. The bases for the trip settings of the RPS are discussed in the bases for Specification 2.2.1.

The measurement of response time at the specified frequencies provides assurance that the protective functions associated with each channel are completed within the time limit assumed in the accident analysis. No credit was taken for those channels with response times indicated as not applicable. Response time may be demonstrated by any series of sequential, overlapping or total channel test measurement, provided such tests demonstrate the total channel response time as defined. Sensor response time verification may be demonstrated by either (1) inplace, onsite or offsite test measurements, or (2) utilizing replacement sensors with certified response times.

3/4.3.4 RECIRCULATION PUMP TRIP ACTUATION INSTRUMENTATION

The anticipated transient without scram (ATWS) recirculation pump trip system provides a means of limiting the consequences of the unlikely occurrence of a failure to scram during an anticipated transient. The response of the plant to this postulated event falls within the envelope of study events in General Electric Company Topical Report NEDO-10349, dated March 1971 and NEDO-24222, dated December, 1979, and Appendix G of the FSAR.

The end-of-cycle recirculation pump trip (EOC-RPT) system is a part of the Reactor Protection System and is an essential safety supplement to the reactor trip. The purpose of the EOC-RPT is to recover the loss of thermal margin which occurs at the end-of-cycle. The physical phenomenon involved is that the void reactivity feedback due to a pressurization transient can add positive reactivity to the reactor system at a faster rate than the control rods add negative scram reactivity. Each EOC-RPT system trips both recirculation pumps, reducing coolant flow in order to reduce the void collapse in the core during two of the most limiting pressurization events. The two events for which the EOC-RPT protective feature will function are closure of the turbine stop valves and fast closure of the turbine control valves.

A generic analysis, which provides for continued operation with one or both trip systems of the EOC-RPT system inoperable, has been performed. The analysis determined bounding cycle independent MINIMUM CRITICAL POWER RATIO (MCPR) Limiting Condition for Operation (LCO) values which must be used if the EOC-RPT system is inoperable. These values ensure that adequate reactivity margin to the MCPR safety limit exists in the event of the analyzed transient with the RPT function inoperable. The analysis results are further discussed in the bases for Specification 3.2.3.

A fast closure sensor from each of two turbine control valves provides input to the EOC-RPT system; a fast closure sensor from each of the other two turbine control valves provides input to the second EOC-RPT system. Similarly, a position switch for each of two turbine stop valves provides input to one EOC-RPT system; a position switch from each of the other two stop valves provides input to the other EOC-RPT system. For each EOC-RPT system, the sensor relay contacts are arranged to form a 2-out-of-2 logic for the fast closure of turbine control valves and a 2-out-of-2 logic for the turbine stop valves. The operation of either logic will actuate the EOC-RPT system and trip both recirculation pumps.

Each EOC-RPT system may be manually bypassed by use of a keyswitch which is administratively controlled. The manual bypasses and the automatic Operating Bypass at less than 30% of RATED THERMAL POWER are annunciated in the control room.

Specified surveillance intervals have been determined in accordance with the following:

1. NEDC-30851P-A, "Technical Specification Improvement Analyses for BWR Reactor Protection System", March 1988.

3/4.3.4 RECIRCULATION PUMP TRIP ACTUATION INSTRUMENTATION

2. GENE-770-06-1-A, "Bases for Changes to Surveillance Test Intervals and Allowed Out-of-Service Times for Selected Instrumentation Technical Specifications", December 1992.

The EOC-RPT system response time is the time assumed in the analysis between initiation of valve motion and complete suppression of the electric arc, i.e., 190 ms, less the time allotted for sensor response, i.e., 10 ms, and less the time allotted for breaker arc suppression determined by test, as correlated to manufacturer's test results, i.e., 83 ms, and plant pre-operational test results.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 95 TO FACILITY OPERATING LICENSE NO. NPF-11 AND

AMENDMENT NO. 79 TO FACILITY OPERATING LICENSE NO. NPF-18

COMMONWEALTH EDISON COMPANY

LASALLE COUNTY STATION, UNITS 1 AND 2

DOCKET NOS. 50-373 AND 50-374

1.0 INTRODUCTION

By letter dated January 28, 1994 (Reference 1) Commonwealth Edison Company (CECo) proposed changes to the functional test interval (FTI) from the existing one-month to a three-month interval for several reactor protection system (RPS) instrumentation channels because testing may cause some RPS channels to become inoperable. This exigent change is requested to avoid Unit 2 shutdown, due to induced inoperability of those RPS channels which could occur if those channels are functionally tested in accordance with the current FTI of the plant Technical Specification (TS). This will allow continued operation of Unit 2 for the next three-months and provide CECo more time to find the root cause of the degraded instruments and determine the appropriate corrective action. CECo's justification for the proposed changes is based on the previously approved topical report on the subject FTI by the NRC staff. This submittal is part of the TS improvement project for both units of the LaSalle County Station.

