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

NOV 0 1 1985

Docket No. 50-373

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

Dear Mr. Farrar:

SUBJECT: ISSUANCE OF AMENDMENT NO. 29 TO FACILITY OPERATING LICENSE NO. NPF-11 - LA SALLE COUNTY STATION, UNIT 1

The Nuclear Regulatory Commission has issued the enclosed Amendment No. 29 to Facility Operating License No. NPF-11 for the La Salle County Station, Unit 1. This amendment is in response to your letter dated July 15, 1985. The amendment incorporates the modification of the automatic depressurization system logic as required by License Condition 2.C.30(1)(b).

A copy of the related safety evaluation supporting Amendment No.29 to Facility Operating License NPF-11 is enclosed.

Sincerely,

Walter R. Butler

Walter R. Butler, Chief
Licensing Branch No. 2
Division of Licensing

Enclosures:
1. Amendment No. 29 to NPF-11
2. Safety Evaluation

cc w/enclosures:
See next page

Mr. Dennis L. Farrar
Commonwealth Edison Company

La Salle County Nuclear Power Station
Units 1 & 2

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

COMMONWEALTH EDISON COMPANY

DOCKET NO. 50-373

LA SALLE COUNTY STATION, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 29
License No. NPF-11

1. The Nuclear Regulatory Commission (the Commission or the NRC) having found that:

   A. The application for amendment filed by the Commonwealth Edison Company, dated July 15, 1985, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's regulations set forth in 10 CFR Chapter I;

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

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

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

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

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and paragraph 2.C.(2) of 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. 29 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 startup following the first refueling outage.

FOR THE NUCLEAR REGULATORY COMMISSION

Walter R. Butler

Walter R. Butler, Chief
Licensing Branch No. 2
Division of Licensing

Enclosure:
Changes to the Technical
Specifications

Date of Issuance: NOV 0 1 1985

ENCLOSURE TO LICENSE AMENDMENT NO. 29
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 3-24	3/4 3-24
3/4 3-25	3/4 3-25
3/4 3-27	3/4 3-27
3/4 3-28	3/4 3-28
3/4 3-29	3/4 3-29
3/4 3-32	3/4 3-32
3/4 3-33	3/4 3-33
3/4 5-1	3/4 5-1
B 3/4 5-1	B 3/4 5-1
B 3/4 5-2	B 3/4 5-2

TABLE 3.3.3-1

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION

TRIP FUNCTION	MINIMUM OPERABLE CHANNELS PER TRIP FUNCTION[a]	APPLICABLE OPERATIONAL CONDITIONS	ACTION
A. DIVISION I TRIP SYSTEM			
1. RHR-A (LPCI MODE) & LPCS SYSTEM			
a. Reactor Vessel Water Level - Low Low Low, Level 1	2[b]	1, 2, 3, 4*, 5*	30
b. Drywell Pressure - High	2[b]	1, 2, 3	30
c. LPCS Pump Discharge Flow-Low (Bypass)	1	1, 2, 3, 4*, 5*	31
d. LPCS and LPCI A Injection Valve Injection Line Pressure-Low (Permissive)	1/valve	1, 2, 3 4*, 5*	32 33
e. LPCS and LPCI A Injection Valve Reactor Pressure-Low (Permissive)	2	1, 2, 3 4*, 5*	38 33
f. LPCI Pump A Start Time Delay Relay	1	1, 2, 3, 4*, 5*	32
g. LPCI Pump A Discharge Flow-Low (Bypass)	1	1, 2, 3, 4*, 5*	31
h. Manual Initiation	1/division	1, 2, 3, 4*, 5*	34
2. AUTOMATIC DEPRESSURIZATION SYSTEM TRIP SYSTEM "A"#			
a. Reactor Vessel Water Level - Low Low Low, Level 1 coincident with	2[b]	1, 2, 3	30
b. Drywell Pressure - High	2[b]	1, 2, 3	30
c. Initiation Timer	1	1, 2, 3	32
d. Reactor Vessel Water Level - Low, Level 3 (Permissive)	1	1, 2, 3	32
e. LPCS Pump Discharge Pressure-High (Permissive)	2	1, 2, 3	32
f. LPCI Pump A Discharge Pressure-High (Permissive)	2	1, 2, 3	32
g. Manual Initiation	1/division	1, 2, 3	34
h. Drywell Pressure Bypass Timer	1	1, 2, 3	32
i. Manual Inhibit	1/division	1, 2, 3	34

