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

PORTLAND GENERAL ELECTRIC COMPANY

THE CITY OF EUGENE, OREGON

PACIFIC POWER AND LIGHT COMPANY

DOCKET NO. 50-344

TROJAN NUCLEAR PLANT

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 76
License No. NPF-1

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

   A. The application for amendment by Portland General Electric Company, the City of Eugene, Oregon, and Pacific Power and Light Company (the licensee) dated May 10, 1982, as supplemented July 7 and 29, 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.

DESIGNATED ORIGINAL

Certified By Patricia J Noonan

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. NPF-1 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 76 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications, except where otherwise stated in specific license conditions.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Robert A. Clark, Chief
Operating Reactors Branch #3
Division of Licensing

Attachment:
Changes to the Technical Specifications

Date of Issuance: August 13, 1982

ATTACHMENT TO LICENSE AMENDMENT

AMENDMENT NO. 76 TO FACILITY OPERATING LICENSE NO. NPF-1

DOCKET NO. 50-344

Revise Appendix A as follows:

Remove Pages	Insert Pages
XVII	XVII
2-2	2-2
B 2-2	B 2-2
3/4 1-5	3/4 1-5
3/4 2-6a	3/4 2-6a
3/4 2-8	3/4 2-8
3/4 2-9	3/4 2-9
3/4 2-9a	3/4 2-9a
3/4 2-9b	3/4 2-9b
B 3/4 2-4	B 3/4 2-4
6-18	6-18
-	6-18a

