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Docket No. 50-335

Florida Power & Light Company
ATTN: Dr. Robert E. Uhrig
Vice President
Nuclear and General Engineering
Post Office Box 3100
Miami, Florida 33101

Gentlemen:

Enclosed is a signed original of the "Order for Modification of License" issued by the Commission for St. Lucie Plant, Unit No. 1. The Order amends Facility Operating License DPR-67 by (1) adding the provision that the reactor shall not be operated with a peak linear heat generation rate in excess of 12.7 kW/ft for all fuel assemblies and (2) a revision restricting power to 90% of full power. Also enclosed is the "Safety Evaluation by the Office of Nuclear Reactor Regulation Supporting an Interim Power Limit of 90% of Full Power". This Order also requires submittal of a corrected ECCS analysis as soon as possible.

A copy of the Order is being filed with the Office of the Federal Register for publication.

Sincerely,

Original signed by
Dennis L. Ziemann

Dennis L. Ziemann, Chief
Operating Reactors Branch #2
Division of Operating Reactors

Enclosures:
1. Order for Modification of License
2. Safety Evaluation

cc w/enclosures:
See next page

DOR:ORB #2
RMDiggs
6/ /76

NOTE: FOR CONCURRENCE, SEE PREVIOUS YELLOW

NRR:DIR
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OFFICE ➤	OR:ORB #2	OR:ORB #2	OR:AD/OR	OELD	OR:DIR	NRR:D/DIR
SURNAME ➤	RSilver:ro	DLZiemann	KRGoller		VStello	EGCase
DATE ➤	6/17/76	6/17/76	6/ /76	6/ /76	6/ /76	6/ /76

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Docket No. 50-335

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BJones (4)
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Vice President
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A copy of the Order is being filed with the Office of the Federal Register for publication.

Sincerely,

Dennis L. Ziemann, Chief
Operating Reactors Branch #2
Division of Operating Reactors

Enclosures:
1. Order for Modification of License
2. Safety Evaluation

cc w/enclosures.
See next page

DOR:ORB #2
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OFFICE➤	DOR:ORB #2	DOR:ORB #2	DOR:AD/ORs	OELD	DOR:DIR	NRR:D/DIR
SURNAME➤	RDSilver:ah	DLZiemann	KRGoller		VStello	EGCase
DATE➤	6/17/76	6/17/76	6/17/76	6/17/76	6/17/76	6/17/76

cc w/enclosures:
Jack R. Newman, Esquire
Lowenstein, Newman, Reis & Axelrad
1025 Connecticut Avenue, N. W.
Washington, D. C. 20036

Norman A. Coll, Esquire
McCarthy, Steel, Hector & Davis
14th Floor, First National Bank Building
Miami, Florida 33131

Mr. John L. McQuigg
Post Office Box 1408
Stuart, Florida 33494

Mr. Ed Maroney
Bureau of Intergovernmental Relations
725 South Bronough Street
Tallahassee, Florida 33450

Mr. Weldon B. Lewis
County Administrator
St. Lucie County
Post Office Box 700
Ft. Pierce, Florida 33450

Indian River Junior College Library
3209 Virginia Avenue
Ft. Pierce, Florida 33450

Mr. Shepard N. Moore, Region IV
Environmental Protection Agency
1421 Peachtree Street, N. E., Suite 300
Atlanta, Georgia 30390

Mr. Bruce Blanchard
Environmental Projects Review
Department of the Interior
Room 5321
18th and C Streets, N. W.
Washington, D. C. 20240

Chief, TIRB (2)
Technology Assessment Division
Office of Radiation Programs
EPA, Room 647A, East Tower
Waterside Mall
401 M Street, S. W.
Washington, D. C. 20460

Weldon B. Lewis
County Administrator
St. Lucie County
Post Office Box 700
Ft. Pierce, Florida 33450

cc w/enclosure and cy of FP&L
filing dtd. 6/15/76:
Mr. Ed Maroney
Bureau of Intergovernmental Relations
725 South Bronough Street
Tallahassee, Florida 32304

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

In the Matter of FLORIDA POWER & LIGHT COMPANY (St. Lucie Plant Unit No. 1)	)))))	Docket No. 50-335

ORDER FOR MODIFICATION OF LICENSE

I.

Florida Power & Light Company, Post Office Box 3100, Miami, Florida 33101 (the Licensee), is the holder of Facility Operating License No. DPR-67 which authorizes the operation of a nuclear power reactor known as St. Lucie Plant Unit No. 1 (the facility) at steady state reactor power levels not in excess of 2560 thermal megawatts (rated power). The facility is a pressurized water reactor (PWR) located at the Licensee's site on Hutchinson Island in St. Lucie County, Florida.

II.

FSAR analyses, setpoint analyses, and Technical Specifications for St. Lucie Unit No. 1 were based on a reactor coolant flow rate of 370,000 gpm. However, hot functional test measurements have indicated that slightly less flow may exist. As a result the Licensee submitted interim limitations and supporting analyses for the purpose of demonstrating that operation at up to 90% of rated power would provide adequate assurance of public health and safety with a minimum reactor coolant flow of 354,000 gpm (some 6% less than the measured flow during flow tests). On the basis of a preliminary assessment of this information Amendment No. 5 to License DPR-67 was issued on April 30, 1976 which limited power to 60% of rated power, under conditions specified therein, pending completion of a more detailed review.

