NRC FORM 195 (2-76)

U.S. NUCLEAR REGULATORY COMMISSION

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TO: MR V STELLO

FROM: FLORIDA POWER & LIGHT CO MIAMI, FLA....... R E UHRIG.......

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CONTROL NUMBER 4273

P. O. BOX 013100, MIAMI, FLORIDA 33101
FLORIDA POWER & LIGHT COMPANY

April 27, 1976
L-76-172

Director of Nuclear Reactor Regulation
Attention: Mr. Victor Stello, Director
Division of Operating Reactors
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Dear Mr. Stello:

Re: St. Lucie Unit 1
Docket No. 50-335
Proposed Amendment to
Facility Operating License DPR-67

In accordance with 10 CFR 50.30, Florida Power and Light Company (FPL) submits herewith three (3) signed originals and forty (40) conformed copies of a request to amend Appendix A of Facility Operating License DPR-67.

Current FSAR analyses, setpoint analyses, and Technical Specifications are based on a reactor coolant flow rate of 370,000 gpm. However, post-core hot functional measurements have not demonstrated at this time that this flow rate exists. As a result, an interim Technical Specification amendment is needed to allow FPL to perform as much of the power ascension test program as possible. This amendment is based on a reactor coolant flow rate of 354,000 gpm (109.0% of design flow). A reanalysis is being performed to support full power operation at the lower flow rate. The reanalysis is expected to result in a proposed permanent change to the Technical Specifications but until then it is emphasized that the change being proposed by this letter is considered temporary.

The proposed interim changes are as described below and as shown on the accompanying Technical Specification pages bearing the date of this letter in the lower right hand corner.

Page 2-2

VESSEL FLOW LESS MEASUREMENT UNCERTAINTIES on Figure 2.1-1 is changed from 370,000 GPM to 354,000 GPM. (Interim change).

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Page 2-4

The note at the bottom of Table 2.2-1 is changed to read "Design reactor coolant flow with 4 pumps operating is 354,000 gpm." In Item 2, 106.5% is changed to 96.5%.(Interim).

Page 2-8

On Figure 2.2-3, the last factor in the P_{VAR}^{TRIP} equation is changed from 6250 to 6150. (Interim change).

Page 2-9

On Figure 2.2-4, the last factor in the P_{VAR}^{TRIP} equation is changed from 6250 to 6150. (Interim change).

Page B2-2

The values of ROD RADIAL PEAK are changed to those indicated on attached Figure B2.1-1. (Interim change).

Page 3/4 2-2

Interim specification 4.2.1.4.b.7 is added, to read as follows:

> "An interim flow adjustment factor of 1.10 is to be implemented if the reactor coolant flow rate is less than 370,000 gpm and greater than 354,000 gpm."

Page 3/4 2-4

Figure 3.2-2 is revised to reflect a power limitation of 90 per cent pending reanalysis at the 354,000 gpm flow rate. (Interim change).

Director of Nuclear Reactor Regulation
Attention: Mr. Victor Stello, Director
Division of Operating Reactors

April 27, 1976

<u>Page 3/4 2-13</u>

In Table 3.2-1, the reactor coolant flow rate is changed from 370,000 gpm to 354,000 gpm. (Interim change).

<u>Page 3/4 2-14</u>

Figure 3.2-4 is revised to reflect a power limitation of 90% pending reanalysis at the 354,000 gpm flow rate. (Interim change).

<u>Page B3/4 2-1</u>

Bases 3/4.2.1 is revised to include mention of the 1.10 interim flow factor. (Interim change).

The proposed interim amendment has been reviewed and the conclusion reached that it does not involve a significant hazards consideration, therefore, prenoticing pursuant to 10 CFR 2.105 should not be required. A written safety evaluation is attached.

Very truly yours,

Robert E. Uhrig

Robert E. Uhrig
Vice President

REU/MAS/cpc

Attachment

cc: Mr. Norman C. Moseley
Jack R. Newman, Esquire

STATE OF FLORIDA)
) SS
COUNTY OF DADE)

Robert E. Uhrig, being first duly sworn, deposes and says:

That he is a Vice President of Florida Power & Light Company, the Licensee herein;

That he has executed the foregoing document; that the statements made in this said document are true and correct to the best of his knowledge, information and belief, and that he is authorized to execute the document on behalf of said Licensee.

Robert E. Uhrig

Robert E. Uhrig

Subscribed and sworn to before me

this 28th day of April, 1976.

Rebecca H. Powell

Notary Public, in and for the County of Dade, State of Florida

My commission expires: NOTARY PUBLIC STATE OF FLORIDA AT LARGE MY COMMISSION EXPIRES APRIL 2, 19[illegible] BONDED THRU MAYNARD BONDING AGENCY

