May 19, 1987

Docket No. 50-374

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

Dear Mr. Farrar:

SUBJECT: CORRECTIONS TO AMENDMENTS NOS. 30 AND 32 TO FACILITY OPERATING LICENSE NO. NPF-18 - LASALLE COUNTY STATION, UNIT 2

On February 9, 1987 and April 16, 1987 the Nuclear Regulatory Commission issued Amendments Nos. 30 and 32, respectively. Amendment No. 30 revised the LaSalle Unit 2 Technical Specifications to permit replacing an existing peripheral locking piston control rod drive module with a Fine Motion Control Rod Drive module during one fuel cycle. Amendment No. 32 revised the LaSalle Unit 2 Technical Specifications to support operation of LaSalle County Station, Unit 2 at full rated power during Cycle 2 operation.

Amendments Nos. 30 and 32 contained several typographical errors on the revised Technical Specifications. A copy of the corrected Technical Specification change pages to Amendments Nos. 30 and 32 is enclosed.

Sincerely,

Original signed by/

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

Enclosures:
As stated

cc: See next page

DISTRIBUTION

Docket File	NRC PDR	Local PDR	PDIII-2 Reading File	GHolahan
EHylton	PShemanski	OGC-Bethesda	DHagen EJordan	JPartlow
TBarnhart (4)	Wanda Jones	EButcher	ACRS (10) GPA/PA	ARM/LFMB
PDIII-2 Plant File				

OFFICIAL RECORD COPY

PDSNP:LA	PDIII-2:PM	PDIII-2:D	~~OGC-Bethesda~~
EHylton	PShemanski:cb	DMuller	
5/19/87	5/18/87	5/18/87	~~5/ /87~~

P.S.

8705260140 870519
PDR ADOCK 05000374
P PDR

CP-1

Mr. Dennis L. Farrar
Commonwealth Edison Company

La Salle County Nuclear Power Station
Units 1 & 2

cc:
Philip P. Steptoe, Esquire
Suite 4200
One First National Plaza
Chicago, Illinois 60603

Assistant Attorney General
188 West Randolph Street
Suite 2315
Chicago, Illinois 60601

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

Chairman
La Salle County Board of Supervisors
La Salle 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

Mr. Gary N. Wright, Manager
Nuclear Facility Safety
Illinois Department of Nuclear Safety
1035 Outer Park Drive, 5th Floor
Springfield, Illinois 62704

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

John W. McCaffrey
Chief, Public Utilities Division
160 North La Salle Street, Room 900
Chicago, Illinois 60601

ENCLOSURE TO LICENSE AMENDMENTS NOS. 30 AND 32

FACILITY OPERATING LICENSE NO. NPF-18

DOCKET NO. 50-374

Replace the following pages of the Appendix "A" Technical Specifications with the enclosed pages.

Remove	Insert
XIX	XIX
B 2-1	B 2-1
3/4 1-13	3/4 1-13
3/4 2-1	3/4 2-1
3/4 3-53	3/4 3-53
3/4 10-11	3/4 10-11
B 3/4 4-1	B 3/4 4-1

