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

May 9, 1986

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. 40 to Facility Operating License No. NPF-11 - La Salle County Station, Unit 1

The Nuclear Regulatory Commission has issued the enclosed Amendment No. 40 to Facility Operating License No. NPF-11 for the La Salle County Station, Unit 1. This amendment is in response to your letter dated October 22, 1985, as supplemented on March 21, 1986.

The amendment revises the Unit 1 Technical Specifications to support operation of La Salle County Station, Unit 1 at full rated power during Cycle 2 operation.

A copy of the related safety evaluation supporting Amendment No. 40 to Facility Operating License No. NPF-11 is enclosed.

Sincerely,

Elinor G. Adensam

Elinor G. Adensam, Director
BWR Project Directorate No. 3
Division of BWR Licensing

Enclosures:
1. Amendment No. 40 to NPF-11
2. Safety Evaluation

cc w/enclosure:
See next page

DESIGNATED ORIGINAL

Certified By

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PDR ADOCK 05000373
P PDR

Mr. Dennis L. Farrar
Commonwealth Edison Company

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Units 1 & 2

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AMENDMENT NO. 40 TO FACILITY OPERATING LICENSE NO. NPF-11 - LA SALLE, UNIT 1

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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. 40
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 (the licensee), dated October 22, 1985, as supplemented on March 21, 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 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. 40, 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.

FOR THE NUCLEAR REGULATORY COMMISSION

Elinor G. Adensam

Elinor G. Adensam, Director
BWR Project Directorate No. 3
Division of BWR Licensing

Enclosure:
Changes to the Technical
Specifications

Date of Issuance: May 9, 1986

ENCLOSURE TO LICENSE AMENDMENT NO. 40

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
VI	VI
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 4-1	3/4 4-1
3/4 4-1a	3/4 4-1a
3/4 4-4b	3/4 4-4b
B3/4 4-1	B3/4 4-1

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

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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

2.0 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. 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 28×10^3 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 28×10^3 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.

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 appply 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
$\leq$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
≤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) vs AVERAGE PLANAR EXPOSURE

