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

INDIANA AND MICHIGAN ELECTRIC COMPANY

DOCKET NO. 50-316

DONALD C. COOK NUCLEAR PLANT UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 48
License No. DPR-74

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

   A. The application for amendment by Indiana and Michigan Electric Company (the licensee) dated April 7, 1982, as supplemented by letters dated June 11 and June 30, 1982, July 8, 1982, September 30, 1982, December 9 and December 22, 1982 and January 12, 1983, 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-74 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 48 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. The license is also amended by the addition of paragraph 2.C.3 (p) to Facility Operating License No. DPR-74 to read as follows:

   "Operation during Cycle 4 with Exxon Nuclear Company 17x17 fuel assemblies is permitted subject to:

   (1) the satisfactory completion by the licensee of the following activities on or before the times indicated:

      i. Complete and submit an analysis within one year from the issuance of this amendment using NRC approved methodology to comply with fuel assembly structural acceptance criteria in Appendix A to SRP-4.2 for the design seismic event.

      ii. Continue to comply with the operating restrictions imposed by the rod drop accident analysis until such time as the generic review of this event has been completed and any analyses required as a result of that review are performed.

      iii. Following NRC approval of the RODEX 2 thermal analysis code, and prior to 10,000 MWD/MTU average fuel assembly burnup of the ENC 17x17 fuel assemblies during Cycle 4 operation, resubmit the cladding strain, oxidation, and pellet/cladding interaction calculations with an approved version of the RODEX 2 code, and

   (2) the following conditions pending receipt and approval of confirmatory and other information on transients and accidents as noted in the Safety Evaluation and Environmental Impact (Report) issued with Amendment No. 48 :

      i. The PTS-PWR2 model, and its adjunct thermal-hydraulic models, cannot be used by the licensee to justify changes to the set points and related uncertainties, and instrumentation response and delay time, for Reactor Protection System (RPS) and Engineered Safeguards Features (ESF) initiation and actuation functions.

ii. The maximum value of $F_Q(Z)$ for the reactor core is to be limited to a maximum value of 2.04 irrespective of any subsequent changes to this value permitted by revisions to LOCA calculations.

iii. No change is allowable to the current Technical Specifications in respect of moderator temperature coefficients.

In addition to the conditions set forth above, the licensee is not authorized to operate in Cycle 5, modes 1 and 2, until it has satisfactorily resolved the issues identified in the Safety Evaluation and Environmental Impact Appraisal (Report) issued with Amendment No. 48 and other Cycle 5 regulatory requirements."

4. Within 30 days after the effective date of this amendment, or such other time as the Commission may specify, the licensee shall satisfy any applicable requirement of P.L. 97-425 related to pursuing an agreement with the Secretary of Energy for the disposal of high-level radioactive waste and spent nuclear fuel.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Gus C. Lainas, Assistant Director
for Operating Reactors
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: January 14, 1983

ATTACHMENT TO LICENSE AMENDMENT

AMENDMENT NO. 48 TO FACILITY OPERATING LICENSE NO. DPR-74

DOCKET NO. 50-316

Revise Appendix A as follows:

Remove Pages	Insert Pages
1-1	1-1
2-7 & 2-8	2-7 & 2-8
2-9	2-9
3/4 2-5 thru 3/4 2-8a	3/4 2-5 thru 3/4 2-8a
---------	3/4 2-8b
3/4 2-9 thru 3/4 2-12	3/4 2-9 thru 3/4 2-12
3/4 2-17	3/4 2-17
3/4 2-18	3/4 2-18
3/4 2-19	3/4 2-19
B2-1 & B2-2	B2-1 & B2-2
B3/4 2-1	B3/4 2-1
B3/4 2-2	B3/4 2-2
B3/4 2-4	B3/4 2-4
2-1 & 2-2	2-1 & 2-2
2-3 & 2-4	2-3 & 2-4

1.0 DEFINITIONS

DEFINED TERMS

1.1 The DEFINED TERMS of this section appear in capitalized type and are applicable throughout these Technical Specifications.

THERMAL POWER

1.2 THERMAL POWER shall be the total reactor core heat transfer rate to the reactor coolant.

RATED THERMAL POWER

1.3 RATED THERMAL POWER shall be a total reactor core heat transfer rate to the reactor coolant of 3411 MWt.

OPERATIONAL MODE

1.4 An OPERATIONAL MODE shall correspond to any one inclusive combination of core reactivity condition, power level and average reactor coolant temperature specified in Table 1.1.

ACTION

1.5 ACTION shall be those additional requirements specified as corollary statements to each principle specification and shall be part of the specifications.

OPERABLE - OPERABILITY

1.6 A system, subsystem, train, component or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified function(s). Implicit in this definition shall be the assumption that all necessary attendant instrumentation, controls, normal and emergency electrical power sources, cooling or seal water, lubrication or other auxiliary equipment that are required for the system, subsystem, train, component or device to perform its function(s) are also capable of performing their related support function(s).

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

NOTATION

NOTE 1: Overtemperature $\Delta T \leq \Delta T_0 \left[K_1 - K_2 \left(\frac{1+t_1 S}{1+\tau_2 S}\right) (T-T') + K_3(P-P') - f_1(\Delta I)\right]$

where: ΔT_0 = Indicated ΔT at RATED THERMAL POWER

T = Average temperature, °F

T' = Indicated T_{avg} at RATED THERMAL POWER $\leq$ 574.0°F

P = Pressurizer pressure, psig

P' = 2235 psig (indicated RCS nominal operating pressure)

$\frac{1+\tau_1 S}{1+\tau_2 S}$ = The function generated by the lead-lag controller for T_{avg} dynamic compensation

τ_1 & τ_2 = Time constants utilized in the lead-lag controller for T_{avg} τ_1 = 33 secs, τ_2 = 4 secs.

S = Laplace transform operator

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

NOTATION (Continued)

Operation with 4 Loops	Operation with 3 Loops
$K_1 = 1.267$	$K_1 = 1.116$
$K_2 = 0.01607$	$K_2 = 0.01607$
$K_3 = 0.000926$	$K_3 = 0.000926$

and f_1 (ΔI) is a function of the indicated difference between top and bottom detectors of the power-range nuclear ion chambers; with gains to be selected based on measured instrument response during plant startup tests such that:

(i) for $q_t - q_b$ between - 40 percent and + 3 percent, f_1 (ΔI) = 0 (where q_t and q_b are percent RATED THERMAL POWER in the top and bottom halves of the core respectively, and $q_t + q_b$ is total THERMAL POWER in percent of RATED THERMAL POWER).

(ii) for each percent that the magnitude of ($q_t - q_b$) exceeds - 40 percent, the ΔT trip setpoint shall be automatically reduced by 1.8 percent of its value at RATED THERMAL POWER.

(iii) for each percent that the magnitude of ($q_t - q_b$) exceeds + 3 percent, the ΔT trip setpoint shall be automatically reduced by 2.2 percent of its value at RATED THERMAL POWER.