The staff has completed a more detailed review of the information, originally submitted by the letters dated April 27 and 30, 1976, and additional information submitted by a letter dated May 14, 1976 regarding the reduced flow ECCS performance analysis and the use of a calorimetric technique to obtain an independent check on the measured value of flow rate.

The Licensee proposed appropriate interim limitations for operation at 90% of full power with a reactor coolant flow rate of at least 354,000 gpm. In support of this evaluation, the Licensee provided an analysis of ECCS performance under the proposed conditions, which indicated that peak clad temperature and cladding oxidation values would be within the limits of 10 CFR §50.46(b) at peak linear heat generation rates of 15.6 kW/ft.

The ECCS performance evaluation submitted by the Licensee was based upon the most current approved ECCS evaluation model developed by Combustion Engineering, Inc. (CE), the designer of the facility, to conform to the requirements of the Commission's ECCS Acceptance Criteria, 10 CFR

Part 50, §50.46 and Appendix K. The evaluation indicated that with peak linear heat generation rate limited as set forth above, and with the other limits set forth in the facility's Technical Specifications, the ECCS cooling performance for the facility would conform to the criteria contained in 10 CFR §50.46(b) which govern calculated peak clad temperature, maximum cladding oxidation, maximum hydrogen generation, coolable geometry and long term cooling.

On June 8, 1976, the NRC staff was informed by CE that several errors had been discovered in STRIKIN-2, the computer code used to calculate peak clad temperature and the clad oxidation percentage in their ECCS model. These errors were discovered by CE during an internal Quality Assurance audit of their LOCA evaluation model codes. While some of these errors have either no significant effect or a conservative effect on the evaluation results, some lead to non-conservative values. Based on a preliminary assessment, including information and supportive calculations by CE, the staff has determined that the following two code errors, when corrected, could produce ECCS evaluation results which would require a reduction in operating limits for Combustion Engineering Plants:

(1) Guide Tube Model - The code treated the control rod guide tube as a solid rod rather than a hollow tube. This resulted in an excess heat storage capacity in the guide tube which then led to excessive thermal radiation cooling from the hot rod to the guide tube.

(2) View Factors for Radiation Cooling Model - The code did not conservatively treat the view factors in the thermal radiation model to account for the possible effect of rupture and ballooning of adjacent fuel rods which contact the hot rod and reduce the surface area available for radiation cooling.

For this reason the staff instructed CE and the Licensee to provide a revised calculation of peak clad temperature for the worst break area identified in previous calculations with the errors properly corrected. The revised ECCS calculations were performed using the current, NRC staff approved, CE ECCS evaluation model, a reactor coolant flow rate value, which was reduced corresponding to current flow-test measurements and a power level of 90% of full power. The code was corrected for the two items discussed above, and with an additional correction of a sign error in the source term of the conduction equations (this latter error produced a conservative effect), the revised calculations demonstrate that for peak linear heat generation rates of 13.7 kW/ft in all fuel assemblies, at a power level of 90% of full power, the peak clad temperature and amount of cladding oxidation remain below the criteria set forth in 10 CFR §50.46(b). The staff expects that when final revised calculations for the facility are submitted using the revised and corrected model they will demonstrate that operation with these peak linear heat generation rates would conform to the criteria of 10 CFR §50.46(b). Such revised calculations fully conforming to the requirements of 10 CFR §50.46 are to be provided for the facility as soon as possible. However, since a revised evaluation for the entire break spectrum for the facility using the new evaluation model properly corrected cannot be completed for several weeks, the staff believes that it is prudent to impose an interim penalty on allowable peak linear heat generation rate to account for uncertainties that may result from the fact that calculations thus far have been made only for the worst case break previously identified. The staff concludes that an additional limitation of 1 kW/ft will eliminate uncertainties resulting from the preliminary limited break spectrum calculations thus far performed, and will assure that ECCS performance at the facility will conform to all the criteria set forth in 10 CFR §50.46(b). These additional limitations will provide reasonable assurance that the public health and safety will not be endangered.

With respect to all other aspects of operation at 90% of full power at a minimum coolant flow rate of 354,000 gpm the staff safety evaluation dated June 17, 1976, indicates that such operation will fully conform to the requirements of the Commission's regulations and will provide reasonable assurance of no undue risk to public health and safety.

Upon notification by the NRC staff on June 11, 1976, the Licensee promptly modified plant setpoints to reduce peak linear heat generation rate by 1 kW/ft to 12.7 kW/ft in all assemblies. This limitation is appropriate for operation at 90% of rated power. The NRC staff believes that the Licensee's action, under the circumstances, is appropriate and that this action should be confirmed by NRC Order.

III.