2.0 EVALUATION

Main steam line isolation valve (MSIV) closure, turbine stop valve (TSV) closure, turbine control valve (TCV) fast closure, and manual scram are functional units in the RPS trip system of both LaSalle Units. The MSIV position is monitored by a limit switch that has a spring loaded arm which automatically returns to its normal position after being toggled during valve strokes. If a limit switch arm is not in the normal position when the valve is next moved from open to close or vice-versa, the limit switch will not toggle. For instance, when an MSIV limit switch arm fails to return to normal after being tripped during the closure of the associated MSIV, the limit switch will not be reset and the associated RPS logic relay will remain de-energized when the MSIV is reopened. This will render the MSIV closure RPS trip inoperable and the limit switch arm will have to be manually brought to "spring return-to-normal" position for resetting the limit switch and thereby making the MSIV closure RPS trip operable. For an inboard MSIV limit switch, this will require containment entry and unit shutdown.

On January 18, 1994, during the Unit 2 shutdown, the licensee performed a functional test of the MSIV closure RPS Trip channels and observed that some limit switch arms did not automatically return to their normal position after being toggled during valve strokes. However, before the unit start-up on January 19, 1994, the licensee verified that all MSIV limit switch arms were in the "spring return-to-normal" position and thus ensured that the MSIV closure RPS trip channels are currently operable. The licensee further stated that the plant design provides continued annunciation of a half scram if the MSIV limit switch arm does not "spring return-to-normal" position after the MSIV closure. The next functional test of these channels is currently scheduled to be performed during plant operation on February 26, 1994. Operability of the limit switches can not be verified after this test unless Unit 2 power is reduced to less than 12 per cent (start-up mode) for containment entry. Since the root cause of the MSIV limit switch inoperability is not yet fully understood, and any corrective action is not yet determined, the licensee has requested the TS FTI for the MSIV closure be changed from the current one-month requirement to three-months to avoid an unnecessary plant shutdown. Additionally, since the TSV closure and TCV fast closure RPS trip channels functional tests are typically performed on the same day and at the same reduced reactor power as that for the MSIV closure RPS trip channels test, the licensee has proposed similar FTI for those channels. The licensee also proposed to change FTI for the manual scram from the current monthly to a weekly test schedule.

The General Electric (GE) Topical Report, "BWR Owner's Group Technical Specification Improvement Analysis for BWR Reactor Protection System" (Reference 2) that provided the basis for making improvements to the BWR RPS TS was submitted to NRC for approval. The analysis concluded that the current weekly and monthly FTI of the RPS channels can be extended to a quarterly schedule for both relay and solid-state plants and the FTI for manual scram from the current monthly to a weekly schedule. The staff approved this report and several BWR licensees have changed their plant TS accordingly. CECo's proposed changes for LaSalle Units 1 and 2 are in accordance with changes approved by the staff in the GE topical report. In addition to referencing the GE topical report, the licensee submitted a plant specific proprietary report (Reference 1) where the BWR generic model is compared with a plant specific model. The report concluded that differences between the two models do not affect the applicability of the generic TS improvements for the LaSalle Nuclear Station, Units 1 and 2. We agree with GE's conclusion and CECo's proposed changes to the FTI for both Units at LaSalle Nuclear Station.

Based on our evaluation of the licensee's submittals and previous staff acceptance of the GE topical report on FTI extension for RPS instrumentation channels in BWRs, we have concluded that CECo's proposed changes of the FTI are acceptable.