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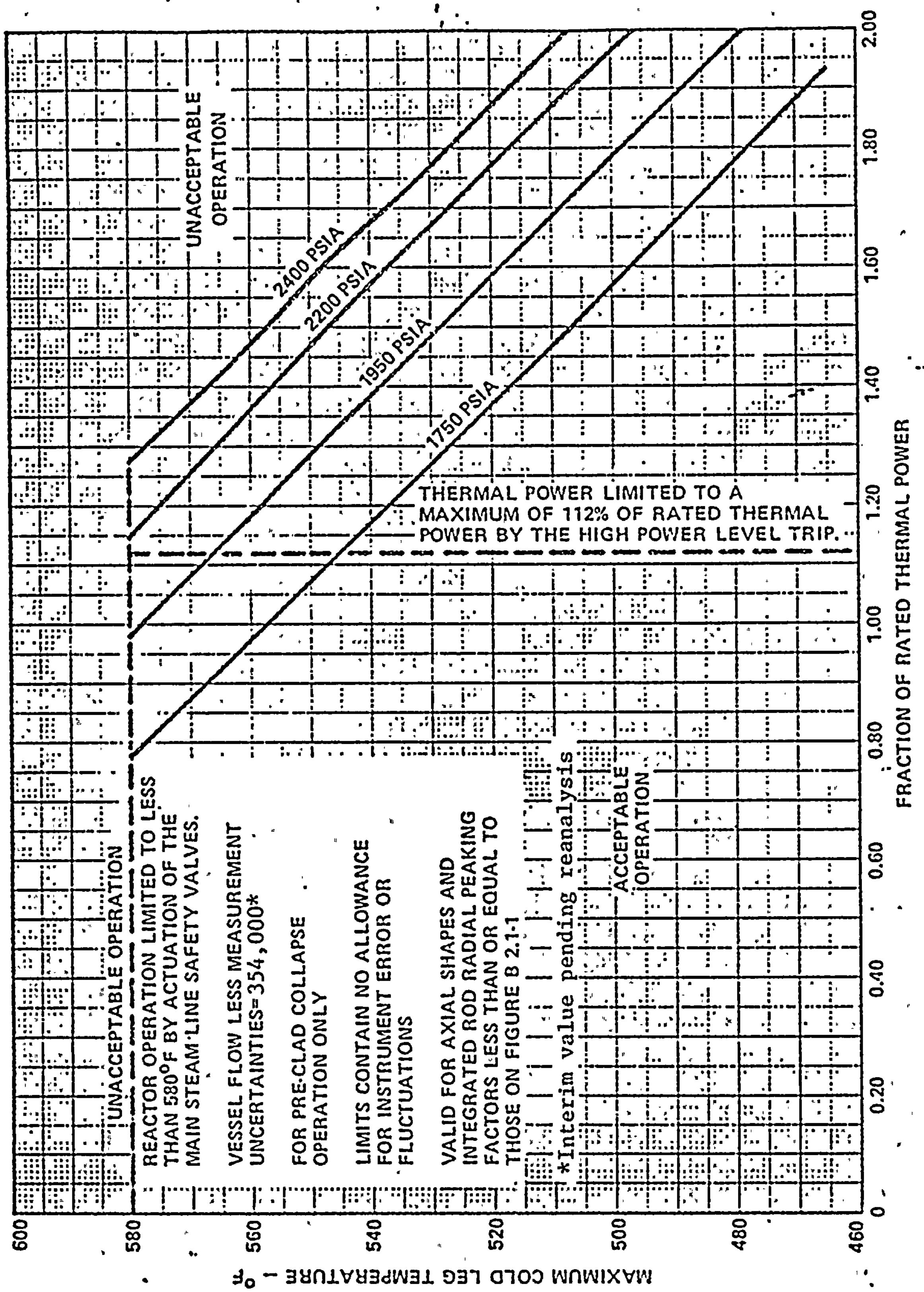

Figure 2.1-1 REACTOR CORE THERMAL MARGIN SAFETY LIMIT – FOUR REACTOR COOLANT PUMPS OPERATING

TABLE 2.2-1

REACTOR PROTECTIVE INSTRUMENTATION TRIP SETPOINT LIMITS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
1. Manual Reactor Trip	Not Applicable	Not Applicable
2. Power Level - High (1)		
Four Reactor Coolant Pumps Operating	≤ 9.61% above THERMAL POWER, with a minimum setpoint of 15% of RATED THERMAL POWER, and a maximum of ≤ 96.5% of RATED THERMAL POWER.	≤ 9.61% above THERMAL POWER, and a minimum setpoint of 15% of RATED THERMAL POWER and a maximum of ≤ 96.5% of RATED THERMAL POWER.
3. Reactor Coolant Flow - Low (1)		
Four Reactor Coolant Pumps Operating	≥ 95% of design reactor coolant flow with 4 pumps operating*	≥ 95% of design reactor coolant flow with 4 pumps operating*
4. Pressurizer Pressure - High	≤ 2400 psia	≤ 2400 psia
5. Containment Pressure - High	≤ 3.9 psig	≤ 3.9 psig
6. Steam Generator Pressure - Low (2)	≥ 485 psig	≥ 485 psig
7. Steam Generator Water Level -Low	≥ 36.3% Water Level - each steam generator	≥ 36.3% Water Level - each steam generator
8. Local Power Density - High (3)	Trip setpoint adjusted to not exceed the limit lines of Figures 2.2-1 and 2.2-2	Trip set point adjusted to not exceed the limit lines of Figures 2.2-1 and 2.2-2.