Copies of the following documents are available for public inspection at the Commission's Public Document Room, 1717 H Street, N. W., Washington, D. C. 20555 and are being placed in the Commission's Local Public Document Room, the Indian River Junior College Library, 3209 Virginia Avenue, Fort Pierce, Florida 33450: (1) Letter dated December 9, 1975 from the NRC staff to Combustion Engineering, and letter dated June 13, 1975 from the NRC staff to Combustion Engineering; (2) Letters dated April 27, April 30, May 14, June 14 and June 15, 1976 from Florida Power & Light Company to the Director of Nuclear Reactor Regulation; (3) Letter dated June 15, 1976 from Combustion Engineering to the NRC staff; (4) This Order for Modification of License, In the Matter of Florida Power & Light Company (St. Lucie Plant Unit No. 1), Docket No. 50-335); and (5) Safety Evaluation by the Office of Nuclear Reactor Regulation Supporting an Interim Power Limit of 90% of Full Power, Florida Power & Light Company, St. Lucie Plant Unit No. 1, Docket No. 50-335, dated June 17, 1976.

IV.

Accordingly, pursuant to the Atomic Energy Act of 1954, as amended, and the Commission's Rules and Regulations in 10 CFR Parts 2 and 50, IT IS ORDERED THAT Facility Operating License No. DPR-67 is hereby amended by adding the following new provisions:

(1) As soon as possible, the Licensee shall submit a re-evaluation of ECCS cooling performance calculated in accordance with Combustion Engineering Company's Evaluation Model approved by the NRC staff on December 9, 1975 and June 13, 1975 and corrected for the errors described herein.

(2) Until further authorization by the Commission, the reactor shall not be operated with a peak linear heat generation rate in excess of 12.7 kW/ft for all fuel assemblies.

(3) Until further authorization by the Commission, operation of the facility shall be limited to 90% of rated power and the following limitation shall apply in lieu of Section K of Enclosure 1 of the license:

"Operation shall be in accordance with the limitations set forth in the Safety Evaluation by the Office of Nuclear Reactor Regulation Supporting An Interim Power Limit Of 90% Of Full Power."

FOR THE NUCLEAR REGULATORY COMMISSION

Ben C. Rusche

Ben C. Rusche, Director
Office of Nuclear Reactor Regulation

Dated in Bethesda, Maryland,
this 17th day of June, 1976.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AN INTERIM POWER LIMIT OF 90% OF FULL POWER

FLORIDA POWER & LIGHT COMPANY

ST. LUCIE PLANT UNIT NO. 1

DOCKET NO. 50-335

INTRODUCTION

By letter dated April 27, 1976, and supplements dated April 30, May 14 and June 15, 1976, Florida Power & Light Company (FPL) requested an amendment to Facility License No. DPR-67 for the St. Lucie Plant Unit No. 1. The amendment request would add interim license requirements which limit power to 90% of rated power. We performed a preliminary review of FPL's analyses and proposed limitations included in their letters of April 27 and April 30, 1976. Based on that review, we issued Amendment No. 5 to License No. DPR-67 on April 30, 1976, which limited power to 60% of rated power until a more detailed review was completed.

We have now completed a more detailed review of the information submitted by the letters dated April 27 and 30, 1976. We have also reviewed some additional information submitted by letters dated May 14 and June 15, 1976, regarding the reduced flow ECCS performance analysis and the use of a calorimetric technique to obtain an independent check on the measured value of flow rate. In addition we were recently informed by Combustion Engineering that several errors were discovered in a computer code used in calculating the ECCS performance for St. Lucie Unit No. 1. The effect of these errors is considered in an Order for Modification of License issued to FPL dated June 17,1976, and is also considered in this evaluation.

EVALUATION

FSAR analysis, setpoint analyses, and Technical Specifications for St. Lucie Unit No. 1 were based on a reactor coolant flow rate of 370,000 gpm. However, hot functional test measurements have indicated that slightly less flow may exist.

To permit continuation of the plant power ascension program, FPL submitted by letters dated April 27 and 30, and May 14 and June 15, 1976, proposed interim limitations and supporting analyses for operation at 90% of rated power assuming a minimum reactor coolant flow rate of 354,000 gpm.

A. The DNB Safety Limit curves would be adjusted to maintain a limiting DNBR of 1.3 based upon the W-3 correlation and a flow rate of 354,000 gpm.

B. The limiting safety system settings would be changed to reflect the reduced reactor coolant flow rate at a power level of 90% of rated power.

C. The thermal margin low pressure trip setpoint equation would be modified in accordance with the reduced coolant flow.

D. The limiting conditions for operation on the axial shape index would be modified to maintain the same steady state operating margin to DNB.

In support of operation at 90% of full power, the licensee has submitted the results of analyses performed to ascertain the impact of the assumed reduced coolant flow on the LOCA and on other accidents and anticipated transients.

