

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

PUBLIC SERVICE ELECTRIC AND GAS COMPANY  
PHILADELPHIA ELECTRIC COMPANY  
DELMARVA POWER AND LIGHT COMPANY  
ATLANTIC CITY ELECTRIC COMPANY

DOCKET NO. 50-272

SALEM NUCLEAR GENERATING STATION, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 52  
License No. DPR-70

1. The Nuclear Regulatory Commission (the Commission) has found that:
  - A. The application for amendment by Public Service Electric and Gas Company, Philadelphia Electric Company, Delmarva Power and Light Company and Atlantic City Electric Company (the licensees) dated October 5, 1982, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and 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 rules and 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;
  - 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.

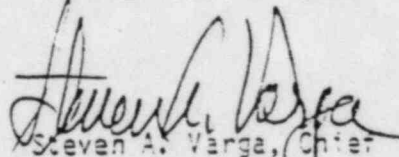
2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. DPR-70 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 52, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

  
Steven A. Varga, Chief  
Operating Reactors Branch #1  
Division of Licensing

Attachment:  
Changes to the Technical  
Specifications

Date of Issuance: May 5, 1983

ATTACHMENT TO LICENSE AMENDMENT NO. 52

FACILITY OPERATING LICENSE NO. DPR-70

DOCKET NO. 50-272

Revise Appendix A as follows:

Remove Pages

3/4 2-9

B 2-2

2-2

Insert Pages

3/4 2-9

B 2-2

2-2

## POWER DISTRIBUTION LIMITS

### NUCLEAR ENTHALPY HOT CHANNEL FACTOR - $F_{\Delta H}^N$

#### LIMITING CONDITION FOR OPERATION

3.2.3  $F_{\Delta H}^N$  shall be limited by the following relationship:

$$F_{\Delta H}^N \leq 1.55 [1.0 + 0.3 (1-P)] [1-RBP (BU)]$$

where:  $P = \frac{\text{THERMAL POWER}}{\text{RATED THERMAL POWER}}$ , and

RBP(BU) = Rod Bow Penalty as a function of region average burnup as shown in Figure 3.2-3, where a region is defined as those assemblies with the same loading date (reloads) or enrichment (first core).

APPLICABILITY: MODE 1

#### ACTION:

With  $F_{\Delta H}^N$  exceeding its limit:

- a. Reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within 2 hours and reduce the Power Range Neutron Flux-High Trip Setpoints to  $\leq 55\%$  of RATED THERMAL POWER within the next 4 hours,
- b. Demonstrate thru in-core mapping that  $F_{\Delta H}^N$  is within its limit within 24 hours after exceeding the limit or reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 2 hours, and
- c. Identify and correct the cause of the out of limit condition prior to increasing THERMAL POWER above the reduced limit required by a. or b. above; subsequent POWER OPERATION may proceed provided that  $F_{\Delta H}^N$  is demonstrated through in-core mapping to be within its limit at a nominal 50% of RATED THERMAL POWER prior to exceeding this THERMAL POWER, at a nominal 75% of RATED THERMAL POWER prior to exceeding this THERMAL power and within 24 hours after attaining 95% or greater RATED THERMAL POWER.

## SAFETY LIMITS

### BASES

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The curves are based on an enthalpy hot channel factor,  $F_{\Delta H}^N$ , of 1.55 and a reference cosine with a peak of 1.55 for axial power shape. An allowance is included for an increase in  $F_{\Delta H}^N$  at reduced power based on the expression:

$$F_{\Delta H}^N = 1.55 [1 + 0.3 (1-P)]$$

where P is the fraction of RATED THERMAL POWER

These limiting heat flux conditions are higher than those calculated for the range of all control rods fully withdrawn to the maximum allowable control rod insertion assuming the axial power imbalance is within the limits of the  $f_1(\Delta I)$  function of the Overtemperature trip. When the axial power imbalance is not within the tolerance, the axial power imbalance effect on the Overtemperature  $\Delta T$  trips will reduce the setpoints to provide protection consistent with core safety limits.

### 2.1.2 REACTOR COOLANT SYSTEM PRESSURE

The restriction of this Safety Limit protects the integrity of the Reactor Coolant System from overpressurization and thereby prevents the release of radionuclides contained in the reactor coolant from reaching the containment atmosphere.