3.0 EXIGENT CIRCUMSTANCES

Due to the problem that has developed with the LaSalle Unit 2 MSIV Limit Switches, LaSalle Unit 2 must shut down to at least the Startup mode prior to the end of the channel functional test interval to verify that the limit switches are in the "Spring Return-to-Normal" position. The nature of the problem is that some of the limit switches may not always automatically return the limit switch arm to the normal position after being toggled during valve strokes. If a limit switch arm is not in the normal position when the valve is next moved from open to closed or vice-versa, the limit switch will not toggle. For the MSIV limit switches that input to the MSIV - Closure Scram RPS logic (MSIV-RPS limit switch), the problem can be readily identified. This failure will occur when a MSIV-RPS limit switch fails to return to normal after tripping during closure of the associated MSIV (only partial closure is required to conduct the functional test). When the MSIV is reopened, the limit switch will not be reset and thus the associated RPS logic relay will remain de-energized. If an MSIV-RPS limit switch returns to normal after being tripped, then the limit switch will reset, and thus re-energize the associated RPS logic relay. However, the limit switch may not spring return to the normal position after being reset, which is not detectable until the next time the MSIV is cycled for a surveillance. During the time interval, an MSIV-RPS limit switch could be inoperable, unable to trip on MSIV closure. Therefore, LaSalle Unit 2 must reduce power to less than 12% (Startup Mode) to de-inert the primary containment for entry to verify proper reset of the inboard MSIVs limit switches to assure RPS Operability.

The licensee verified by visual inspection that all MSIV limit switches were in the "Spring Return-to-Normal" position when LaSalle Unit 2 was started up on January 19, 1994, and therefore the RPS limit switches for the MSIV - Closure Scram are currently operable. However, the position of the MSIV limit switches would be suspect without visual observation after cycling the MSIVs. Therefore, LaSalle Unit 2 must be shut down to at least the Startup mode in order to allow personnel access for visual verification of proper MSIV limit switch position in the "Spring Return-to-Normal" position following the functional test. The monthly channel functional test is due again at a maximum of 1.25 times the current 31-day surveillance test interval.

There are three Technical Specification Table 4.3.1.1-1 reactor protection system instrumentation surveillance requirements that the licensee typically performs on the same day, typically on the night shift (shift 1). These surveillances meet the channel functional test requirements for item 5, Main Steam Line Isolation Valve (MSIV) - Closure; item 9, Turbine Stop Valve (TSV) - Closure; and item 10, Turbine Control Valve (TCV) Fast Closure, Valve Trip System Oil Pressure - Low. The channel functional tests for these valves require the valves to be cycled and a reactor power decrease to ≤ 90% thermal power is required prior to cycling MSIVs or TCVs. Due to the requirement to drop power and the similarity of the surveillance, these surveillance requirements are satisfied together (sequentially, not simultaneously). The EOC-RPT system instrumentation channel functional tests are also met by the same tests on the TSVs and TCVs. Therefore, although the urgency for this amendment request is for the MSIV - Closure channel functional test, the frequencies of the TSV and TCV closure scram and EOC-RPT TSV and TCV channel functional tests are also requested to be changed to quarterly. This will minimize the number of times LaSalle Units 1 and 2 will be required to reduce thermal power to less than or equal to 90% power and keep these related surveillance together.

Without this Technical Specification amendment, LaSalle Unit 2 will need to be placed in the Startup mode to verify the proper position of MSIV limit switches for RPS instrumentation operability following performance of the above mentioned surveillance requirements.

The circumstances leading to the exigent relief request could not be avoided, because this problem with the LaSalle Unit 2 MSIV limit switches was not identified for Unit 2 startup until January 17, 1994. The determination that LaSalle Unit 2 would be required to shut down to at least the Startup mode the next time the channel functional test for the MSIV - closure scram was due was not made until January 18, 1994 as a result of an operability evaluation. The possibility of a change to the STI from monthly to quarterly was determined on January 19, 1994. Due to the due date of the MSIV - Closure channel functional test that must be performed by February 26, 1994 at 1000 hours (10:00 A.M. CST), the 30-day period for public comment required for a normal Technical Specification amendment per 10 CFR 50.91(a)(2) cannot be met. Therefore, this request for amendment was submitted by the licensee as an exigent request per 10 CFR 50.91(a)(6).

4.0 FINAL NO SIGNIFICANT HAZARDS DETERMINATION

The Commission's regulations in 10 CFR 50.92 state that the Commission may make a final determination that a license amendment involves no significant hazards considerations if operation of the facility in accordance with the amendment would not: (1) Involve a significant increase in the probability or consequences of an accident previously evaluated; or (2) create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) involve a significant reduction in a margin of safety.