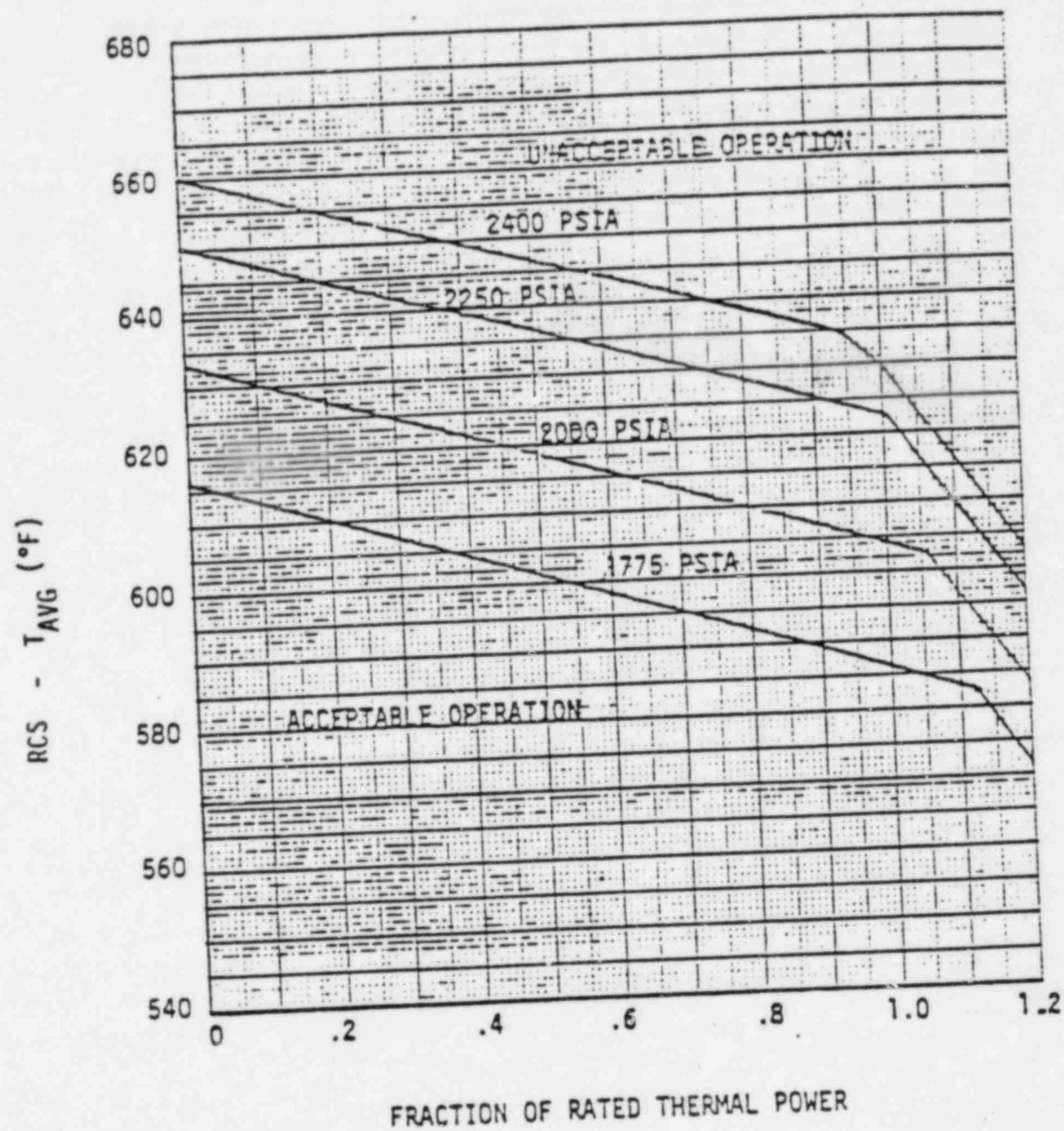

FIGURE 2.1-1 REACTOR CORE SAFETY LIMIT - FOUR LOOPS IN OPERATION

The curves are based on an enthalpy hot channel factor, $F^N_{\Delta H}$, of 1.49 and a reference cosine with a peak of 1.55 for axial power shape. An allowance is included for an increase in $F^N_{\Delta H}$ at reduced power based on the expression:

$$F^N_{\Delta H} = 1.49\ [1 + 0.3\ (1-P)]$$

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(ΔI) function of the Overtemperature ΔT trip. When the axial power imbalance is not within the tolerance, the axial power imbalance effect on the Overtemperature ΔT 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, valves and fittings, are designed to ANSI B 31.7-1969, which permits a maximum transient pressure of 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 to demonstrate integrity prior to initial operation.

REACTIVITY CONTROL SYSTEMS

MODERATOR TEMPERATURE COEFFICIENT

LIMITING CONDITION FOR OPERATION

3.1.1.4 The moderator temperature coefficient (MTC) shall be:

a. $\leq 0.5 \times 10^{-4}$ $\Delta k/k/°F$ below 70 percent RATED THERMAL POWER.

$\leq 0.0 \times 10^{-4}$ $\Delta k/k/°F$ at or above 70 percent RATED THERMAL POWER.

b. Less negative than -5.36×10^{-4} $\Delta k/k/°F$ at RATED THERMAL POWER.

APPLICABILITY: MODES 1 and 2*#

ACTION:

With the moderator temperature coefficient outside any one of the above limits, be in HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

4.1.1.4.1 The MTC shall be determined to be within its limits by confirmatory measurements. MTC measured values shall be extrapolated and/or compensated to permit direct comparison with the above limits.

4.1.1.4.2 The MTC shall be determined at the following frequencies and THERMAL POWER conditions during each fuel cycle:

a. Prior to initial operation above 5% of RATED THERMAL POWER, after each fuel loading.

b. At any THERMAL POWER, within 7 EFPD after reaching a RATED THERMAL POWER equilibrium boron concentration of 300 ppm.

*With $K_{eff} \geq 1.0$.

#See Special Test Exception 3.10.4.

b) At least once per 31 EFPD, whichever occurs first.

2. When the F_{xy}^{C} is less than or equal to the F_{xy}^{RTP} limit for the appropriate measured core plane, additional power distribution maps shall be taken and F_{xy}^{C} compared to F_{xy}^{RTP} and F_{xy}^{L} at least once per 31 EFPD.

e. Changes in the F_{xy} limits for RATED THERMAL POWER (F_{xy}^{RTP}) shall be provided for all core planes containing bank "D" control rods and all unrodded core planes in a Radial Peaking Factor Limit Report per Specification 6.9.1.10.

f. The F_{xy} limits of e, above, are not applicable in the following core plane regions as measured in percent of core height from the bottom of the fuel:

1. Lower core region from 0 to 15%, inclusive.
2. Upper core region from 85 to 100% inclusive.
3. Grid plane regions at 17.8 $\pm$ 2%, 32.1 $\pm$ 2%, 47.4 $\pm$ 2%, 60.6 $\pm$ 2% and 74.9 $\pm$ 2%, inclusive.
4. Core plane regions within $\pm$ 2% of core height ($\pm$ 2.88 inches) about the bank demand position of the bank "D" rods.

g. Evaluating the effects of F_{xy} on $F_Q(Z)$ to determine if $F_Q(Z)$ is within its limit whenever F_{xy}^{C} exceeds F_{xy}^{L}.