*Design reactor coolant flow with 4 pumps operating is 354,000**

**Interim value pending reanalysis.

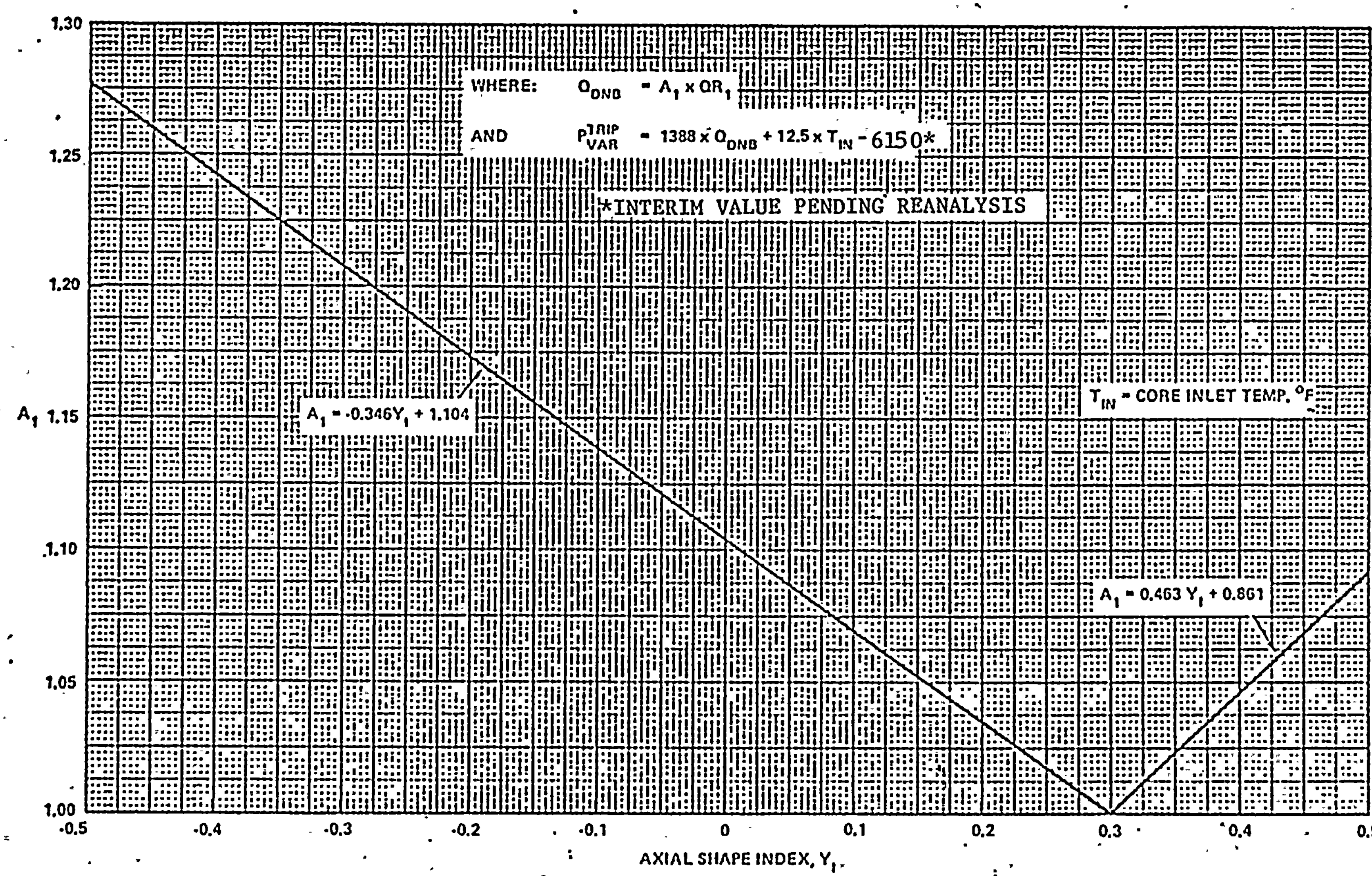

FIGURE 2.2-3
Thermal Margin/Low Pressure Trip Setpoint
Part 1 (Y_1 Versus A_1)