The reactor pressure vessel and pressurizer are designed to Section III of the ASME Code for Nuclear Power Plant which permits a maximum transient pressure of 110% (2735 psig) of design pressure. The Reactor Coolant System piping and fittings are designed to ANSI B 31.1 1955 Edition while the valves are designed to ANSI B 16.5, MSS-SP-66-1964, or ASME Section III-1968, which permit maximum transient pressures of up to 120% (2985 psig) of component design pressure. The Safety Limit of 2735 psig is therefore consistent with the design criteria and associated code requirements.

The entire Reactor Coolant System is hydrotested at 3107 psig, 125% of design pressure, to demonstrate integrity prior to initial operation.



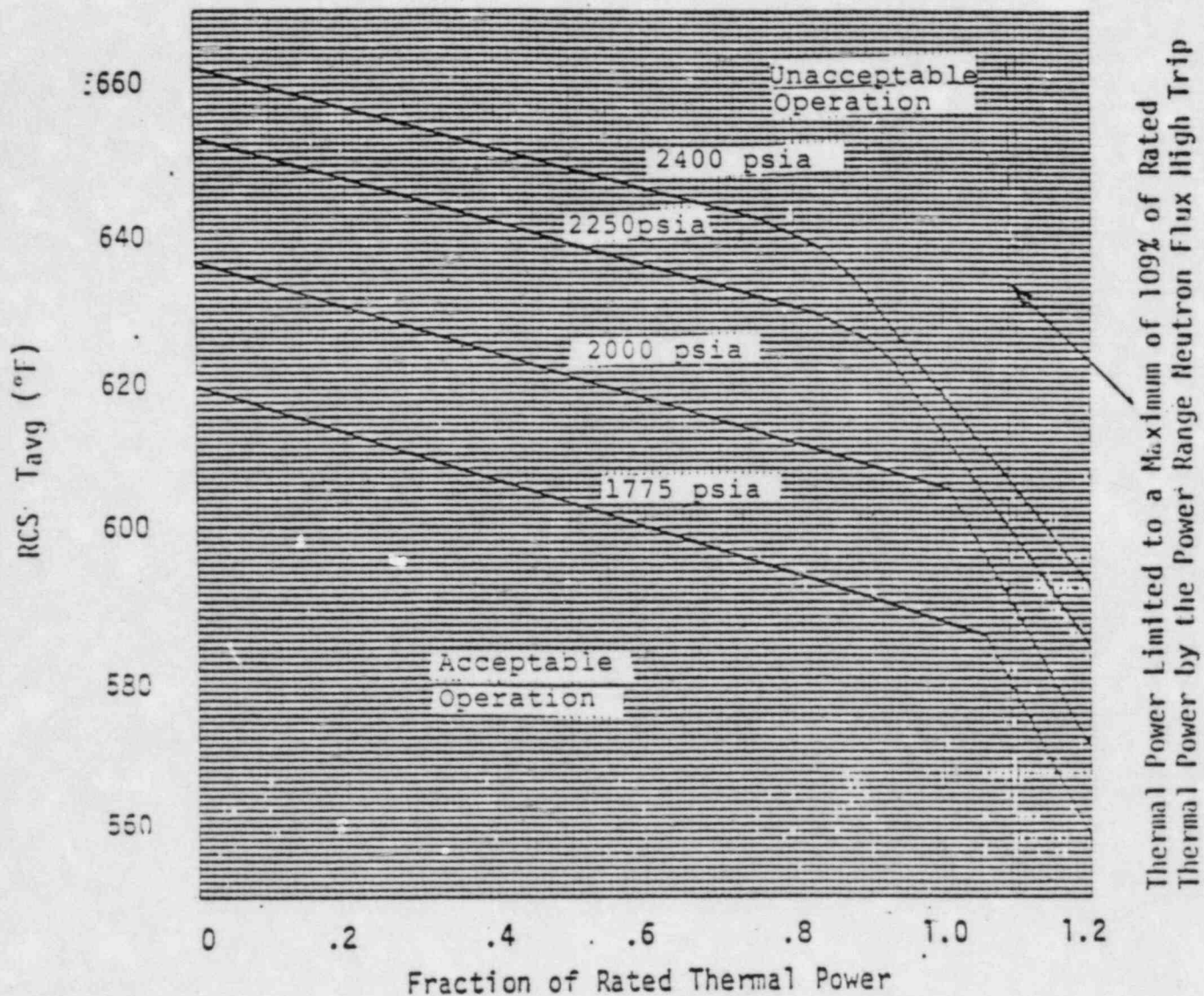


FIGURE 2.1-1 REACTOR CORE SAFETY LIMIT - FOUR LOOPS IN OPERATION



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DOCKET NO. 50-311

SALEM NUCLEAR GENERATING STATION, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 20  
License No. DPR-75

1. The Nuclear Regulatory Commission (the Commission) has found that:
  - A. The application for amendment by Public Service Electric and Gas Company, Philadelphia Electric Company, Delmarva Power and Light Company and Atlantic City Electric Company (the licensees) dated October 5, 1982, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and 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 rules and 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;
  - 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.

