4-16-87

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: Issuance of Amendment No. 32 to Facility Operating License No. NPF-18 - La Salle County Station, Unit 2

The Nuclear Regulatory Commission has issued the enclosed Amendment No. 32 to Facility Operating License No. NPF-18 for the La Salle County Station, Unit 2. This amendment is in response to your letter dated December 9, 1986.

The amendment revises the La Salle Unit 2 Technical Specifications to support operation of La Salle County Station, Unit 2 at full rated power during Cycle 2 operation.

A copy of the related safety evaluation supporting Amendment No. 32 ~~to~~ Facility Operating License No. NPF-18 is enclosed.

Sincerely,

/s/

Daniel R. Muller, Director
Project Directorate III-2
Division of Reactor Projects

Enclosures:
1. Amendment No 32 to NPF-18
2. Safety Evaluation

cc w/enclosure:
See next page

8704200319 870416
PDR ADOCK 05000374
P PDR

*Previously Concurred

BWD-3:DBL	LA:BWD3:DBL	D:BWD-3:DBL	DRSP:PD III-2
ABournia/hmc*	EHylton*	EAdensam*	DMuller
4/06/87	3/06/87	4/06/87	4/16/87

CP1

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

April 16, 1987

Docket No. 50-374

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Director of Licensing
Commonwealth Edison Company
P.O. Box 767
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2. Safety Evaluation

cc w/enclosure:
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. 50-374

LA SALLE COUNTY STATION, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 32
License No. NPF-18

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

   A. The application for amendment filed by Commonwealth Edison Company (the licensee), dated December 9, 1986, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's regulations set forth in 10 CFR Chapter I;

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

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

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

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

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

   (2) Technical Specifications and Environmental Protection Plan

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 32, 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.

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3. This amendment is effective upon startup following the first refueling.

FOR THE NUCLEAR REGULATORY COMMISSION

Daniel R. Muller

Daniel R. Muller, Director
Project Directorate III-2
Division of Reactor Projects

Enclosure:
Changes to the Technical
Specifications

Date of Issuance: April 16, 1987

ENCLOSURE TO LICENSE AMENDMENT NO. 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. The revised pages are identified by Amendment number and contain a vertical line indicating the area of change.

REMOVE	INSERT
XIX	XIX
2-1	2-1
B2-1	B2-1
B2-4	B2-4
B2-5	B2-5
B2-6	B2-6
B2-7	B2-7
3/4 2-1	3/4 2-1
3/4 2-2	3/4 2-2
	3/4 2-2(a)
3/4 2-4	3/4 2-4
3/4 2-5	3/4 2-5
3/4 2-6	3/4 2-6
3/4 3-39	3/4 3-39
3/4 3-53	3/4 3-53
3/4 4-1	3/4 4-1
3/4 4-2	3/4 4-2
	3/4 4-2(a)
B3/4 4-1	B3/4 4-1

LIST OF FIGURES

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMITS

THERMAL POWER, Low Pressure or Low Flow

2.1.1 THERMAL POWER shall not exceed 25% of RATED THERMAL POWER with the reactor vessel steam dome pressure less than 785 psig or core flow less than 10% of rated flow.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

With THERMAL POWER exceeding 25% of RATED THERMAL POWER and the reactor vessel steam dome pressure less than 785 psig or core flow less than 10% of rated flow, be in at least HOT SHUTDOWN within 2 hours and comply with the requirements of Specification 6.4.

THERMAL POWER, High Pressure and High Flow

2.1.2 The MINIMUM CRITICAL POWER RATIO (MCPR) shall not be less than 1.07 with two recirculation loop operation and shall not be less than 1.08 with single recirculation loop operation with the reactor vessel steam dome pressure greater than 785 psig and core flow greater than 10% of rated flow.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

With MCPR less than 1.07 with two recirculation loop operation or less than 1.08 with single recirculation loop operation and the reactor vessel steam dome pressure greater than 785 psig and core flow greater than 10% of rated flow, be in at least HOT SHUTDOWN within 2 hours and comply with the requirements of Specification 6.4.

REACTOR COOLANT SYSTEM PRESSURE

2.1.3 The reactor coolant system pressure, as measured in the reactor vessel steam dome, shall not exceed 1325 psig.

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

ACTION:

With the reactor coolant system pressure, as measured in the reactor vessel steam dome, above 1325 psig, be in at least HOT SHUTDOWN with reactor coolant system pressure less than or equal to 1325 psig within 2 hours and comply with the requirements of Specification 6.4.

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 103 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 103 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.

BASES TABLE B2.1.2-1

UNCERTAINTIES USED IN THE DETERMINATION OF THE FUEL CLADDING SAFETY LIMIT*

QUANTITY	STANDARD DEVIATION (% of Point)
Feedwater Flow	1.76
Feedwater Temperature	0.76
Reactor Pressure	0.5
Core Inlet Temperature	0.2
Core Total Flow	
Two Recirculation Loop Operation	2.5
Single Recirculation Loop Operation	6.0
Channel Flow Area	3.0
Friction Factor Multiplier	10.0
Channel Friction Factor Multiplier	5.0
TIP Readings	
Two Recirculation Loop Operation	8.7
Single Recirculation Loop Operation	6.8
R Factor	1.6
Critical Power	3.6

*The uncertainty analysis used to establish the core wide Safety Limit MCPR is based on the assumption of quadrant power symmetry for the reactor core. The values herein apply to both two recirculation loop operation and single recirculation loop operation, except as noted.