WHERE: $A_1 \times QR_1 = Q_{DNB}$

AND $P_{VAR}^{TRIP} = 1388 \times Q_{DNB} + 12.5\, T_{IN} - 6150^*$

*INTERIM VALUE PENDING REANALYSIS

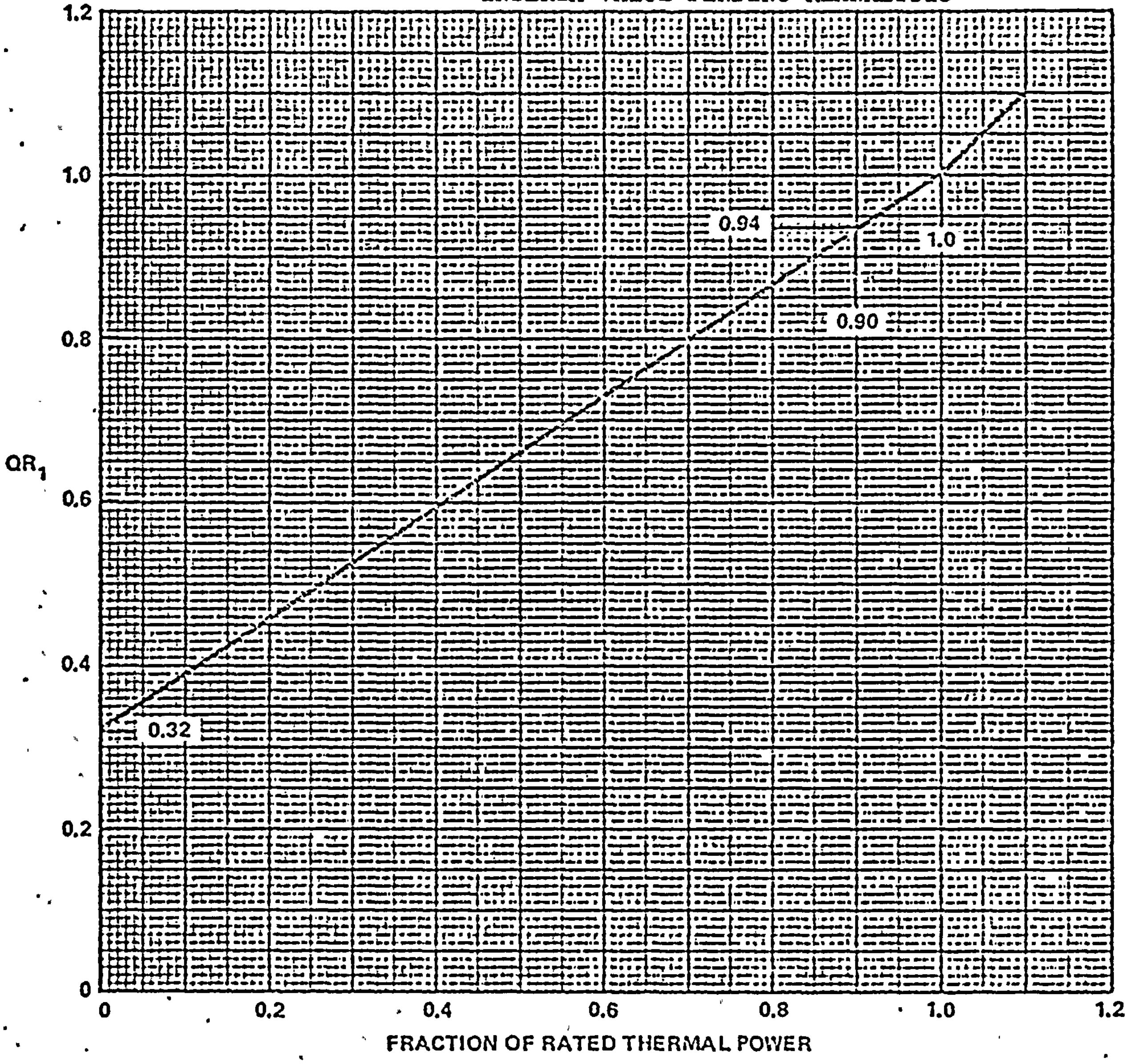

FIGURE 2.2-4

Thermal Margin/Low Pressure Trip Setpoint
Part 2 (Fraction of RATED THERMAL POWER Versus QR_1)