The assumed flow rate is approximately 6% less than the flow measured during Byron Jackson Tests (377,491 gpm) of the St. Lucie Unit No. 1 reactor coolant pumps and is conservative. LOCA analyses at the reduced flow rate (354,000 gpm) and at peak linear heat generation rates of 15.6 and 14.2 kW/ft resulted in maintaining peak clad temperature and clad oxidation values within acceptable limits. However, the NRC staff was recently informed by Combustion Engineering that several errors were discovered during a code audit of STRIKIN-2, a computer code which was used in calculating the ECCS performance of St. Lucie Unit No. 1. Subsequent calculations performed with a corrected version of STRIKIN-2 and for the previously determined worst break indicate that the peak linear heat generation rate should be reduced to 13.7 kW/ft. Since corrected calculations have only been performed for the previously determined worst break, an interim additional limitation of 1 kW/ft is being applied by FPL resulting in modified plant setpoints to limit peak linear heat generation rates to 12.7 kW/ft. The power limitation of 90% of full power and the peak linear heat generation rate limit of 12.7 kW/ft, confirmed in the Order for Modification of License issued to FPL dated June 17, 1976, provide sufficient margin to assure acceptable ECCS performance. In all other respects, operation of the facility at 90% of full power under the conditons described herein and in the Order for Modification of License, fully conform to the Commission's regulations. A reanalysis of the most limiting transient for DNBR, the loss of flow, was performed at the reduced coolant flow rate with a resulting minimum DNBR of 1.58. Thus, sufficient margin exists to the limiting value of 1.30 for the reduced flow condition. In addition, our independent evaluation indicates that a power reduction of less than 5% is required to maintain the same DNBR at the reduced flow rate as at the original value of 370,000 gpm. Therefore, a 10% power reduction will provide additional margin over that required. Based on the licensee's calculations and our evaluations, we conclude that power operation at 90% of rated power will provide safety margins to the limits associated with plant transient and LOCA response which are acceptable with peak linear heat generation rates limited to 12.7 kW/ft and with operations in accordance with the limitations proposed in FPL letter L-76-172 of April 27, 1976, as modified by FPL letter L-76-223 of June 15, 1976. The modification to the Technical Specifications reflect these limitations and are set forth in Attachment 1 hereto.

At our request, the licensee included in his May 14, 1976 submittal a reference for the detailed methodology for determining the reactor coolant flow rate. The reference, "Reactor Coolant Pump (RCP) Flow Test Report," letter from D. C. Switzer to USNRC, March 24, 1976, was submitted on the Millstone Unit 2 Docket No. 50-336, which is also applicable to St. Lucie Unit No. 1. The cited reference includes an error analysis of flow measurement uncertainty which confirms the validity of the 3.5% in flow measurement uncertainty assumed in the analyses and Technical Specifications for St. Lucie Unit No. 1. We conclude that this is acceptable. In addition, the May 14, 1976 submittal includes a brief description of the use of calorimetric techniques to obtain an independent check on the measured value of flow rate. The method utilizes easily measured parameters such as temperature and pressure to perform a heat balance which provides an independent estimate of reactor coolant flow rate. The information provided was mainly descriptive and did not include an error analysis or data demonstrating the accuracy of calorimetrics at different reactor power levels. While redundant flow measurement instrumentation will provide an acceptable means for reactor coolant flow rate determination, we will require that additional detail, including an error analysis, be provided for the technique of calorimetrics prior to allowing operation at 100% of full power. This information is necessary if credit is to be given for the use of calorimetric techniques as an independent check of measured flow rate. Because of the large margins of safety at 90% power, the additional information regarding the calorimetric techniques is not required prior to operation at 90% power.

ENVIRONMENTAL CONSIDERATION

We have determined that the amendment does not authorize a change in effluent types or total amounts nor an increase in power level and will not result in any significant environmental impact. Having made this determination, we have further concluded that the amendment involves an action which is insignificant from the standpoint of environmental impact and pursuant to 10 CFR §51.5(d)(4) that an environmental statement, negative declaration, or environmental impact appraisal need not be prepared in connection with the issuance of this amendment.

CONCLUSION

We have concluded, based on the considerations discussed above, that: (1) because the changes do not involve a significant increase in the probability or consequences of accidents previously considered and do not involve a significant decrease in a safety margin, the changes do 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.

Enclosure: Attachment I

Date: June 17, 1976

ATTACHMENT I
TO
SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION
SUPPORTING AN INTERIM POWER LIMIT OF 90% OF FULL POWER