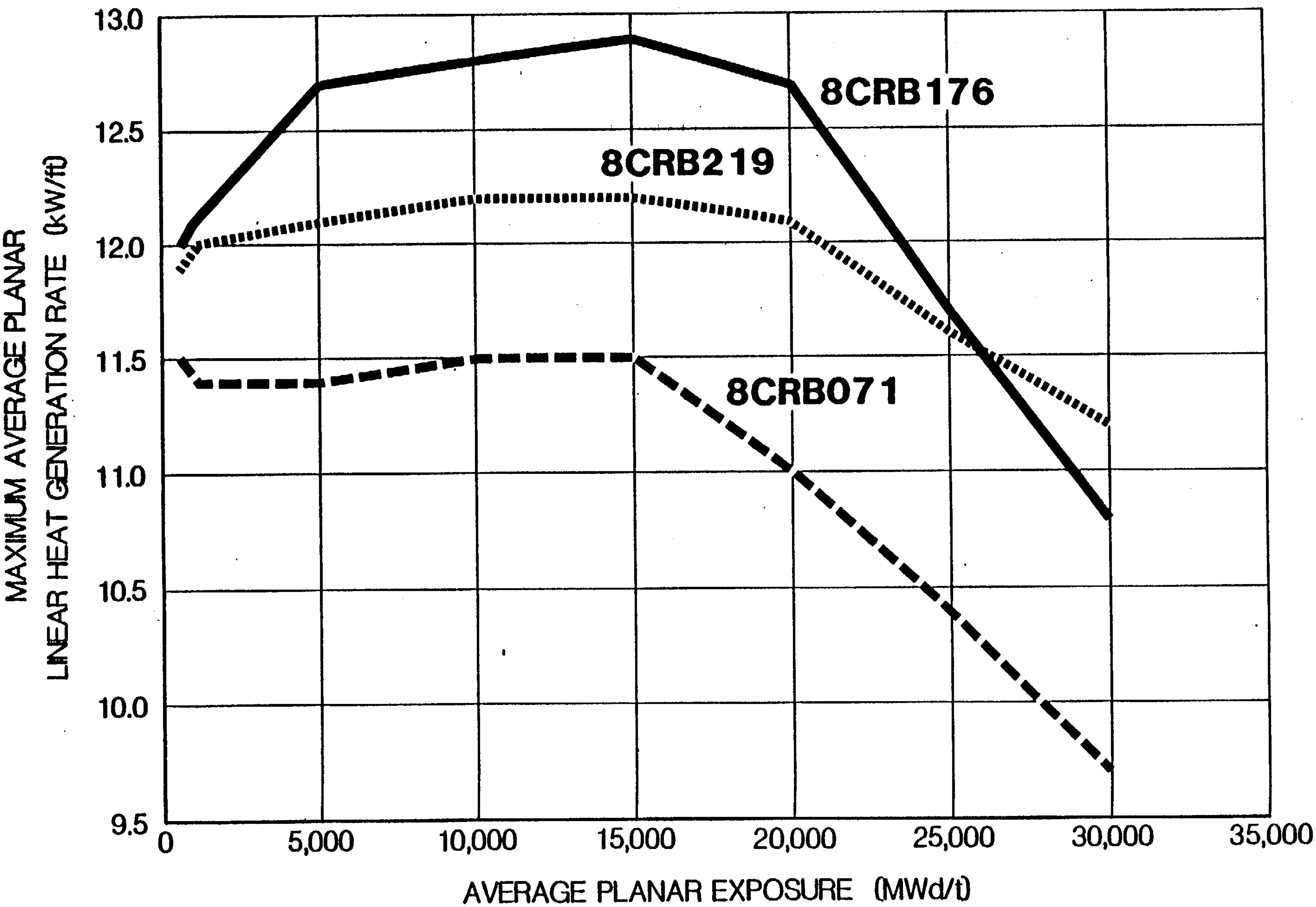


Figure 3.2.1-1

MAPLHGR vs AVERAGE PLANAR EXPOSURE

FUEL TYPE BP8CRB299L