Bases Table B2.1.2-2

NOMINAL VALUES OF PARAMETERS USED IN

THE STATISTICAL ANALYSIS OF FUEL CLADDING INTEGRITY SAFETY LIMIT

THERMAL POWER	3293 MW
Core Flow	102.5 Mlb/hr
Dome Pressure	1010.4 psig
R-Factor	1.038 - 0 GWD/t 1.031 - 7 GWD/t 1.030 - 15 GWD/t 1.033 - 20 GWD/t

Bases Table B2.1.2-3

RELATIVE BUNDLE POWER DISTRIBUTION

USED IN THE GETAB STATISTICAL ANALYSIS

Range of Relative Bundle Power	Percent of Fuel Bundles Within Power Interval
1.375 to 1.425	5.1
1.325 to 1.375	7.3
1.275 to 1.325	7.8
1.225 to 1.275	9.8
1.175 to 1.225	7.3
1.125 to 1.175	11.8
1.075 to 1.125	4.7
1.025 to 1.075	4.7
≤1.025	41.5
	100.0

Bases Table B2.1.2-4

R-FACTOR DISTRIBUTION USED IN GETAB STATISTICAL ANALYSIS

8x8 Rod Array

R-Factor	Rod Sequence No.
1.038	1
1.038	2
1.037	3
1.037	4
1.035	5
1.035	6
1.030	7
$\leq$ 1.030	8 through 64

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.