ST. LUCIE - UNIT 1

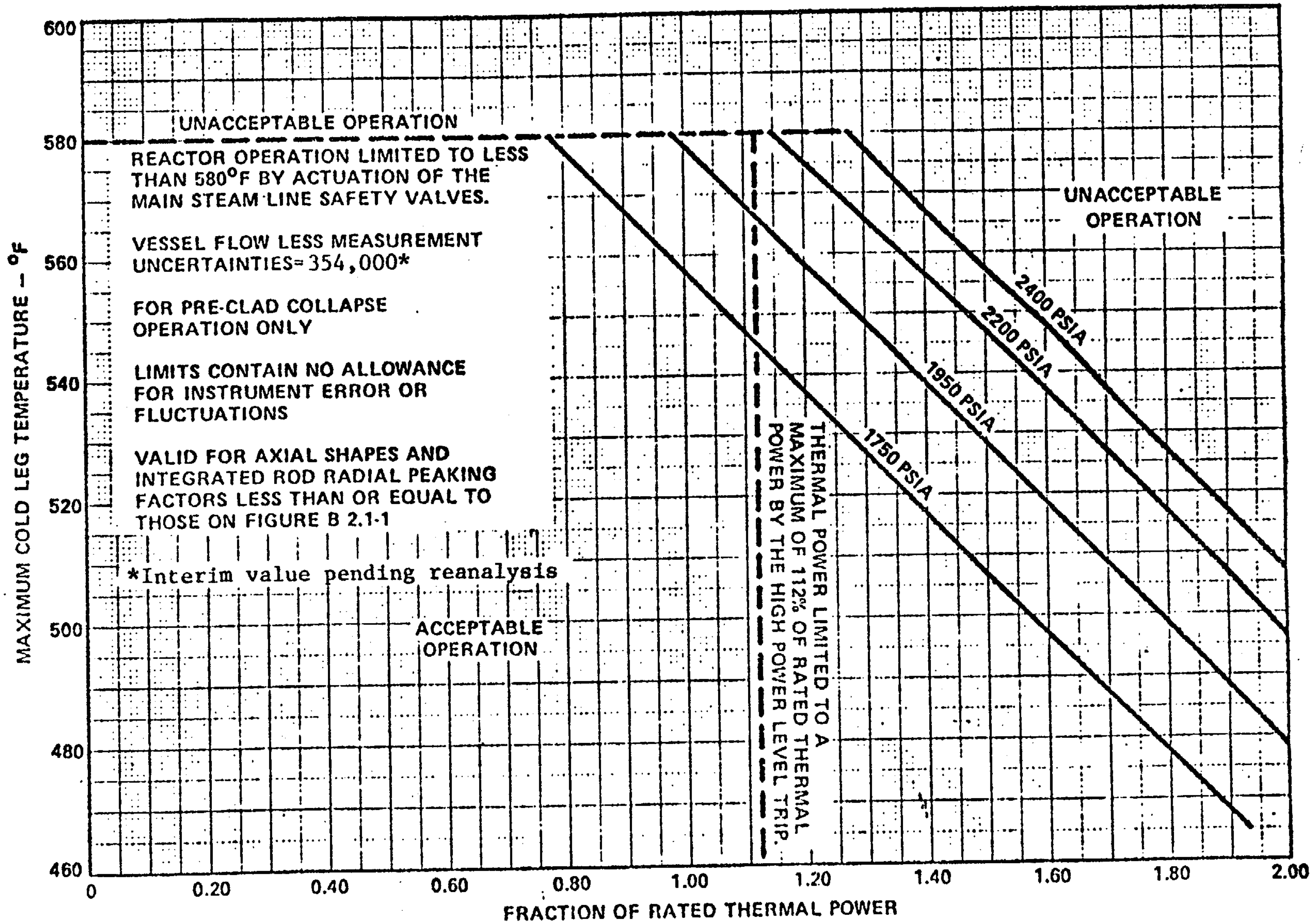

Figure 2.1-1 REACTOR CORE THERMAL MARGIN SAFETY LIMIT – FOUR REACTOR COOLANT PUMPS OPERATING

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	$\leq$ 9.61% above THERMAL POWER, with a minimum setpoint of 15% of RATED THERMAL POWER, and a maximum of $\leq$ 96.5% of RATED THERMAL POWER.	$\leq$ 9.61% above THERMAL POWER, and a minimum setpoint of 15% of RATED THERMAL POWER and a maximum of $\leq$ 96.5% of RATED THERMAL POWER.
3. Reactor Coolant Flow - Low (1)		
Four Reactor Coolant Pumps Operating	$\geq$ 95% of design reactor coolant flow with 4 pumps operating*	$\geq$ 95% of design reactor coolant flow with 4 pumps operating*
4. Pressurizer Pressure - High	$\leq$ 2400 psia	$\leq$ 2400 psia
5. Containment Pressure - High	$\leq$ 3.9 psig	$\leq$ 3.9 psig
6. Steam Generator Pressure - Low (2)	$\geq$ 485 psig	$\geq$ 485 psig
7. Steam Generator Water Level -Low	$\geq$ 36.3% Water Level - each steam generator	$\geq$ 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.