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
B. DIVISION 2 TRIP SYSTEM			
1. RHR B & C (LPCI MODE)			
a. Reactor Vessel Water Level - Low, Low Low, Level 1	2[b]	1, 2, 3, 4*, 5*	30
b. Drywell Pressure - High	2[b]	1, 2, 3	30
c. LPCI B and C Injection Valve Injection Line Pressure-Low (Permissive)	1/valve	1, 2, 3 4*, 5*	32 33
d. LPCI Pump B Start Time Delay Relay	1	1, 2, 3, 4*, 5*	32
e. LPCI Pump Discharge Flow - Low (Bypass)	1/pump	1, 2, 3, 4*, 5*	31
f. Manual Initiation	1/division	1, 2, 3, 4*, 5*	34
g. LPCI B and C Valve Reactor Pressure-Low (Permissive)	2	1, 2, 3 4*, 5*	38 33
2. AUTOMATIC DEPRESSURIZATION SYSTEM TRIP SYSTEM "B"#			
a. Reactor Vessel Water Level - Low Low Low, Level 1 coincident with	2[b]	1, 2, 3	30
b. Drywell Pressure - High	2[b]	1, 2, 3	30
c. Initiation Timer	1	1, 2, 3	32
d. Reactor Vessel Water Level - Low, Level 3 (Permissive)	1	1, 2, 3	32
e. LPCI Pump B and C Discharge Pressure - High (Permissive)	2/pump	1, 2, 3	32
f. Manual Initiation	1/division	1, 2, 3	34
g. Drywell Pressure Bypass Timer	1	1, 2, 3	32
h. Manual Inhibit	1/division	1, 2, 3	34

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 - With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip Function requirement, place at least one inoperable channel in the tripped condition within one hour* or declare the HPCS system inoperable.

ACTION 37 - With the number of OPERABLE channels less than the Total Number of Channels, 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

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
A. DIVISION 1 TRIP SYSTEM		
1. RHR-A (LPCI MODE) AND LPCS SYSTEM		
a. Reactor Vessel Water Level - Low Low Low, Level 1	$\geq$ -129 inches*	$\geq$ -136 inches*
b. Drywell Pressure - High	$\leq$ 1.69 psig	$\leq$ 1.89 psig
c. LPCS Pump Discharge Flow-Low	$\geq$ 750 gpm	$\geq$ 640 gpm
d. LPCS and LPCI A Injection Valve Injection Line-Low Pressure Interlock	500 psig	500 ± 20 psig
e. LPCS and LPCI A Injection Valve Reactor Pressure-Low Pressure Interlock	500 psig	500 ± 20 psig
f. LPCI Pump A Start Time Delay Relay	$\leq$ 5 seconds	$\leq$ 6 seconds
g. LPCI Pump A Discharge Flow-Low	$\geq$ 1000 gpm	$\geq$ 550 gpm
h. Manual Initiation	NA	NA
2. AUTOMATIC DEPRESSURIZATION SYSTEM TRIP SYSTEM "A"		
a. Reactor Vessel Water Level - Low Low Low, Level 1	$\geq$ -129 inches*	$\geq$ -136 inches*
b. Drywell Pressure - High	$\leq$ 1.69 psig	$\leq$ 1.89 psig
c. Initiation Timer	$\leq$ 105 seconds	$\leq$ 117 seconds
d. Reactor Vessel Water Level-Low, Level 3	$\geq$ 12.5 inches*	$\geq$ 11 inches*
e. LPCS Pump Discharge Pressure-High	$\geq$ 146 psig, increasing	$\geq$ 136 psig, increasing
f. LPCI Pump A Discharge Pressure-High	$\geq$ 119 psig, increasing	$\geq$ 106 psig, increasing
g. Manual Initiation	NA	NA
h. Drywell Pressure Bypass Timer	$\leq$ 9.0 minutes	Footnote (a)
i. Manual Inhibit	NA	NA

(a) The sum of the time delays associated with the ADS initiation timer and the drywell pressure bypass time shall be less than or equal to 687 seconds.

TABLE 3.3.3-2 (Continued)

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
B. DIVISION 2 TRIP SYSTEM		
1. RHR B AND C (LPCI MODE)		
a. Reactor Vessel Water Level - Low Low Low, Level 1	≥ -129 inches*	≥ -136 inches*
b. Drywell Pressure - High	≤ 1.69 psig	≤ 1.89 psig
c. LPCI B and C Injection Valve Injection Line-Low Pressure Interlock	500 psig	500 psig ±20 psig
d. LPCI Pump B Start Time Delay Relay	≤ 5 seconds	≤ 6 seconds
e. LPCI Pump Discharge Flow-Low	≥ 1000 gpm	≥ 550 gpm
f. Manual Initiation	NA	NA
g. LPCI B and C Injection Valve Reactor Pressure Low Pressure Interlock	500 psig	500 ± 20 psig
2. AUTOMATIC DEPRESSURIZATION SYSTEM TRIP SYSTEM "B"		
a. Reactor Vessel Water Level - Low Low Low, Level 1	≥ -129 inches*	≥ -136 inches*
b. Drywell Pressure - High	≤ 1.69 psig	≤ 1.89 psig
c. Initiation Timer	≤ 105 seconds	≤ 117 seconds
d. Reactor Vessel Water Level-Low, Level 3	≥ 12.5 inches*	≥ 11 inches*
e. LPCI Pump B amd C Discharge Pressure-High	≥ 119 psig, increasing	≥ 106 psig, increasing
f. Manual Initiation	NA	NA
g. Drywell Pressure Bypass Timer	≤ 9.0 minutes	Footnote (a)
h. Manual Inhibit	NA	NA

(a) The sum of the time delays associated with the ADS initiation timer and the drywell pressure bypass time shall be less than or equal to 687 seconds.