4.2.2.3 $F_Q(Z)$ shall be measured at least once per 31 EFPD. When $F_Q(Z)$ is measured, an overall measured value shall be obtained from a power distribution map and increased by 3% to account for manufacturing tolerances and further increased by 5% to account for measurement uncertainty.

RCS FLOWRATE AND F_R

LIMITING CONDITION FOR OPERATION

3.2.3 The combination of indicated Reactor Coolant System (RCS) total flow rate and F_R shall be maintained within the region of allowable operation (above and to the left of the line) shown on Figures 3.2-3 and 3.2-4 for 4- and 3-loop operation, respectively.

Where:

a. $F_R = \dfrac{F^N_{\Delta H}}{1.49\ \{1.0 + 0.3\ (1.0 - P)\}}$, and

b. $P = \dfrac{\text{THERMAL POWER}}{\text{RATED THERMAL POWER}}$.

APPLICABILITY: MODE 1

ACTION:

With the combination of RCS total flow rate and F_R outside the region of acceptable operation shown on Figure 3.2-3 or 3.2-4 (as applicable):

a. Within 2 hours:

1. Either restore the combination of RCS flow rate and F_R 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 $\leq$ 55% of RATED THERMAL POWER within the next 4 hours.

b. Within 24 hours of initially being outside the above limits, verify through incore flux mapping and RCS total flow rate comparison that the combination of F_R and RCS total flow rate are restored to within the above limits, or reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 2 hours.

ACTION: (Continued)

c. Identify and correct the cause of the out-of-limit condition prior to increasing THERMAL POWER above the reduced THERMAL POWER limit required by ACTION items a.2 and/or b above; subsequent POWER OPERATION may proceed provided that the combination of R and indicated RCS total flow rate are demonstrated, through incore flux mapping and RCS total flow rate comparison, to be within the region of acceptable operation shown on Figure 3.2-3 or 3.2-4 (as applicable) prior to exceeding the following THERMAL POWER levels:

1. A nominal 50% of RATED THERMAL POWER,

2. A nominal 75% of RATED THERMAL POWER, and

3. Within 24 hours of attaining $\geq$ 95% of RATED THERMAL POWER.

SURVEILLANCE REQUIREMENTS

4.2.3.1 The provisions of Specification 4.0.4 are not applicable.

4.2.3.2 The combination of indicated RCS total flow rate and F_R shall be determined to be within the region of acceptable operation of Figure 3.2-3 or 3.2-4 (as applicable):

a. Prior to operation above 75% of RATED THERMAL POWER after each fuel loading, and

b. At least once per 31 Effective Full Power Days.

Where:

$$F_R = \frac{F^N_{\Delta H}}{1.49\ \{1.0 + 0.3\ (1.0 - P)\}}, \text{ and}$$

$F^N_{\Delta H}$ = Measured values of $F^N_{\Delta H}$ obtained by using the movable incore detectors to obtain a power distribution map. The measured values of $F^N_{\Delta H}$ shall be used to calculate F_R since Figures 3.2-3 and 3.2-4 include measurement calculational uncertainties of 3.5% for flow and 4% for incore measurement of $F^N_{\Delta H}$.