LIST OF FIGURES

FIGURE		PAGE
3.1.5-1	SODIUM PENTABORATE SOLUTION TEMPERATURE/ CONCENTRATION REQUIREMENTS	3/4 1-21
3.1.5-2	SODIUM PENTABORATE ($Na_2B_{10}O_{16} \bullet 10\ H_2O$) VOLUME/CONCENTRATION REQUIREMENTS	3/4 1-22
3.2.1-1	MAXIMUM AVERAGE PLANAR LINEAR HEAT GENERATION RATE (MAPLHGR) VERSUS AVERAGE PLANAR EXPOSURE, INITIAL CORE FUEL TYPES 8CRB176, 8CRB219, and 8CRB071	3/4 2-2
3.2.1-2	MAXIMUM AVERAGE PLANAR LINEAR HEAT GENERATION RATE (MAPLHGR) VERSUS AVERAGE PLANAR EXPOSURE, FUEL TYPE BP8CRB299L.	3/4 2-2(a)
3.2.3-1	MINIMUM CRITICAL POWER RATIO (MCPR) VERSUS τ AT RATED FLOW	3/4 2-5
3.2.3-2	K_f FACTOR	3/4 2-6
3.4.1.1-1	CORE THERMAL POWER (% OF RATED) VERSUS TOTAL CORE FLOW (% OF RATED)	3/4 4-2a
3.4.6.1-1	MINIMUM REACTOR VESSEL METAL TEMPERATURE VS. REACTOR VESSEL PRESSURE	3/4 4-19
4.7-1	SAMPLE PLAN 2) FOR SNUBBER FUNCTIONAL TEST	3/4 7-33
B 3/4 3-1	REACTOR VESSEL WATER LEVEL	B 3/4 3-7
B 3/4.4.6-1	CALCULATED FAST NEUTRON FLUENCE (E>1MeV) at 1/4 T AS A FUNCTION OF SERVICE LIFE	B 3/4 4-7
5.1.1-1	EXCLUSION AREA AND SITE BOUNDARY FOR GASEOUS AND LIQUID EFFLUENTS	5-2
5.1.2-1	LOW POPULATION ZONE	5-3
6.1-1	CORPORATE MANAGEMENT	6-11
6.1-2	UNIT ORGANIZATION	6-12
6.1-3	MINIMUM SHIFT CREW COMPOSITION	6-13

8705260281 870519
PDR ADOCK 05000374
P PDR

2.1 SAFETY LIMITS

BASES

The fuel cladding, reactor pressure vessel, and primary system piping are the principal barriers to the release of radioactive materials to the environs. Safety Limits are established to protect the integrity of these barriers during normal plant operations and anticipated transients. The fuel cladding integrity Safety Limit is set such that no fuel damage is calculated to occur if the limit is not violated. Because fuel damage is not directly observable, a step-back approach is used to establish a Safety Limit such that the MCPR is not less than 1.07 for two recirculation loop operation and 1.08 for single recirculation loop operation. MCPR greater than 1.07 for two recirculation loop operation and 1.08 for single recirculation loop operation represents a conservative margin relative to the conditions required to maintain fuel cladding integrity. The fuel cladding is one of the physical barriers which separate the radioactive materials from the environs. The integrity of this cladding barrier is related to its relative freedom from perforations or cracking. Although some corrosion or use related cracking may occur during the life of the cladding, fission product migration from this source is incrementally cumulative and continuously measurable. Fuel cladding perforations, however, can result from thermal stresses which occur from reactor operation significantly above design conditions and the Limiting Safety System Settings. While fission product migration from cladding perforation is just as measurable as that from use related cracking, the thermally caused cladding perforations signal a threshold beyond which still greater thermal stresses may cause gross rather than incremental cladding deterioration. Therefore, the fuel cladding Safety Limit is defined with a margin to the conditions which would produce onset of transition boiling, MCPR of 1.0. These conditions represent a significant departure from the condition intended by design for planned operation.

2.1.1 THERMAL POWER, Low Pressure or Low Flow

The use of the GEXL correlation is not valid for all critical power calculations at pressures below 785 psig or core flows less than 10% of rated flow. Therefore, the fuel cladding integrity Safety Limit is established by other means. This is done by establishing a limiting condition on core THERMAL POWER with the following basis. Since the pressure drop in the bypass region is essentially all elevation head, the core pressure drop at low power and flows will always be greater than 4.5 psi. Analyses show that with a bundle flow of 28 x 10^3 lbs/hr, bundle pressure drop is nearly independent of bundle power and has a value of 3.5 psi. Thus, the bundle flow with a 4.5 psi driving head will be greater than 28 x 10^3 lbs/hr. Full scale ATLAS test data taken at pressures from 14.7 psia to 800 psia indicate that the fuel assembly critical power at this flow is approximately 3.35 MWt. With the design peaking factors, this corresponds to a THERMAL POWER of more than 50% of RATED THERMAL POWER. Thus, a THERMAL POWER limit of 25% of RATED THERMAL POWER for reactor pressure below 785 psig is conservative.