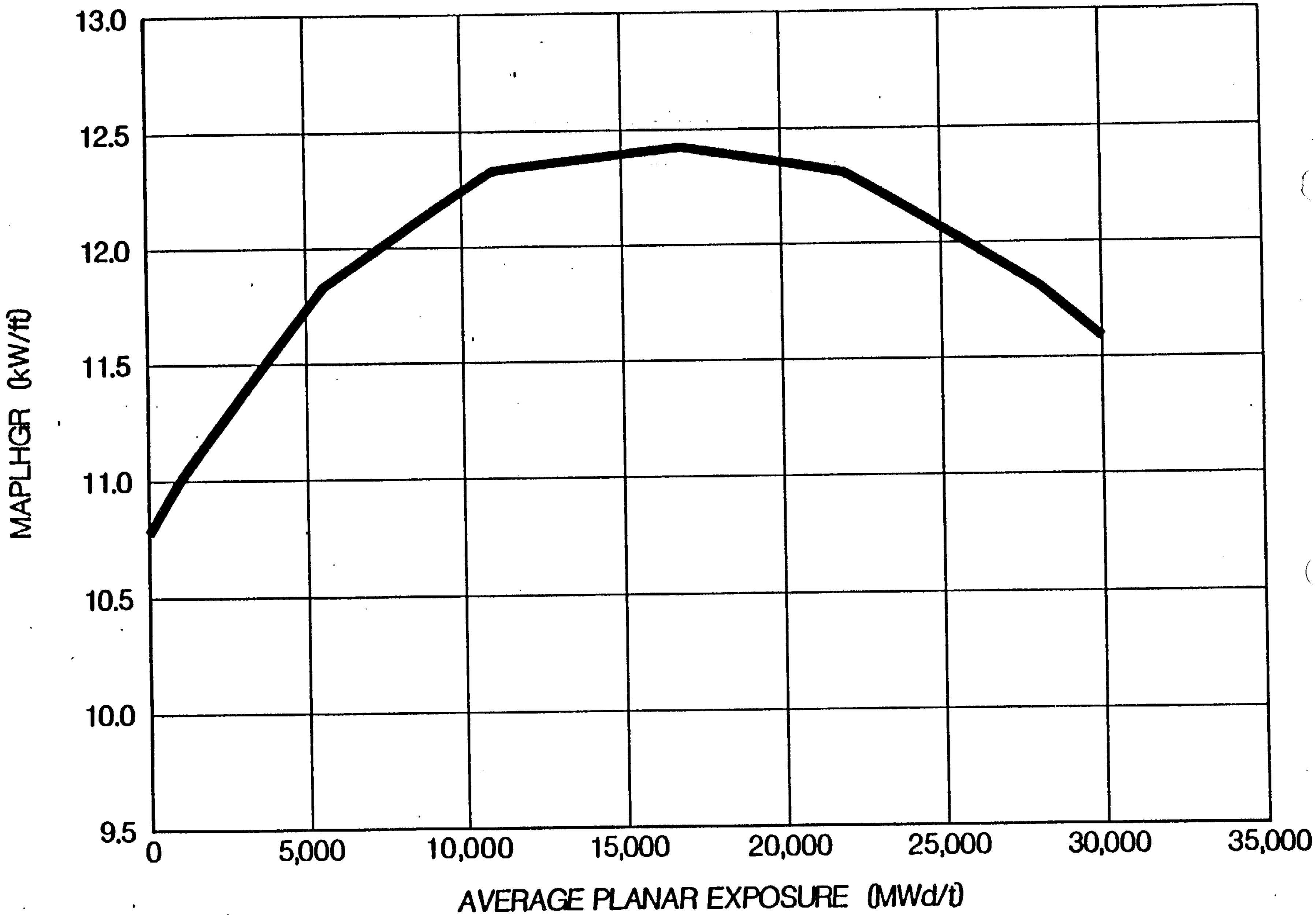


Figure 3.2.1-2

POWER DISTRIBUTION LIMITS

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:

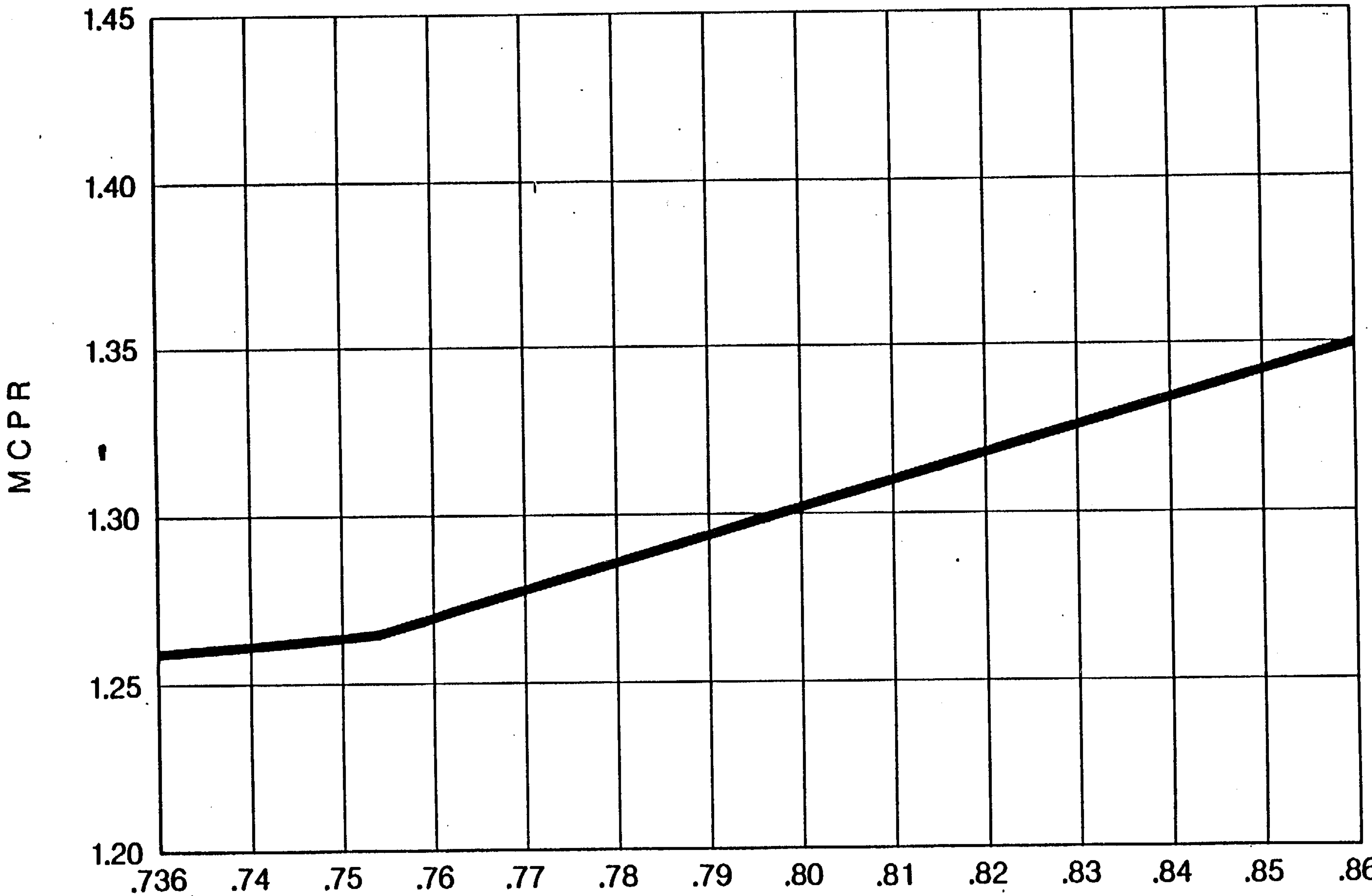
a. At least once per 24 hours,

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.

