November 27, 1991

Docket Nos. 50-373
and 50-374

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Mr. Thomas J. Kovach
Nuclear Licensing Manager
Commonwealth Edison Company-Suite 300
OPUS West III
1400 OPUS Place
Downers Grove, Illinois 60515

Dear Mr. Kovach:

SUBJECT: ISSUANCE OF AMENDMENTS (TAC NOS. M77812 AND M77813)

The Commission has issued the enclosed Amendment No. 81 to Facility Operating License No. NPF-11 and Amendment No. 65 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 dated October 10, 1990, as supplemented October 16, 1991.

The amendments modify the Technical Specification (TS) requirements for the High Pressure Core Spray (HPCS) system by permanently aligning the HPCS suction to the suppression pool and removing all TS requirements for aligning the HPCS suction to the condensate storage tank. In addition, containment isolation requirements are added for the new Reactor Core Isolation Cooling (RCIC) full flow test line to the suppression pool.

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:

Byron L. Siegel, Project Manager
Project Directorate III-2
Division of Reactor Projects - III/IV/V
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 81 to NPF-11
2. Amendment No. 65 to NPF-18
3. Safety Evaluation

cc w/enclosures:
See next page

OFFICIAL RECORD COPY
DOCUMENT NAME: [AMENDMENT 77812/13]

Office:	LA/PDIII-2	PE/PDIII-2	PM/PDIII-2	PD/PDIII-2	OGC-WF1
Surname:	CMoore	RElliott:jar	BSiegel	RBarrett	
Date:	11/ /91	11/13/91	11/ /91	11/ /91	11/ /91

Mr. Thomas J. Kovach
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, Il'ᶦ ois 62704

Regional Administrator, Region III
U. S. Nuclear Regulatory Commission
799 Roosevelt Road, Bldg. #4
Glen Ellyn, Illinois 60137

Robert Neumann
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, Esq.
Sidley and Austin
One First National Plaza
Chicago, Illinois 60690

Mr. G. Diederich
LaSalle Station Manager
LaSalle County Station
Rural Route 1
P. O. Box 220
Marseilles, IL 61341

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

COMMONWEALTH EDISON COMPANY

DOCKET NO. 50-373

LASALLE COUNTY STATION, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 81
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 October 10, 1990, as amended October 16, 1991, 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) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A, as revised through Amendment No. 81 , 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 to be implemented prior to startup following the L1R05 refueling outage.

FOR THE NUCLEAR REGULATORY COMMISSION

Richard J. Barrett, 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: November 27, 1991

ATTACHMENT TO LICENSE AMENDMENT NO. 81

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 3-26	3/4 3-26
3/4 3-27	3/4 3-27
3/4 3-30	3/4 3-30
3/4 3-30a	3/4 3-30a
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 5-5	3/4 5-5
3/4 5-6	3/4 5-6
3/4 5-7	3/4 5-7
3/4 5-8	3/4 5-8
3/4 5-9	3/4 5-9
3/4 6-33	3/4 6-33
3/4 6-34	3/4 6-34
---	3/4 6-34a
3/4 8-30	3/4 8-30
B 3/4 5-1	B 3/4 5-1
B 3/4 5-2	B 3/4 5-2
B 3/4 6-3a	B 3/4 6-3a

TABLE 3.3.3-1 (Continued)

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION

TRIP FUNCTION	MINIMUM OPERABLE CHANNELS PER TRIP FUNCTION[a]	APPLICABLE OPERATIONAL CONDITIONS	ACTION
C. DIVISION 3 TRIP SYSTEM			
1. HPCS SYSTEM			
a. Reactor Vessel Water Level - Low, Low, Level 2	4[b]	1, 2, 3, 4*, 5*	35
b. Drywell Pressure - High	4[b]	1, 2, 3	35
c. Reactor Vessel Water Level-High, Level 8	2[c]	1, 2, 3, 4*, 5*	32
d. Deleted			
e. Deleted			
f. Pump Discharge Pressure-High (Bypass)	1	1, 2, 3, 4*, 5*	31
g. HPCS System Flow Rate-Low (Permissive)	1	1, 2, 3, 4*, 5*	31
h. Manual Initiation	1/division	1, 2, 3, 4*, 5*	34

D. LOSS OF POWER	TOTAL NO. OF INSTRUMENTS	INSTRUMENTS TO TRIP	MINIMUM OPERABLE INSTRUMENTS[a]	APPLICABLE OPERATIONAL CONDITIONS	ACTION
1. 4.16 kv Emergency Bus Undervoltage (Loss of Voltage)	2/bus	2/bus	2/bus	1, 2, 3, 4**, 5**	37
2. 4.16 kv Emergency Bus Undervoltage (Degraded Voltage)	2/bus	2/bus	2/bus	1, 2, 3, 4**, 5**	37

(a) A channel instrument may be placed in an inoperable status for up to 2 hours during periods of required surveillance without placing the trip system/channel/instrument in the tripped condition provided at least one other OPERABLE channel/instrument in the same trip system is monitoring that parameter.
(b) Also actuates the associated division diesel generator.
(c) Provides signal to close HPCS pump discharge valve only on 2-out-of-2 logic.
* Applicable when the system is required to be OPERABLE per Specification 3.5.2 or 3.5.3.
** Required when ESF equipment is required to be OPERABLE.
\# Not required to be OPERABLE when reactor steam dome pressure is ≤ 122 psig.

TABLE 3.3.3-1 (Continued)

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION

ACTION

ACTION 30 - With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip Function requirement:

a. With one channel inoperable, place the inoperable channel in the tripped condition within one hour* or declare the associated system inoperable.

b. With more than one channel inoperable, declare the associated system inoperable.

ACTION 31 - With the number of OPERABLE channels less than required by the Minimum OPERABLE channels per Trip Function, place the inoperable channel in the tripped condition within one hour; restore the inoperable channel to OPERABLE status within 7 days or declare the associated system inoperable.

ACTION 32 - With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip Function requirement, declare the associated ADS trip system or ECCS inoperable.

ACTION 33 - With the number of OPERABLE channels less than the Minimum OPERABLE Channels per Trip Function requirement, place the inoperable channel in the tripped condition within one hour.

ACTION 34 - With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip Function requirement, restore the inoperable channel to OPERABLE status within 8 hours or declare the associated ADS trip system or ECCS inoperable.

ACTION 35 - With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip Function requirement

a. For one trip system, place that trip system in the tripped condition within one hour* or declare the HPCS system inoperable.

b. For both trip systems, declare the HPCS system inoperable.

ACTION 36 - Deleted

ACTION 37 - With the number of OPERABLE instruments less than the Minimum Operable Instruments, place the inoperable instrument(s) in the tripped condition within 1 hour* or declare the associated emergency diesel generator inoperable and take the ACTION required by Specification 3.8.1.1 or 3.8.1.2 as appropriate.

*The provisions of Specification 3.0.4 are not applicable.

TABLE 3.3.3-2 (Continued)

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
C. DIVISION 3 TRIP SYSTEM		
1. HPCS SYSTEM		
a. Reactor Vessel Water Level - Low Low, Level 2	≥ -50 inches*	≥ -57 inches*
b. Drywell Pressure - High	≤ 1.69 psig	≤ 1.89 psig
c. Reactor Vessel Water Level - High, Level 8	≤ 55.5 inches*	≤ 56 inches*
d. Deleted		
e. Deleted		
f. Pump Discharge Pressure - High	≥ 120 psig	≥ 110 psig
g. HPCS System Flow Rate - Low	≥ 1000 gpm	≥ 900 gpm
h. Manual Intiation	NA	NA
D. LOSS OF POWER		
1. 4.16 kV Emergency Bus Undervoltage (Loss of Voltage)#		
a. 4.16 kV Buses		
1) Divisions 1 and 2	2625 ± 131 volts with ≤ 10 seconds time delay	2625 ± 262 volts with ≤ 11 seconds time delay
	2496 ± 125 volts with ≥ 4 seconds time delay	2496 ± 250 volts with ≥ 3 seconds time delay
2) Division 3	2870 ± 143 volts with ≤ 10 seconds time delay	2870 ± 287 volts with ≤ 11 seconds time delay

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

#These are inverse time delay voltage relays or instantaneous voltage relays with a time delay. The voltages shown are the maximum that will not result in a trip. Lower voltage conditions will result in decreased trip times.

TABLE 3.3.3-2 (Continued)

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
2. 4.16 kV Emergency Bus Undervoltage (Degraded Voltage)		
a. 4.16 kV Buses		
1) Divisions 1, 2 and 3	3814 ± 76 volts with 10 ± 1 seconds time delay with LOCA signal or 5 ± 0.5 minutes time delay without LOCA signal	3814 ± 76 volts with 10 ± 1 seconds time delay with LOCA signal or 5 ± 0.5 minutes time delay without LOCA signal

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 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. Initiation 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
h. Drywell Pressure Bypass Timer	NA	M	Q	1, 2, 3
i. Manual Inhibit	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. Initiation Timer	NA	M	Q	1, 2, 3
d. Reactor Vessel Water Level - Low, Level 3	NA	M	R	1, 2, 3
e. LPCS Pump B and C Discharge Pressure-High	NA	M	Q	1, 2, 3
f. Manual Initiation	NA	R	NA	1, 2, 3
h. Drywell Pressure Bypass Timer	NA	M	Q	1, 2, 3
i. Manual Inhibit	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. Deleted				
e. Deleted				
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)	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.