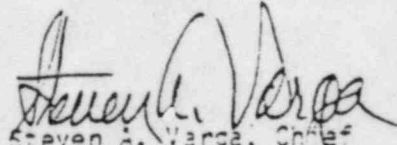
2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. DPR-75 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 20, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

  
Steven A. Varga, Chief  
Operating Reactors Branch #1  
Division of Licensing

Attachment:  
Changes to the Technical  
Specifications

Date of Issuance: May 5, 1983



ATTACHMENT TO LICENSE AMENDMENT NO. 20

FACILITY OPERATING LICENSE NO. DPR-75

DOCKET NO. 50-311

Revise Appendix A as follows:

Remove Pages

3/4 2-9

3/4 2-11

B 2-1

2-2

Insert Pages

3/4 2-9

3/4 2-11

B 2-1

2-2

## POWER DISTRIBUTION LIMITS

### 3/4.2.3 RCS FLOW RATE AND R

#### LIMITING CONDITION FOR OPERATION

3.2.3 The combination of indicated Reactor Coolant System (RCS) total flow rate and  $R_1$ ,  $R_2$  shall be maintained within the region of allowable operation shown on Figure 3.2-3 for 4 loop operation.

Where:

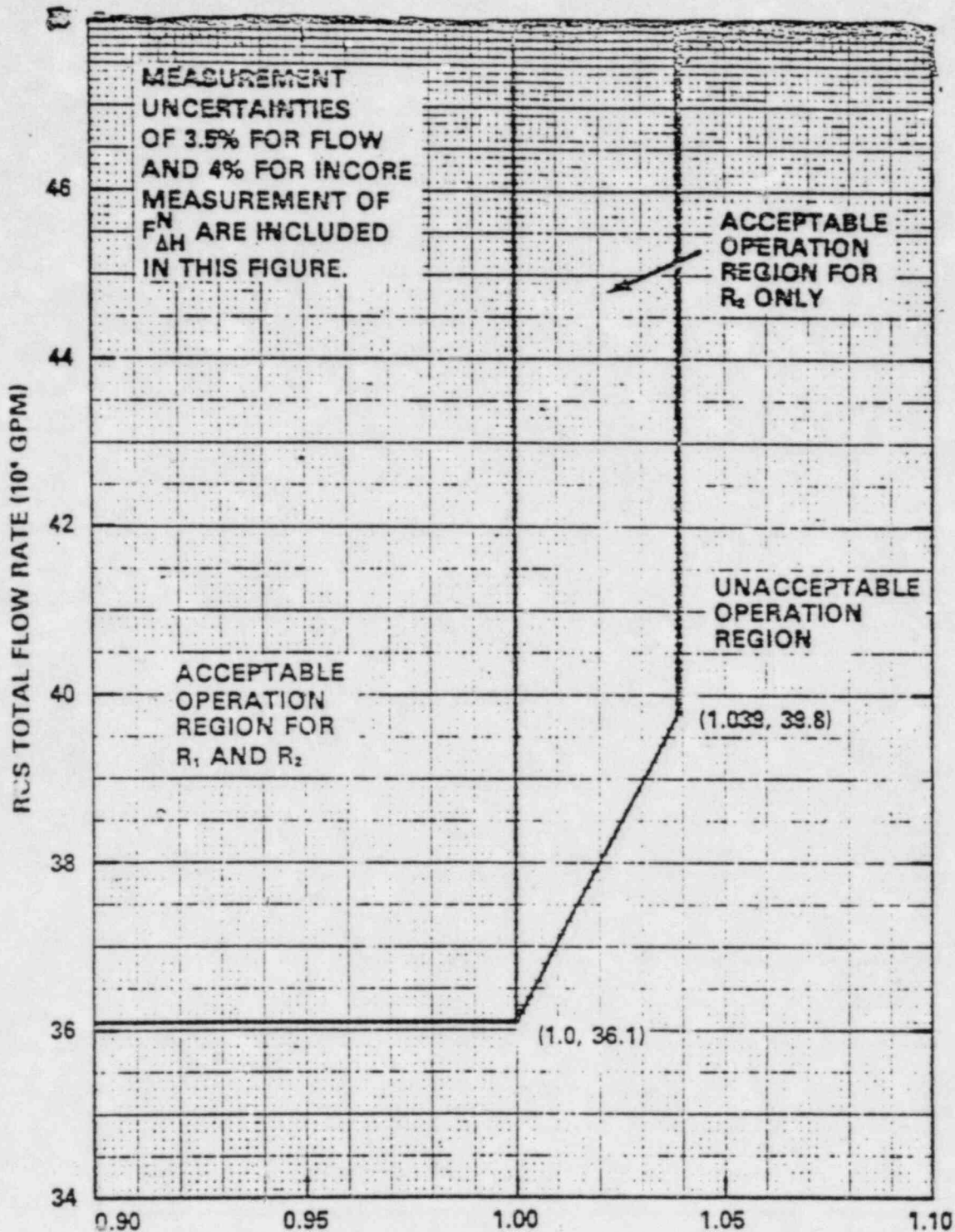
- a.  $R_1 = \frac{F_{\Delta H}^N}{1.49 [1.0 + 0.3 (1.0 - P)]}$ ,
- b.  $R_2 = \frac{R_1}{[1 - RBP(BU)]}$ ,
- c.  $P = \frac{\text{THERMAL POWER}}{\text{RATED THERMAL POWER}}$ , and
- d.  $F_{\Delta H}^N$  = Measured values of  $F_{\Delta H}^N$  obtained by using the movable incore detectors to obtain a power distribution map. The measured values of  $F_{\Delta H}^N$  shall be used to calculate R since Figure 3.2-3 includes measurement uncertainties of 3.5% for flow and 4% for incore measurement of  $F_{\Delta H}^N$ .
- e. RBP (BU) = Rod Bow Penalty as a function of region average burnup as shown in Figure 3.2-4, where a region is defined as those assemblies with the same loading date (reloads) or enrichment (first core).