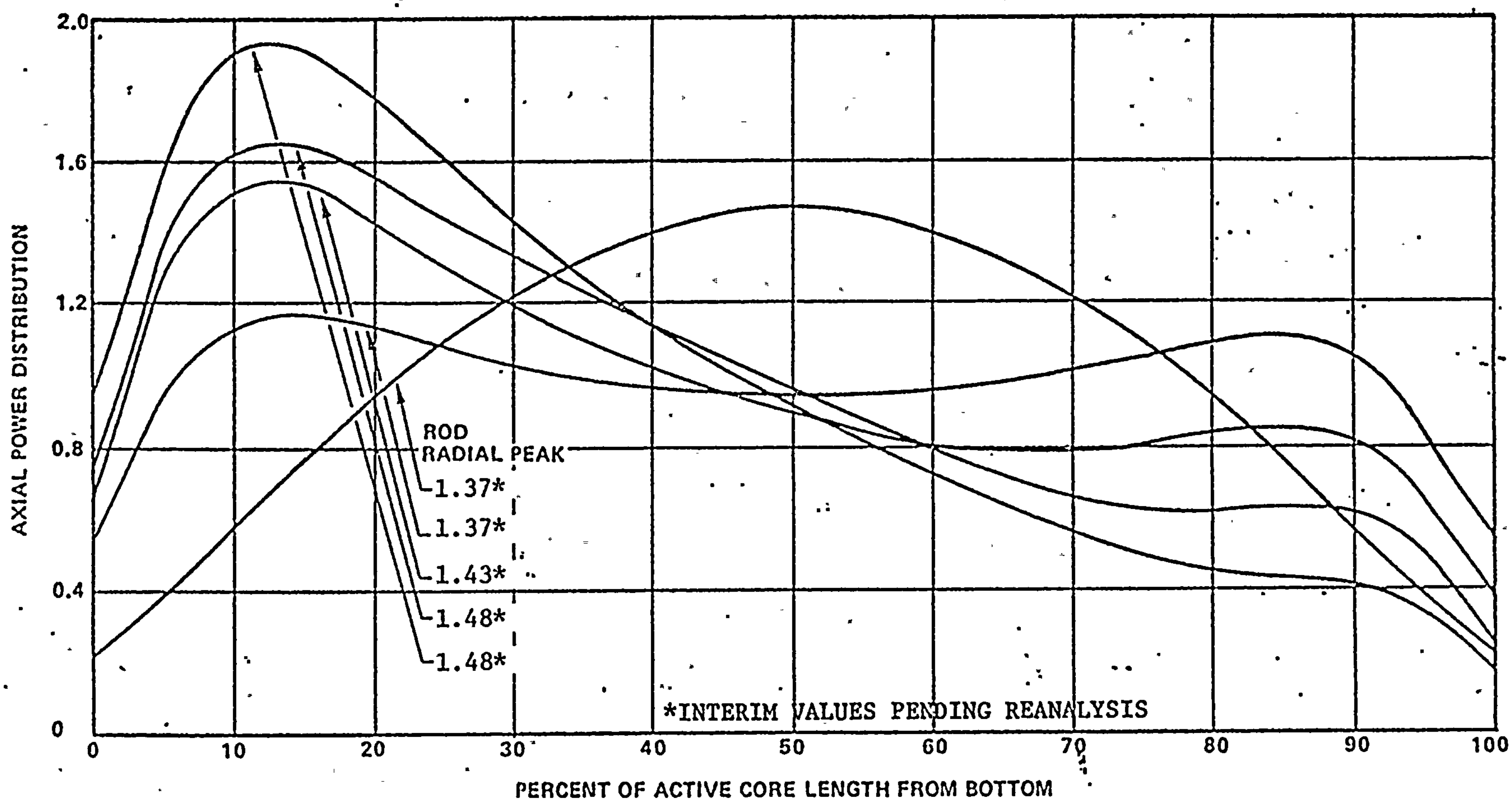

Figure B2.1-1 Axial Power Distribution for Thermal Margin Safety Limits

SURVEILLANCE REQUIREMENTS (Continued)

c. Verifying at least once per 31 days that the THERMAL POWER does not exceed the value determined by the following relationship:

$$\frac{L}{17.85} \times M$$

where:

1. L is the maximum allowable linear heat rate as determined from Figure 3.2-1 and is based on the core average burnup at the time of the latest incore flux map.

2. M is the maximum allowable THERMAL POWER level for the existing Reactor Coolant Pump combination.

4.2.1.4 Incore Detector Monitoring System - The incore detector monitoring system may be used for monitoring the core power distribution by verifying that the incore detector Local Power Density alarms:

a. Are adjusted to satisfy the requirements of the core power distribution map which shall be updated at least once per 31 days.

b. Have their alarm setpoint adjusted to less than or equal to the limits shown on Figure 3.2-1 when the following factors are appropriately included in the setting of these alarms:

   1. Flux peaking augmentation factors as shown in Figure 4.2-1,

   2. A measurement-calculational uncertainty factor of 1.10,

   3. An engineering uncertainty factor of 1.03,

   4. A linear heat rate uncertainty factor of 1.01 due to axial fuel densification and thermal expansion,

   5. A THERMAL POWER measurement uncertainty factor of 1.02, and

   6. A rod bow penalty factor of 1.05.

   7. An interim flow adjustment factor of 1.10 is to be implemented if the reactor coolant flow rate is less than 370,000 gpm and greater than 354,000 gpm.*

*Interim specification pending reanalysis.