4.2.3.3 The RCS total flow rate indicators shall be subjected to a CHANNEL CALIBRATION at least once per 18 months.

4.2.3.4 The RCS total flow rate shall be determined by measurement at least once per 18 months.

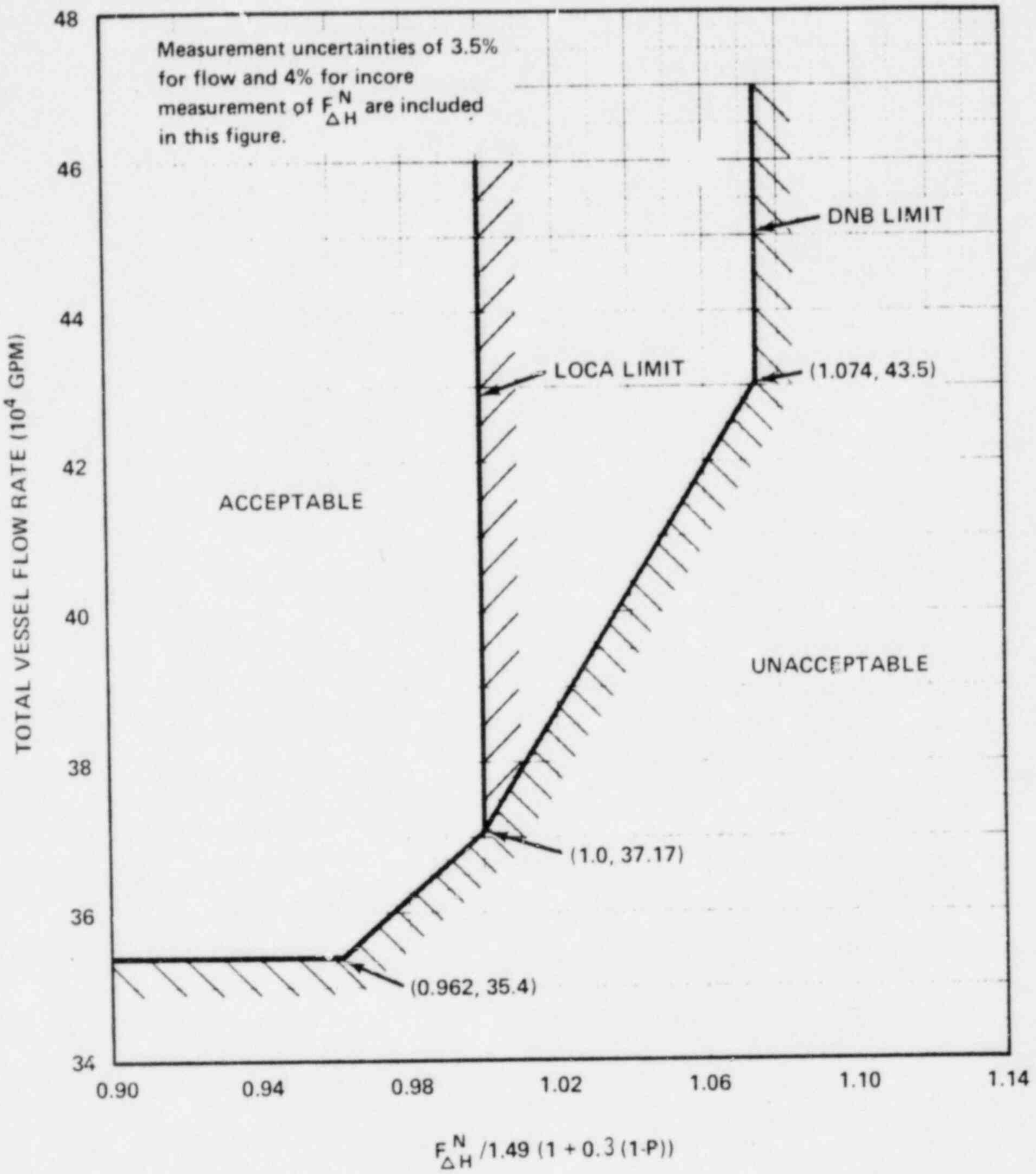

Figure 3.2-3 Flow vs. $F_{\Delta H}^{N}$ Limit for 4 Loops in Operation.