POWER DISTRIBUTION LIMITS

M C P R



MINIMUM CRITICAL POWER RATIO (MCPR) VERSUS τ AT RATED FLOW

Figure 3.2.3-1

POWER DISTRIBUTION LIMITS

Kf FACTOR

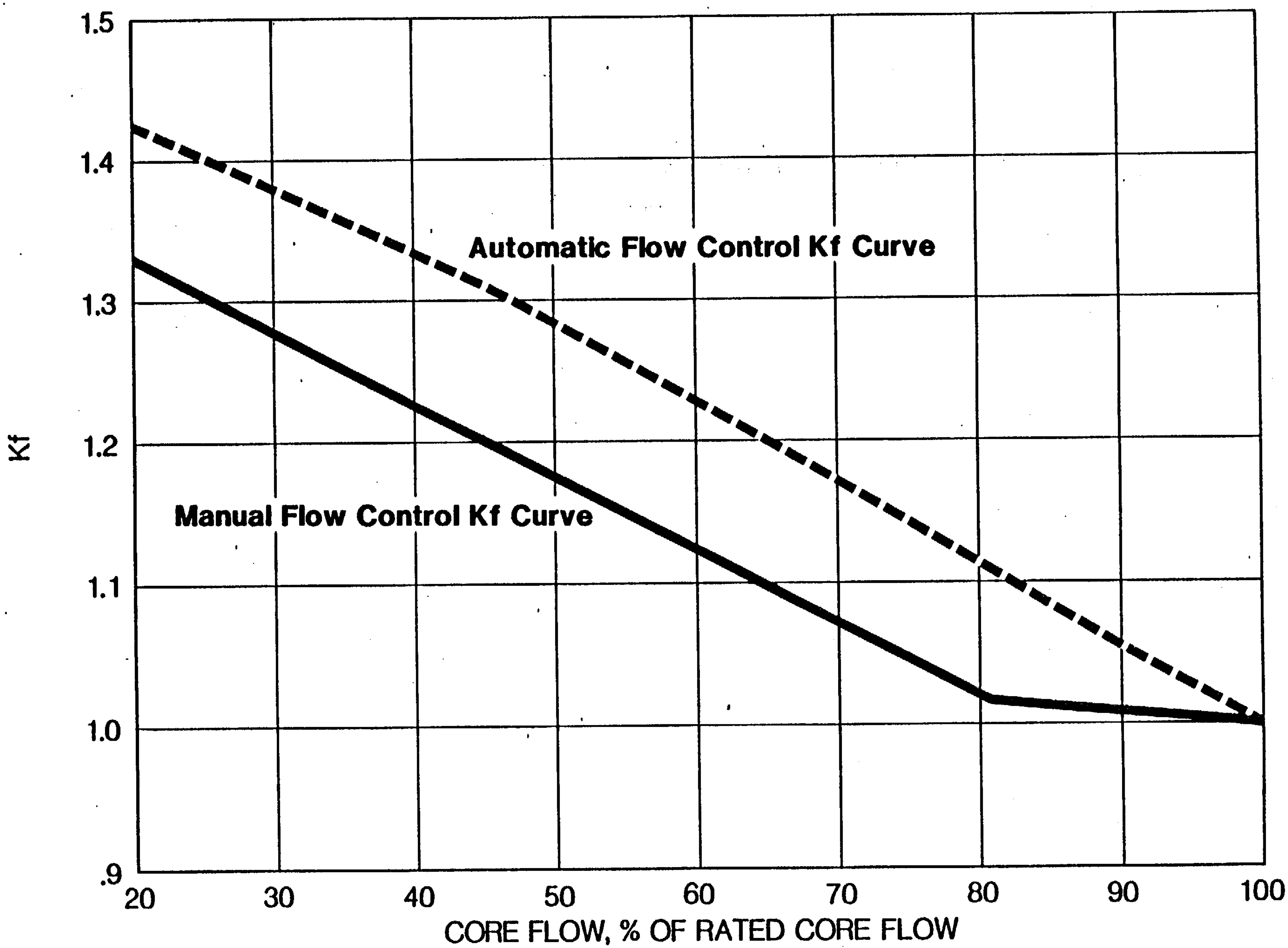


Figure 3.2.3-2

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% of RATED THERMAL POWER within the next 6 hours.

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 to 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:

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.

*See Special Test Exception 3.10.4.

#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.

LIMITING CONDITION FOR OPERATION

ACTION: (Continued)

3. The provisions of Specification 3.0.4 are not applicable.

4. Otherwise, be ina 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 units, 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 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.