SURVEILLANCE REQUIREMENTS (Continued)

(a) LPCS system to be $\leq$ 500 psig and $\geq$ 55 psig, respectively.

(b) LPCI subsystems to be $\leq$ 400 psig and $\geq$ 55 psig, respectively.

2) Low pressure setpoint of the HPCS system to be $\geq$ 63 psig.

b) Header delta P instrumentation and verifying the setpoint of the:

1) LPCS system and LPCI subsystems to be ± 1 psid.

2) HPCS system to be 5 ± 2.0 psid greater than the normal indicated ΔP.

3. Deleted.

4. Visually inspecting the ECCS corner room watertight door seals and room penetration seals and verifying no abnormal degradation, damage, or obstructions.

d. For the ADS by:

1. At least once per 31 days, performing a CHANNEL FUNCTIONAL TEST of the accumulator backup compressed gas system low pressure alarm system.

2. At least once per 18 months:

a) Performing a system functional test which includes simulated automatic actuation of the system throughout its emergency operating sequence, but excluding actual valve actuation.

b) Manually opening each ADS valve and observing the expected change in the indicated valve position.

c) Performing a CHANNEL CALIBRATION of the accumulator backup compressed gas system low pressure alarm system and verifying an alarm setpoint of 500 + 40, - 0 psig on decreasing pressure.

EMERGENCY CORE COOLING SYSTEMS

3/4.5.2 ECCS - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.5.2 At least two of the following shall be OPERABLE:

a. The low pressure core spray (LPCS) system with a flow path capable of taking suction from the suppression chamber and transferring the water through the spray sparger to the reactor vessel.

b. Low pressure coolant injection (LPCI) subsystem "A" of the RHR system with a flow path capable of taking suction from the suppression chamber upon being manually realigned and transferring the water to the reactor vessel.

c. Low pressure coolant injection (LPCI) subsystem "B" of the RHR system with a flow path capable of taking suction from the suppression chamber upon being manually realigned and transferring the water to the reactor vessel.

d. Low pressure coolant injection (LPCI) subsystem "C" of the RHR system with a flow path capable of taking suction from the suppression chamber upon being manually realigned and transferring the water to the reactor vessel.

e. The high pressure core spray (HPCS) system with a flow path capable of taking suction from the suppression pool and transferring the water through the spray sparger to the reactor vessel.

APPLICABILITY: OPERATIONAL CONDITION 4 or 5*.

ACTION:

a. With one of the above required subsystems/systems inoperable, restore at least two subsystems/systems to OPERABLE status within 4 hours or suspend all operations that have a potential for draining the reactor vessel.

b. With both of the above required subsystems/systems inoperable, suspend CORE ALTERATIONS and all operations that have a potential for draining the reactor vessel. Restore at least one subsystem/system to OPERABLE status within 4 hours or establish SECONDARY CONTAINMENT INTEGRITY within the next 8 hours.

*The ECCS is not required to be OPERABLE provided that the reactor vessel head is removed, the cavity is flooded, the spent fuel pool gates are removed, and water level is maintained within the limits of Specifications 3.9.8 and 3.9.9.

SURVEILLANCE REQUIREMENTS

4.5.2.1 At least the above required ECCS shall be demonstrated OPERABLE per Surveillance Requirement 4.5.1, except that the header delta P instrumentation is not required to be OPERABLE.

EMERGENCY CORE COOLING SYSTEMS

3/4.5.3 SUPPRESSION CHAMBER#

LIMITING CONDITION FOR OPERATION

3.5.3 The suppression chamber shall be OPERABLE:

a. In OPERATIONAL CONDITION 1, 2, or 3 with a contained water volume of at least 128,800 ft^3, equivalent to a level of -4 1/2 inches.**

b. In OPERATIONAL CONDITION 4 or 5* with a contained water volume of at least 70,000 ft^3, equivalent to a level of -12 feet 7 inches.**

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, 3, 4, and 5*.

ACTION:

a. In OPERATIONAL CONDITION 1, 2, or 3 with the suppression chamber water level less than the above limit, restore the water level to within the limit within 1 hour or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. In OPERATIONAL CONDITION 4 or 5* with the suppression chamber water level less than the above limit, suspend CORE ALTERATIONS and all operations that have a potential for draining the reactor vessel and lock the reactor mode switch in the Shutdown position. Establish SECONDARY CONTAINMENT INTEGRITY within 8 hours.

#See Specification 3.6.2.1 for pressure suppression requirements.

*The suppression chamber is not required to be OPERABLE provided that the reactor vessel head is removed, the cavity is flooded or being flooded from the suppression pool, the spent fuel pool gates are removed when the cavity is flooded, and the water level is maintained within the limits of Specifications 3.9.8 and 3.9.9.

**Level is referenced to a plant elevation of 699 feet 11 inches (See Figure B 3/4.6.2-1).

LIMITING CONDITION FOR OPERATION (Continued)

ACTION: (Continued)

c. With one suppression chamber water level instrumentation channel inoperable, restore the inoperable channel to OPERABLE status within 7 days or verify the suppression chamber water level to be greater than or equal to -4 1/2 inches** or -12 feet 7 inches**, as applicable, at least once per 12 hours by local indication.

d. With both suppression chamber water level instrumentation channels inoperable, restore at least one inoperable channel to OPERABLE status within 8 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours and verify the suppression chamber water level to be greater than or equal to -4 1/2 inches** or -12 feet 7 inches**, as applicable, at least once per 12 hours by local indication.

SURVEILLANCE REQUIREMENTS

4.5.3.1 The suppression chamber shall be determined OPERABLE by verifying:

a. The water level to be greater than or equal to, as applicable:

1. -4 1/2 inches** at least once per 24 hours.
2. -12 feet 7 inches** at least once per 12 hours.

b. Two suppression chamber water level instrumentation channels OPERABLE by performance of a:

1. CHANNEL CHECK at least once per 24 hours,
2. CHANNEL FUNCTIONAL TEST at least once per 31 days, and
3. CHANNEL CALIBRATION at least once per 18 months,

with the low water level alarm setpoint at greater than or equal to -3 inches.**

4.5.3.2 With the suppression chamber level less than the above limit in OPERATIONAL CONDITION 5*, at least once per 12 hours verify footnote conditions* to be satisfied.

*The suppression chamber is not required to be OPERABLE provided that the reactor vessel head is removed, the cavity is flooded or being flooded from the suppression pool, the spent fuel pool gates are removed when the cavity is flooded, and the water level is maintained within the limits of Specifications 3.9.8 and 3.9.9.

**Level is referenced to a plant elevation of 699 feet 11 inches (See Figure B 3/4.6.2-1).

TABLE 3.6.3-1 (Continued)

PRIMARY CONTAINMENT ISOLATION VALVES

VALVE FUNCTION AND NUMBER

Other Isolation Valves (Continued)

4. Low Pressure Core Spray System

1E21-F005
1E21-F001[j]
1E21-F012[j]
1E21-F011[j]
1E21-F018[j]
1E21-F031[j]
1E21-F006[k]

5. High Pressure Core Spray System

1E22-F004
1E22-F015[j]
1E22-F023[j]
1E22-F012[j]
1E22-F014[j]
1E22-F005[k]

6. Reactor Core Isolation Cooling System

1E51-F013
1E51-F069
1E51-F028
1E51-F068
1E51-F040
1E51-F031[j]
1E51-F019[j]
1E51-F065[k]
1E51-F066[k]
1E51-F059[m]
1E51-F022[m]
1E51-F362[n]
1E51-F363[n]

TABLE 3.6.3-1 (Continued)

PRIMARY CONTAINMENT ISOLATION VALVES

VALVE FUNCTION AND NUMBER

Other Isolation Valves (Continued)

7. Post LOCA Hydrogen Control

1HG001A, B
1HG002A, B
1HG005A, B
1HG006A, B

8. Standby Liquid Control System

1C41-F004A, B
1C41-F007

9. Reactor Recirculation Seal Injection

1B33-F013A, B(j)
1B33-F017A, B(j)

10. Drywell Pneumatic System

1IN018

TABLE 3.6.3-1 (Continued)

PRIMARY CONTAINMENT ISOLATION VALVES

---

* But ≥ 3 seconds.

(a) See Specification 3.3.2, Table 3.3.2-1, for isolation signal(s) that operates each valve group.

(b) Not included in total sum of Type B and C tests.

(c) May be opened on an intermittent basis under administrative control.

(d) Not closed by SLCS actuation.

(e) Not closed by Trip Functions 5a, b or c, Specification 3.3.2, Table 3.3.2-1.

(f) Not closed by Trip Functions 4a, c, d, e or f of Specification 3.3.2, Table 3.3.2-1.

(g) Not subject to Type C leakage test.

(h) Opens on an isolation signal. Valves will be open during Type A test. No Type C test required.

(i) Also closed by drywell pressure-high signal.

(j) Hydraulic leak test at 43.6 psig.

(k) Not subject to Type C leakage test - leakage rate tested per Specification 4.4.3.2.2.

(l) These penetrations are provided with removable spools outboard of the outboard isolation valve. During operation, these lines will be blind flanged using a double O-ring and a type B leak test. In addition, the packing of these isolation valves will be soap-bubble tested to ensure insignificant or no leakage at the containment test pressure each refueling outage.