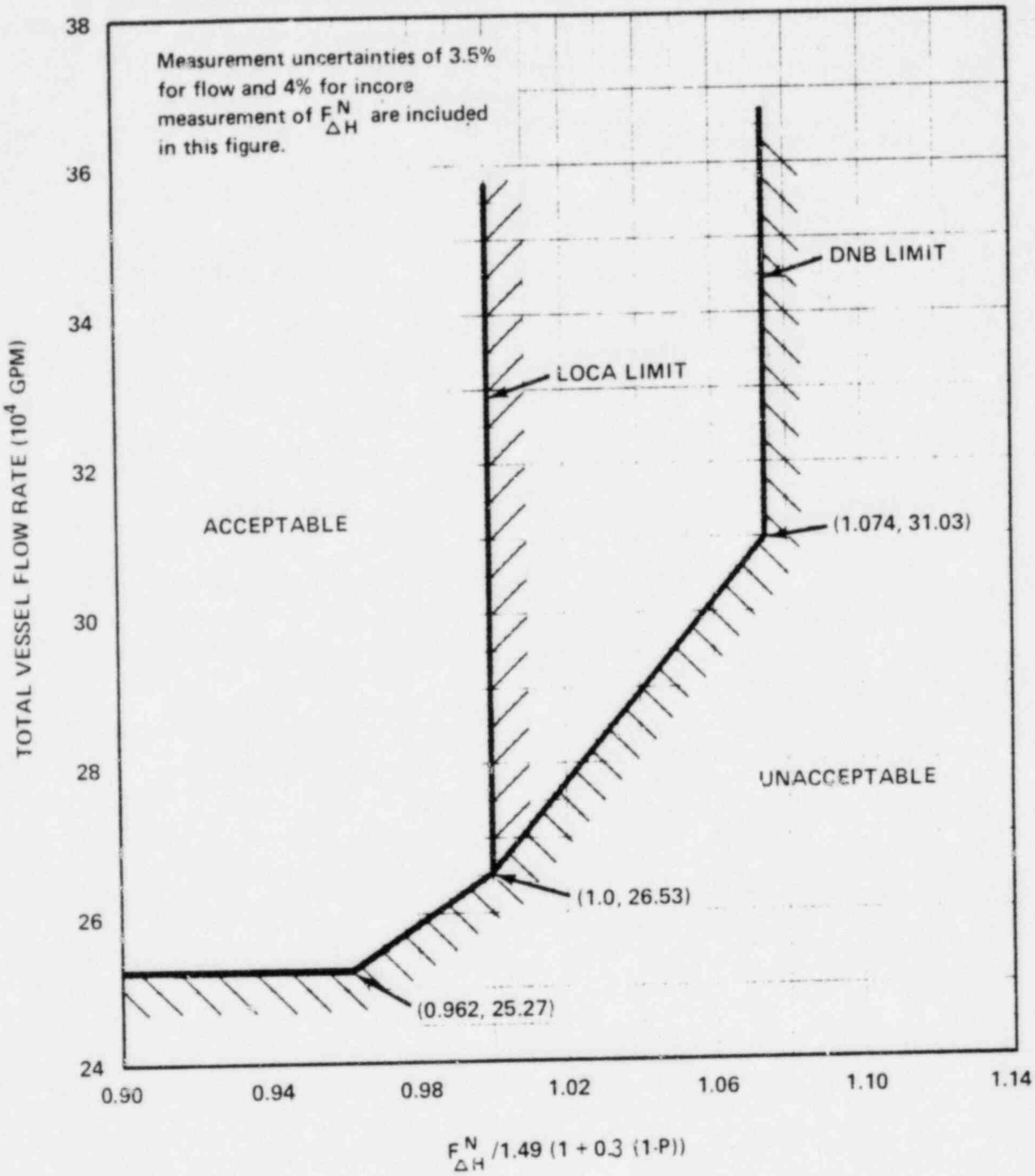

Figure 3.2-4 Flow vs. $F^N_{\Delta H}$ Limit for 3 Loops in Operation.

3/4.2.2 and 3/4.2.3 HEAT FLUX HOT CHANNEL FACTOR, RCS FLOWRATE, AND NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR

The limits on heat flux hot channel factor, RCS flowrate, and nuclear enthalpy rise hot channel factor ensure that 1) the design limits on peak local power density and minimum DNBR are not exceeded and 2) in the event of a LOCA the peak fuel clad temperature will not exceed the 2200°F ECCS acceptance criteria limit.

Each of these is measurable but will normally only be determined periodically as specified in Specifications 4.2.2 and 4.2.3. This periodic surveillance is sufficient to insure that the limits are maintained provided:

a. Control rods in a single group move together with no individual rod insertion differing by more than ± 12 steps from the group demand position.

b. Control rod groups are sequenced with overlapping groups as described in Specification 3.1.3.5.

c. The control rod insertion limits of Specification 3.1.3.5 are maintained.

d. The axial power distribution, expressed in terms of AXIAL FLUX DIFFERENCE, is maintained within the limits.