CONTROL ROD POSITION INDICATION

LIMITING CONDITION FOR OPERATION

3.1.3.7 The control rod position indication system shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1#, 2# and 5*#.

ACTION:

a. In OPERATIONAL CONDITION 1 or 2 with one or more control rod position indicators inoperable within one hour:

1. Determine the position of the control rod by:
   (a) Moving the control rod, by single notch movement, to a position with an OPERABLE position indicator,
   (b) Returning the control rod, by single notch movement, to its original position, and
   (c) Verifying no control rod drift alarm at least once per 12 hours, or
2. Move the control rod to a position with an OPERABLE position indicator, or
3. When THERMAL POWER is:
   (a) Within the low power setpoint of the RSCS:
      (1) Declare the control rod inoperable,
      (2) Verify the position and bypassing of control rod with inoperable "Full in" and/or "Full out" position indicators by a second licensed operator or other technically qualified member of the unit technical staff.
   b) Greater than the low power setpoint of the RSCS, declare the control rod inoperable, insert the control rod and disarm the associated directional control valves** either:
      (1) Electrically, or
      (2) Hydraulically by closing the drive water and exhaust water isolation valves.
4. Otherwise, be in at least HOT SHUTDOWN within the next 12 hours.

*At least each withdrawn control rod. Not applicable to control rods removed per Specification 3.9.10.1 or 3.9.10.2.
**May be rearmed intermittently, under administrative control, to permit testing associated with restoring the control rod to OPERABLE status.
#See Special Test Exception 3.10.10.

3/4.2 POWER DISTRIBUTION LIMITS

3/4.2.1 AVERAGE PLANAR LINEAR HEAT GENERATION RATE

LIMITING CONDITION FOR OPERATION

3.2.1 All AVERAGE PLANAR LINEAR HEAT GENERATION RATES (APLHGRs) for each type of fuel as a function of AVERAGE PLANAR EXPOSURE shall not exceed the limits shown in Figures 3.2.1-1 and 3.2.1-2. The limits of Figures 3.2.1-1 and 3.2.1-2 shall be reduced to a value of 0.85 times the two recirculation loop operation limit when in single recirculation loop operation.

APPLICABILITY: OPERATIONAL CONDITION 1, when THERMAL POWER is greater than or equal to 25% of RATED THERMAL POWER.

ACTION:

With an APLHGR exceeding the limits of Figures 3.2.1-1 and 3.2.1-2, initiate corrective action within 15 minutes and restore APLHGR to within the required limits within 2 hours or reduce THERMAL POWER to less than 25% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.1 All APLHGRs shall be verified to be equal to or less than the limits determined from Figures 3.2.1-1 and 3.2.1-2:

a. At least once per 24 hours,

b. Within 12 hours after completion of a THERMAL POWER increase of at least 15% of RATED THERMAL POWER, and

c. Initially and at least once per 12 hours when the reactor is operating with a LIMITING CONTROL ROD PATTERN for APLHGR.

TABLE 3.3.6-2

CONTROL ROD WITHDRAWAL BLOCK INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
1. ROD BLOCK MONITOR		
a. Upscale		
1) Two Recirculation Loop Operation	≤0.66 W + 38%	≤0.66 W + 41%
2) Single Recirculation Loop Operation	<0.66W + 32.7%	<0.66W + 35.7%
b. Inoperative	N.A.	N.A.
c. Downscale	≥5% of RATED THERMAL POWER	≥3% of RATED THERMAL POWER
2. APRM		
a. Flow Biased Simulated Thermal Power-Upscale		
1) Two Recirculation Loop Operation	≤0.66 W + 42%*	≤0.66 W + 45%*
2) Single Recirculation Loop Operation	<0.66W + 36.7%*	<0.66W + 39.7%*
b. Inoperative	N.A.	N.A.
c. Downscale	≥5% of RATED THERMAL POWER	≥3% of RATED THERMAL POWER
d. Neutron Flux-High	≤12% of RATED THERMAL POWER	≤14% of RATED THERMAL POWER
3. SOURCE RANGE MONITORS		
a. Detector not full in	N.A.	N.A.
b. Upscale	<2 x 10^5 cps	<5 x 10^5 cps
c. Inoperative	N.A.	N.A.
d. Downscale	≥0.7 cps	≥0.5 cps
4. INTERMEDIATE RANGE MONITORS		
a. Detector not full in	N.A.	N.A.
b. Upscale	<108/125 of full scale	<110/125 of full scale
c. Inoperative	N.A.	N.A.
d. Downscale	≥5/125 of full scale	≥3/125 of full scale