(m) If valves 1E51-F362 and 1E51-F363 are locked closed and acceptably leak rate tested, then valves 1E51-F059 and 1E51-F022 are not considered to be primary containment isolation valves and are not required to be leak rate tested.

(n) Either the 1E51-F362 or the 1E51-F363 valve may be open when the RCIC system is in the standby mode of operation, and both valves may be open during operation of the RCIC system in the full flow test mode, providing that:

1) valve 1E51-F022 is acceptably leak rate tested, and

2) valve 1E51-F059 is deactivated, locked closed and acceptably leak rate tested, and

3) the spectacle flange, installed immediately downstream of the 1E51-F059 valve, is closed and acceptably leak rate tested.

TABLE 3.8.3.3-1 (Continued)

MOTOR OPERATED VALVES THERMAL OVERLOAD PROTECTION

	VALVE NUMBER	BYPASS DEVICE (Continuous)(Accident Conditions)	SYSTEM(S) AFFECTED
l.	1E32 - F001A	Accident Conditions	MSIV-LCS
	1E32 - F002A	Accident Conditions	
	1E32 - F003A	Accident Conditions	
	1E32 - F001E	Accident Conditions	
	1E32 - F002E	Accident Conditions	
	1E32 - F003E	Accident Conditions	
	1E32 - F001J	Accident Conditions	
	1E32 - F002J	Accident Conditions	
	1E32 - F003J	Accident Conditions	
	1E32 - F001N	Accident Conditions	
	1E32 - F002N	Accident Conditions	
	1E32 - F003N	Accident Conditions	
	1E32 - F006	Accident Conditions	
	1E32 - F007	Accident Conditions	
	1E32 - F008	Accident Conditions	
	1E32 - F009	Accident Conditions	
m.	1E22 - F004	Accident Conditions	HPCS system
	1E22 - F012	Accident Conditions	
	1E22 - F015	Continuous	
	1E22 - F023	Accident Conditions	

BASES

3/4.5.1 and 3/4.5.2 ECCS - OPERATING and SHUTDOWN

ECCS Division 1 consists of the low pressure core spray system, low pressure coolant injection subsystem "A" of the RHR system, and the automatic depressurization system (ADS) as actuated by ADS trip system "A". ECCS Division 2 consists of low pressure coolant injection subsystems "B"and "C" of the RHR system and the automatic depressurization system as actuated by ADS trip system "B".

The low pressure core spray (LPCS) system is provided to assure that the core is adequately cooled following a loss-of-coolant accident and provides adequate core cooling capacity for all break sizes up to and including the double-ended reactor recirculation line break, and for transients or smaller breaks following depressurization by the ADS.

The LPCS is a primary source of emergency core cooling after the reactor vessel is depressurized and a source for flooding of the core in case of accidental draining.

The surveillance requirements provide adequate assurance that the LPCS system will be OPERABLE when required. Although all active components are testable and full flow can be demonstrated by recirculation through a test loop during reactor operation, a complete functional test requires reactor shutdown. The pump discharge piping is maintained full to prevent water hammer damage to piping and to start cooling at the earliest moment.

The low pressure coolant injection (LPCI) mode of the RHR system is provided to assure that the core is adequately cooled following a loss-of-coolant accident. Three subsystems, each with one pump, provide adequate core flooding for all break sizes up to and including the double-ended reactor recirculation line break, and for transients or small breaks following depressurization by the ADS.

The surveillance requirements provide adequate assurance that the LPCI system will be OPERABLE when required. Although all active components are testable and full flow can be demonstrated by recirculation through a test loop during reactor operation, a complete functional test requires reactor shutdown. The pump discharge piping is maintained full to prevent water hammer damage to piping and to start cooling at the earliest moment.

ECCS Division 3 consists of the high pressure core spray system. The high pressure core spray (HPCS) system is provided to assure that the reactor core is adequately cooled to limit fuel clad temperature in the event of a small break in the reactor coolant system and loss of coolant which does not result in rapid depressurization of the reactor vessel. The HPCS system permits the reactor to be shut down while maintaining sufficient reactor vessel water level inventory until the vessel is depressurized. The HPCS system operates over a range of 1160 psid, differential pressure between reactor vessel and HPCS suction source, to 0 psid.

The capacity of the HPCS system is selected to provide the required core cooling. The HPCS pump is designed to deliver greater than or equal to 516/1550/6200 gpm at differential pressures of 1160/1130/200 psid. Water is taken from the suppression pool and injected into the reactor.

ECCS-OPERATING and SHUTDOWN (Continued)

With the HPCS system inoperable, adequate core cooling is assured by the OPERABILITY of the redundant and diversified automatic depressurization system and both the LPCS and LPCI systems. In addition, the reactor core isolation cooling (RCIC) system, a system for which no credit is taken in the hazards analysis, will automatically provide makeup at reactor operating pressures on a reactor low water level condition. The HPCS out-of-service period of 14 days is based on the demonstrated OPERABILITY of redundant and diversified low pressure core cooling systems.

The surveillance requirements provide adequate assurance that the HPCS system will be OPERABLE when required. Although all active components are testable and full flow can be demonstrated by recirculation through a test loop during reactor operation, a complete functional test with reactor vessel injection requires reactor shutdown. The pump discharge piping is maintained full to prevent water hammer damage and to provide cooling at the earliest moment.

Upon failure of the HPCS system to function properly, if required, the automatic depressurization system (ADS) automatically causes selected safety-relief valves to open, depressurizing the reactor so that flow from the low pressure core cooling systems can enter the core in time to limit fuel cladding temperature to less than 2200°F. ADS is conservatively required to be OPERABLE whenever reactor vessel pressure exceeds 122 psig even though low pressure core cooling systems provide adequate core cooling up to 350 psig.

ADS automatically controls seven selected safety-relief valves. Six valves are required to be OPERABLE since the LOCA analysis assumes 6 ADS valves in addition to a single failure. It is therefore appropriate to permit one of the required valves to be out-of-service for up to 14 days without materially reducing system reliability.

3/4.5.3 SUPPRESSION CHAMBER

The suppression chamber is also required to be OPERABLE as part of the ECCS to ensure that a sufficient supply of water is available to the HPCS, LPCS and LPCI systems in the event of a LOCA. This limit on suppression chamber minimum water volume ensures that sufficient water is available to permit recirculation cooling flow to the core (See Figure B 3/4.6.2-1). The OPERABILITY of the suppression chamber in OPERATIONAL CONDITIONS 1, 2 or 3 is required by Specification 3.6.2.1.

Repair work might require making the suppression chamber inoperable. This specification will permit those repairs to be made and at the same time give assurance that the irradiated fuel has an adequate cooling water supply when the suppression chamber must be made inoperable in OPERATIONAL CONDITION 4 or 5.

In OPERATIONAL CONDITION 4 and 5 the suppression chamber minimum required water volume is reduced because the reactor coolant is maintained at or below 200°F. Since pressure suppression is not required below 212°F, the minimum water volume is based on NPSH, recirculation volume, vortex prevention plus a 2'-4" safety margin for conservatism.

Suppression Chamber Level	Plant Elevation	Control Room/Local Indication
26' 10"	700' 2"	+3" High Level LCO (Volume 131,900 ft^3)
26' 9"	700' 1"	+2" High Level Alarm
26' 7"	699' 11"	0" Instrument Zero
26' 4"	699' 8"	-3" Low Level Alarm
26' 2 1/2"	699' 6 1/2"	-4 1/2" Low Level LCO Operational Condition 1, 2, or 3 (Volume 128,800 ft^3)
14'	687' 4"	-12' 7" Low Level LCO Operational Condition 4 or 5 (Volume 70,000 ft^3)

SUPPRESSION POOL LEVEL SETPOINTS

BASES FIGURE B 3/4.6.2-1

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

COMMONWEALTH EDISON COMPANY

DOCKET NO. 50-374

LASALLE COUNTY STATION, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 65
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 October 10, 1990, as amended October 16, 1991, 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) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A, as revised through Amendment No. 65 , 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 to be implemented prior to startup following the L2R04 refueling outage.

FOR THE NUCLEAR REGULATORY COMMISSION

RJBarrett

Richard J. Barrett, 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: November 27, 1991

ATTACHMENT TO LICENSE AMENDMENT NO. 65

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. Pages indicated by an asterisk are provided for convenience.