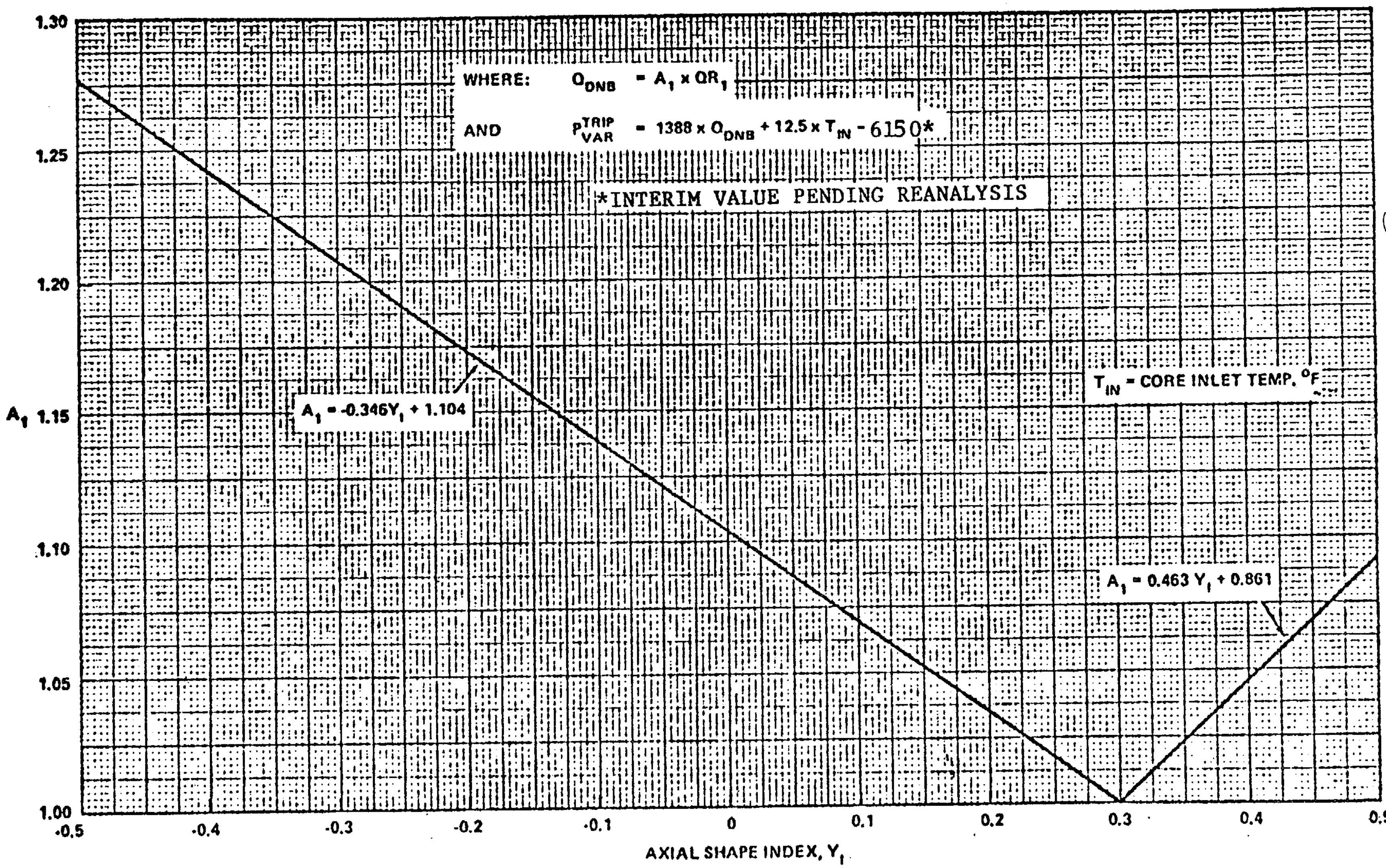

FIGURE 2.2-3

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

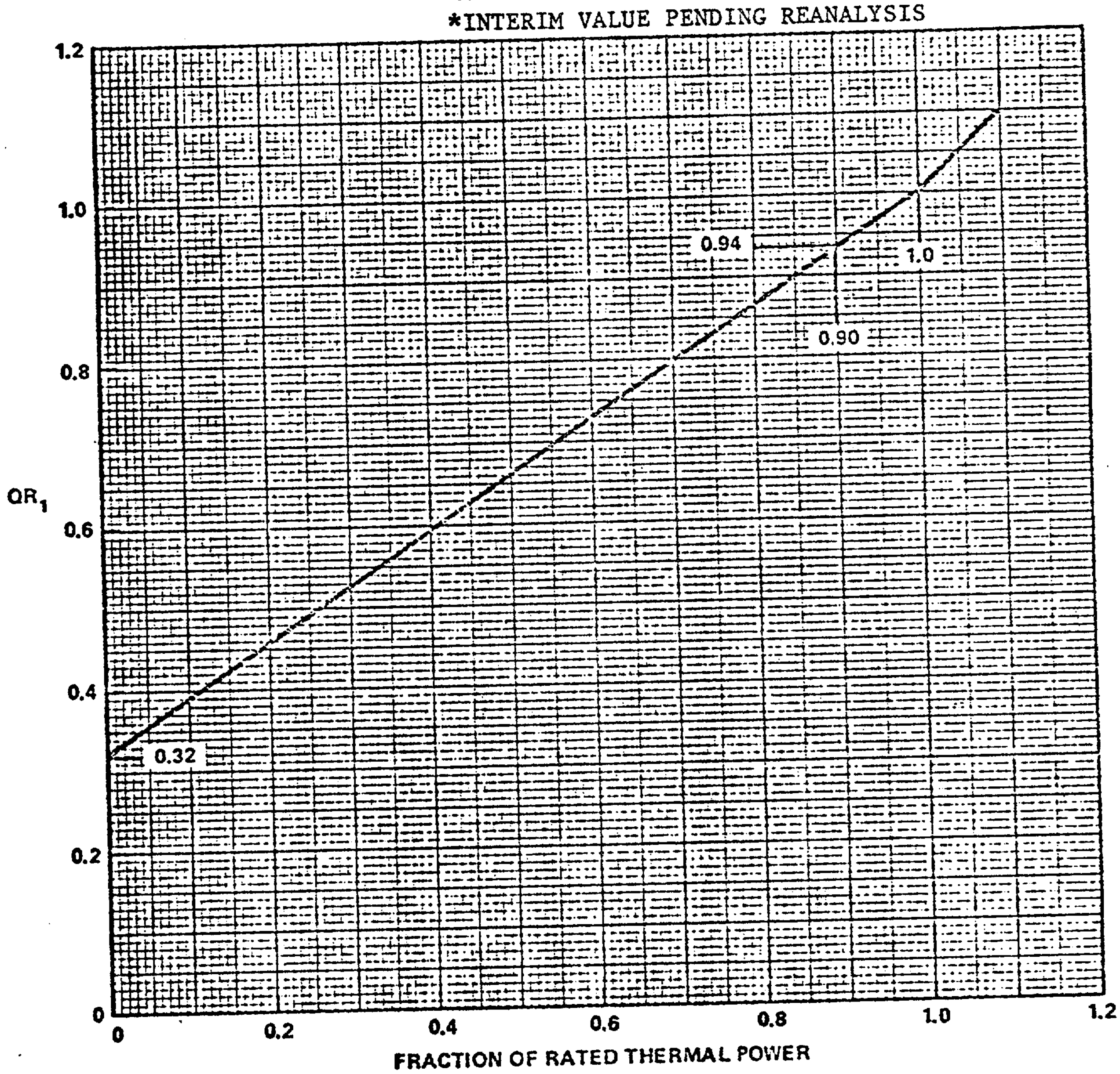

FIGURE 2.2-4