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

NOTATION (Continued)

Note 2: Overpower $\Delta T \leq \Delta T_o[K_4 - K_5 \left(\frac{\tau_3 S}{1+\tau_3 S}\right) T - K_6 (T-T'') - f_2(\Delta I)]$

where:

ΔT_o	=	Indicated ΔT at rated power
T	=	Average temperature, °F
T''	=	Indicated T_{avg} at RATED THERMAL POWER $\leq$ 574.0°F
K_4	=	1.078
K_5	=	0.02/°F for increasing average temperature and 0 for decreasing average temperature
K_6	=	0.00197 for $T > T''$; $K_6 = 0$ for $T \leq T''$
$\frac{\tau_3 S}{1+\tau_3 S}$	=	The function generated by the rate lag controller for T_{avg} dynamic compensation
τ_3	=	Time constant utilized in the rate lag controller for T_{avg} τ_3 = 10 secs.
S	=	Laplace transform operator
$f_2(\Delta I)$	=	0 for all ΔI

Note 3: The channel's maximum trip point shall not exceed its computed trip point by more than 4 percent.

POWER DISTRIBUTION LIMITS

HEAT FLUX HOT CHANNEL FACTOR - $F_Q(Z)$

LIMITING CONDITION FOR OPERATION

3.2.2 $F_Q(Z)$ shall be limited by the following relationships:

Westinghouse Fuel	Exxon Nuclear Co. Fuel	
$F_Q(Z) \leq \frac{[1.97]}{P} [K(Z)]$	$F_Q(Z) \leq \frac{[2.04]}{P} [K(Z)]$	$P > 0.5$
$F_Q(Z) \leq [3.94] [K(Z)]$	$F_Q(Z) \leq [4.08] [K(Z)]$	$P \leq 0.5$

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

and K(Z) is the function obtained from Figure 3.2-2 for Westinghouse fuel and Figure 3.2-2(a) for Exxon Nuclear Company fuel.

APPLICABILITY: MODE 1

ACTION:

With $F_Q(Z)$ exceeding its limit:

a. Comply with either of the following ACTIONS:

1. Reduce THERMAL POWER at least 1% for each 1% $F_Q(Z)$ exceeds the limit within 15 minutes and similiarly reduce the Power Range Neutron Flux-High Trip Setpoints within the next 4 hours; POWER OPERATION may proceed for up to a total of 72 hours; subsequent POWER OPERATION may proceed provided the Overpower ΔT Trip Setpoints have been reduced at least 1% for each 1% $F_Q(Z)$ exceeds the limit. The Overpower ΔT Trip Setpoint reduction shall be performed with the reactor in at least HOT STANDBY.

2. Reduce THERMAL POWER as necessary to meet the limits of Specification 3.2.6 using the APDMS with the latest incore map and updated $\overline{R}$.

b. Identify and correct the cause of the out of limit condition prior to increasing THERMAL POWER above the reduced limit required by a, above; THERMAL POWER may then be increased provided $F_Q(Z)$ is demonstrated through incore mapping to be within its limit.

SURVEILLANCE REQUIREMENTS

4.2.2.1 The provisions of Specification 4.0.4 are not applicable.

4.2.2.2 $F_Q(Z)$ shall be determined to be within its limit by:

a. Using the movable incore detectors to obtain a power distribution map at any THERMAL POWER greater than 5% of RATED THERMAL POWER.

b. Increasing the measured $F_Q(Z)$ component of the power distribution map by 3% to account for manufacturing tolerances and further increasing the value by 5% to account for measurement uncertainties. This product defined is $F_Q^M(Z)$.

c. Satisfying the following relationships at the time of the target flux determination.

Westinghouse Fuel	Exxon Nuclear Co. Fuel	
$F_Q^M(Z) \leq \left[\frac{1.97}{P}\right] \left[\frac{K(Z)}{V(Z)}\right]$	$F_Q^M(Z) \leq \left[\frac{2.04}{P}\right] \left[\frac{K(Z)}{V(Z)}\right]$	$P > .5$
$F_Q^M(Z) \leq [3.94] \left[\frac{K(Z)}{V(Z)}\right]$	$F_Q^M(Z) \leq [4.08] \left[\frac{K(Z)}{V(Z)}\right]$	$P \leq .5$

where

$F_Q^M(Z)$ is the measured total peaking as a function of core height.

V(Z) is the function defined in Figure 3.2-3 which corresponds to the target band, K(Z) is defined in Figure 3.2-2 for Westinghouse fuel and Figure 3.2-2(a) for Exxon Nuclear Co. fuel, P is the fraction of RATED THERMAL POWER.

d. Measuring $F_Q(Z)$ in conjunction with the target flux difference and target band determination, according to the following schedule:

1. Upon achieving equilibrium conditions after exceeding by 10% or more of RATED THERMAL POWER, the THERMAL POWER at which $F_Q(Z)$ was last determined*, or

2. At least once per 31 effective full power days, whichever occurs first.

*During power escalation at the beginning of each cycle, the design target may be used until a power level for extended operation has been achieved.

SURVEILLANCE REQUIREMENTS

e. With successive measurements indicating an increase in peak pin power, $F_{\Delta H}$, with exposure, either of the following additional actions shall be taken.

1. $F_Q^M(Z)$ shall be increased by 2% over that specified in 4.2.2.2.c, or

2. $F_Q^M(Z)$ shall be measured and a target axial flux difference reestablished at least once per 7 effective full power days until 2 successive maps indicate that the peak pin power, $F_{\Delta H}$, is not increasing.

f. With the relationship specified in 4.2.2.2.c not being satisfied either of the following actions shall be taken.

1. Place the core in an equilibrium condition where the limit in 4.2.2.2.c is satisfied and remeasure the target axial flux difference.

2. Comply with the requirements of Specification 3.2.2 for $F_Q(Z)$ exceeding its limit by the maximum percent calculated with the following expressions with V(Z) corresponding to the target band and P $\geq$ .5:

$$\left[\left[\text{max. over Z of } \frac{F_Q^M(Z) \times V(Z)}{\frac{1.97}{P} \times [K(Z)]}\right] -1\right] \times 100 \qquad \text{Westinghouse Fuel}$$

$$\left[\left[\text{max. over Z of } \frac{F_Q^M(Z) \times V(Z)}{\frac{2.04}{P} \times [K(Z)]}\right] -1\right] \times 100 \qquad \text{Exxon Nuclear Company Fuel}$$

g. The limits specified in 4.2.2.2.c and 4.2.2.2.f above are not applicable in the following core plane regions:

1. Lower core region 0 to 10% inclusive.
2. Upper core region 90% to 100% inclusive.

4.2.2.3 When $F_Q(Z)$ is measured for reasons other than meeting the requirements of Specification 4.2.2.2, an overall measured $F_Q(Z)$ 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.