REMOVE	INSERT
3/4 3-26	3/4 3-26
3/4 3-27	3/4 3-27
3/4 3-30	3/4 3-30
3/4 3-30a	3/4 3-30a
3/4 3-34	3/4 3-34
3/4 5-5	3/4 5-5
3/4 5-6	3/4 5-6
3/4 5-7	3/4 5-7
3/4 5-8	3/4 5-8
3/4 5-9	3/4 5-9
3/4 6-36	3/4 6-36
3/4 6-37	3/4 6-37
---	3/4 6-37a
* 3/4 6-38	* 3/4 6-38
3/4 8-30	3/4 8-30
B 3/4 5-1	B 3/4 5-1
B 3/4 5-2	B 3/4 5-2
B 3/4 6-3a	B 3/4 6-3a

TABLE 3.3.3-1 (Continued)

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION

TRIP FUNCTION	MINIMUM OPERABLE CHANNELS PER TRIP FUNCTION(a)	APPLICABLE OPERATIONAL CONDITIONS	ACTION
C. DIVISION 3 TRIP SYSTEM			
1. HPCS SYSTEM			
a. Reactor Vessel Water Level - Low, Low, Level 2	4(b)	1, 2, 3, 4*, 5*	35
b. Drywell Pressure - High	4(b)	1, 2, 3	35
c. Reactor Vessel Water Level-High, Level 8	2(c)	1, 2, 3, 4*, 5*	32
d. Deleted			
e. Deleted			
f. Pump Discharge Pressure-High (Bypass)	1	1, 2, 3, 4*, 5*	31
g. HPCS System Flow Rate-Low (Permissive)	1	1, 2, 3, 4*, 5*	31
h. Manual Initiation	1/division	1, 2, 3, 4*, 5*	34

D. LOSS OF POWER	TOTAL NO. OF INSTRUMENTS	INSTRUMENTS TO TRIP	MINIMUM OPERABLE INSTRUMENTS(a)	APPLICABLE OPERATIONAL CONDITIONS	ACTION
1. 4.16 kV Emergency Bus Undervoltage (Loss of Voltage)	2/bus	2/bus	2/bus	1, 2, 3, 4**, 5**	37
2. 4.16 kV Emergency Bus Undervoltage (Degraded Voltage)	2/bus	2/bus	2/bus	1, 2, 3, 4**, 5**	37

TABLE NOTATION

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

(b) Also actuates the associated division diesel generator.

(c) Provides signal to close HPCS pump discharge valve only on 2-out-of-2 logic.

* Applicable when the system is required to be OPERABLE per Specification 3.5.2 or 3.5.3.

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

\# Not required to be OPERABLE when reactor steam dome pressure is $\leq$ 122 psig.

TABLE 3.3.3-1 (Continued)

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION

ACTION

ACTION 30 - With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip Function requirement:

a. With one channel inoperable, place the inoperable channel in the tripped condition within one hour* or declare the associated system inoperable.

b. With more than one channel inoperable, declare the associated system inoperable.

ACTION 31 - With the number of OPERABLE channels less than required by the Minimum OPERABLE channels per Trip Function, place the inoperable channel in the tripped condition within one hour; restore the inoperable channel to OPERABLE status within 7 days or declare the associated system inoperable.

ACTION 32 - With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip Function requirement, declare the associated ADS trip system or ECCS inoperable.

ACTION 33 - With the number of OPERABLE channels less than the Minimum OPERABLE Channels per Trip Function requirement, place the inoperable channel in the tripped condition within one hour.

ACTION 34 - With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip Function requirement, restore the inoperable channel to OPERABLE status within 8 hours or declare the associated ADS trip system or ECCS inoperable.

ACTION 35 - With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip Function requirement

a. For one trip system, place that trip system in the tripped condition within one hour* or declare the HPCS system inoperable.

b. For both trip systems, declare the HPCS system inoperable.

ACTION 36 - Deleted

ACTION 37 - With the number of OPERABLE instruments less than the Minimum OPERABLE INSTRUMENTS, place the inoperable instrument(s) in the tripped condition within 1 hour* or declare the associated emergency diesel generator inoperable and take the ACTION required by Specification 3.8.1.1 or 3.8.1.2 as appropriate.

*The provisions of Specification 3.0.4 are not applicable.

TABLE 3.3.3-2 (Continued)

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
C. DIVISION 3 TRIP SYSTEM		
1. HPCS SYSTEM		
a. Reactor Vessel Water Level - Low Low, Level 2	≥ - 50 inches*	≥ - 57 inches*
b. Drywell Pressure - High	≤ 1.69 psig	≤ 1.89 psig
c. Reactor Vessel Water Level - High, Level 8	≤ 55.5 inches*	≤ 56 inches*
d. Deleted		
e. Deleted		
f. Pump Discharge Pressure - High	≥ 120 psig	≥ 110 psig
g. HPCS System Flow Rate - Low	≥ 1000 gpm	≥ 900 gpm
h. Manual Intiation	N.A.	N.A.
D. LOSS OF POWER		
1. 4.16 kV Emergency Bus Undervoltage (Loss of Voltage)#		
a. 4.16 kV Buses		
1) Divisions 1 and 2	2625 ± 131 volts with ≤ 10 second time delay	2625 ± 262 volts with ≤ 11 second time delay
	2496 ± 125 volts with ≥ 4 second time delay	2496 ± 250 volts with ≥ 3 second time delay
2) Division 3	2870 ± 143 volts with ≤ 10 second time delay	2870 ± 287 volts with ≤ 11 second time delay

TABLE NOTATIONS

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

#These are inverse time delay voltage relays or instantaneous voltage relays with a time delay. The voltages shown are the maximum that will not result in a trip. Lower voltage conditions will result in decreased trip times.

N.A. Not Applicable

TABLE 3.3.3-2 (Continued)

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
D. LOSS OF POWER (Continued)		
2. 4.16 kV Emergency Bus Undervoltage (Degraded Voltage)		
a. 4.16 kV Buses		
1) Divisions 1, 2 and 3	3814 ± 76 volts with 10 ± 1 seconds time delay with LOCA signal or 5 ± 0.5 minutes time delay without LOCA signal	3814 ± 76 volts with 10 ± 1 seconds time delay with LOCA signal or 5 ± 0.5 minutes time delay without LOCA signal

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. Deleted				
e. Deleted				
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)	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.

SURVEILLANCE REQUIREMENTS (Continued)

2. Performing a CHANNEL CALIBRATION of the:

   a) Discharge line "keep filled" pressure alarm instrumentation and verifying the:

      1) High pressure setpoint and the low pressure setpoint of the:

         (a) LPCS system to be $\leq$ 500 psig and $\geq$ 55 psig, respectively.

         (b) LPCI subsystems to be $\leq$ 400 psig and $\geq$ 55 psig, respectively.

      2) Low pressure setpoint of the HPCS system to be $\geq$ 63 psig.

   b) Header delta P instrumentation and verifying the setpoint of the:

      1) LPCS system and LPCI subsystems to be ± 1 psid.

      2) HPCS system to be 5 ± 2.0 psid greater than the normal indicated ΔP.

3. Deleted

4. Visually inspecting the ECCS corner room watertight door seals and room penetration seals and verifying no abnormal degradation, damage, or obstructions.

d. For the ADS by:

1. At least once per 31 days, performing a CHANNEL FUNCTIONAL TEST of the accumulator backup compressed gas system low pressure alarm system.

2. At least once per 18 months:

   a) Performing a system functional test which includes simulated automatic actuation of the system throughout its emergency operating sequence, but excluding actual valve actuation.

   b) Manually opening each ADS valve and observing the expected change in the indicated valve position.

   c) Performing a CHANNEL CALIBRATION of the accumulator backup compressed gas system low pressure alarm system and verifying an alarm setpoint of 500 + 40, - 0 psig on decreasing pressure.

3/4.5.2 ECCS - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.5.2 At least two of the following shall be OPERABLE:

a. The low pressure core spray (LPCS) system with a flow path capable of taking suction from the suppression chamber and transferring the water through the spray sparger to the reactor vessel.

b. Low pressure coolant injection (LPCI) subsystem "A" of the RHR system with a flow path capable of taking suction from the suppression chamber upon being manually realigned and transferring the water to the reactor vessel.

c. Low pressure coolant injection (LPCI) subsystem "B" of the RHR system with a flow path capable of taking suction from the suppression chamber upon being manually realigned and transferring the water to the reactor vessel.

d. Low pressure coolant injection (LPCI) subsystem "C" of the RHR system with a flow path capable of taking suction from the suppression chamber upon being manually realigned and transferring the water to the reactor vessel.

e. The high pressure core spray (HPCS) system with a flow path capable of taking suction from the suppression pool and transferring the water through the spray sparger to the reactor vessel.

APPLICABILITY: OPERATIONAL CONDITION 4 or 5*.

ACTION:

a. With one of the above required subsystems/systems inoperable, restore at least two subsystems/systems to OPERABLE status within 4 hours or suspend all operations that have a potential for draining the reactor vessel.

b. With both of the above required subsystems/systems inoperable, suspend CORE ALTERATIONS and all operations that have a potential for draining the reactor vessel. Restore at least one subsystem/system to OPERABLE status within 4 hours or establish SECONDARY CONTAINMENT INTEGRITY within the next 8 hours.

---

*The ECCS is not required to be OPERABLE provided that the reactor vessel head is removed, the cavity is flooded, the spent fuel pool gates are removed, and water level is maintained within the limits of Specifications 3.9.8 and 3.9.9.

SURVEILLANCE REQUIREMENTS

4.5.2.1 At least the above required ECCS shall be demonstrated OPERABLE per Surveillance Requirement 4.5.1, except that the header delta P instrumentation is not required to be OPERABLE.

3/4.5.3 SUPPRESSION CHAMBER#

LIMITING CONDITION FOR OPERATION

3.5.3 The suppression chamber shall be OPERABLE:

a. In OPERATIONAL CONDITION 1, 2, or 3 with a contained water volume of at least 128,800 ft^3, equivalent to a level of -4 1/2 inches.**

b. In OPERATIONAL CONDITION 4 or 5* with a contained water volume of at least 70,000 ft^3, equivalent to a level of -12 feet 7 inches.**

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, 3, 4, and 5*.