The staff reviewed and presented the licensee's analysis of the issue of no significant hazards consideration (provided below) and, based on that review, proposed a no significant hazards consideration in the Federal Register (59 FR 6062) on February 9, 1994. There were no public comments in response to the notice published in the Federal Register.

Based on the criteria for defining a significant hazards consideration established in 10 CFR 50.92, the licensee provided its analysis of the issue of no significant hazards consideration which states that the proposed amendment will not:

Involve a significant increase in the probability or consequences of an accident previously evaluated because:

The proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated. The proposed changes increase the STI for actuation instrumentation supporting RPS and EOC-RPT trip functions. There are no changes in any of the affected systems themselves. Because of this there is no change in the probability of occurrence of an accident or the consequences of an accident or the consequences of malfunction of equipment. With respect to the malfunction of equipment, topical reports prepared by GE demonstrated that there is a reduction in scram frequency for the RPS. This offsets the slight increase in trip function unavailability determined by GE. This was judged acceptable by GE. The NRC concurred with this conclusion in its review of the topical reports (NEDC-30851P-A). For EOC-RPT GE demonstrated that the trip function unavailability when the surveillance interval is extended from 1 to 3 months is lower for the turbine stop valve trip function and slightly higher for the turbine control valve trip function than the same trip functions for RPS-scram. However, GE concluded that the small increase in EOC-RPT unavailability (represented by small increased risk of a MCPR violation) is offset by the benefits associated with the similar approved STI and AOT changes for the RPS-scram function. Therefore, GE concluded that the STI changes for EOC-RPT trip function are bounded by the approved RPS analysis (Reference 3). The NRC accepted the conclusions of GE by a SER included in Reference 4. The proposed changes are consistent with the Safety Evaluation Reports issued in these topical reports. The proposed changes therefore do not involve a significant increase in the probability or consequences of an accident previously evaluated.

Create the possibility of a new or different kind of accident from any accident previously evaluated because:

The proposed changes do not create the possibility for an accident or malfunction of a different type than any evaluated previously in the UFSAR. The proposed changes increase the STI for the RPS and EOC-RPT Instrumentation. There are no changes in the instrumentation of these systems. Since there are no such changes there is no possibility for an accident or malfunction of a different type than any previously evaluated.

Involve a significant reduction in the margin of safety because:

The proposed changes do not reduce the margin of safety as defined in the basis for any Technical Specification. The proposed changes do not change any setpoints in the above mentioned systems or their levels of redundancy. Setpoints are based upon the drift occurring during an 18 month calibration interval. The bases in the Technical Specifications either do not discuss STI, or state "...one channel may be inoperable for brief intervals to conduct required surveillance." The proposed changes are bounded by the analyses of References 3 and 4. These analyses, which were prepared by GE and approved by the NRC, examined the effects of extending STI and found that the proposed changes would not involve a significant reduction in a margin of safety. LaSalle Station Units 1 and 2 RPS and EOC-RPT systems have been compared to the generic analyses and verified to be bounded.

The staff has completed its review of the licensee's proposed no significant hazards consideration and concludes that the amendments meet the three standards of 10 CFR 50.92(c). Therefore, the staff has made a final determination that the proposed amendment does not involve a significant hazards consideration.

5.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Illinois State official was notified of the proposed issuance of the amendments. The State official had no comments.

6.0 ENVIRONMENTAL CONSIDERATION

The amendments change a requirement with respect to the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the 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 the amendments involve no significant hazards consideration, and there has been no public comment on such finding (59 FR 6062). Accordingly, the 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 the amendments.

7.0 CONCLUSION

The Commission 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, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributor: Iqbal Ahmed, HICB

Date: February 25, 1994

8.0 REFERENCES

1. CECo Letter (Gary G. Benes) to NRC (Document Control Desk) dated January 28, 1994.

2. General Electric Topical Report, "Technical Specification Improvement Analyses for BWR Reactor Protection System," NEDC-30851P, DRF A00-02119-A, May 1985.

3. General Electric Topical Report, "Technical Specification Improvement Analyses for BWR Reactor Protection System," NEDC-30851P-A, DRF A00-02119-A, March 1988.

4. General Electric Topical Report, "Bases for Changes to Surveillance Test Intervals and Allowed Out-of-Service Times for Selected Instrumentation Technical Specifications", GENE-770-06-1-A, December 1992.