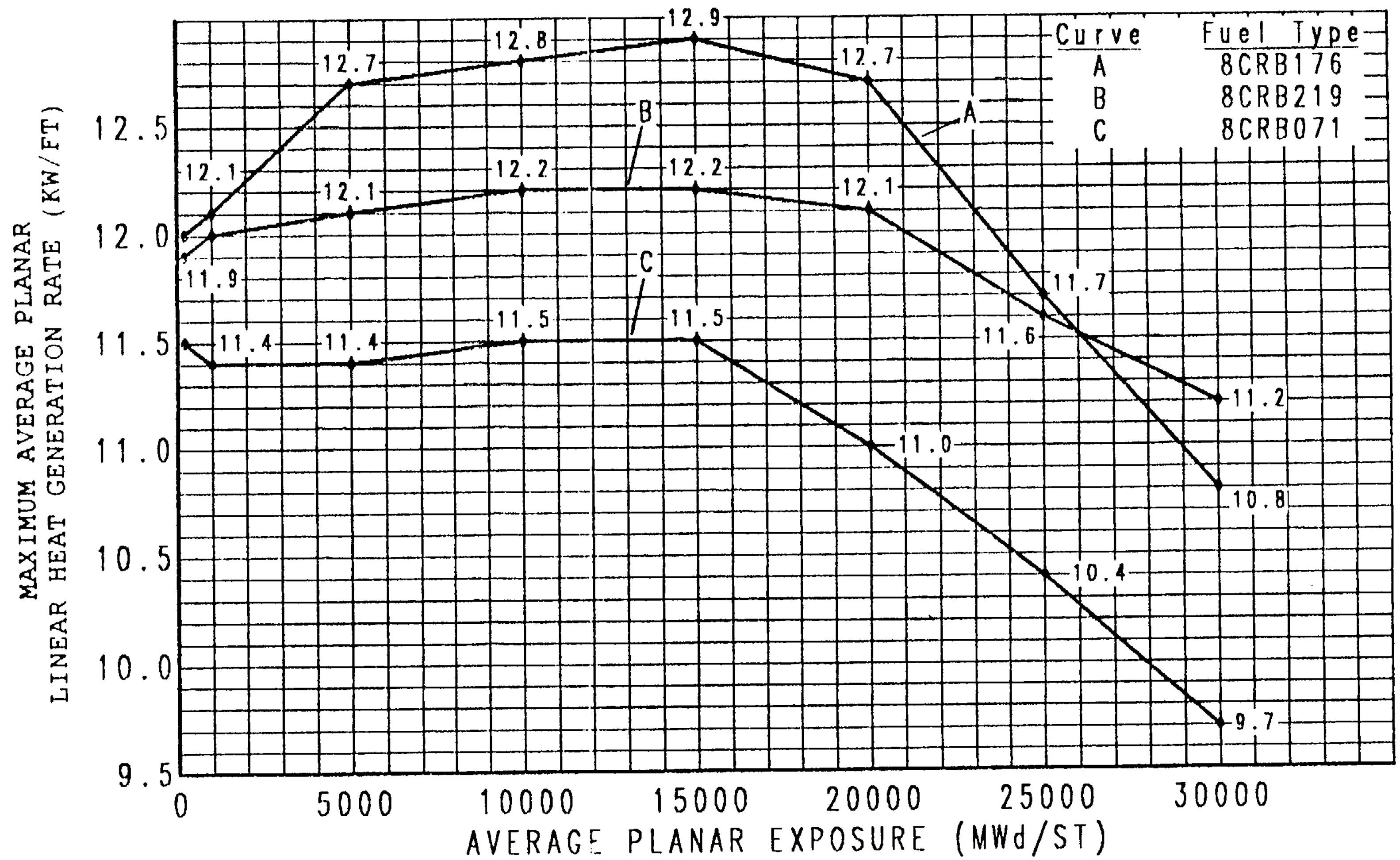
SURVEILLANCE REQUIREMENTS

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,

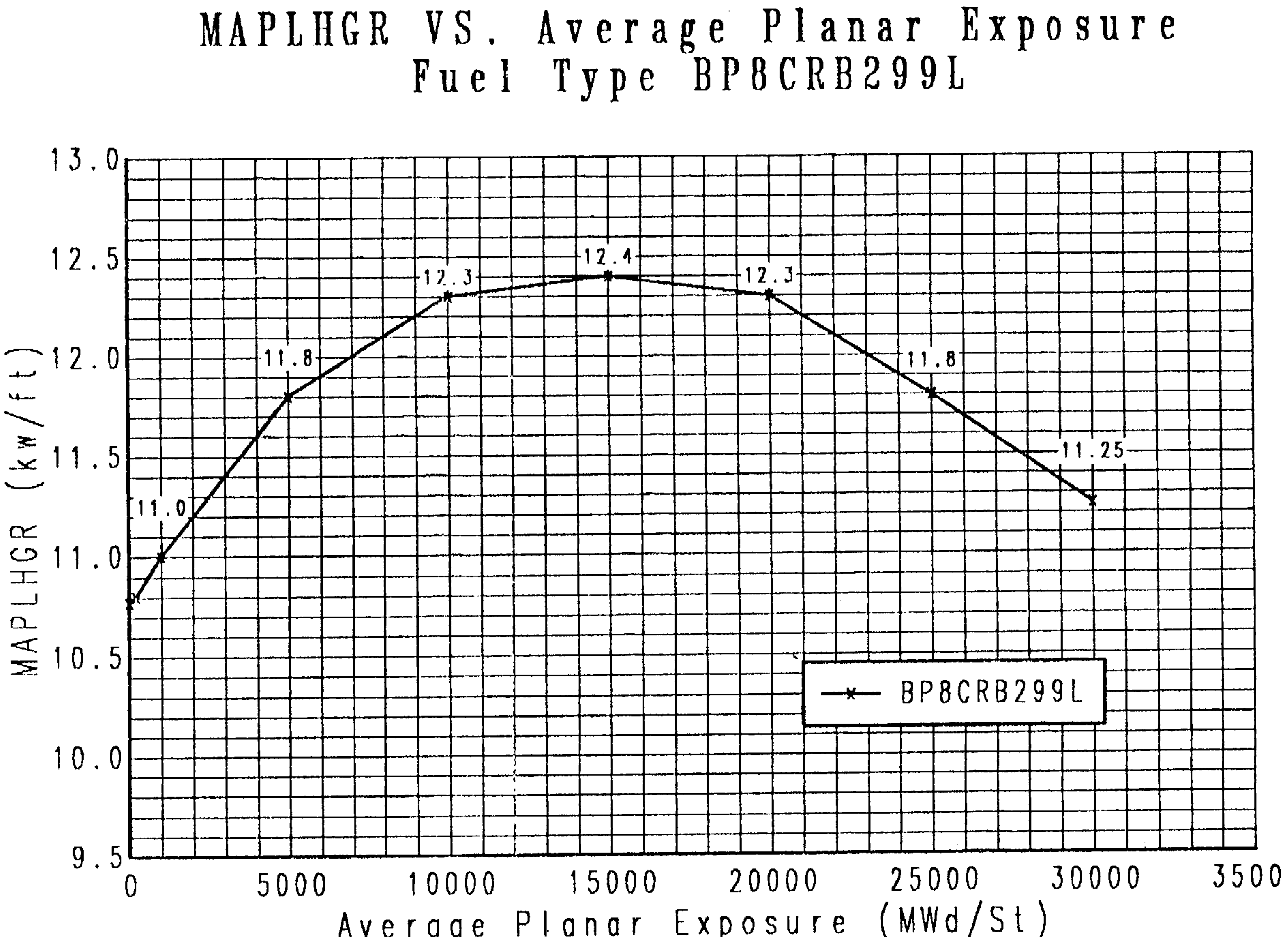
b. Within 12 hours after completion of a THERMAL POWER increase of at least 15% of RATED THERMAL POWER, and

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



MAXIMUM AVERAGE PLANAR LINEAR HEAT GENERATION RATE (MAPLHGR) VERSUS AVERAGE PLANAR EXPOSURE INITIAL CORE FUEL TYPES 8CRB176, 8CRB219, AND 8CRB071

FIGURE 3.2.1-1



MAXIMUM AVERAGE PLANAR LINEAR HEAT GENERATION RATE (MAPLHGR) VERSUS AVERAGE PLANAR EXPOSURE, FUEL TYPE BP8CRB299L

FIGURE 3.2.1-2

3/4.2.3 MINIMUM CRITICAL POWER RATIO

LIMITING CONDITION FOR OPERATION

3.2.3 The MINIMUM CRITICAL POWER RATIO (MCPR) shall be equal to or greater than the MCPR limit determined from Figure 3.2.3-1 times the K_f determined from Figure 3.2.3-2 for two recirculation loop operation and shall be equal to or greater than the MCPR limit determined from Figure 3.2.3-1 + 0.01 times the K_f determined from Figure 3.2.3-2 for single recirculation loop operation.

APPLICABILITY:

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

ACTION

With MCPR less than the applicable MCPR limit determined from Figures 3.2.3-1 and 3.2.3-2, initiate corrective action within 15 minutes and restore MCPR to within the required limit within 2 hours or reduce THERMAL POWER to less than 25% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.3 MCPR, with:

a. τ_{ave} = 0.86 prior to performance of the initial scram time measurements for the cycle in accordance with Specification 4.1.3.2, or

b. τ_{ave} determined within 72 hours of the conclusion of each scram time surveillance test required by Specification 4.1.3.2,

shall be determined to be equal to or greater than the applicable MCPR limit determined from Figures 3.2.3-1 and 3.2.3-2:

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 MCPR.