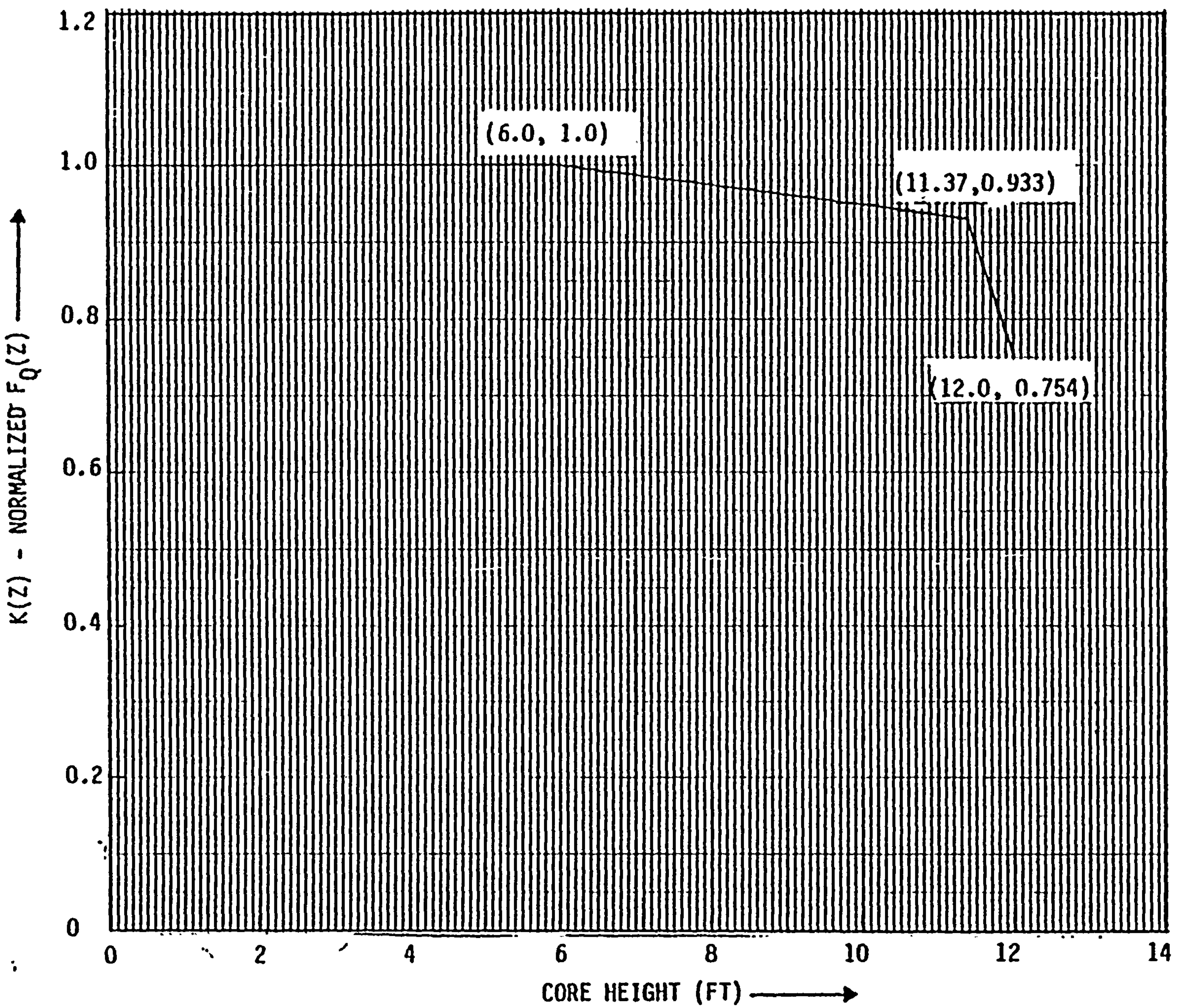


FIGURE 3.2-2 D. C. COOK UNIT 2, K(Z)-NORMALIZED $F_Q(Z)$ AS A FUNCTION OF CORE HEIGHT FOR WESTINGHOUSE FUEL

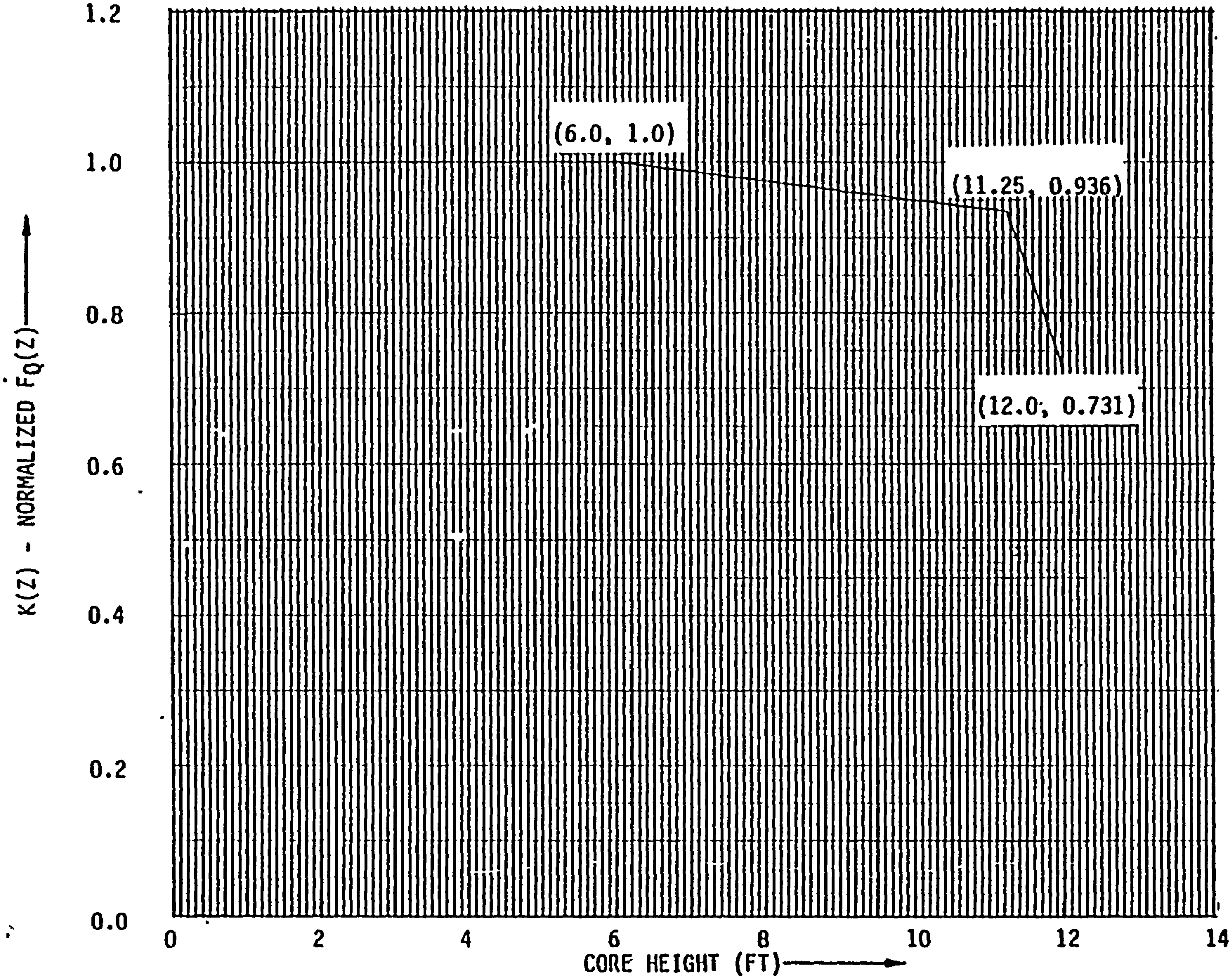


FIGURE 3.2-2(a) D. C. COOK UNIT 2, K(Z) - NORMALIZED F_Q (Z) AS A FUNCTION OF CORE HEIGHT FOR EXXON NUCLEAR CO. FUEL

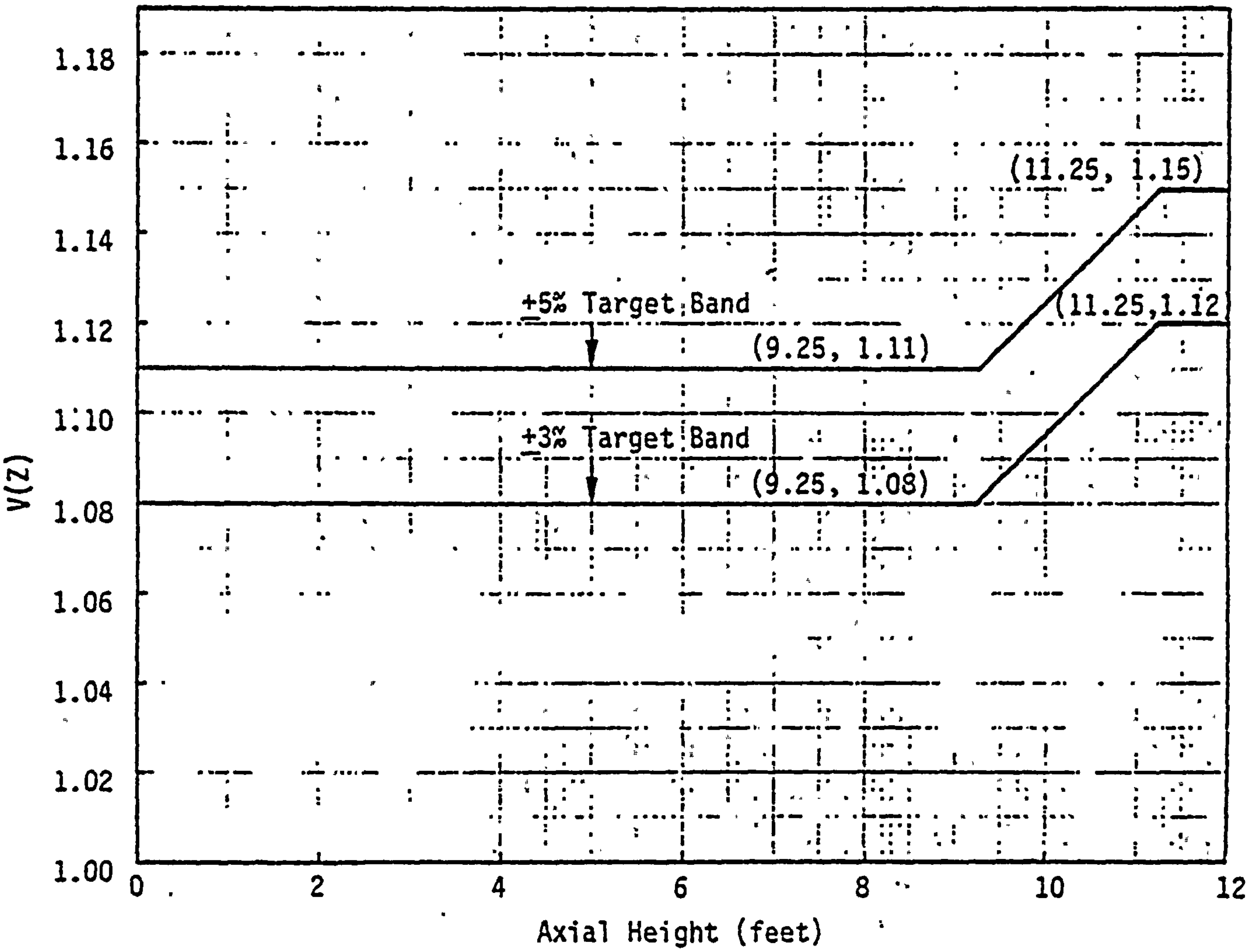


Figure 3.2-3 V(Z) As A Function of Core Height

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 shall be maintained within the region of allowable operation shown on Figures 3.2-4 and 3.2-5 for 4 and 3 loop operation, respectively.

Where: Westinghouse Fuel / Exxon Nuclear Company Fuel

a. Westinghouse Fuel: $R = \dfrac{F^N_{\Delta H}}{1.48\,[1.0 + 0.2\,(1.0 - P)]}$; Exxon Nuclear Company Fuel: $R = \dfrac{F^N_{\Delta H}}{1.49\,[1.0 + 0.2\,(1.0 - P)]}$

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

APPLICABILITY: MODE 1.

ACTION:

With the combination of RCS total flow rate and R outside the region of acceptable operation shown on Figure 3.2-4 or 3.2-5 (as applicable):

a. Within 2 hours:

1. Either restore the combination of RCS total flow rate and 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 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.

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

ACTION: (Continued)

flow rate comparison, to be within the region of acceptable operation shown on Figure 3.2-4 or 3.2-5 (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 R shall be determined to be within the region of acceptable operation of Figure 3.2-4 or 3.2-5 (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: Westinghouse Fuel

$$R = \frac{F^N_{\Delta H}}{1.48\ [1.0 + 0.2\ (1.0 - P)]}$$

Exxon Nuclear Company Fuel

$$R = \frac{F^N_{\Delta H}}{1.49\ [1.0 + 0.2\ (1.0 - P)]}$$

$F^N_{\Delta H}$ = 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 R since Figures 3.2-4 and 3.2-5 include measurement 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.