$F^N_{\Delta H}$ will be maintained within its limits provided conditions a. through d. above are maintained. As noted on Figures 3.2-3 and 3.2-4, RCS flow rate and $F^N_{\Delta H}$ may be "traded off" against one another (i.e., a low measured RCS flow rate is acceptable if the measured $F^N_{\Delta H}$ is also low) to ensure that the calculated DNBR will not be below the design DNBR value. This tradeoff is allowed up to a maximum $F^N_{\Delta H}$ of 1.49 (1+0.3(1-P)) which is consistent with the initial conditions assumed for the LOCA analysis. The relaxation of $F^N_{\Delta H}$ as a function of THERMAL POWER allows changes in the radial power shape for all permissible rod insertion limits.

When an F_Q measurement is taken, both experimental error and manufacturing tolerance must be allowed for. 5% is the appropriate allowance for a full core map taken with the incore detector flux mapping system and 3% is the appropriate allowance for manufacturing tolerance. Application of these two penalties in a multiplication fashion is sufficient to provide a correction for the effect of rod bow on F_Q, which has been conservatively estimated as 5% in WCAP-8692, "Fuel Rod Bowing". The appropriate statistical combination of local power, manufacturing tolerance and rod bow uncertainties, results in a penalty on F_Q of 7.68%, whereas multiplying measured values of F_Q by 1.03 x 1.05 results in a penalty of 8.15%.

additional narrative material to provide complete explanation of the circumstances surrounding the event.

a. Reactor protection system or engineered safety feature instrument settings which are found to be less conservative than those established by the technical specifications but which do not prevent the fulfillment of the functional requirements of affected systems.

b. Conditions leading to operation in a degraded mode permitted by a limiting condition for operation or plant shutdown required by a limiting condition for operation.

c. Observed inadequacies in the implementation of administrative or procedural controls which threaten to cause reduction of degree of redundancy provided in reactor protection systems or engineered safety feature systems.

d. Abnormal degradation of systems other than those specified in 6.9.1.8.c above designed to contain radioactive material resulting from the fission process.

RADIAL PEAKING FACTOR LIMIT REPORT

6.9.1.10 The F_{xy} limits for Rated Thermal Power (F_{xy}^{RTP}) for all core planes containing bank "D" control rods and all unrodded core planes and the plot of predicted ($F_q^T \cdot P_{Rel}$) vs Axial Core Height with the limit envelope shall be provided to the NRC Regional Administrator with a copy to Director of Nuclear Reactor Regulation, Attention: Chief, Core Performance Branch, U. S. Nuclear Regulatory Commission, Washington, D. C. 20555 at least 60 days prior to each cycle initial criticality unless otherwise approved by the Commission by letter.

In addition, in the event that the limit should change requiring a new submittal or an amended submittal to the Peaking Factor Limit Report, it will be submitted 60 days prior to the date the limit would become effective unless otherwise approved by the Commission by letter.

Any information needed to support F_{xy}^{RTP} will be by request from the NRC and need not be included in this report.

SPECIAL REPORTS

6.9.2 Special reports shall be submitted to the Director of the Office of Inspection and Enforcement Regional Office within the time period specified for each report. These reports shall be submitted covering the activities identified below pursuant to the requirements of the applicable reference specification:

a. Inoperable Seismic Monitoring Instrumentation, Specification 3.3.3.3.

b. Inoperable Meteorological Monitoring Instrumentation, Specification 3.3.3.4.

c. Inservice Inspection Program Reviews, Specifications 4.4.10.1 and 4.4.10.2.

d. ECCS Actuation, Specifications 3.5.2 and 3.5.3.

e. Sealed Source Leakage in excess of limits, Specification 4.7.7.1.3.

f. Seismic Event Analysis, Specification 4.3.3.3.2.

g. Fire Detection Instrumentation, Specification 3.3.3.7.

h. Fire Suppression Systems, Specifications 3.7.8.1 and 3.7.8.2.

i. Accident Monitoring Instrumentation, Specification 3.3.3.9.

j. Control Building Modification Connection Bolts, Specifications 3.7.11 and 4.7.11.1.