APPLICABILITY: MODE 1.

#### ACTION:

With the combination of RCS total flow rate and  $R_1$ ,  $R_2$  outside the region of acceptable operation shown on Figure 3.2-3:

- a. Within 2 hours:
  - 1. Either restore the combination of RCS total flow rate and  $R_1$ ,  $R_2$  to within the above limits, or
  - 2. Reduce THERMAL POWER to less than 50% of RATED THERMAL POWER and reduce the Power Range Neutron Flux - High trip setpoint to less than or equal to 55% of RATED THERMAL POWER within the next 4 hours.



$$R_1 = F_{\Delta H}^N / 1.49 [1.0 + 0.3 (1.0 - P)]$$

$$R_2 = R_1 / [1 - RBP(BU)]$$

Figure 3.2-3  
RCS TOTAL FLOWRATE VERSUS R — FOUR LOOPS  
IN OPERATION

## 2.1 SAFETY LIMITS

### BASES

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#### 2.1.1 REACTOR CORE

The restrictions of this safety limit prevent overheating of the fuel and possible cladding perforation which would result in the release of fission products to the reactor coolant. Overheating of the fuel cladding is prevented by restricting fuel operation to within the nucleate boiling regime where the heat transfer coefficient is large and the cladding surface temperature is slightly above the coolant saturation temperature.

Operation above the upper boundary of the nucleate boiling regime could result in excessive cladding temperatures because of the onset of departure from nucleate boiling (DNB) and the resultant sharp reduction in heat transfer coefficient. DNB is not a directly measurable parameter during operation and therefore THERMAL POWER and Reactor Coolant Temperature and Pressure have been related to DNB through the W-3 correlation. The W-3 DNB correlation has been developed to predict the DNB flux and the location of DNB for axially uniform and non-uniform heat flux distributions. The local DNB heat flux ratio, DNBR, defined as the ratio of the heat flux that would cause DNB at a particular core location to the local heat flux, is indicative of the margin to DNB.

The minimum value of the DNBR during steady state operation, normal operational transients, and anticipated transients is limited to 1.30. This value corresponds to a 95 percent probability at a 95 percent confidence level that DNB will not occur and is chosen as an appropriate margin to DNB for all operating conditions.

The curves of Figures 2.1-1 and 2.1-2 show the loci of points of THERMAL POWER, Reactor Coolant System pressure and average temperature for which the minimum DNBR is no less than 1.30, or the average enthalpy at the vessel exit is equal to the enthalpy of saturated liquid.