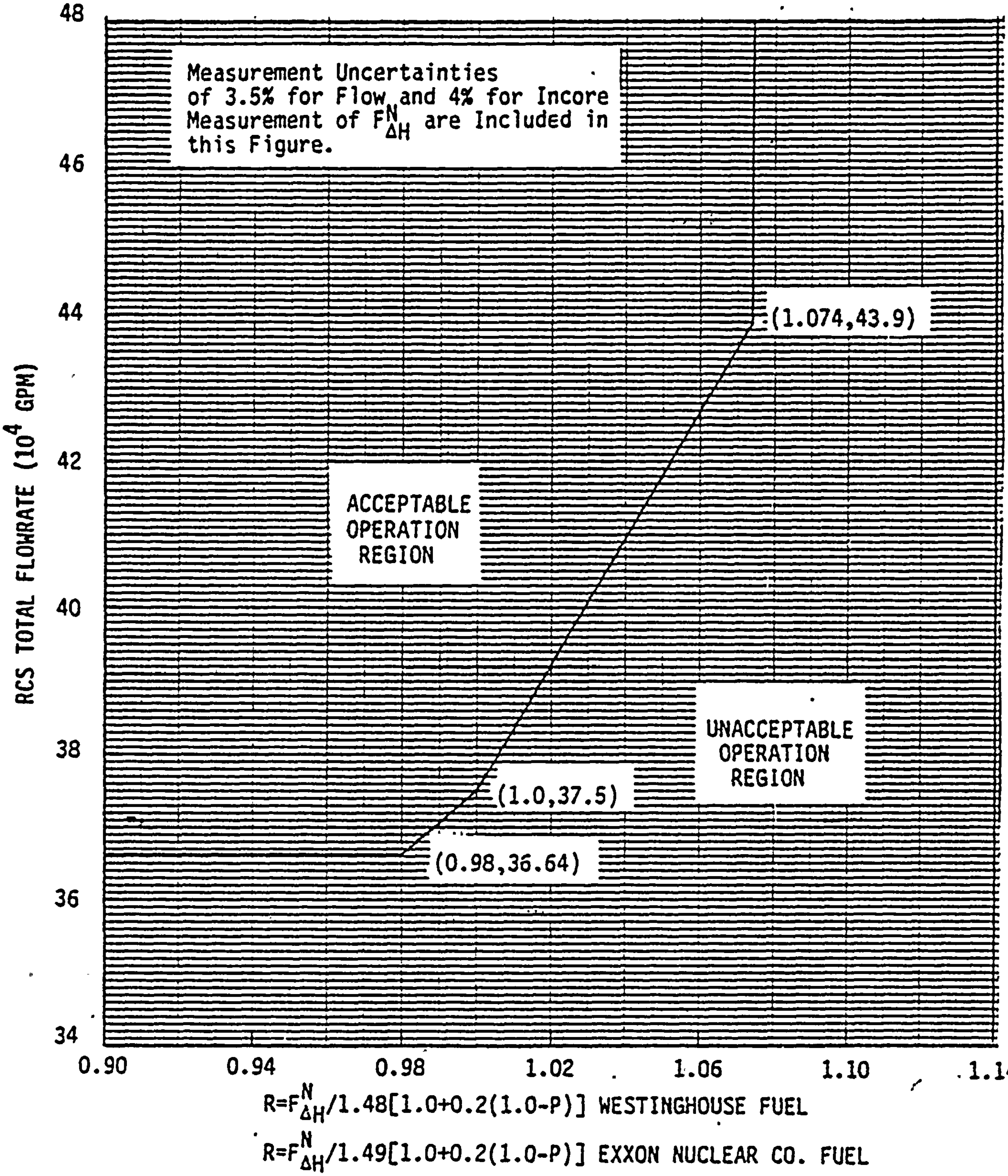
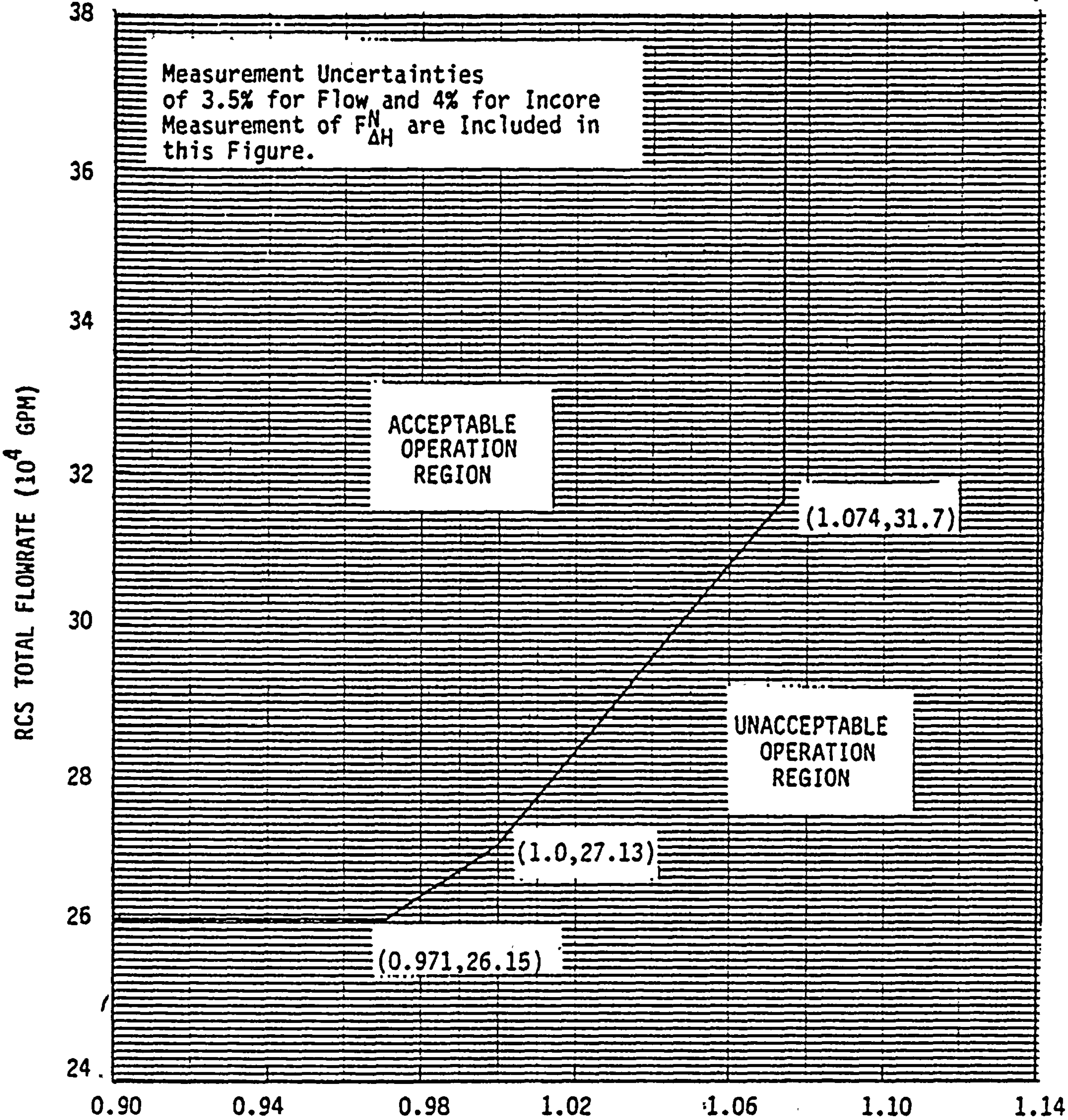