ACTION:

a. In OPERATIONAL CONDITION 1, 2, or 3 with the suppression chamber water level less than the above limit, restore the water level to within the limit within 1 hour or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. In OPERATIONAL CONDITION 4 or 5* with the suppression chamber water level less than the above limit, suspend CORE ALTERATIONS and all operations that have a potential for draining the reactor vessel and lock the reactor mode switch in the Shutdown position. Establish SECONDARY CONTAINMENT INTEGRITY within 8 hours.

#See Specification 3.6.2.1 for pressure suppression requirements.
*The suppression chamber is not required to be OPERABLE provided that the reactor vessel head is removed, the cavity is flooded or being flooded from the suppression pool, the spent fuel pool gates are removed when the cavity is flooded, and the water level is maintained within the limits of Specifications 3.9.8 and 3.9.9.
**Level is referenced to a plant elevation of 699 feet 11 inches (see Figure B 3/4.6.2-1).

LIMITING CONDITION FOR OPERATION (Continued)

ACTION: (Continued)

c. With one suppression chamber water level instrumentation channel inoperable, restore the inoperable channel to OPERABLE status within 7 days or verify the suppression chamber water level to be greater than or equal to -4 1/2 inches** or -12 feet 7 inches**, as applicable, at least once per 12 hours by local indication.

d. With both suppression chamber water level instrumentation channels inoperable, restore at least one inoperable channel to OPERABLE status within 8 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours and verify the suppression chamber water level to be greater than or equal to -4 1/2 inches** or -12 feet 7 inches**, as applicable, at least once per 12 hours by local indication.

SURVEILLANCE REQUIREMENTS

4.5.3.1 The suppression chamber shall be determined OPERABLE by verifying:

a. The water level to be greater than or equal to, as applicable:

1. -4 1/2 inches** at least once per 24 hours.

2. -12 feet 7 inches** at least once per 12 hours.

b. Two suppression chamber water level instrumentation channels OPERABLE by performance of a:

1. CHANNEL CHECK at least once per 24 hours,

2. CHANNEL FUNCTIONAL TEST at least once per 31 days, and

3. CHANNEL CALIBRATION at least once per 18 months,

with the low water level alarm setpoint at greater than or equal to -3 inches.**

4.5.3.2 With the suppression chamber level less than the above limit in OPERATIONAL CONDITION 5*, at least once per 12 hours verify footnote conditions* to be satisfied.

*The suppression chamber is not required to be OPERABLE provided that the reactor vessel head is removed, the cavity is flooded or being flooded from the suppression pool, the spent fuel pool gates are removed when the cavity is flooded, and the water level is maintained within the limits of Specifications 3.9.8 and 3.9.9.

**Level is referenced to a plant elevation of 699 feet 11 inches (See Figure B 3/4.6.2-1).

TABLE 3.6.3-1 (Continued)

PRIMARY CONTAINMENT ISOLATION VALVES

VALVE FUNCTION AND NUMBER

Other Isolation Valves (Continued)

4. Low Pressure Core Spray System

2E21-F005
2E21-F001(j)
2E21-F012(j)
2E21-F011(j)
2E21-F018(j)
2E21-F031(j)
2E21-F006(k)

5. High Pressure Core Spray System

2E22-F004
2E22-F015(j)
2E22-F023(j)
2E22-F012(j)
2E22-F014(j)
2E22-F005(k)

6. Reactor Core Isolation Cooling System

2E51-F013
2E51-F069
2E51-F028
2E51-F068
2E51-F040
2E51-F031(j)
2E51-F019(j)
2E51-F065(k)
2E51-F066(k)
2E51-F059(m)
2E51-F022(m)
2E51-F362(n)
2E51-F363(n)

TABLE 3.6.3-1 (Continued)

PRIMARY CONTAINMENT ISOLATION VALVES

VALVE FUNCTION AND NUMBER

Other Isolation Valves (Continued)

7. Post LOCA Hydrogen Control

   2HG001A, B
   2HG002A, B
   2HG005A, B
   2HG006A, B

8. Standby Liquid Control System

   2C41-F004A, B
   2C41-F007

9. Reactor Recirculation Seal Injection

   2B33-F013A, B[j]
   2B33-F017A, B[j]

10. Drywell Pneumatic Valves

   2IN018

TABLE 3.6.3-1 (Continued)

PRIMARY CONTAINMENT ISOLATION VALVES

TABLE NOTATIONS

*But ≥ 3 seconds.

(a) See Specification 3.3.2, Table 3.3.2-1, for isolation signal(s) that operates each valve group.
(b) Not included in total sum of Type B and C tests.
(c) May be opened on an intermittent basis under administrative control.
(d) Not closed by SLCS actuation.
(e) Not closed by Trip Functions 5a, b, or c, Specification 3.3.2, Table 3.3.2-1.
(f) Not closed by Trip Functions 4a, c, d, e, or f of Specification 3.3.2, Table 3.3.2-1.
(g) Not subject to Type C leakage test.
(h) Opens on an isolation signal. Valves will be open during Type A test. No Type C test required.
(i) Also closed by drywell pressure-high signal.
(j) Hydraulic leak test at 43.6 psig.
(k) Not subject to Type C leakage test - leakage rate tested per Specification 4.4.3.2.2.
(l) These penetrations are provided with removable spools outboard of the outboard isolation valve. During operation, these lines will be blind flanged using a double O-ring and a type B leak test. In addition, the packing of these isolation valves will be soap-bubble tested to ensure insignificant or no leakage at the containment test pressure each refueling outage.
(m) If valves 2E51-F362 and 2E51-F363 are locked closed and acceptably leak rate tested, then valves 2E51-F059 and 2E51-F022 are not considered to be primary containment isolation valves and are not required to be leak rate tested.

(n) Either the 2E51-F362 or the 2E51-F363 valve may be open when the RCIC system is in the standby mode of operation, and both valves may be open during operation of the RCIC system in the full flow test mode, providing that:

1) valve 2E51-F022 is acceptably leak rate tested, and

2) valve 2E51-F059 is deactivated, locked closed and acceptably leak rate tested, and

3) the spectacle flange, installed immediately downstream of the 2E51-F059 valve, is closed and acceptably leak rate tested.

CONTAINMENT SYSTEMS

3/4.6.4 VACUUM RELIEF

LIMITING CONDITION FOR OPERATION

3.6.4 All suppression chamber - drywell vacuum breakers shall be OPERABLE and closed.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

a. With one suppression chamber - drywell vacuum breaker inoperable and/or open, within 4 hours close the manual isolation valves on both sides of the inoperable and/or open vacuum breaker. Restore the inoperable and/or open vacuum breaker to OPERABLE and closed status within 72 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. With one position indicator of any OPERABLE suppression chamber - drywell vacuum breaker inoperable, restore the inoperable position indicator to OPERABLE status within 14 days or visually verify the vacuum breaker to be closed at least once per 24 hours. Otherwise, declare the vacuum breaker inoperable.

SURVEILLANCE REQUIREMENTS

4.6.4.1 Each suppression chamber - drywell vacuum breaker shall be:

a. Verified closed at least once per 7 days.

b. Demonstrated OPERABLE:

   1. At least once per 31 days and within 12 hours after any discharge of steam to the suppression chamber from the safety-relief valves, by cycling each vacuum breaker through at least one complete cycle of full travel.

   2. At least once per 31 days by verifying both position indicators OPERABLE by performance of a CHANNEL FUNCTIONAL TEST.

3. At least once per 18 months by;

   a) Verifying the force required to open the vacuum breaker, from the closed position, to be less than or equal to 0.5 psid, and

   b) Verifying both position indicators OPERABLE by performance of a CHANNEL CALIBRATION.

TABLE 3.8.3.3-1 (Continued)

MOTOR OPERATED VALVES THERMAL OVERLOAD PROTECTION

	VALVE NUMBER	BYPASS DEVICE (Continuous)(Accident Conditions)	SYSTEM(S) AFFECTED
	2E32 - F003N	Accident Conditions	
	2E32 - F006	Accident Conditions	
	2E32 - F007	Accident Conditions	
	2E32 - F008	Accident Conditions	
	2E32 - F009	Accident Conditions	
m.	2E22 - F004	Accident Conditions	HPCS system
	2E22 - F012	Accident Conditions	
	2E22 - F015	Continuous	
	2E22 - F023	Accident Conditions	

BASES

3/4.5.1 and 3/4.5.2 ECCS - OPERATING and SHUTDOWN

ECCS Division 1 consists of the low pressure core spray system, low pressure coolant injection subsystem "A" of the RHR system, and the automatic depressurization system (ADS) as actuated by ADS trip system "A". ECCS Division 2 consists of low pressure coolant injection subsystems "B"and "C" of the RHR system and the automatic depressurization system as actuated by ADS trip system "B".

The low pressure core spray (LPCS) system is provided to assure that the core is adequately cooled following a loss-of-coolant accident and provides adequate core cooling capacity for all break sizes up to and including the double-ended reactor recirculation line break, and for transients or smaller breaks following depressurization by the ADS.

The LPCS is a primary source of emergency core cooling after the reactor vessel is depressurized and a source for flooding of the core in case of accidental draining.

The surveillance requirements provide adequate assurance that the LPCS system will be OPERABLE when required. Although all active components are testable and full flow can be demonstrated by recirculation through a test loop during reactor operation, a complete functional test requires reactor shutdown. The pump discharge piping is maintained full to prevent water hammer damage to piping and to start cooling at the earliest moment.