POWER vs FLOW

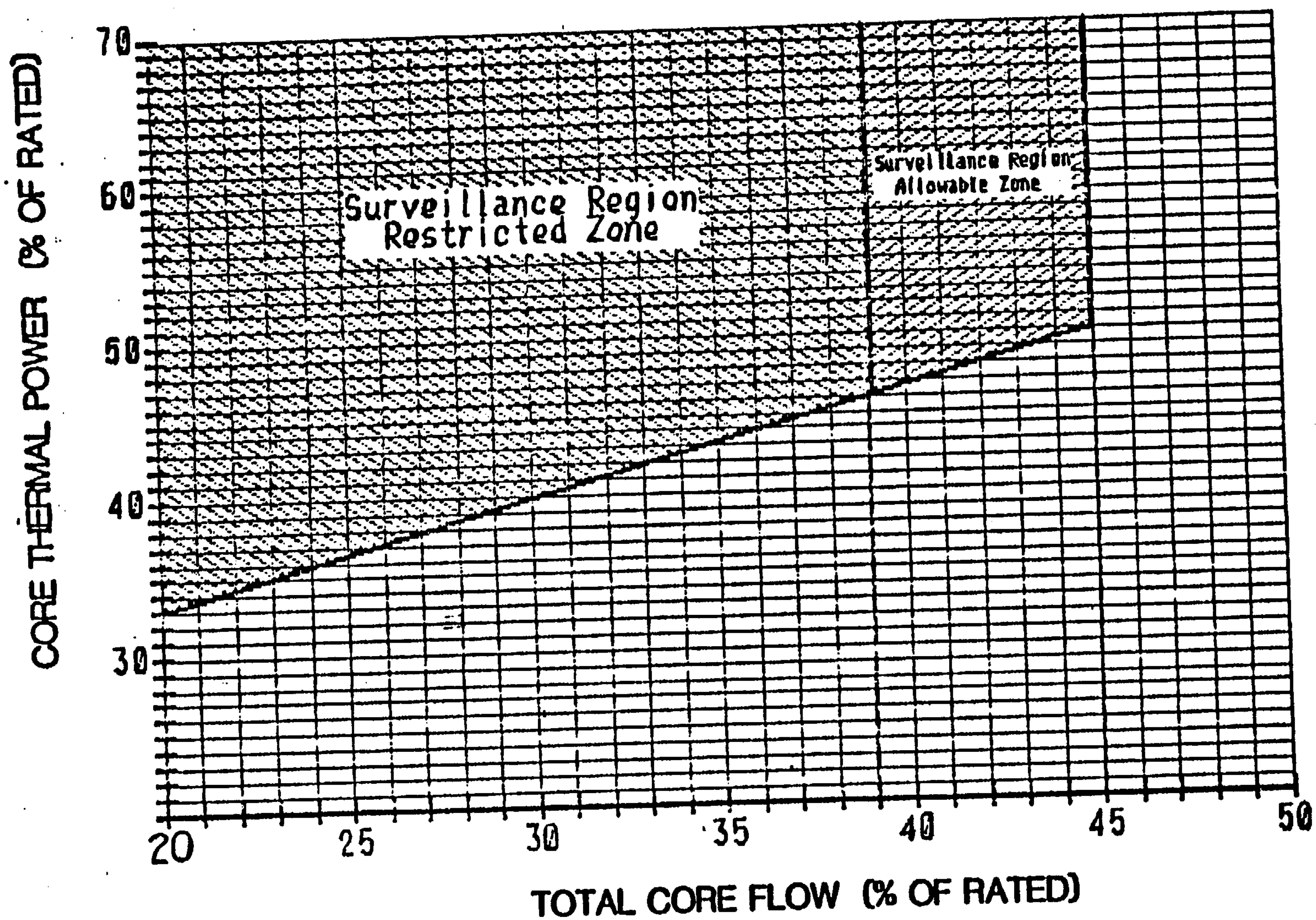


Figure 3.4.1.1 -1

3/4.4.1 RECIRCULATION SYSTEM

Operation with one reactor core coolant 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 maybe 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. 40 TO FACILITY OPERATING LICENSE NO. NPF-11

COMMONWEALTH EDISON COMPANY

LA SALLE COUNTY STATION, UNIT 1

DOCKET NO. 50-373

1.0 INTRODUCTION

By letter from H. Massin, Commonwealth Edison, to H. Denton, NRC, dated October 22, 1985 (Reference 1), Technical Specification changes were proposed for the operation of La Salle County Station Unit 1 for Cycle 2 (LSIC2) with a reload using General Electric (GE) manufactured fuel assemblies and GE analyses and methodologies. Enclosed were the requested Technical Specification changes and reports (including Reference 2) discussing the reload and analyses done to support and justify the second cycle operation. There was also a subsequent letter (Reference 3) revising part of the requested Technical Specification changes relating to single loop operation thermal-hydraulic stability.

2.0 EVALUATION

2.1 RELOAD DESCRIPTIONS

The LSIC2 reload will retain 100 8CRB176 and 432 8CRB219 fuel assemblies from the first cycle and add 232 new BP8CRB299L GE fuel assemblies (about 30% of the fuel). The reload is based on a Cycle 1 exposure of 10.3 to 10.5 Giga watt days/short ton (GWD/ST) and a Cycle 2 exposure of 6.8 GWD/ST. The loading will be a conventional scatter pattern with low reactivity fuel on the periphery. This is generally a normal reload with no unusual core feature or characteristics. Technical Specification changes are few and primarily related to Maximum Average Planar Linear Heat Generation Rate and Minimum Critical Power Ratio (MAPLHGR and MCPR) limits for the new fuel and Cycle 2 core and transient parameters.

2.2 FUEL DESIGN

The new fuel assembly to be used for LSIC2, BP8CRB299L, was not specifically listed in the latest GESTAR II Revision (Reference 4) or in NRC approved amendments at the time of the LSIC2 submittal (Reference 1). Thus the submittal presented a description and information on the assembly usually included in GESTAR II. The assembly is a BP8x8R fuel design, a type approved by the NRC, and has been analyzed for this application with the approved methods and meets the approved limits discussed in GESTAR II. The fuel enrichment and gadolinium patterns, generally considered a non-safety related change, are acceptable. The BP8CRB299L assembly has subsequently been included in Amendment 13 to GESTAR II and this section of the amendment has been reviewed and approved by the staff. (Amendment 13 approval is in final processing.) Thus the new fuel is acceptable for LSIC2.