*The Average Power Range Monitor rod block function is varied as a function of recirculation loop flow (W). The trip setting of this function must be maintained in accordance with Specification 3.2.2.

SPECIAL TEST EXCEPTIONS

3/4.10.10 CONTROL RODS

LIMITING CONDITION FOR OPERATION

3.10.10 The provisions of Specifications 3.1.3.1 thru 3.1.3.7 may be suspended for control rod 02-43 during the second fuel cycle to allow the demonstration of a fine motion control rod drive installed at this control rod location, provided conditions of 3.10.9 are satisfied.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 5.

ACTION:

With the requirements of 3.10.9 not satisfied, immediately insert control rod 02-43 and disarm the drive motor electrically.

SURVEILLANCE REQUIREMENTS

4.10.10 The provisions of Specification 4.1.3.1 thru 4.1.3.7 may be suspended for control rod 02-43 during the second fuel cycle to allow the demonstration of a fine motion control rod drive installed at this location.

3/4.4.1 RECIRCULATION SYSTEM

Operation with one reactor recirculation loop inoperable has been evaluated and been found to be acceptable, provided the unit is operated in accordance with the single recirculation loop operation Technical Specifications herein.

An inoperable jet pump is not, in itself, a sufficient reason to declare a recirculation loop inoperable, but it does present a hazard in case of a design-basis-accident by increasing the blowdown area and reducing the capability of reflooding the core, thus, the requirement for shutdown of the facility with a jet pump inoperable. Jet pump failure can be detected by monitoring jet pump performance on a prescribed scheduled for significant degradation.

Recirculation loop flow mismatch limits are in compliance with the ECCS LOCA analysis design criterion. The limits will ensure an adequate core flow coastdown from either recirculation loop following a LOCA. Where the recirculation loop flow mismatch limits can not be maintained during the recirculation loop operation, continued operation is permitted in the single recirculation loop operation mode.

In order to prevent undue stress on the vessel nozzles and bottom head region, the recirculation loop temperatures shall be within 50°F of each other prior to startup of an idle loop. The loop temperature must also be within 50°F of the reactor pressure vessel coolant temperature to prevent thermal shock to the recirculation pump and recirculation nozzles. Since the coolant in the bottom of the vessel is at a lower temperature than the water in the upper regions of the core, undue stress on the vessel would result if the temperature difference was greater than 145°F.

The possibility of thermal hydraulic instability in a BWR has been investigated since the startup of early BWRs. Based on tests and analytical models, it has been identified that the high power-low flow corner of the power-to-flow map is the region of least stability margin. This region may be encountered during startups, shutdowns, sequence exchanges, and as a result of a recirculation pump(s) trip event.

To ensure stability, single loop operation is limited in a designated restricted region (Figure 3.4.1.1-1) of the power-to-flow map. Single loop operation with a designated surveillance region (Figure 3.4.1.1-1) of the power-to-flow map requires monitoring of APRM and LPRM noise levels.

3/4.4.2 SAFETY/RELIEF VALVES

The safety valve function of the safety/relief valves operate to prevent the reactor coolant system from being pressurized above the Safety Limit of 1325 psig in accordance with the ASME Code. A total of 18 OPERABLE safety/relief valves is required to limit reactor pressure to within ASME III allowable values for the worst case upset transient.

Demonstration of the safety/relief valve lift settings will occur only during shutdown and will be performed in accordance with the provisions of Specification 4.0.5.