The low pressure coolant injection (LPCI) mode of the RHR system is provided to assure that the core is adequately cooled following a loss-of-coolant accident. Three subsystems, each with one pump, provide adequate core flooding for all break sizes up to and including the double-ended reactor recirculation line break, and for transients or small breaks following depressurization by the ADS.

The surveillance requirements provide adequate assurance that the LPCI system will be OPERABLE when required. Although all active components are testable and full flow can be demonstrated by recirculation through a test loop during reactor operation, a complete functional test requires reactor shutdown. The pump discharge piping is maintained full to prevent water hammer damage to piping and to start cooling at the earliest moment.

ECCS Division 3 consists of the high pressure core spray system. The high pressure core spray (HPCS) system is provided to assure that the reactor core is adequately cooled to limit fuel clad temperature in the event of a small break in the reactor coolant system and loss of coolant which does not result in rapid depressurization of the reactor vessel. The HPCS system permits the reactor to be shut down while maintaining sufficient reactor vessel water level inventory until the vessel is depressurized. The HPCS system operates over a range of 1160 psid, differential pressure between reactor vessel and HPCS suction source, to 0 psid.

The capacity of the HPCS system is selected to provide the required core cooling. The HPCS pump is designed to deliver greater than or equal to 516/1550/6200 gpm at differential pressures of 1160/1130/200 psid. Water is taken from the suppression pool and injected into the reactor.

ECCS-OPERATING and SHUTDOWN (Continued)

With the HPCS system inoperable, adequate core cooling is assured by the OPERABILITY of the redundant and diversified automatic depressurization system and both the LPCS and LPCI systems. In addition, the reactor core isolation cooling (RCIC) system, a system for which no credit is taken in the hazards analysis, will automatically provide makeup at reactor operating pressures on a reactor low water level condition. The HPCS out-of-service period of 14 days is based on the demonstrated OPERABILITY of redundant and diversified low pressure core cooling systems.

The surveillance requirements provide adequate assurance that the HPCS system will be OPERABLE when required. Although all active components are testable and full flow can be demonstrated by recirculation through a test loop during reactor operation, a complete functional test with reactor vessel injection requires reactor shutdown. The pump discharge piping is maintained full to prevent water hammer damage and to provide cooling at the earliest moment.

Upon failure of the HPCS system to function properly, if required, the automatic depressurization system (ADS) automatically causes selected safety-relief valves to open, depressurizing the reactor so that flow from the low pressure core cooling systems can enter the core in time to limit fuel cladding temperature to less than 2200°F. ADS is conservatively required to be OPERABLE whenever reactor vessel pressure exceeds 122 psig even though low pressure core cooling systems provide adequate core cooling up to 350 psig.

ADS automatically controls seven selected safety-relief valves. Six valves are required to be OPERABLE since the LOCA analysis assumes 6 ADS valves in addition to a single failure. It is therefore appropriate to permit one of the required valves to be out-of-service for up to 14 days without materially reducing system reliability.

3/4.5.3 SUPPRESSION CHAMBER

The suppression chamber is also required to be OPERABLE as part of the ECCS to ensure that a sufficient supply of water is available to the HPCS, LPCS and LPCI systems in the event of a LOCA. This limit on suppression chamber minimum water volume ensures that sufficient water is available to permit recirculation cooling flow to the core (See Figure B 3/4.6.2-1). The OPERABILITY of the suppression chamber in OPERATIONAL CONDITIONS 1, 2 or 3 is required by Specification 3.6.2.1.

Repair work might require making the suppression chamber inoperable. This specification will permit those repairs to be made and at the same time give assurance that the irradiated fuel has an adequate cooling water supply when the suppression chamber must be made inoperable in OPERATIONAL CONDITION 4 or 5.

In OPERATIONAL CONDITION 4 and 5 the suppression chamber minimum required water volume is reduced because the reactor coolant is maintained at or below 200°F. Since pressure suppression is not required below 212°F, the minimum water volume is based on NPSH, recirculation volume, vortex prevention plus a 2'-4" safety margin for conservatism.

Suppression Chamber Level	Plant Elevation	Control Room/Local Indication
26' 10"	700' 2"	+3" High Level LCO (Volume 131,900 ft^3)
26' 9"	700' 1"	+2" High Level Alarm
26' 7"	699' 11"	0" Instrument Zero
26' 4"	699' 8"	-3" Low Level Alarm
26' 2 1/2"	699' 6 1/2"	-4 1/2" Low Level LCO Operational Condition 1, 2, or 3 (Volume 128,800 ft^3)
14'	687' 4"	-12' 7" Low Level LCO Operational Condition 4 or 5 (Volume 70,000 ft^3)

SUPPRESSION POOL LEVEL SETPOINTS

BASES FIGURE B 3/4.6.2-1

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 81 TO FACILITY OPERATING LICENSE NO. NPF-11 AND

AMENDMENT NO. 65 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 BACKGROUND

Commonwealth Edison Company (CECo) submitted by letter dated October 10, 1990, an application to amend the Technical Specifications (TS) for the LaSalle County Station, Units 1 and 2, to modify their existing High Pressure Core Spray (HPCS) and Reactor Core Isolation Cooling (RCIC) systems due to two underground piping failures they had experienced in 1985. The initial submittal was reviewed by the staff and in response to the deficiencies noted by the NRC staff and the licensee's own technical staff, the licensee submitted an amended application by letter dated October 16, 1991. This letter provided additional clarifying information that did not change the initial proposed no significant hazards consideration determination.

The normal surveillance piping for both the RCIC and HPCS systems takes suction from the Condensate Storage Tank (CST) and discharges back to the CST via a full flow test return line. Both the suction and return lines pass underground. Both systems also have the ability to take suction from the suppression pool. In addition, the HPCS system has a full flow test return line to the suppression pool that can be used as an alternate means of testing the HPCS system. The RCIC system is not equipped with this feature. The suction and return piping to the suppression pool does not run below ground on either system.

During a preservice inspection and hyrdrostatic test, a failure was discovered on an above ground elbow joint weld located immediately adjacent to the CST on the Unit 2 RCIC system. This failure was attributed to intergranular corrosion. The licensee was concerned about the integrity of the underground piping and decided to replace the underground portions of the system with a heavier schedule pipe wrapped in a material designed to reduce the potential for intergranular corrosion at the weld joints.

The second failure was discovered in May 1985 and involved the HPCS piping. The failure was detected when water seepage was observed at ground level above the HPCS full flow return line to the CST. This failure was attributed to a form of microbiological corrosion which primarily affects the weld material. The problem was temporarily corrected by isolating the failed line and aligning the HPCS system to the suppression pool.

Due to the failure of the underground HPCS piping, the licensee performed an evaluation to determine the effect on plant operations of a similar failure on the RCIC piping. It was determined that there would be no way to perform the quarterly surveillance test without directly injecting water into the reactor vessel. Based on the results of this evaluation, the licensee decided on the following corrective actions for both systems:

1. Periodic tests are being performed to monitor the integrity of the underground RCIC piping.

2. Improvements to the current cathodic protection system were made and plans to install additional deep anode systems that will help to minimize corrosion damage to the underground piping over the lifetime of the plant have been initiated.

3. The licensee decided to permanently isolate and abandon in place the HPCS system suction and full flow test return lines to the CST (see Figure 1).

4. A modification which will give full flow test capability to the suppression pool for the RCIC system will be installed during each unit's refueling outage (see Figure 2). This modification would be used should a future failure occur to the underground RCIC piping.

As a result of the HPCS system modifications and planned modifications to the RCIC system, the licensee determined that changes to the TS were required to achieve consistency between the system design modifications and the TS requirements. These changes are contained in the licensee's October 10, 1990, and October 16, 1991 submittals.

2.0 DISCUSSION

2.1 Proposed HPCS TS Changes

Since the discovery of leakage from the HPCS system suction and full flow test lines to the CST, these lines have been isolated and administratively controlled. The licensee's HPCS system modifications will permanently isolate the suction and test lines. As a result, the current TS requirements for the HPCS system suction flowpath transfer will no longer be required. The licensee proposed the following changes to the TS in their October 10, 1990, submittal.

- Remove instrumentation requirements for the CST low level trip and suppression pool high water level trip from the ESF Division 3 requirements in Tables 3.3.3-1 (including ACTION 36), 3.3.3-2 and 4.3.3.1-1. Since, the HPCS system suction is permanently lined up to the suppression pool, neither of these trips are required for safe plant operation.

- Remove Surveillance Requirement 4.5.1.c.3 which requires verification of the HPCS system suction flow path automatic transfer from the CST to the suppression pool. Since, the HPCS system suction flow is permanently lined up to the suppression pool there is no longer any need to test the automatic transfer capability.

- Remove references to the CST from the HPCS system flow path requirements in TS Limiting Condition For Operation (LCO) 3.5.2.e. The CST flow path is not required for plant safety but was allowed to be used during shutdown as an alternate water source when the suppression pool was drained. This option is no longer available and will no longer be used.

- Remove Surveillance Requirement 4.5.2.2 for verification of HPCS system operability by verifying the acceptability of the CST water level. With the LCO deleted this surveillance would no longer be required.