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

3/4.5 EMERGENCY CORE COOLING SYSTEMS

3/4.5.1 ECCS - OPERATING

LIMITING CONDITION FOR OPERATION

3.5.1 ECCS divisions 1, 2 and 3 shall be OPERABLE with:

a. ECCS division 1 consisting of:

1. The OPERABLE 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.

2. The OPERABLE low pressure coolant injection (LPCI) subsystem "A" of the RHR system with a flow path capable of taking suction from the suppression chamber and transferring the water to the reactor vessel.

3. At least 6 OPERABLE** ADS valves.

b. ECCS division 2 consisting of:

1. The OPERABLE low pressure coolant injection (LPCI) subsystems "B" and "C" of the RHR system, each with a flow path capable of taking suction from the suppression chamber and transferring the water to the reactor vessel.

2. At least 6 OPERABLE ADS valves.

c. ECCS division 3 consisting of the OPERABLE high pressure core spray (HPCS) 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.

APPLICABILITY: OPERATIONAL CONDITION 1, 2*# and 3*.

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*The ADS is not required to be OPERABLE when reactor steam dome pressure is less than or equal to 122 psig.

**See Specification 3.3.3 for trip system operability.

#See Special Test Exception 3.10.6.

3/4.5 EMERGENCY CORE COOLING SYSTEM

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. Initially, water from the condensate storage tank is used instead of injecting water from

ECCS-OPERATING and SHUTDOWN (Continued)

the suppression pool into the reactor, but no credit is taken in the hazards analyses for the condensate storage tank water.

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 although the hazards analysis only takes credit for five valves. 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. 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, including draining, 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.

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

SAFETY EVALUATION

AMENDMENT NO.29 TO NPF-11

LA SALLE COUNTY STATION, UNIT 1

DOCKET NO. 50-373

Introduction

Item II.K.3.18, "Modification of Automatic Depressurization System Logic - Feasibility for Increased Diversity for Some Event Sequences" of NUREG-0737 required the licensee to modify the automatic depressurization system (ADS) actuation logic in order to eliminate the need for manual actuation to assure adequate core cooling. To ensure satisfactory resolution of the outstanding concern, La Salle Unit 1, Operating License bears License Condition 2.C(30)(1)(b):

> Prior to start up after the first refueling outage, the licensee shall implement the approved alternative logic modification of the automatic depressurization system.

By letters dated May 4, 1981 and May 6, 1981, Commonwealth Edison Company (CECo) endorsed the BWR Owner's Group study results as applicable to La Salle County Station and committed to implement ADS modifications recommended in the study. The logic will be modified by either: (1) eliminating the high drywell pressure trip or (2) bypassing the high drywell pressure trip after runout of a timer started at the reactor vessel level 1. The staff indicated that either of these modifications were acceptable. In Amendment 62 to the La Salle Final Safety Analysis Report, CECo opted to modify the ADS logic to bypass the high drywell pressure trip after a sustained low water level signal; and to add a manual switch which may be used to inhibit the ADS action, if necessary. By letter dated July 15, 1985, CECo provided a description of the logic modifications and elementary diagrams of the modified ADS actuation circuitry and requested a revision to the technical specifications which reflects the change to the ADS logic.

The automatic depressurization system, through seven safety/relief valves, functions as a backup to the operation of the high pressure coolant systems by depressurizing the reactor vessel so that the low pressure systems may inject water for core cooling. The present ADS design is activated automatically upon coincident signals of low reactor water level, high drywell pressure and low pressure emergency core cooling system (ECCS) pumps running. A time delay of approximately 2 minutes after receipt of these signals allows time for the water level to be restored by the high pressure coolant systems prior to automatic blowdown. For transient events which do not directly produce a high drywell pressure signal and are further degraded by a loss of all high pressure coolant systems, adequate core cooling is assured by manual depressurization of the reactor vessel.