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

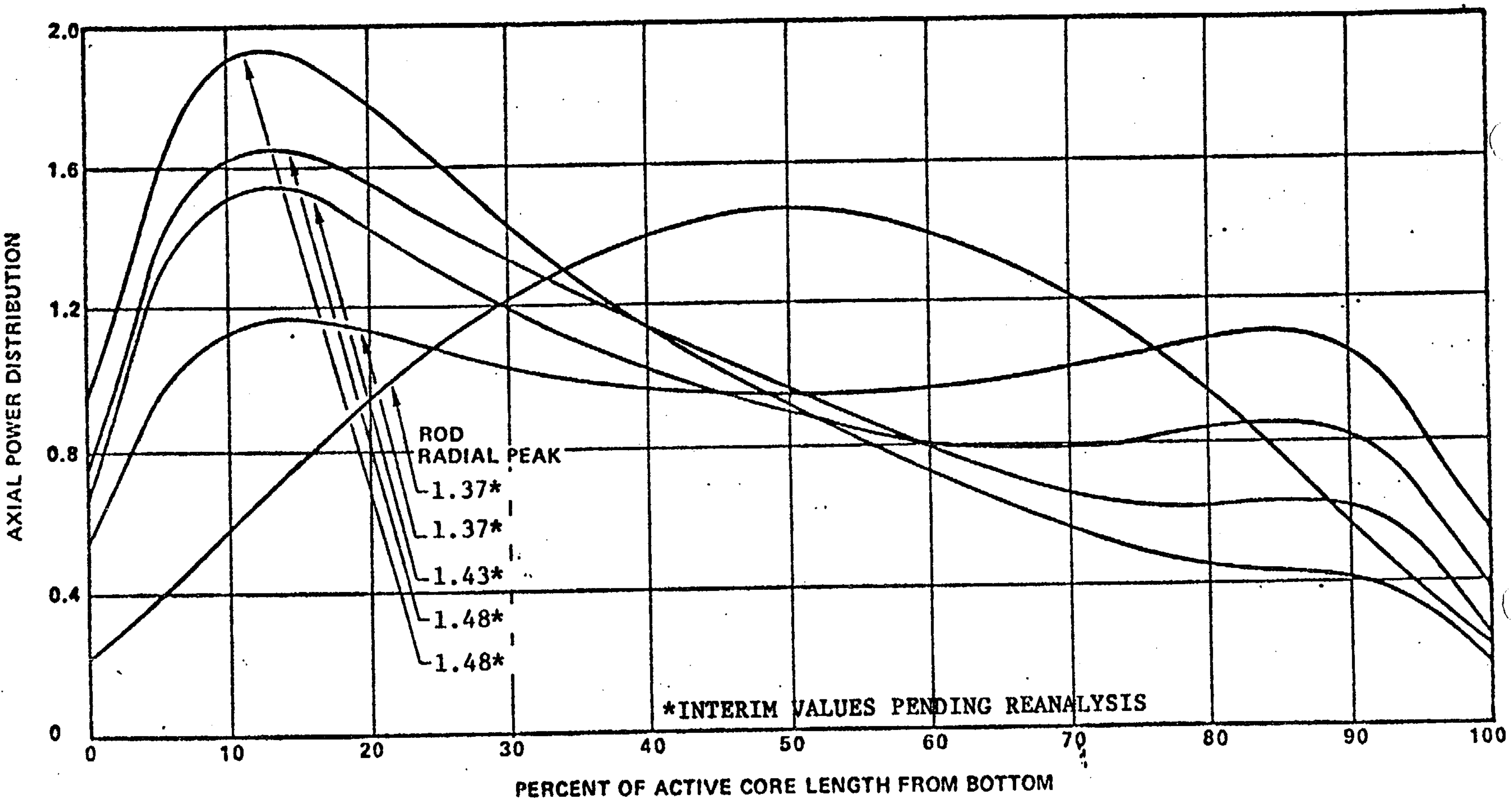

Figure B2.1-1 Axial Power Distribution for Thermal Margin Safety Limits