4/27/76

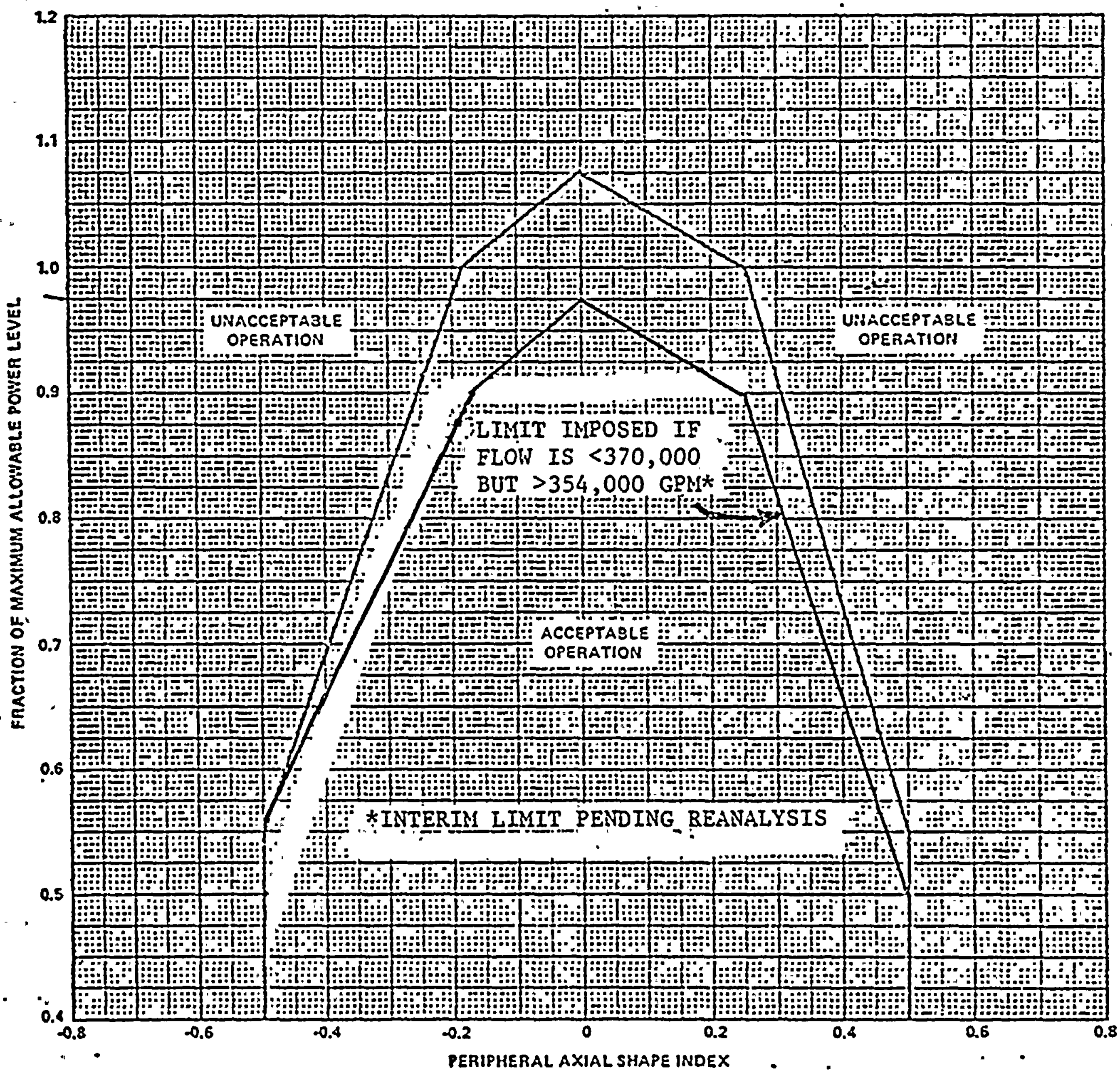

Figure 3.2-2

AXIAL SHAPE INDEX vs Fraction of Maximum Allowable Power Level per Specification 4.2.1.3

TABLE 3.2-1

DNB MARGIN

LIMITS

Parameter	Four Reactor Coolant Pumps Operating
Cold Leg Temperature	≤ 542°F
Pressurizer Pressure	≥ 2225 psia*
Reactor Coolant Flow Rate	≥ 354,000 gpm**
AXIAL SHAPE INDEX	Figure 3.2-4

*Limit not applicable during either a THERMAL POWER ramp increase in excess of 5% of RATED THERMAL POWER or a THERMAL POWER step increase of greater than 10% of RATED THERMAL POWER.

**Interim value pending reanalysis.

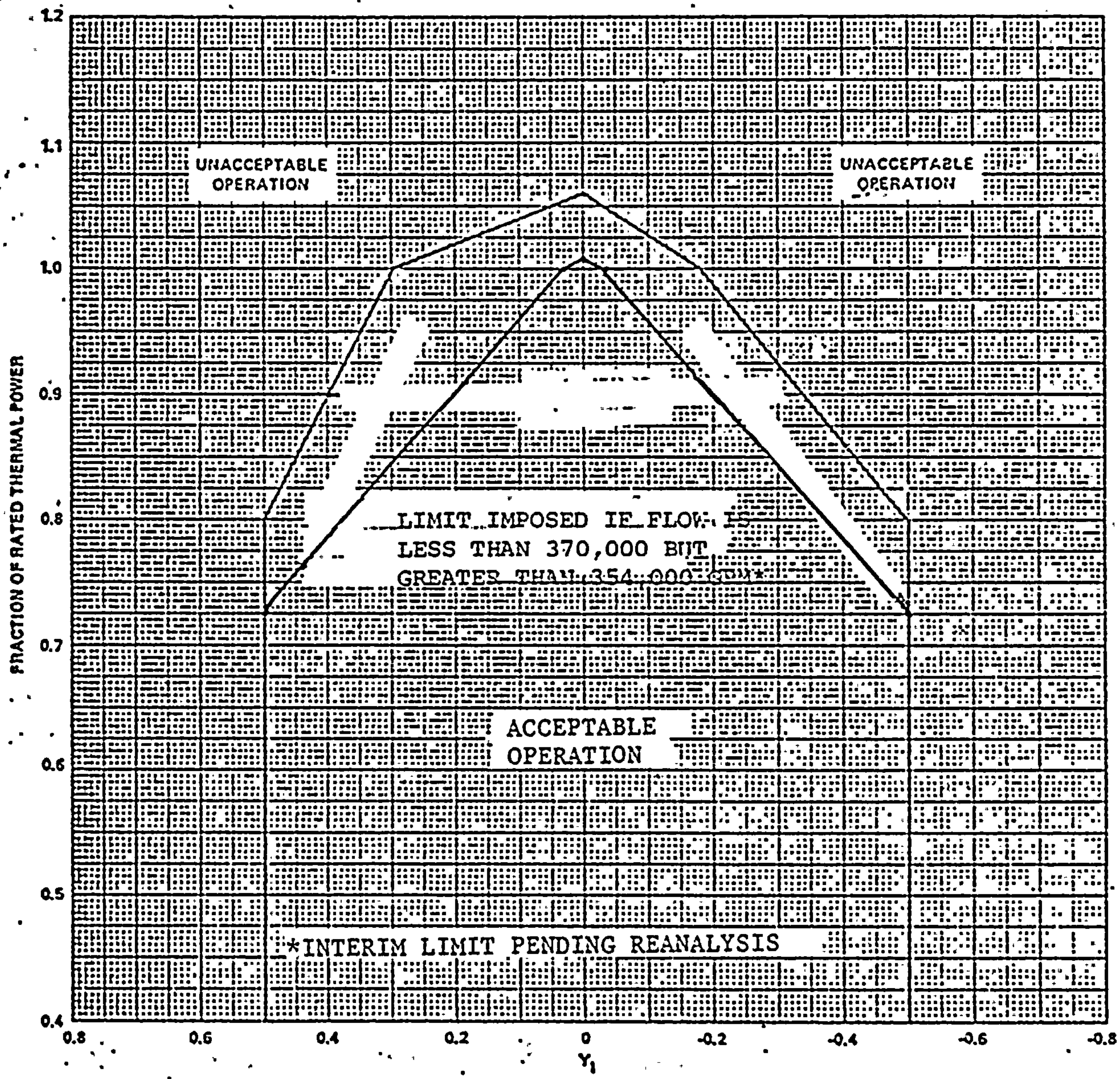

FIGURE 3.2-4
AXIAL SHAPE INDEX Operating Limits with 4 Reactor Coolant Pumps Operating