2.3 NUCLEAR DESIGN

The nuclear design for LSIC2 has been performed with the methodology described in GESTAR II (Reference 4). The results of these analyses are given in Reference 2 in standard GESTAR II format. The results are within the range of those usually encountered for BWR reloads. In particular, the shutdown margin is 1.9 and 1.5% Δk at beginning of life (BOL) and at the exposure of minimum shutdown margin respectively, thus fully meeting the required 0.38% Δk. The Standby Liquid Control System also meets shutdown requirements with a shutdown margin of 3.8% Δk. Since these and other LSIC2 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 LSIC2 has been performed with the methodology described in GESTAR II (Reference 4) and the results are given in Reference 2. The parameters used for the analyses are those approved in Reference 4 for the La Salle class BWR-5.

The Safety Limit MCPR (SLMCPR) values are increased by 0.01 which is standard 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 LSIC2, via the ODYN Option B approach, provide new Cycle 2 Technical Specification values of OLMCPR as a function of average scram time, τ. At (and below) a τ of 0.736, RWE and FWCF both provide the limit at a MCPR of 1.26. FWCF is limiting above 0.736 until above a τ of 0.755, where LRWBP is the limiting event. Approved methods (Reference 4) 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 and MAPLHGR limits and scram and rod block setpoints on going from two to one (recirculation) loop operation remain the same as for Cycle 1. These changes for one loop operation, which have been determined by approved methods, continue to be acceptable. The Technical Specifications for thermal-hydraulic stability have been changed and will be discussed later. It is only the stability aspects of one loop operation which require reapproval for second cycle one loop operation. With the approval of these specifications, one loop operation is approved for La Salle 1 within the limits given in the relevant Technical Specifications.

2.5 TRANSIENT AND ACCIDENT ANALYSES

The transient and accident analysis methodologies used for LSIC2 are described in GESTAR II (Reference 4). 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 4. 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 a OLMCPR of 1.26. The fuel assembly misloading and misorientation events were not analyzed for LSIC2. As approved via Reference 4, 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. The local event analyses are thus acceptable.

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

Loss of coolant accident (LOCA) analyses, using approved methodologies and parameters (Reference 4), 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 LSIC2. 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 that with a high degree of confidence the peak fuel enthalpy would not approach the NRC required limit of 280 cal/gm for this event. This approach and analysis has been approved by the NRC (Reference 4). This approach is acceptable for LSIC2.

2.6 THERMAL-HYDRAULIC STABILITY

The original submittal for LSIC2 (Reference 1) presented several changes and additions to the Technical Specification relating to Thermal-Hydraulic Stability (THS) for both two and one (recirculation) loop operation. There followed a number of discussions with the staff on the acceptability of these changes for LSIC2. As a result of these discussions, Commonwealth Edison submitted (Reference 3) a revised approach and new specifications in this area. This review will discuss only the revised approach.

The NRC has recently published two Generic Letters (Reference 5 and 6) relating to THS. The following is a brief summary of the staff position indicated in these letters as pertaining in particular to approved GE stability analysis methodology and the La Salle reactor characteristics for LSIC2.

(1) Approved GE analysis methods may be uncertain in the calculation of the THS decay ratio by up to 20%.

(2) Therefore, analyses (for approved fuel) which result in a decay ratio of 0.8 or greater may not be able to comply with General Design Criteria 10 and 12. Decay ratios well below 0.8 would indicate acceptable stability without further requirements (for two loop operation), but stability calculations would be required for every cycle.

(3) Where further Technical Specification requirements are needed, regions in the power-flow operating map should be defined in which operating restrictions or specific oscillation surveillance and suppression requirements must be met. These regions and surveillance should be consistent with the recommendations of the GE Service Information Letter (SIL) - 380 (Reference 7).

(4) For single loop operation (SLO), these types of restrictions and surveillance should be instituted without specific regard to the decay ratio.