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.

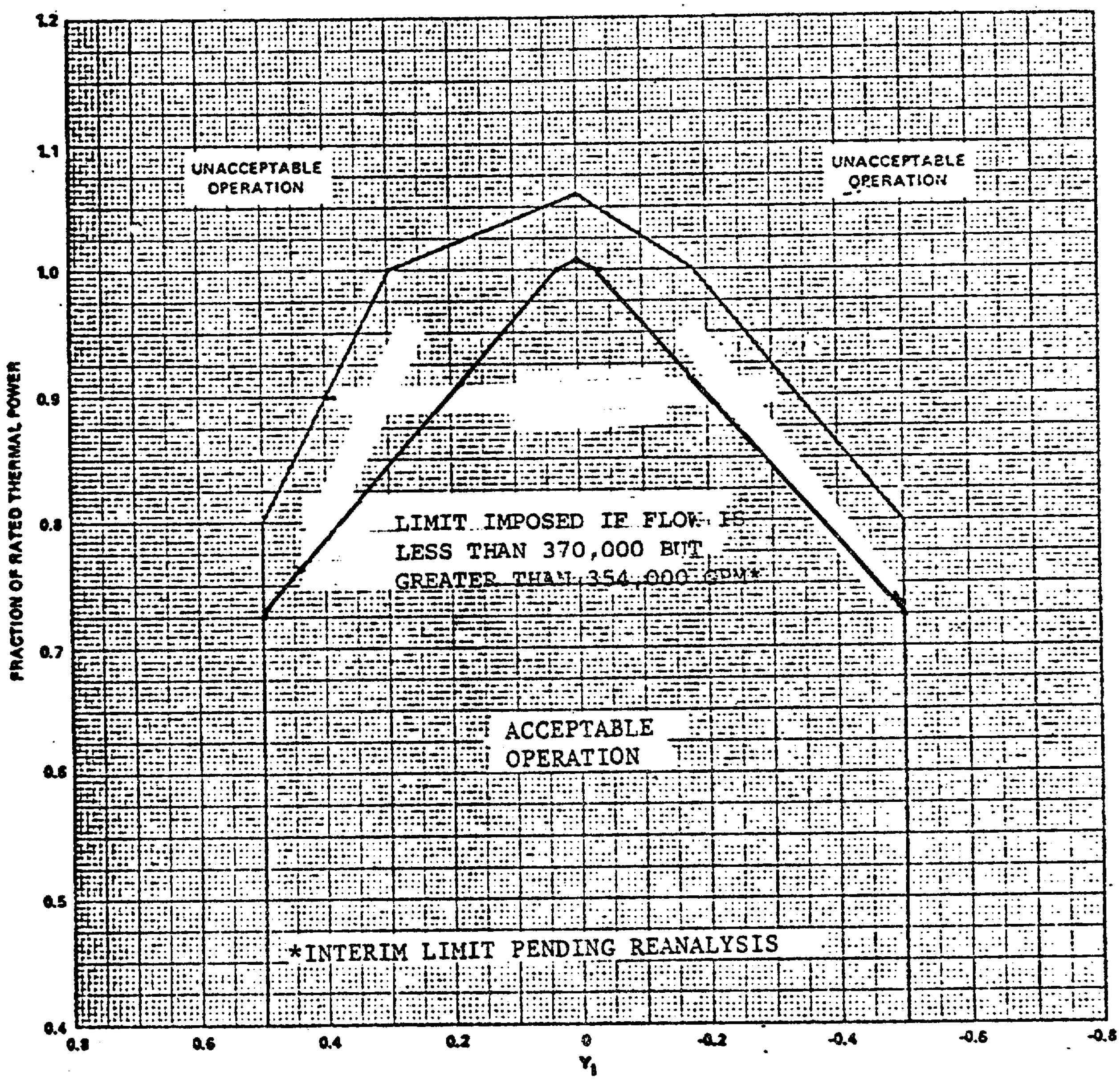

FIGURE 3.2-4
AXIAL SHAPE INDEX Operating Limits with 4 Reactor Coolant Pumps Operating