FIGURE 3.2-4 RCS TOTAL FLOWRATE VERSUS R - FOUR LOOPS IN OPERATION



$R=F^N_{\Delta H}/1.48[1.0+0.2(1.0-P)]$ WESTINGHOUSE FUEL

$R=F^N_{\Delta H}/1.49[1.0+0.2(1.0-P)]$ EXXON NUCLEAR CO. FUEL

FIGURE 3.2-5 RCS TOTAL FLOWRATE VERSUS R - THREE LOOPS IN OPERATION

AXIAL POWER DISTRIBUTION

LIMITING CONDITION FOR OPERATION

3.2.6 The axial power distribution shall be limited by the following relationship:

Westinghouse Fuel

$$[F_j(Z)]_S = \frac{[1.97]\,[K(Z)]}{(\overline{R}_j)(P_L)(1.03)(1+\sigma_j)(1.07)}$$

Exxon Nuclear Company Fuel

$$[F_j(Z)]_S = \frac{[2.04]\,[K(Z)]}{(\overline{R}_j)(P_L)(1.03)(1+\sigma_j)(1.07)}$$

Where:

a. $F_j(Z)$ is the normalized axial power distribution from thimble j at core elevation Z.

b. P_L is the fraction of RATED THERMAL POWER.

c. K(Z) is the function obtained from Figure 3.2-2 for Westinghouse Fuel and Figure 3.2-2(a) for Exxon Nuclear Company Fuel for a given core height location.

d. $\overline{R}_j$, for thimble j, is determined from at least n=6 in-core flux maps covering the full configuration of permissible rod patterns above 100% or APL (whichever is less) of RATED THERMAL POWER in accordance with:

$$\overline{R}_j = \frac{1}{n} \sum_{i=1}^{n} R_{ij}$$

Where:

$$R_{ij} = \frac{F_{Qi}^{Meas}}{[F_{ij}(Z)]_{max}}$$

and $[F_{ij}(Z)]_{max}$ is the maximum value of the normalized axial distribution at elevation Z from thimble j in map i which has a measured peaking factor without uncertainties or densification allowance of F_Q^{Meas}.

σ_j is the standard deviation associated with thimble j, expressed as a fraction or percentage of $\overline{R}_j$, and is derived from n flux maps from the relationship below, or 0.02, (2%) whichever is greater.

$$\sigma_j = \frac{\left[\frac{1}{n-1} \sum_{i=1}^{n} (\overline{R}_j - R_{ij})^2\right]^{1/2}}{\overline{R}_j}.$$

The factor 1.07 is comprised of 1.02 and 1.05 to account for the axial power distribution instrumentation accuracy and the measurement uncertainty associated with F_Q using the movable detector system respectively.

The factor 1.03 is the engineering uncertainty factor.

APPLICABILITY: Mode 1 above the minimum percent of RATED THERMAL POWER indicated by the relationships. #

$$\text{APL} = \text{min over Z of } \frac{1.97\ K(Z)}{F_Q(Z) \times V(Z)} \times 100\% \qquad \text{Westinghouse Fuel}$$

$$\text{APL} = \text{min over Z of } \frac{2.04\ K(Z)}{F_Q(Z) \times V(Z)} \times 100\% \qquad \text{Exxon Nuclear Company Fuel}$$

where $F_Q(Z)$ is the measured $F_Q(Z)$, including a 3% manufacturing tolerance uncertainty and a 5% measurement uncertainty, at the time of target flux determination from a power distribution map using the movable incore detectors. V(Z) is the function defined in Figure 3.2-3 which corresponds to the target band. The above limit is not applicable in the following core plane regions.

1) Lower core region 0% to 10% inclusive.
2) Upper core region 90% to 100% inclusive.

ACTION:

a. With a $F_j(Z)$ factor exceeding $[F_j(Z)]_S$ by ≤ 4 percent, reduce THERMAL POWER one percent for every percent by which th $F_j(Z)$ factor exceeds its limit within 15 minutes and within the next two hours either reduce the $F_j(Z)$ factor to within its limit or reduce THERMAL POWER to APL or less of RATED THERMAL POWER.

The APDMS may be out of service when surveillance for determining power distribution maps is being performed.

LIMITING CONDITIONS FOR OPERATION (Continued)

b. With a $F_j(Z)$ factor exceeding $[F_j(Z)]_S$ by $\geq$4 percent, reduce THERMAL POWER to APL or less of RATED THERMAL POWER within 15 minutes.

SURVEILLANCE REQUIREMENTS

4.2.6.1 $F_j(Z)$ shall be determined to be within its limit by:

a. Either using the APDMS to monitor the thimbles required per Specification 3.3.3.7 at the following frequencies.

1. At least once per 8 hours, and

2. Immediately and at intervals of 10, 30, 60, 90, 120, 240 and 480 minutes following:

a) Increasing the THERMAL POWER above APL of RATED THERMAL POWER, or

b) Movement of control bank "D" more than an accumulated total of 5 steps in any one direction.

b. Or using the movable incore detectors at the following frequencies when the APDMS is inoperable:

1. At least once per 8 hours, and

2. At intervals of 30, 60, 90, 120, 240 and 480 minutes following:

a) Increasing the THERMAL POWER above APL of RATED THERMAL POWER, or

b) Movement of control bank "D" more than an accumulated total of 5 steps in any one direction.

4.2.6.2 When the movable incore detectors are used to monitor $F_j(Z)$, at least 2 thimbles shall be monitored and an $F_j(Z)$ accuracy equivalent to that obtained from the APDMS shall be maintained.

2.1 SAFETY LIMITS

BASES

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. This relation 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 DNB design basis is as follows: there must be at least a 95 percent probability that the minimum DNBR of the limiting rod during Condition I and II events is greater than or equal to the DNBR limit of the DNB correlation being used (the XNB correlation in this application). The correlation DNBR limit is established based on the entire applicable experimental data set such that there is a 95 percent probability with 95 percent confidence that DNB will not occur when the minimum DNBR is at the DNBR limit.

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 below which the calculated DNBR is no less than the correlation DNBR limit value or the average enthalpy at the vessel exit is less than the enthalpy of saturated liquid. Uncertainties in primary system pressure, core temperature, core thermal power, primary coolant flow rate, and fuel fabrication tolerances have been included in the analyses from which Figures 2.1-1 and 2.1-2 are derived.

The curves are based on a nuclear enthalpy rise 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:

$$F^N_{\Delta H} = 1.48\ [1 + 0.2\ (1-P)] \quad \text{(Westinghouse Fuel)}$$

$$F^N_{\Delta H} = 1.49\ [1 + 0.2\ (1-P)] \quad \text{(Exxon Nuclear Company Fuel)}$$

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 (ΔI) function of the Overtemperature 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 31.1 1967 Edition, 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, 125% of design pressure, to demonstrate integrity prior to initial operation.

3/4.2 POWER DISTRIBUTION LIMITS

BASES

The specifications of this section provide assurance of fuel integrity during Condition I (Normal Operation) and II (Incidents of Moderate Frequency) events by: (a) maintaining the calculated DNBR in the core at or above design during normal operation and in short term transients, and (b) limiting the fission gas release, fuel pellet temperature and cladding mechanical properties to within assumed design criteria. In addition, limiting the peak linear power density during Condition I events provides assurance that the initial conditions assumed for the LOCA analyses are met and the ECCS acceptance criteria limit of 2200°F is not exceeded.