3/4.2 POWER DISTRIBUTION LIMITS

BASES

3/4.2.1 LINEAR HEAT RATE

The limitation on linear heat rate ensures that in the event of a LOCA, the peak temperature of the fuel cladding will not exceed 2200°F.

Either of the two core power distribution monitoring systems, the Excore Detector Monitoring System and the Incore Detector Monitoring System, provide adequate monitoring of the core power distribution and are capable of verifying that the linear heat rate does not exceed its limits. The Excore Detector Monitoring System performs this function by continuously monitoring the AXIAL SHAPE INDEX with the OPERABLE quadrant symmetric excore neutron flux detectors and verifying that the AXIAL SHAPE INDEX is maintained within the allowable limits of Figure 3.2-2. In conjunction with the use of the excore monitoring system and in establishing the AXIAL SHAPE INDEX limits, the following assumptions are made: 1) the CEA insertion limits of Specifications 3.1.3.2, 3.1.3.5 and 3.1.3.6 are satisfied, 2) the flux peaking augmentation factors are as shown in Figure 4.2-1, 3) the AZIMUTHAL POWER TILT restrictions of Specification 3.2.3 are satisfied, and 4) the TOTAL RADIAL PEAKING FACTOR does not exceed the limits of Specification 3.2.2.

The Incore Detector Monitoring System continuously provides a direct measure of the peaking factors and the alarms which have been established for the individual incore detector segments ensure that the peak linear heat rates will be maintained within the allowable limits of Figure 3.2-1. The setpoints for these alarms include allowances, set in the conservative directions, for 1) flux peaking augmentation factors as shown in Figure 4.2-1, 2) a measurement-calculational uncertainty factor of 1.10, 3) an engineering uncertainty factor of 1.03, 4) an allowance of 1.01 for axial fuel densification and thermal expansion, 5) a THERMAL POWER measurement uncertainty factor of 1.02, 6) a rod bow penalty factor of 1.05, and 7) an interim flow adjustment factor is to be implemented if the reactor coolant flow rate is ≤370,000 but >354,000 gpm.

3/4.2.2 and 3/4.2.3 TOTAL RADIAL PEAKING FACTOR - F_r^T AND AZIMUTHAL POWER TILT - T_q

The limitations on F_r^T and T_q are provided to ensure that the assumptions used in the analysis for establishing the DNB Margin, Linear Heat Rate, Thermal Margin/Low Pressure and Local Power Density - High LCOs and LSSS setpoints remain valid during operation at the various allowable CEA group insertion limits. If either F_r^T or T_q exceed their basic limitations, operation may continue under the additional restrictions imposed by the

SAFETY EVALUATION

1. DNB Safety Limit

   a. Figure 2.1-1 (Page 2-2)

      Change 370,000 GPM to 354,000 GPM as indicated on Figure 2.1-1.

   b. Figure B2.1-1 (Page B2-2)

      Change the values of the rod radial peaks to those indicated on Figure B2.1-1.

Basis

The present Reactor Core Thermal Margin Safety Limit shown by Figure 2.1-1, page 2-2, represents a locus of points corresponding to a limit of 1.3 DNBR (W-3) for a reactor coolant flow of 370,000 GPM and for the rod radial peaks and axial power distributions shown in Figure B2.1-1, page B2-2. If reactor coolant flow is decreased, there is a corresponding increase in the enthalpy rise across the core. This results in an increase in the quality in the hot channels, and the limit of 1.3 DNBR is reached at a lower value of rod radial peak. The values of rod radial peak indicated on the revised Figure B2.1-1 have been reduced to a factor of 0.95 to accommodate a 5% reduction in the reactor coolant flow. Thus for the revised road radial peaks, and the axial power distributions in the revised Figure B2.1-1 and for the reduced reactor coolant flow of 354,000 GPM, the Thermal Margin Safety Limit in Figure 2.1-1, page 2-2, continues to represent a locus of points corresponding to a limit of 1.3 DNBR (W-3).

2. Limiting Safety System Settings

   a. Table 2.2-1 (Page 2-4)

      Change note on bottom of page to read:
      "Design reactor coolant flow with 4 pumps operating is 354,000 GPM."

   b. Table 2.2-1 (Page 2-4)

      Item 2, Power Level-High, under Trip Setpoint and Allowable Values, change the maximum limit from ≤106.5% to ≤96.5% of Rated Thermal Power.