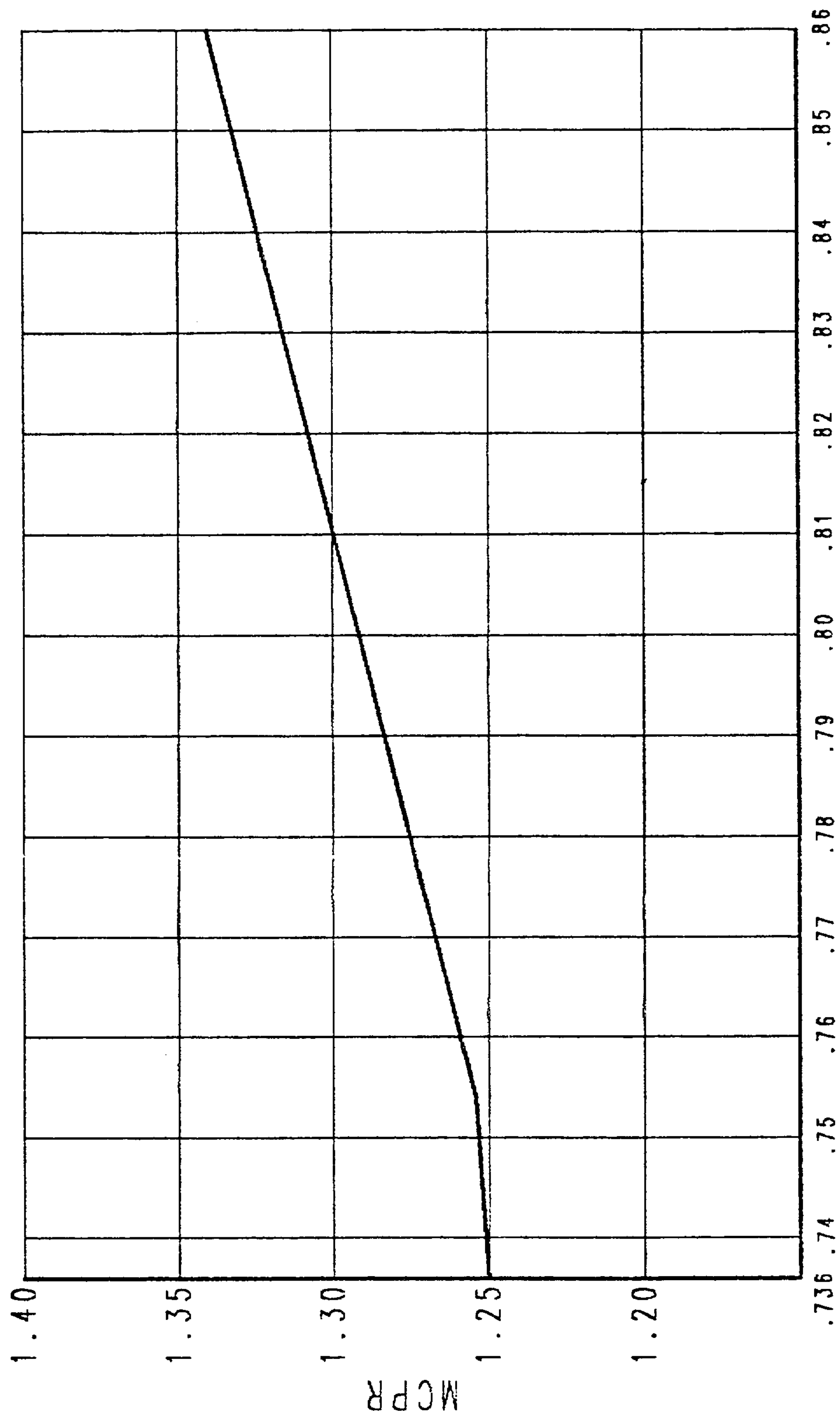
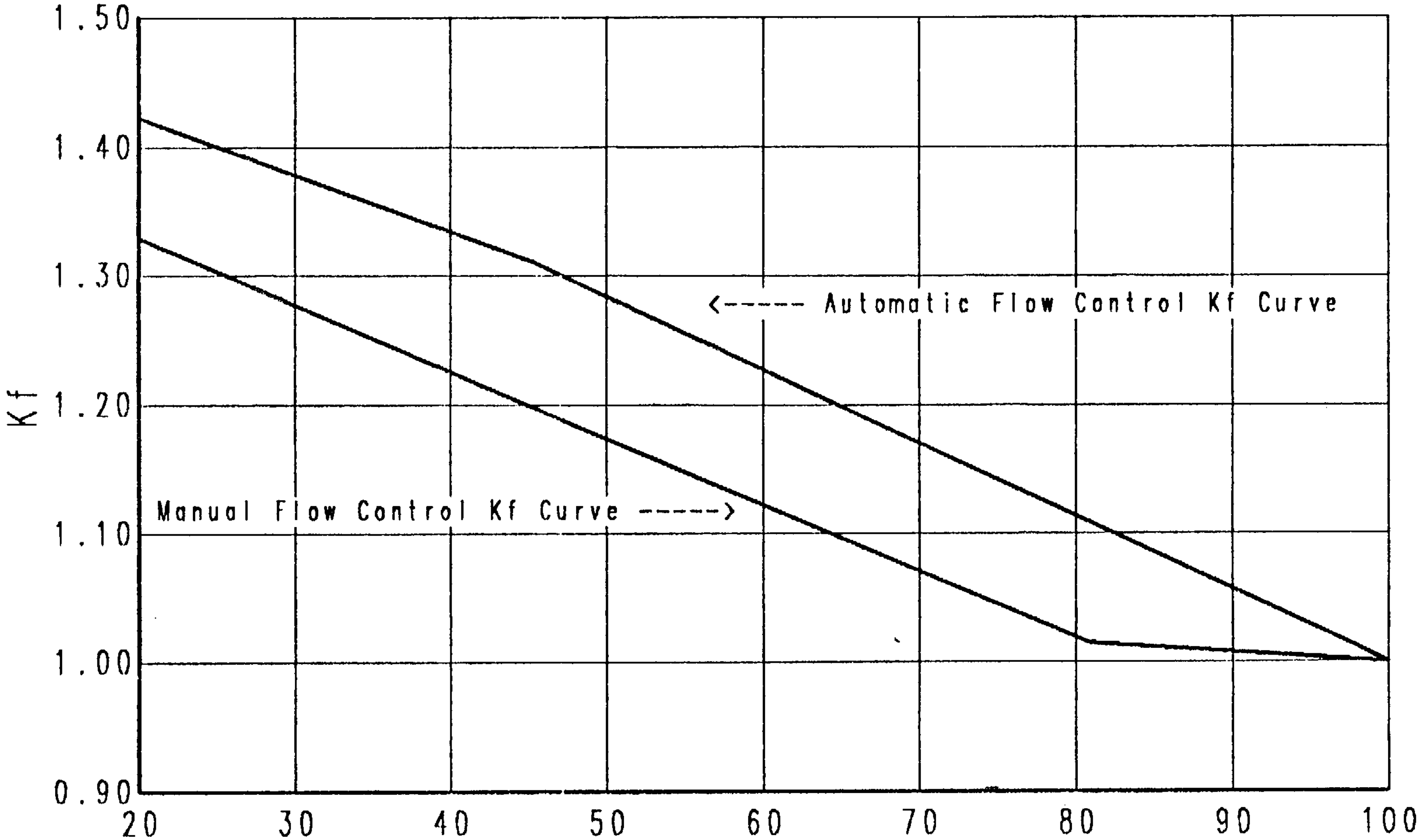