- Remove the references to the CST from the suppression pool volume requirements in TS LCO 3.5.3.b. Since the CST is no longer available via the HPCS system flow path, exceptions to the suppression pool level requirements that rely on the additional water volume contained in the CST can no longer be allowed. Surveillance Requirement 4.5.3.2 must be amended to remove Paragraph "a" which refers back to the conditions being deleted from TS 3.5.3.b. Since the remainder of Surveillance Requirement 4.5.3.2 is applicable only in Operational Condition 5, the reference to Operational Condition 4 is being removed.

- Remove the reference to the CST from the bases of TS 3/4.5.1 and 3/4.5.2.

- Remove the reference to the HPCS suction valve transfer from Bases Figure B 3/4.6.2-1.

Evaluation

The HPCS system is designed to be normally aligned to take suction from the CST, which provides an additional source of primary system water. The suction valves to the suppression pool would normally be closed in this configuration. The UFSAR states that credit for this water is not taken in the accident analysis; therefore, the CST is only expected to function during normal plant conditions and performs no safety function. Since no credit for the water in the CST is taken in the LaSalle accident analysis, the staff has concluded that removal of the associated requirements in the TS is acceptable. On this basis the staff has reviewed the proposed TS changes for removal of CST requirements described above and finds them acceptable.

The licensee's addendum of October 16, 1991, requested that Table 3.8.3.3-1 be additionally revised to change the thermal overload bypass device on motor operated valve (MOV) 1(2)E22-F015 from accident to continuous operating conditions. This valve is located on the HPCS suction line from the suppression pool. In the new configuration, this valve will normally be open. Closure of this valve would occur only in the event of a suction line break on the

HPCS system suction line, and for maintenance. The licensee has stated that since this valve is not manipulated, thermal overload protection during normal operations is not needed. However, a bypass of the thermal overload protection to prevent this circuit from inhibiting the valve from performing its safety function is required.

Evaluation

Regulatory Guide 1.106 (RG 1.106) provides the staff position on thermal overload protection for electric motors on MOVs. In order to ensure that safety-related MOVs which are equipped with thermal overload protective devices will perform their safety function, two types of bypasses may be used. The thermal overload protection can either be continuously bypassed or bypassed under accident conditions only. RG 1.106 states that provided that the completion of the safety function is not jeopardized or that other safety systems are not degraded, then the thermal overload protection devices should be continuously bypassed and temporarily placed in force only when the valve motors are undergoing periodic or maintenance testing. Accident condition bypasses should be used where the thermal overload protection is required for valves that are regularly manipulated during normal plant operations. Since this valve will not be manipulated during normal operations, an accident condition bypass is not required for the HPCS suction valve. On this basis, the staff finds the use of a continuous bypass acceptable.

2.2 Proposed RCIC TS Changes

The licensee's planned modification to install an RCIC system full flow test line to the suppression chamber is being performed to minimize the impact on plant operations of any future RCIC underground piping failure. TS 3.7.3 requires the RCIC system to be operable and to have an operable flow path from the suppression pool to the reactor vessel. A failure of the RCIC underground lines to the CST would not necessarily render the system inoperable immediately because the specified flow path from the suppression pool to the reactor would be unaffected. However, if a failure involved the full flow test line to the CST, performance of the quarterly pump performance surveillance would be difficult (Surveillance Requirement 4.7.3.b). Under these conditions the only means of performing this test would be by actual injection to the reactor vessel. Since an RCIC system injection causes a trip of the main turbine and the feedwater pump turbines, reactor power would have to be reduced and the turbines would have to be shut down prior to performing the surveillance.

This would be a costly option since it would require a power reduction outage every three months until repairs or modifications to the underground piping could be completed. The licensee's preferred and planned option (refer to Figure 2) is to install (1) an additional full flow test line to the suppression pool for the RCIC system, and (2) to install a flanged joint downstream of the 1(2)E51-F059 valve where a blind flange can be installed. This option will increase the RCIC system flexibility, such that an underground piping failure would have a minimal impact on plant operation and safety. This modification is scheduled to be installed during the next refueling outage for each unit. (Unit 1, October 1992, Unit 2, January 1992.)

As a result of these planned modifications, the licensee has proposed TS changes which add the RCIC system full flow test valves 1(2)E51-F059, 1(2)E51-F022, 1(2)E51-F362, and 1(2)E51-F363 to Table 3.6.3-1, "Primary Containment Isolation Valves." The licensee proposed the following changes to the TS in their October 10, 1990, submittal to be implemented prior to startup from the next refueling outage for each unit.

- Add the following valves and associated footnotes to the Primary Containment Isolation Valves list, Table 3.6.3-1:

  1(2)E51-F059 footnote m
  1(2)E51-F022 footnotes j and m
  1(2)E51-F362 footnote n
  1(2)E51-F363 footnote n

- Relocate existing footnotes from Table 3.6.3-1 to new page 3/4 6-34a for Unit 1 and 3/4 6-37a for Unit 2. Add new footnotes m and n as follows:

  m. If valves 1(2)E51-F362 and 1(2)E51-F363 are locked closed and acceptably leak rate tested, then valves 1(2)E51-F059 and 1(2)E51-F022 are not considered to be primary containment isolation valves and are not required to be leak rate tested.

  n. If valve 1(2)E51-F059 is deactivated and locked closed with the line blind flanged downstream of the valve and acceptably leakage rate tested, valves 1(2)E51-F362 and 1(2)E51-F363 are not considered primary containment isolation valves and are not subject to leakage rate testing requirements.

- Remove motor operated valves 1(2)E22-F001, 1(2)E22-F010, and 1(2)E22-F011 from Table 3.8.3.3-1 since these valves are to be permanently out-of-service closed.

Evaluation

As a result of these modifications to the RCIC system, the licensee proposed that valves E51-F022, E51-F059, E51-F362, and E51-F363 be added to Table 3.6.3-1, "Primary Containment Isolation Valves." Since this will ensure that primary containment integrity is maintained by instituting appropriate controls for operation and testing of these valves as required, the staff finds this acceptable.

The licensee also proposed to add footnotes m and n to TS 3.6.3-1. If the system is lined up to CST, valves E51-F362 and E51-F363 will be considered to be primary containment isolation valves and will be locked closed. In that case, footnote m will be applicable which states, "if valves E51-F362 and

E51-F363 are locked closed and acceptably leak rate tested, then valves E51-F059 and E51-F022 are not considered to be primary containment isolation valves and are not required to be leak rate tested." The staff finds the above footnote m acceptable as double locked closed manual isolation valves meet the requirements of General Design Criteria (GDC) 56, for primary containment isolation.

The licensee proposed that if the system were lined up to the suppression pool for test purposes, footnote n would be applicable. Footnote n as proposed stated "if valve E51-F059 is deactivated and locked closed with the line blind flanged downstream of the valve and acceptably leak rate tested, valves E51-F362 and E51-F363 are not considered primary containment isolation valves and are not subjected to leakage rate testing requirements." The licensee indicated that containment isolation would be provided by motor-operated isolation valve E51-F022 and the RCIC piping boundary. The staff considered that a single isolation valve was not acceptable for a new line and that if the system were lined up to the suppression pool, the licensee should designate one of the two manual isolation valves E51-F362 or E51-F363 as an isolation valve and keep it locked closed except during surveillance testing, when it would be under administrative controls. In a conference call with the staff on July 25, 1991, the licensee agreed to the above requirement and in letter dated October 16, 1991, revised the wording in footnote n of Table 3.6.3-1 to read as follows:

n. Either the 1(2)E51-F362 or 1(2)E51-F363 valve may be open when the RCIC system is in the standby mode of operation, and both valves may be open during operation of the RCIC system in the full flow test mode, providing that:

   1) valve 1(2)E51-F022 is acceptably leak rate tested, and

   2) valve 1(2)E51-F059 is deactivated, locked closed and acceptably leak rate tested, and

   3) the spectacle flange, installed immediately downstream of the 1(2)E51-F059 valve, is closed and acceptably leak rate tested.

The staff has concluded that the revised footnote n meets the containment isolation requirements of ´)C 56 and is, therefore, acceptable.

2.3 Proposed Adminstrative TS Changes

The licensee's October 10, 1990, and October 16, 1991, submittals also proposed several administrative TS changes and correction of typographical errors. The most significant of these changes included:

- Deletion of Footnote "***" located in Tables 4.3.3.1-1 and 3.6.3-1 of the Unit 1 TS. These footnotes allowed several 18-month interval surveillance requirements to be waived during the first cycle of operation.

- Removal of the reference to footnote "j" from valve 1(2)E51-F022 in Table 3.6.3-1 which was inadvertently included in the October 10, 1990 submittal.

- The inadvertent omission of the deletion of footnote (d) from page 3/4 3-26 for Unit 1.

The staff has reviewed these changes and determined that they are administrative in nature and have no impact on plant safety. On this basis, the staff finds them acceptable.

3.0 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.

4.0 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 and change surveillance requirements. 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 (55 FR 47569). Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Other changes are administrative and are eligible for categorical enclosure under 10 CFR 51.22(c)(10). 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.

5.0 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 op- ation 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.