The definitions of certain hot channel and peaking factors as used in these specifications are as follows:

$F_Q(Z)$ Heat Flux Hot Channel Factor, is defined as the maximum local heat flux on the surface of a fuel rod at core elevation Z divided by the average fuel rod heat flux, allowing for manufacturing tolerances on fuel pellets and rods.

$F^N_{\Delta H}$ Nuclear Enthalpy Rise Hot Channel Factor, is defined as the ratio of the integral of linear power along the rod with the highest integrated power to the average rod power.

The limits on $F_Q(Z)$ and $F^N_{\Delta H}$ for Westinghouse supplied fuel at a core average power of 3411 MWt are 1.97 and 1.48, respectively, which assure consistency with the allowable heat generation rates developed for a core average thermal power of 3391 MWt. The limits on $F_Q(Z)$ and $F^N_{\Delta H}$ for ENC supplied fuel have been established for a core thermal power of 3425 MWt and are 2.04 and 1.49, respectively.

3/4.2.1 AXIAL FLUX DIFFERENCE (AFD)

The limits on AXIAL FLUX DIFFERENCE assure that the $F_Q(Z)$ upper bound envelope is not exceeded during either normal operation or in the event of xenon redistribution following power changes. The $F_Q(Z)$ upper bound envelope is 1.97 times the average fuel rod heat flux for Westinghouse supplied fuel and 2.04 times the average fuel rod heat flux for Exxon Nuclear Company supplied fuel.

Target flux difference is determined at equilibrium xenon conditions. The full length rods may be positioned within the core in accordance with their respective insertion limits and should be inserted near their normal position for steady state operation at high power levels. The value of the target flux difference obtained under these conditions divided by the fraction of RATED THERMAL POWER is the target flux difference at RATED THERMAL POWER for the associated core burnup conditions. Target flux differences for other THERMAL POWER levels are obtained by multiplying the RATED THERMAL POWER value by the appropriate fractional THERMAL POWER level. The periodic updating of the target flux difference value is necessary to reflect core burnup considerations.

Although it is intended that the plant will be operated with the AXIAL FLUX DIFFERENCE within the target band about the target flux difference, during rapid plant THERMAL POWER reductions, control rod motion will cause the AFD to deviate outside of the target band at reduced THERMAL POWER levels. This deviation will not affect the xenon redistribution sufficiently to change the envelope of peaking factors which may be reached on a subsequent return to RATED THERMAL POWER (with the AFD within the target band) provided the time duration of the deviation is limited. Accordingly, a 1 hour penalty deviation limit cumulative during the previous 24 hours is provided for operation outside of the target band but within the limits of Figure 3.2-1 while at THERMAL POWER levels above 50% of RATED THERMAL POWER. For THERMAL POWER levels between 15% and 50% of RATED THERMAL POWER, deviations of the AFD outside of the target band are less significant. The penalty of 2 hours actual time reflects this reduced significance.

Provisions for monitoring the AFD on an automatic basis are derived from the plant process computer through the AFD Monitor Alarm. The computer determines the one minute average of each of the OPERABLE excore detector outputs and provides an alarm message if the AFD for at least 2 of 4 or 2 of 3 OPERABLE excore channels are outside the target band and the THERMAL POWER is greater than 90% or 0.9 x APL of RATED THERMAL POWER (whichever is less). During operation at THERMAL POWER levels between 50% and 90% or 0.9 x APL of RATED THERMAL POWER (whichever is less) and between 15% and 50% RATED THERMAL POWER, the computer outputs an alarm message when the penalty deviation accumulates beyond the limits of 1 hour and 2 hours, respectively.

Figure B 3/4 2-1 shows a typical monthly target band.

The basis and methodology for establishing these limits is presented in topical report XN-NF-77-57, "Exxon Nuclear Power Distribution Control for PWRs - Phase II" and Supplements 1 and 2 to that report.

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 ensure 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 3.1.3.6.

c. The control rod insertion limits of Specifications 3.1.3.5 and 3.1.3.6 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 3.2-4 and 3.2-5, 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. 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.

When RCS flow rate and $F^N_{\Delta H}$ are measured, no additional allowances are necessary prior to comparison with the limits of Figures 3.2-4 and 3.2-5. Measurement errors of 3.5% for RCS flow total flow rate and 4% for $F^N_{\Delta H}$ have been allowed for in determination of the design DNBR value.

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMITS

REACTOR CORE

2.1.1 The combination of THERMAL POWER, pressurizer pressure, and the highest operating loop coolant temperature (T_{avg}) shall not exceed the limits shown in Figures 2.1-1 and 2.1-2 for 4 and 3 loop operation, respectively.

APPLICABILITY: MODES 1 and 2.

ACTION:

Whenever the point defined by the combination of the highest operating loop average temperature and THERMAL POWER has exceeded the appropriate pressurizer pressure line, be in HOT STANDBY within 1 hour.

REACTOR COOLANT SYSTEM PRESSURE

2.1.2 The Reactor Coolant System pressure shall not exceed 2735 psig.

APPLICABILITY: MODES 1, 2, 3, 4 and 5.

ACTION:

MODES 1 and 2

Whenever the Reactor Coolant System pressure has exceeded 2735 psig, be in HOT STANDBY with the Reactor Coolant System pressure within its limit within 1 hour.

MODES 3, 4 and 5

Whenever the Reactor Coolant System pressure has exceeded 2735 psig, reduce the Reactor Coolant System pressure to within its limit within 5 minutes.