Figure 3.2.3-1 MINIMUM CRITICAL POWER RATIO (MCPR) VERSUS τ AT RATED FLOW



Kf Factor

Figure 3.2.3-2

END-OF-CYCLE RECIRCULATION PUMP TRIP SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.4.2 The end-of-cycle recirculation pump trip (EOC-RPT) system instrumentation channels shown in Table 3.3.4.2-1 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3.4.2-2 and with the END-OF-CYCLE RECIRCULATION PUMP TRIP SYSTEM RESPONSE TIME as shown in Table 3.3.4.2-3.

APPLICABILITY: OPERATIONAL CONDITION 1, when THERMAL POWER is greater than or equal to 30% of RATED THERMAL POWER.

ACTION:

a. With an end-of-cycle recirculation pump trip system instrumentation channel trip setpoint less conservative than the value shown in the Allowable Values column of Table 3.3.4.2-2, declare the channel inoperable until the channel is restored to OPERABLE status with the channel setpoint adjusted consistent with the Trip Setpoint value.

b. With the number of OPERABLE channels one less than required by the Minimum OPERABLE Channels per Trip System requirement for one or both trip systems, place the inoperable channel(s) in the tripped condition within 1 hour.

c. With the number of OPERABLE channels two or more less than required by the Minimum OPERABLE Channels per Trip System requirement(s) for one trip system and:

    1. If the inoperable channels consist of one turbine control valve channel and one turbine stop valve channel, place both inoperable channels in the tripped condition within 1 hour.

    2. If the inoperable channels include two turbine control valve channels or two turbine stop valve channels, declare the trip system inoperable.

d. With one trip system inoperable, restore the inoperable trip system to OPERABLE status within 72 hours or reduce THERMAL POWER to less than 30% of RATED THERMAL POWER within the next 6 hours.

e. With both trip systems inoperable, restore at least one trip system to OPERABLE status within 1 hour or reduce THERMAL POWER to less than 30% RATED THERMAL POWER within the next 6 hours.

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

3/4.4 REACTOR COOLANT SYSTEM

3/4.4.1 RECIRCULATION SYSTEM

RECIRCULATION LOOPS

LIMITING CONDITION FOR OPERATION

3.4.1.1 Two reactor coolant system recirculation loops shall be in operation.

APPLICABILITY: OPERATIONAL CONDITIONS 1* and 2*.

ACTION:

a. With one reactor coolant system recirculation loop not in operation:

1. Within 4 hours:

   a) Place the recirculation flow control system in the Master Manual mode, and

   b) Increase the MINIMUM CRITICAL POWER RATIO (MCPR) Safety Limit by 0.01 to 1.08 per Specification 2.1.2, and,

   c) Increase the MINIMUM CRITICAL POWER RATIO (MCPR) Limiting Condition for Operation by 0.01 per Specification 3.2.3, and,

   d) Reduce the MAXIMUM AVERAGE PLANAR LINEAR HEAT GENERATION RATE (MAPLHGR) limit to a value of 0.85 times the two recirculation loop operation limit per Specification 3.2.1, and,

   e) Reduce the Average Power Range Monitor (APRM) Scram and Rod Block and Rod Block Monitor Trip Setpoints and Allowable Values to those applicable for single loop recirculation loop operation per Specifications 2.2.1, 3.2.2, and 3.3.6.

2. When operating within the surveillance region specified in Figure 3.4.1.1-1:

   a) With core flow less than 39% of rated core flow, initiate action within 15 minutes to either:

      1) Leave the surveillance region within 4 hours, or

      2) Increase core flow to greater than or equal to 39% of rated flow within 4 hours.

   b) With the APRM and LPRM# neutron flux noise level greater than three (3) times their established baseline noise levels:

*See Special Test Exception 3.10.4.