(5) THS Technical Specifications acceptable to the staff have recently been approved. Reference 6 indicates, in particular, that Technical Specifications similar to those approved for Duane Arnold should be submitted.

(6) These Technical Specifications generally call for restrictions or surveillance above the 80% rod line (in the power-flow map) and below 45% flow; surveillance above (approximately) 39% flow, no operation below 39%. Surveillance is by observation of the noise level of the Average Power Range Monitor (APRM) and selected Local Power Range Monitor (LPRM) detectors. Noise levels greater than 3 times base levels require noise suppression activity, e.g. lower power level.

For LSIC2 the stability has been analyzed using the approved methods (and fuel) of Reference 4. The result is a decay ratio of 0.60. Thus, no increased stability restrictions or surveillance is required for Cycle 2 normal (two loop) operation and Commonwealth Edison has elected to make no Technical Specification changes relating to stability, at this time, for two loop operation. This is acceptable provided that stability analyses are performed for all subsequent cycles and THS technical specifications are implemented when needed. For SLO, Commonwealth Edison has provided new Technical Specifications in Reference 3 (as changes and additions to 3/4.4.1.1 and Bases). These changes are in general accord with the specifications approved recently for other reactors (e.g., Duane Arnold). They provide for the establishment of regions above the 80 percent rod line where; (a) below 39% flow action must be taken to leave the region and (b) above 39% flow and below 45% flow action must by taken to monitor

APRM-LPRM noise and to reduce the noise or leave the region if the noise is greater than 3 times the baseline. They 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 aims of SIL-380. These are acceptable to the staff. Thus, one loop operation is generally acceptable for La Salle without restrictions other than those presented in Specification 3/4.1.1.

2.7 TECHNICAL SPECIFICATIONS

The Technical Specification changes are for the most part minor and provide for MCPR and MAPLHGR changes due to second cycle parameter changes and a new fuel assembly, no End of Cycle - Reactor Pump Trip (EOC-RPT) inoperable analysis for the cycle, and for a change in k_f. The primary change is for the one loop operation stability specification. The bases for most of these changes have already been discussed. The Specification changes are as follows:

(1) 2.1.2 and Bases and Table B 2.1.2 - 1 thru - 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 4. These various changes are acceptable.

(2) 3.2.1 and Figures:
A new MAPLHGR curve is provided for the new fuel and a fuel assembly designation change is made. These are acceptable.

(3) 3.2.3 and Figures:
The EOC-RPT inoperable analyses was not performed for Cycle 2 and thus the provision for this condition 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 4. This is acceptable.

(4) 3.3.4.2; Action d. and e.:
Changes were made to make this specification compatible with the elimination of the EOC-RPT inoperable provision of 3.2.3. These changes, including the indicated power reduction are reasonable and acceptable.

(5) 3.4.1.1 and Bases:
These changes are for the Thermal-Hydraulic Stability for single loop operation. They have been discussed in Section 2.6. They are acceptable.

2.8 REFERENCES

1. Letter from H. Massin, Commonwealth Edison, to H. Denton, NRC, "LaSalle County Station Unit 1, Proposed Amendment to Technical Specification for Facility Operating License NPF-11- Reload Licensing Package for Cycle 2," October 22, 1985

2. General Electric Report - 23A1843, June 1985, "Supplemental Reload Licensing Submittal for LaSalle County Station Unit 1, Reload 1 (Cycle 2)"

3. Letter from C. M. Allen, Commonwealth Edison, To H. Denton, NRC, "Supplemental Information," March 21, 1986

4. NEDE-24011-A-7, August 1985, "General Electric Standard Application for Reactor Fuel," (GESTAR II)

5. Generic Letter No. 86-02, "Technical Resolution of Generic Issue B-19-Thermal Hydraulic Stability," January 23, 1986

6. Generic Letter No. 86-09, "Technical Resolution of Generic Issue No. B-59-(N-1) Loop Operation in BWRs and PWRs", March 31, 1986

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

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 initially published in the Federal Register (50 FR 47859) on November 20, 1985, and a renotice which was published in the Federal Register (51 FR 12225) on April 9, 1986. No public comments were received on either notice, and the state of Illinois did not have any comments.

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.

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