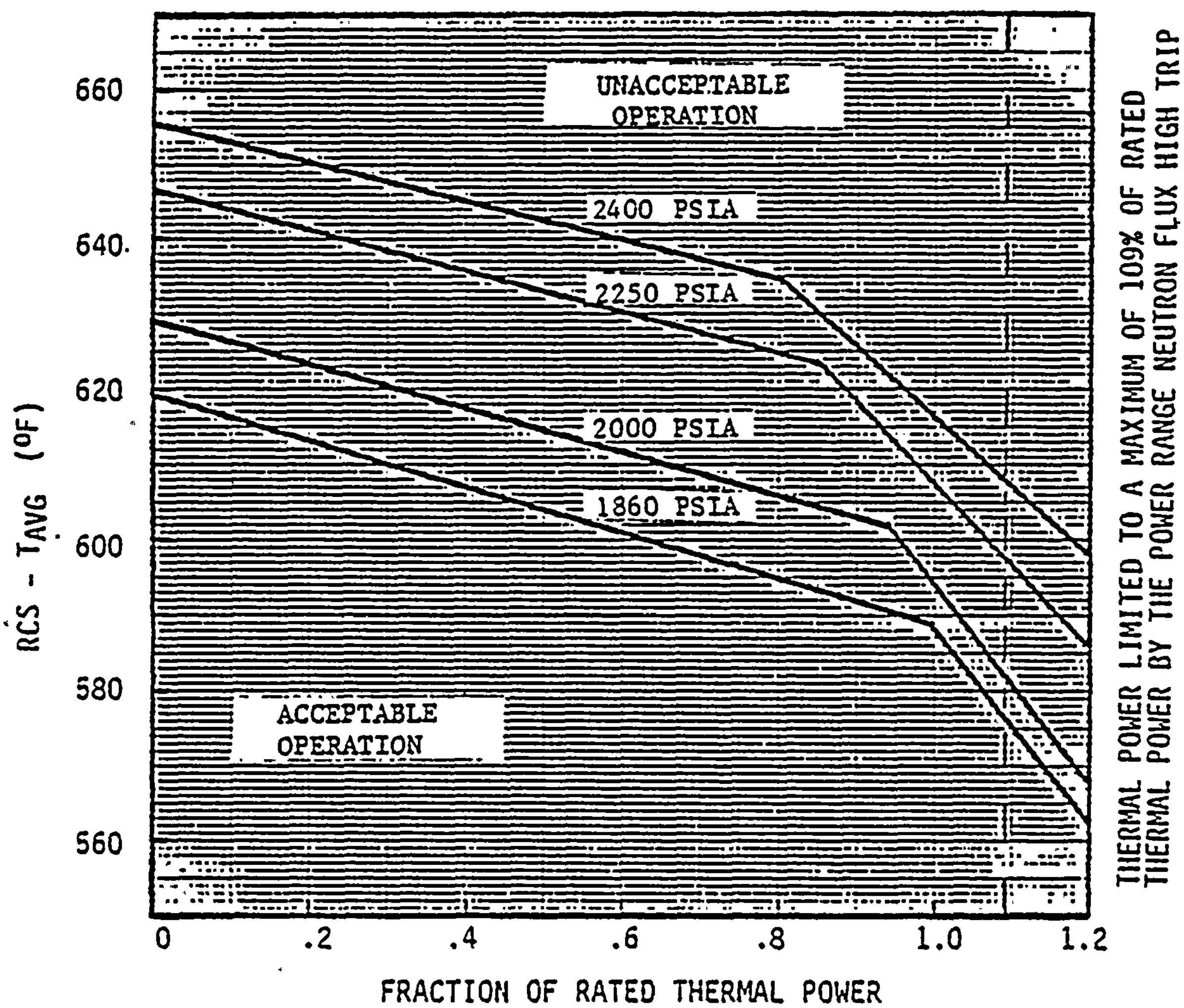


Figure 2.1-1 Reactor Core Safety Limits - Four Loops in Operation

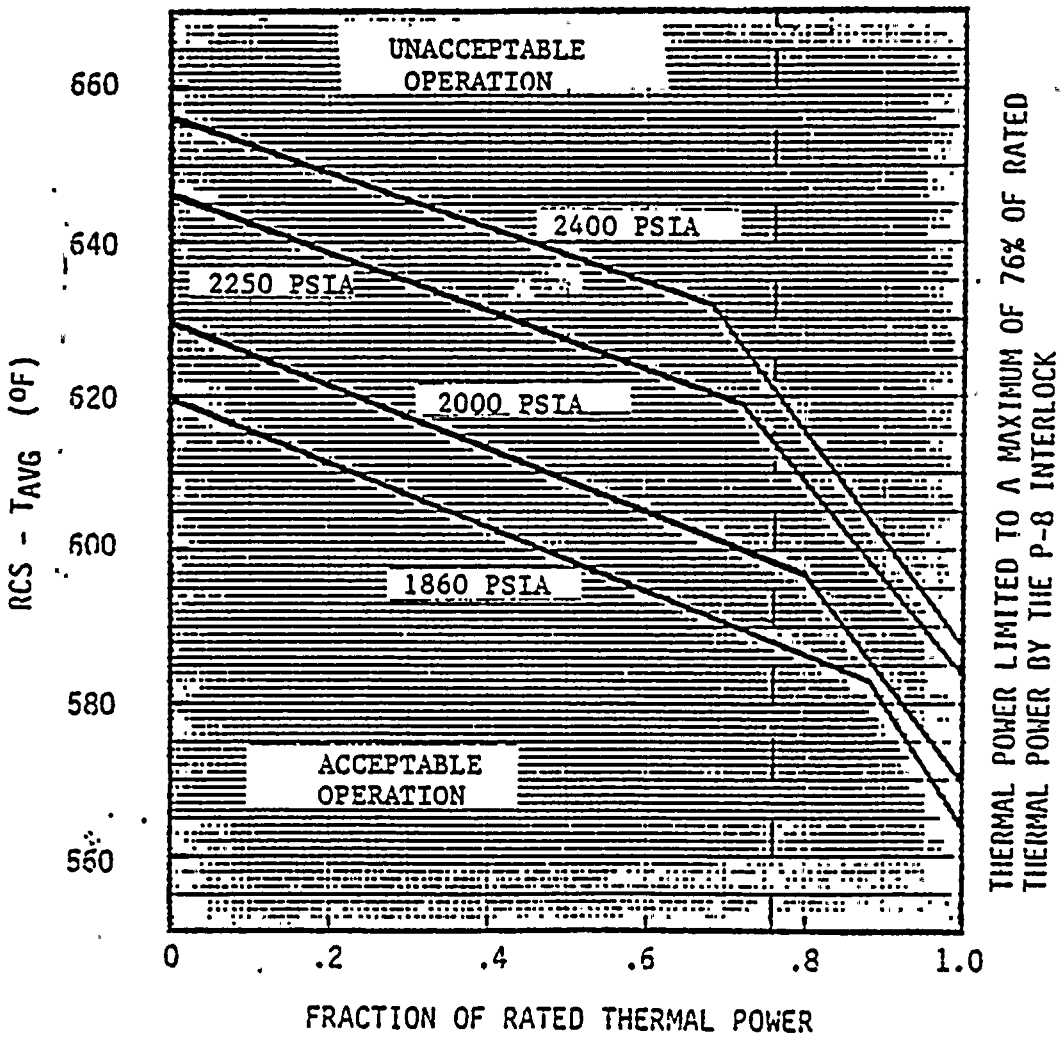


Figure 2.1-2 Reactor Core Safety Limit - Three Loops in Operation

2.2 LIMITING SAFETY SYSTEM SETTINGS

REACTOR TRIP SYSTEM INSTRUMENTATION SETPOINTS

2.2.1 The reactor trip system instrumentation setpoints shall be set consistent with the Trip Setpoint values shown in Table 2.2-1.

APPLICABILITY: As shown for each channel in Table 3.3-1.

ACTION:

With a reactor trip system instrumentation setpoint less conservative than the value shown in the Allowable Values column of Table 2.2-1, declare the channel inoperable and apply the applicable ACTION statement requirement of Specification 3.3.1.1 until the channel is restored to OPERABLE status with its trip setpoint adjusted consistent with the Trip Setpoint value.