#Detector levels A anc C of one LPRM string per core octant plus detector levels A and C of one LPRM string in the center region of the core should be monitored.

LIMITING CONDITION FOR OPERATION (Continued)

ACTION: (Continued)

1) Initiate corrective action within 15 minutes to restore the noise levels to within the required limit within 2 hours, otherwise

2) Leave the surveillance region specified in Figure 3.4.1.1-1 within the next 2 hours.

3. The provisions of Specification 3.0.4 are not applicable.

4. Otherwise, be in at least HOT SHUTDOWN within the next 12 hours.

b. With no reactor coolant system recirculation loops in operation, immediately initiate measures to place the unit in at least HOT SHUTDOWN within the next 6 hours.

SURVEILLANCE REQUIREMENTS

4.4.1.1 Each reactor coolant system recirculation loop flow control valve shall be demonstrated OPERABLE at least once per 18 months by:

a. Verifying that the control valve fails "as is" on loss of hydraulic pressure at the hydraulic power unit, and

b. Verifying that the average rate of control valve movement is:

1. Less than or equal to 11% of stroke per second opening, and

2. Less than or equal to 11% of stroke per second closing.

4.4.1.2 With one reactor coolant system recirculation loop not in operation:

a. Establish baseline APRM and LPRM# neutron flux noise level values within 4 hours upon entering the surveillance region of Figure 3.4.1.1-1 provided that the baseline values have not been established since last refueling.

b. When operating in the surveillance region of Figure 3.4.1.1-1, verify that the APRM and LPRM# neutron flux noise levels are less than or equal to three (3) times the baseline values:

1. At least once per 12 hours, and

2. Within 1 hour after completion of a THERMAL POWER increase of at least 5% of RATED THERMAL POWER, initiating the surveillance within 15 minutes of completion of the increase.

c. When operating in the surveillance region of Figure 3.4.1.1-1, verify that core flow is greater than or equal to 39% of rated core flow at least once per 12 hours.

#Detector levels A and C of one LPRM string per core octant plus detector levels A and C of one LPRM string in the center region of the core should be monitored.

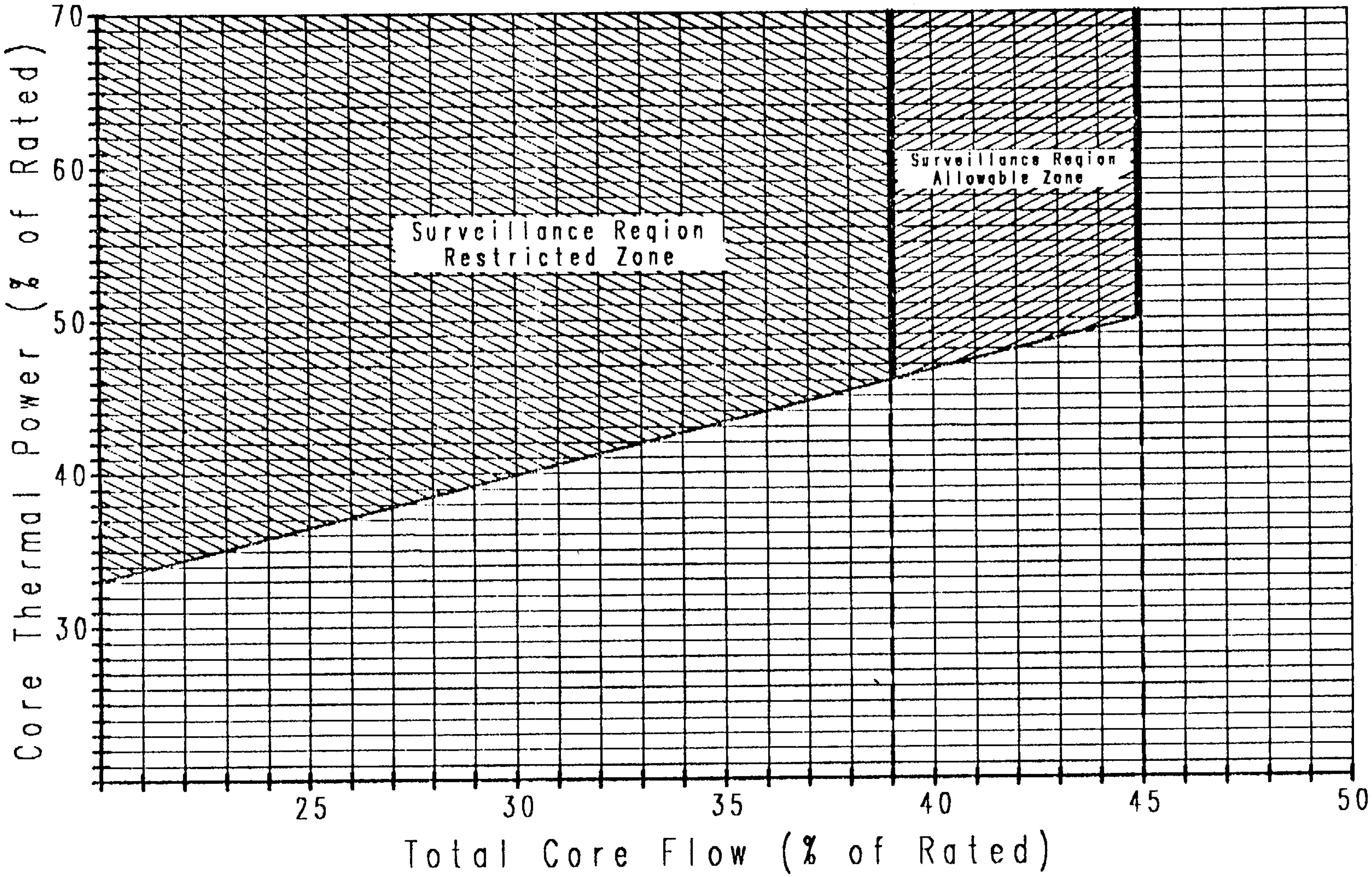


Figure 3.4.1.1-1

CORE THERMAL POWER (% OF RATED) VERSUS
TOTAL CORE FLOW (% OF RATED)