Attachment: Figure 1, HPCS System Modifications
Figure 2, RCIC System Modifications

Principal Contributor: B. Siegel/R. Elliott/R. Goel/M. Razzaque

Date: November 27, 1991

FIGURE 1

HPCS System Current Alignment

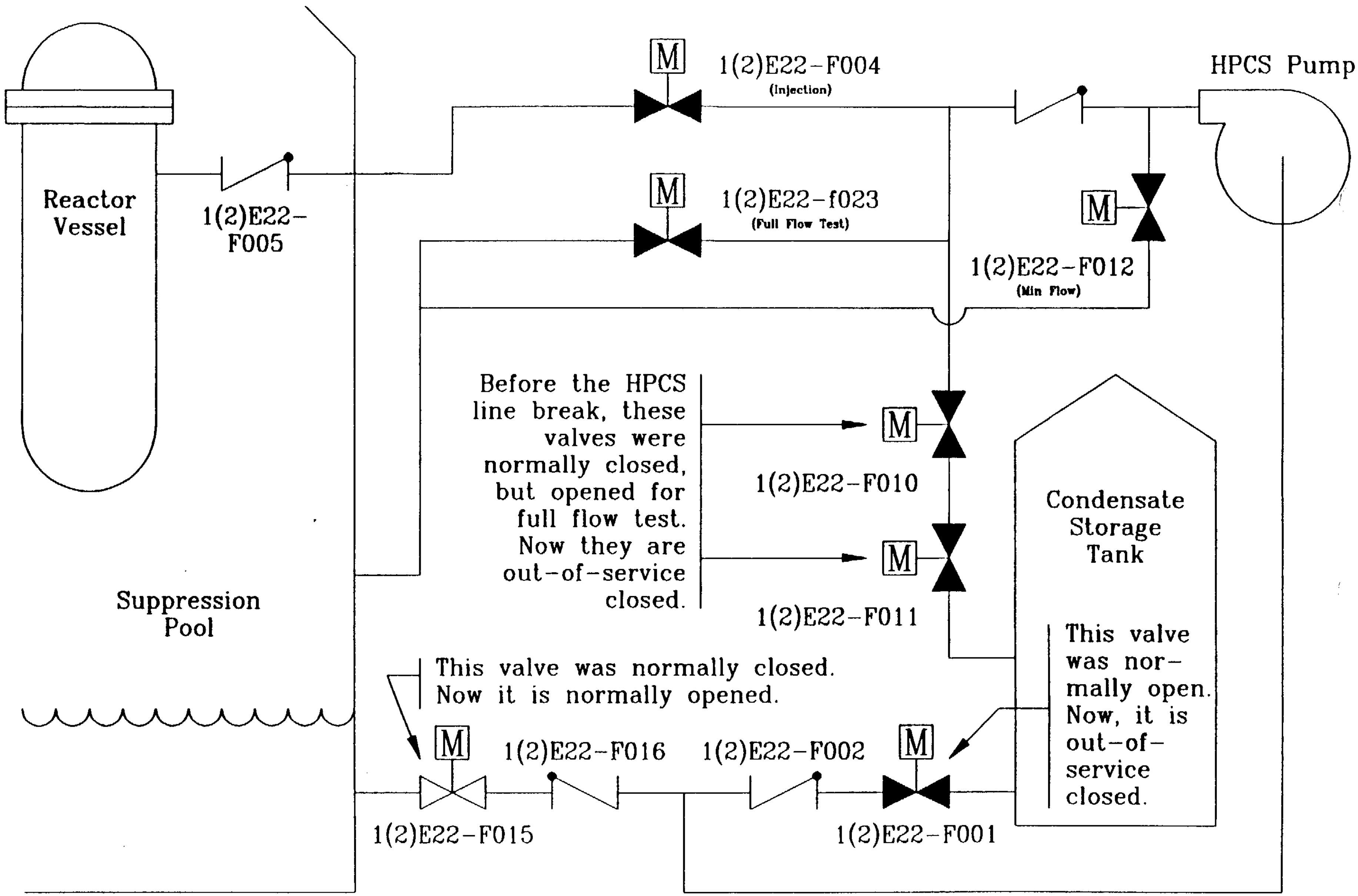
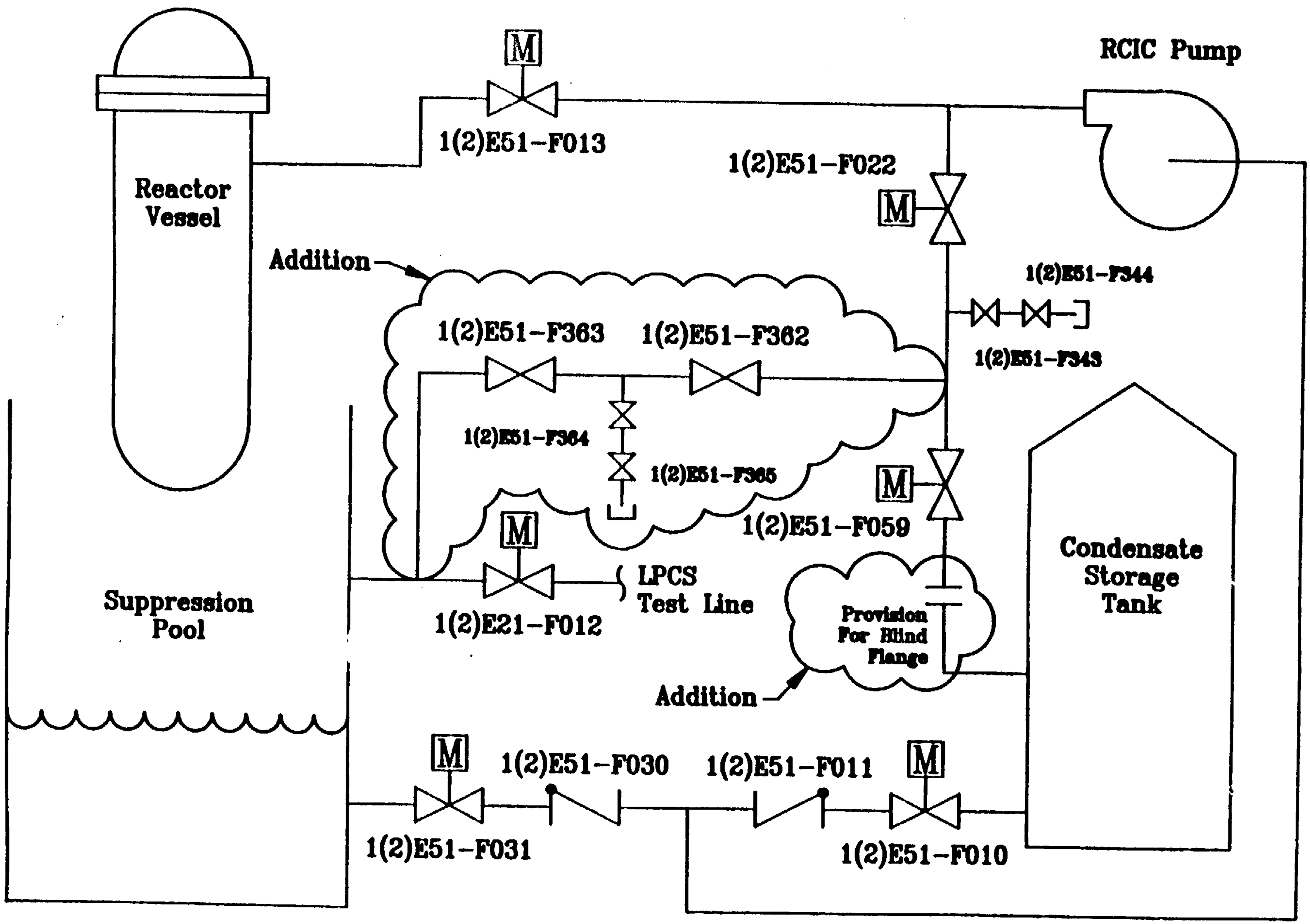


Figure 2

RCIC Full Flow Test Return To Suppression Pool



Docket File

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

November 27, 1991

MEMORANDUM FOR: Sholly Coordinator

FROM: Byron L. Siegel, Project Manager
Project Directorate III-2
Division of Reactor Projects - III/IV/V

SUBJECT: REQUEST FOR PUBLICATION IN BIWEEKLY FR NOTICE - NOTICE OF ISSUANCE OF AMENDMENTS TO FACILITY OPERATING LICENSES (TAC NOS. M77812 AND M77813)

Commonwealth Edison Company, Docket Nos. 50-373 and 50-374, LaSalle County Station, Units 1 and 2, LaSalle County, Illinois

Date of application for amendments: October 10, 1990, as amended by letter dated October 16, 1991.

Brief description of amendments: This amendment modifies the Technical Specification (TS) requirements for the High Pressure Core Spray (HPCS) system by permanently aligning the HPCS suction to the suppression pool and removing all TS requirements for aligning the HPCS suction to the condensate storage tank. In addition, containment isolation requirements are added for the new Reactor Core Isolation Cooling (RCIC) full flow test line to the suppression pool.

Date of issuance: November 27, 1991

Effective date: November 27, 1991

Amendment Nos.: 81 and 65

Facility Operating License Nos. NPF-11 and NPF-18. The amendments revised the Technical Specifications.

Date of initial notice in FEDERAL REGISTER: November 14, 1990 (55 FR 47569) The October 16, 1991, letter provided additional clarifying information that did not change the initial proposed no significant hazards consideration.

DF01

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated November 27, 1991.

No significant hazards consideration comments received: No

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

Original Signed By:

Byron L. Siegel, Project Manager
Project Directorate III-2
Division of Reactor Projects - III/IV/V

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Date:	11/18/91	11/13/91	11/17/91	11/14/91