The curves are based on an enthalpy hot channel factor,  $F_{\Delta H}^N$ , of 1.55 and a reference cosine with a peak of 1.55 for axial power shape. An allowance is included for an increase in  $F_{\Delta H}^N$  at reduced power based on the expression:

$$F_{\Delta H}^N = 1.55 [1 + 0.3(1-P)]$$

where P is the fraction of RATED THERMAL POWER

These limiting heat flux conditions are higher than those calculated for the range of all control rods fully withdrawn to the maximum allowable control rod insertion assuming the axial power imbalance is within the limits of the  $f_1(\Delta I)$  function of the Overtemperature trip. When the axial power



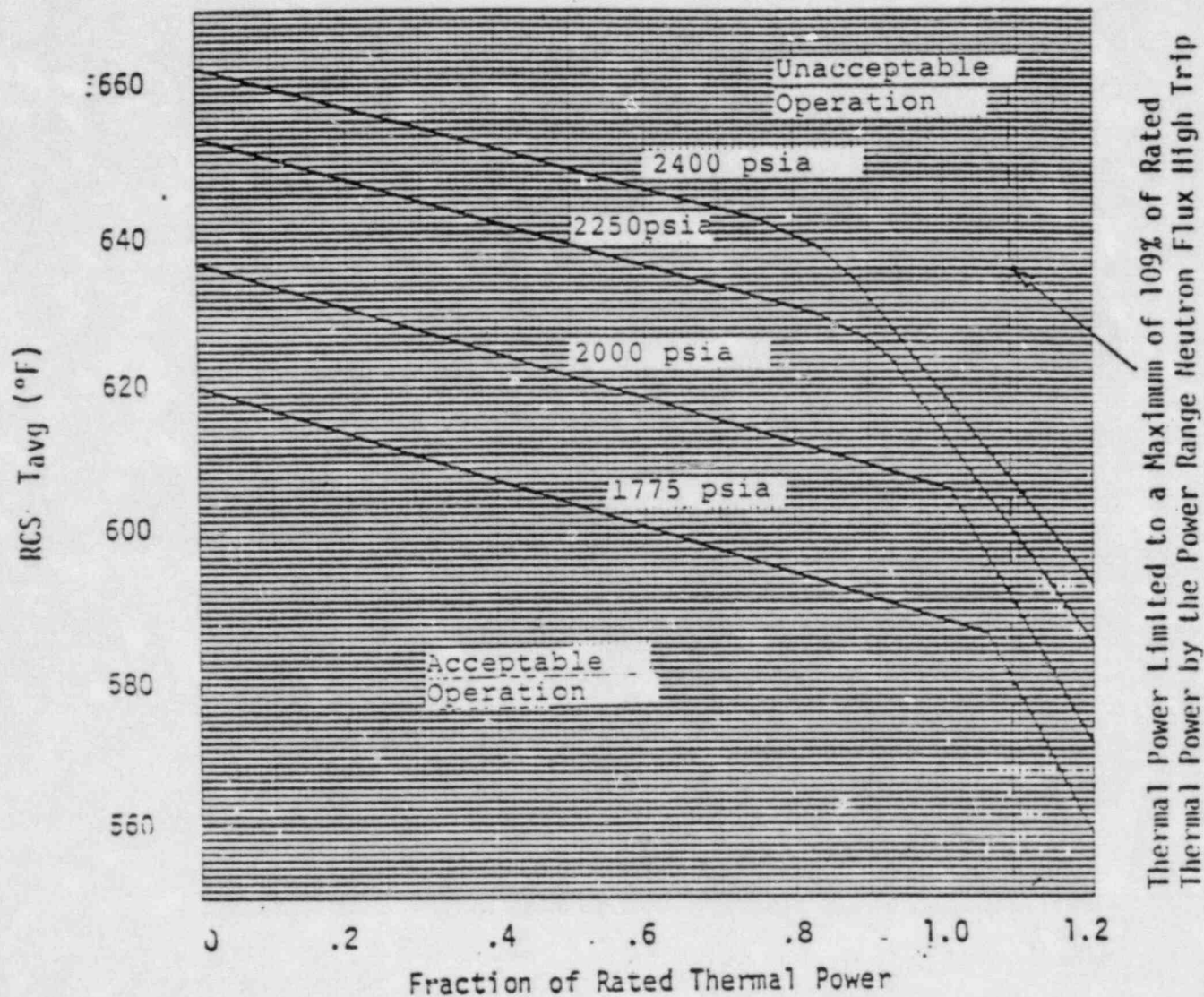


FIGURE 2.1-1 REACTOR CORE SAFETY LIMIT - FOUR LOOPS IN OPERATION