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.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 32 TO FACILITY OPERATING LICENSE NO. NPF-18

COMMONWEALTH EDISON COMPANY

LA SALLE COUNTY STATION, UNIT 2

DOCKET NO. 50-374

1.0 INTRODUCTION

By letter from C. M. Allen, Commonwealth Edison (CE), to H. R. Denton, NRC, dated December 9, 1986 (Reference 1), Technical Specification changes were proposed for the operation of La Salle County Station Unit 2 for Cycle 2 (LS2C2) with a reload using General Electric (GE) manufactured fuel assemblies and GE analyses and methodologies. The requested Technical Specification changes and reports (including Reference 2) discussing the reload and analyses done to support and justify the second cycle operation were included with the submittal. There was also an attachment B to Reference 1 proposing Technical Specification changes related to single loop operation.

2.0 EVALUATION

2.1 RELOAD DESCRIPTION

The LS2C2 reload will retain 108 8CRB176 and 432 8CRB219 fuel assemblies from the first cycle and add 224 new BP8CRB299L GE fuel assemblies. The reload is based on the Cycle 1 exposure of 10.016 to 10.216 Giga Watt Days/Short Ton (GWD/ST) and a Cycle 2 exposure of 6.775 GWD/ST. The loading will be a conventional scatter pattern with low reactivity fuel on the periphery.

2.2 FUEL DESIGN

The new fuel assembly to be used for LS2C2, BP8CRB299L, has been generically approved structually in the staff's review of NEDE-24011, GESTAR II (Amendment 13). This fuel type has been analyzed for generic application with approved methods and meets the approved limits of GESTAR II (Reference 5). Therefore, the new fuel is acceptable for LS2C2.

2.3 NUCLEAR DESIGN

The nuclear design for LS2C2 has been performed with the methodology described in GESTAR II (Reference 3). The results of these analyses are given in Reference 2. The shutdown margin is 2 percent Delta K at the beginning of cycle and 1.4 percent Delta K at the exposure of minimum shutdown margin. Therefore, it meets the required 0.38 percent Delta K shutdown margin. The Standby Liquid Control system also meets the shutdown requirements with a

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shutdown margin of 3.9 percent Delta K. Since these and other LS2C2 nuclear design parameters have been obtained with previously approved methods and fall within expected ranges, the nuclear design is acceptable.

2.4 THERMAL-HYDRAULIC DESIGN

The thermal-hydraulic design for LS2C2 has been performed with the methodology described in GESTAR II (Reference 3), and the results are given in Reference 2. The parameters used for the analyses are those approved in Reference 3 for the La Salle class BWR-5.

The Safety Limit Minimum Critical Power Ratio (SLMCPR) values are increased by 0.01 to reflect the approved limits when going from first cycle to reload cores. These SLMCPR values are 1.07 and 1.08 for two and one loop operation, respectively. The Operating Limit MCPR (OLMCPR) values are determined by the limiting transients, Rod Withdrawal Error (RWE), Feedwater Controller Failure (FWCF) and Load Rejection Without Bypass (LRWBP). The analysis of these events for LS2C2, via the ODYN Option B approach, provides new Cycle 2 Technical Specification values of OLMCPR as a function of τ. At (and below) a τ of 0.736, RWE provides the limit at a MCPR of 1.25. FWCF is limiting above 0.736 until above a τ of 0.754, where LRWBP is the limiting event. Approved methods (Reference 3) were used to analyze these events (and others which could be limiting), and the analyses and results are acceptable and fall within expected ranges.

The changes in MCPR limits, Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) limits, and the Average Power Range Monitor (APRM) setpoints, when going from two to one (recirculation) loop operation, remain the same for Cycle 2 as they were for Cycle 1. These changes for one loop operation, which have been approved previously, continue to be acceptable.

The thermal-hydraulic stability of the Cycle 2 core has been analyzed using the approved methods (Reference 3). The result is a decay ratio of 0.60 at the intersection of the natural circulation line and the 105 percent rod line. Existing Technical Specifications do not allow continued operation in natural circulation. Operation at the combination of low flow and high power sufficient to produce high decay ratio is thus limited. Although the cycle specific analysis predicts adequate stability, proposed Technical Specification changes to assure stability during single loop operation (SLO) have been included in Attachment B of Reference 1. These changes are in general accord with the specifications approved recently for other reactors (e.g., Duane Arnold 50-331). These specifications provide for the establishment of regions above the 80 percent rod line where: (a) below 39 percent flow, action must be taken to leave the region; and (b) above 39 percent flow and below 45 percent flow, action must be taken to monitor Average Power Range Monitor (APRM) - Local Power Range Monitor (LPRM) noise and to reduce the noise or leave the region if the noise is greater than 3 times the baseline. These specifications also provide for the establishment of baseline noise levels when entering the surveillance region (if not previously accomplished in the cycle). These action and surveillance requirements (including the LPRM specification) and the times for accomplishing them are comparable to other recently approved specifications and meet the objectives of GE Service Information Letter 380 (Reference 4); and are, therefore, acceptable. Thus one loop operation is, generally, acceptable for La Salle without restrictions other than those presented in Specification 3/4.4.1.