EVALUATION

In order to eliminate the need for manual ADS actuation to ensure adequate core cooling, the licensee has provided four bypass timers which will automatically bypass the four drywell high pressure initiation signals after a 9.5 minute time delay. Each time delay is activated on its corresponding level 1 initiation signal. An alarm is provided in the control room to indicate that the bypass logic has been activated. After the 9.5 minutes have elapsed, the high drywell pressure is no longer required for ADS actuation and the 105 second ADS actuation timer is initiated. Another alarm indicates that this timer has started. After this time delay, the ADS solenoids are energized provided that at least one low pressure pump in that division is operating. The 105 second timer allows the operator to manually inhibit ADS actuation if desired. The sum of the time delays associated with the ADS actuation timer and the drywell pressure bypass timer is less than or equal to 687 seconds (9.5 minutes + 117 seconds allowable value for the ADS initiation timer). The bypass timers automatically reset when the low water level signal clears or the reset button is depressed. The settings for the pressure bypass timers are determined based upon minimizing the operator actions related to the ADS logic, minimizing the chance for unintentional ADS, and limiting the impact on the design basis ECCS analyses. In addition, a manual inhibit switch is also included to allow the operator to inhibit ADS operation without repeatedly pressing the reset push-button. An indicating light and an annunciator alert the operator of the inhibit condition. The pressure relief function and the manual ADS or individual safety relief valve control is not affected by this manual inhibit switch. This modification also allows the operator to manually initiate ADS without the low pressure ECCS pumps in that division operating. This provides additional flexibility in the use of the Emergency Operating Procedures (EOP) where the operator, for example, is instructed to perform an "emergency depressurization" to prevent containment damage. The seal-in feature has also been revised such that the seal-in occurs only if the final relays which energize the ADS solenoids are actually energized. This eliminates the possibility of inadvertent depressurization of the reactor pressure vessel if a low pressure ECCS pump is started and a previous seal-in has not been reset.

The licensee proposed a revision to the Technical Specifications Tables 3.3.3-1, 3.3.3-2 and 4.3.3.1-1 to add drywell pressure bypass timers and manual inhibit switches. The inscription of 'ADS Timer' in these tables is changed to 'Initiation Timer.' Since Specification 3.3.3 is concerned with the instrumentation which actuates the ADS system and not the ADS valves, the word "valve" is changed to "trip system" in ACTION 34 of Specification Table 3.3.3-1. To clarify the distinction between ADS valve(s) and ADS trip system, foot note ** has been added to Specification 3.5.1 to refer to Technical Specification 3.3.3 if an ADS trip system is inoperable. The staff has reviewed the licensee's submittal and concludes that the LaSalle Unit 1 ADS design conforms to the requirements of TMI Action Plan Item II.K.3.18 regarding ADS automatic actuation to ensure adequate core cooling and is, therefore, acceptable. The staff has also reviewed the licensee's proposed Technical Specification changes and finds that they appropriately address the actuation instrumentation setpoints and surveillance requirements for the changes made to the ADS logic and are, therefore, acceptable. Since the logic required by these proposed changes is not yet installed for Unit 1, the request that these Technical Specification changes not be made effective until the first refueling outage should be granted. This effective date is, also, consistent with the License Condition 2.C.(30)(1)(b).

Environmental Consideration

This amendment involves a change in the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that this amendment involves no significant hazards consideration and there has been no public comment on such finding. Accordingly, this amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of this amendment.

Conclusion

The Commission made a proposed determination that the amendment involves no significant hazards consideration which was published in the Federal Register (50 FR 34935) on August 28, 1985. No public comments were received.

We have concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (2) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Dated: NOV 0 1 [illegible]

Docket No. 50-373

NOV 01 1985

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

Dear Mr. Farrar:

SUBJECT: ISSUANCE OF AMENDMENT NO. 29 TO FACILITY OPERATING LICENSE NO. NPF-11 - LA SALLE COUNTY STATION, UNIT 1

The Nuclear Regulatory Commission has issued the enclosed Amendment No. 29 to Facility Operating License No. NPF-11 for the La Salle County Station, Unit 1. This amendment is in response to your letter dated July 15, 1985. The amendment incorporates the modification of the automatic depressurization system logic as required by License Condition 2.C.30(1)(b).

A copy of the related safety evaluation supporting Amendment No. 29 to Facility Operating License NPF-11 is enclosed.

Sincerely,

Original signed by:

Walter R. Butler, Chief
Licensing Branch No. 2
Division of Licensing

Enclosures:
1. Amendment No. 29 to NPF-11
2. Safety Evaluation

cc w/enclosures:
See next page

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EHylton	ABournia:dh	Woodhead	WRButler
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3. This amendment is effective upon startup following the first refueling outage.

FOR THE NUCLEAR REGULATORY COMMISSION

Original signed by:

Walter R. Butler, Chief
Licensing Branch No. 2
Division of Licensing

Enclosure:
Changes to the Technical
Specifications

Date of Issuance: NOV 21 1985

LB#2/DL/LA	LB#2/DL/PM	OELD	LB#2/DL/BC	AD/L/DL
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