Basis

Item (a) is modified to be consistent with the reduced reactor coolant flow rate. Item (b), the variable portion of the high power trip, remains at a maximum allowable setting of 10% above indicated power since this parameter is not sensitive to reactor coolant flow rate. The upper limit of the high power trip has been reduced to a value of 96.5% of Rated Thermal Power. This is consistent with an operating limit power level of 90% of Rated Power.

c. Figure 2.2-3 (Page 2-8)

Change P_{var}^{trip} equation to read:

$$P_{var}^{trip} = 1388 \times Q_{DNB} + 12.5 \times T_{in} - 6150$$

d. Figure 2.2-4 (Page 2-9)

Change P_{var}^{trip} equation to read as specified in item 2.c above.

Basis

The Thermal Margin/Low Pressure trip is provided to prevent operation when the DNBR (W-3) is less than 1.30. The available margin to the limit of 1.30 DNBR previously calculated for a wide range of core conditions has been reduced to a factor of 0.95 to accommodate a 5% reduction in the reactor coolant flow. The revised LSSS implement this reduction by modifying the computed value of pressure at which a reactor trip is initiated. This modification is shown in the P_{var}^{trip} equation in the revised Figures 2.2-3 and 2.2-4. The revised thermal margin LSSS provide at least the same margin of safety as that provided with the present LSSS.

3. Limiting Conditions for Operation

a. Table 3.2-1 (Page 3/4 2-13)

Change reactor coolant flow rate to read: $\geq$ 354,000 GPM

b. Figure 3.2-4 (Page 3/4 2-14)

Change to implement the revised Figure 3.2-4.

c. Figure 3.2-2 (Page 3/4 2-4)

Change to implement the revised Figure 3.2-2.

Basis

Item 3.a. is modified to be consistent with the reduced reactor coolant flow rate. Item 3.b. refers to the Limiting Condition for Operation (LCO) on the axial shape index to maintain the required steady state operating margin to DNB. The available margin to a 1.3 DNBR calculated for a wide range of core conditions was reduced to a factor of 0.95 to accommodate a 5% reduction in reactor coolant flow. The revised LCO was established consistent with this reduction. The transient analyses of the design basis LCO and the initial assumptions in the FSAR have been reviewed, and it was concluded that the revised LCO in Figure 3.2-4 provides at least the same margin of safety as that provided with the present LCO.

Item 3.c. refers to the LCO on axial shape index to maintain the require steady-state operating margin on linear heat rate when the ex-core detectors are used to monitor this limit. Since the limiting value of kw/ft has at present been established by the LOCA analysis, Figure 3.2-2 has been reduced to a factor of 1.10 to accommodate the 5% reduction in reactor coolant flow. The basis for this reduction is provided after item 3.e below.

d. Specification 4.2.1.4 (Page 3/4 2-2)

Add item 7 under part b to read:

7. A flow adjustment factor of 1.10 is to be implemented if the reactor coolant flow rate is less than 370,000 GPM and greater than 354,000 GPM.

e. Basis 3/4.2.1 (Page B3/4 2-1)

Add to the paragraph on incore detectors the statement in modification 3.d.above.

Basis — St. Lucie 1 Reduced Flow LCOS Performance Results

The results of a St. Lucie 1 break spectrum analysis, using the approved Combustion Engineering large break evaluation model (2) are reported in Reference 1. This analysis, which employed a system flow rate of 139.44 x 10^6 lbs/hr, demonstrated that the LOCA Acceptance Criteria (3) were met at a peak linear heat generation rate (PLHGR) of 15.8 kw/ft.

Subsequent to the analysis reported in Reference 1, a conservative re-analysis has been performed to determine the allowable PLHGR when the system flow rate is reduced to 134.06 x 10^6 lbs/hr which is 4% less than the flow rate used in Reference 1. The allowable PLHGR at the reduced flow was found to be 15.6 kw/ft. In order to determine sensitivity to linear heat rate, a second reduced flow case was examined, at a PLHGR of 14.2 kw/ft. The results of this case, as well as those discussed above, are summarized below:

Worst Break* LOCA Results

System Flow (X10^6 lb/hr)	139.44	134.06	134.06
PLHGR (kw/ft)	15.8	15.6	14.2
Peak Clad Temperature (^{o}F)	2192	2189	1956
Peak Local Clad Oxidation (%)	10.42	9.71	5.32
Peak Core-Wide Clad Oxidation (%)	<.787	<.787	<.787

These results show that all LOCA acceptance criteria (3) are met at a PLHGR of 15.6 kw/ft, and there is substantial margin at 14.2 kw/ft.

*The worst break is the 0.8 Double-Ended Guillotine at the Pump Discharge (0.8 DEG/PD)

Based on these considerations, (1) the proposed change does not increase the probability or consequences of accidents or malfunctions of equipment important to safety and does not reduce the margin of safety as defined in the basis for any technical specification, therefore, the change does not involve a significant hazards consideration, (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (3) 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.

REFERENCES

1. Safety Injection System Analysis of St. Lucie 1 for FSAR, Section 6.3.3.6 as amended by Revision. 55, 2-9-76.

2. CENPD-132, "Calculative Methods for the C-E Large Break LOCA Evaluation Model", August, 1974 (Proprietary).

   CENPD-132, Supplement 1, "Updated Calculative Methods for the C-E Large Break LOCA Evaluation Model," December, 1974 (Proprietary).

   CENPD-132, Supplement 2, "Calculational Methods for the C-E Large Break LOCA Evaluation Model", July, 1975.

3. Acceptance Criteria for Emergency Core Cooling Systems for Light-Water-Cooled Nuclear Power Reactors, Federal Register, Vol. 39, No. 3 - Friday, January 4, 1974.