2.5 TRANSIENT AND ACCIDENT ANALYSES

The transient and accident analysis methodologies used for LS2C2 are described in GESTAR II (Reference 3). Generally, the ODYN Option B approach, was used for transient analyses. The Loss of Feedwater Heating event was analyzed with the GE BWR Simulator Code, approved in Reference 3. The limiting MCPR events have been previously indicated in Section 2.4. The core wide transient analyses methodologies and results are acceptable and fall within expected ranges.

The RWE was analyzed on a plant and cycle specific basis (as opposed to the statistical approach), and a rod block setpoint of 1.07 was selected to provide an OLMCPR of 1.25. The fuel assembly misloading and misorientation events were not analyzed for LS2C2. As approved via Reference 3, the mislocated assembly is not analyzed for reload cores on the basis of studies indicating the small probability of an event exceeding MCPR limits. The misorientation event is not of concern to C lattice cores (i.e., La Salle) because of the symmetry of the fuel bundle, gaps and power distribution. These local event analyses are, thus, acceptable.

The limiting pressurization event, the Main Steam Isolation Valve closure with flux scram, was analyzed with standard GESTAR II methods. The results for peak steam dome and vessel pressures were well under required limits. These are acceptable methodologies and results.

Loss of Coolant Accident (LOCA) analyses, using approved methodologies and parameters (Reference 3), were performed to provide MAPLHGR values for the new reload fuel assemblies (BP8CRB299L). These analyses and results are acceptable.

The Rod Drop Accident (RDA) was not specifically analyzed for LS2C2. La Salle uses a Banked Position Withdrawal Sequence for control rod withdrawal. For plants using this system, the RDA event has been statistically analyzed generically; and it was found, with a high degree of confidence, that the peak fuel enthalpy would not approach the NRC required limit of 280 cal/gm for this event. This approach and analysis have been approved by the NRC (Reference 3) and are acceptable for LS2C2.

2.6 TECHNICAL SPECIFICATIONS

The Technical Specification changes are for the most part minor and provide for MCPR changes due to second cycle parameter changes, MAPLHGR limits for a new fuel type, elimination of End of Cycle - Recirculation Pump Trip (EOC-RPT) inoperable provision for the cycle, and for a change in k_f. In addition, a Technical Specification providing for extended operation in the SLO mode has been added. Details of the specification changes follow:

(1) Specification 2.1 and Bases and Tables B2.1.2-1 through B2.1.2-4:

The SLMCPR for two and one (recirculation) loop operations were increased by 0.01 to 1.07 and 1.08. This is standard practice for second cycles and is based on parameter changes for reload cores given in the changes in the Bases Tables. These changes are taken from Reference 3. These various changes are acceptable.

(2) Specification 3/4.2.1 and Figures 3.2.1-1 and 3.2.1-2:

A new MAPLHGR curve is provided for the new fuel and a fuel assembly designation change is made. These are acceptable.

(3) Specification 3/4.2.3 and Figures 3.2.2-1 and 3.2.3-2:

The provision for the EOC-RPT inoperable condition in Specification 3/4.2.3 was removed. This is acceptable.

The MCPR vs τ curve is changed to reflect the new transient analyses as previously discussed. The change is acceptable.

The k_f factor curve was changed to be compatible with the standard La Salle power and flow values as given in Reference 3. This is acceptable.

(4) Actions d and e of Specification 3.3.4.2:

Changes were made to make this specification compatible with the elimination of EOC-RPT inoperable provision of Specification 3.2.3. These changes, including the indicated power reduction, are reasonable and acceptable.

(5) Specification 3/4.4.1 and Bases for 3/4.4.1:

These changes are to assure thermal-hydraulic stability for single loop operation. They have been discussed in Section 2.4 of this evaluation and are acceptable.

(6) Table 3.3.6-2:

Table 3.3.6-2 is revised to incorporate the required RBM setpoint change (RBM setpoint of 107 percent). This revision is acceptable.

3.0 ENVIRONMENTAL CONSIDERATION

This amendment involves a change in the installation and use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes in surveillance requirements. The staff has determined that this 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.

4.0 CONCLUSION

The Commission made a proposed determination that the amendment involves no significant hazards consideration which was published in the Federal Register (52 FR 2877) on January 28, 1987, and consulted with the state of Illinois. No public comments were received, and the state of Illinois did not have any comments.

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

5.0 REFERENCES

1. Letter from C. M. Allen, to H. R. Denton, NRC, "La Salle County Station Unit 2, Proposed Amendment to Technical Specification for Facility Operating License NPF-18-Reload Licensing Package for Cycle 2," December 9, 1986.

2. General Electric Report - 23A4735, June 1986, "Supplemental Reload Licensing Submittal for La Salle County Station, Unit 2, Reload 1 (Cycle 2)."

3. NEDE-24011-PA-8, May 1986, "General Electric Standard Application for Reactor Fuel," (GESTAR II).

4. General Electric Service Information Letter No. 380, Revision 1, February 10, 1984.

5. Letter, G. Lainas, NRC, to J. S. Charnley, GE, dated March 26, 1986, Acceptance for Referencing of Licensing Topical Report NEDE-24011-P-A Amendment 13, Rev. 6, "General Electric Standard Application for Reactor Fuel."

Principal Contributor: T. Huang, NRR

Dated: April 16, 1987

AMENDMENT NO. 32 TO FACILITY OPERATING LICENSE NO. NPF-18 - LA SALLE, UNIT 2

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