ENCLOSURE 2

NORTH ANNA UNITS 1 & 2
587.8°F REACTOR COOLANT SYSTEM
STONE & WEBSTER/BOP
SAFETY EVALUATION SUMMARY

STONE & WEBSTER ENGINEERING CORPORATION
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A. OBJECTIVE

The objective of this report is to provide a technical basis for determining that the proposed 7.5°F increase in reactor coolant system Tavg does not involve an unreviewed safety question in accordance with the requirements of 10CFR50.59. This review is limited to systems within Stone & Webster Engineering Corporation's original scope of work. The NSSS and Turbine-Generator review has been performed by Westinghouse and is documented as Enclosure 1.

Our evaluation used the following parameters which bound or are equivalent to the proposed uprated conditions:

Main Steam Pressure 100% Power	930 psia
Main Steam Temp. No-Load	547°F
Main Steam Pressure No-Load	1020 psia
RCS Tavg	587.8°F
Steam Flow 10^6 lb/hr Total	12.25
Reactor Power MW_t	2775.
NSSS Power MW_t	2787.

B. CONCLUSION

The proposed change in reactor coolant system average temperature has been reviewed and evaluated with respect to the following:

1. Accident Analysis and Environmental Qualification
2. Pipe Stress and Supports
3. Major Equipment Supports and Pipe Rupture Restraints
4. Structures
5. BOP System and NSSS Interfaces
6. Heat Balance Calculations
7. Control Systems and Instrumentation
8. Electrical Systems
9. Review of Technical Specifications

Based on the results of our review it has been concluded that the proposed 7.5°F increase in Tavg does not represent an unreviewed safety question as defined in 10CFR50.59. The summary of the analyses related to the above are attached.

1. It has been determined that the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety and previously evaluated in the Safety Analysis Report has not been increased.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report has not been created.

3. The margin of safety as defined in the basis for any Technical Specification has not been reduced.

C. ACCIDENT ANALYSIS AND ENVIRONMENTAL QUALIFICATIONS

1. Containment Loss of Coolant Accident (LOCA)

An analysis has been performed to investigate the effect of the 7.5°F uprate on containment integrity and Net Positive Suction Head Available (NPSHA) for the Recirculation Spray (RS) and Low Head Safety Injection (LHSI) pumps. The following are the results:

1. Containment Integrity

Table 1 shows the effect on containment peak pressure, subatmospheric peak pressure and depressurization time due to the uprate. The pressures increase slightly but are still within the North Anna acceptance criteria and containment design pressure rating.

2. NPSHA for RS and LHSI Pumps

Table 2 shows that there is no decrease in NPSHA for the inside and outside RS pumps. There is a slight decrease in the LHSI pump NPSHA but it remains above NPSH requirements at design flow.

2. Containment Main Steam Line Break (MSLB) Analysis

The basis of the steam line break analysis is the full guillotine main steam line break at the no-load (hot shutdown) condition. The no-load Reactor Coolant System and Steam Generator temperature and pressure remain unchanged subsequent to the uprate. Although there are additional energy releases at power operating conditions due to increased (uprated) Steam Generator pressure and temperature, the no-load condition remains the limiting case. Therefore the Main Steam Line Break post-accident conditions remain as previously analyzed.

3. Subcompartment Analysis

Subcompartment analyses were performed and are documented in the UFSAR for the reactor cavity, steam generator cubicle and pressurizer cubicle. Our calculations have confirmed that for the subcooled reactor coolant system, mass and energy releases decrease with increased reactor coolant temperature. The analyses documented in the UFSAR are therefore bounding for the uprate.

4. Equipment Qualification

1. Inside Containment

Equipment qualification inside the containment is based on the Main Steam Line Break and LOCA post-accident environmental conditions. Containment design pressure is used for qualification. The current Main Steam Line Break and LOCA analyses are either unchanged or negligibly impacted for the uprate conditions as discussed in Sections 1 and 2. Post-accident containment pressure remains below the design pressure.

2. Outside Containment

Post-accident environments outside containment which are used to generate equipment qualification envelopes are based on the following high energy line breaks:

a. Primary System Branch Line Break

The Letdown Line Break is part of the basis for the environmental qualification in the charging pump cubicle in the auxiliary building. The letdown line temperature increases one degree above the normal operating temperature used in the original analysis. The resulting environmental temperature due to a postulated line break is not significantly affected by this change.

b. Secondary System Break

The Main Steam Line Break affects the Main Steam Valve House, the Service Water Valve Pit and the Turbine Building. The Main Steam Valve House environmental envelope is based on the no-load power condition which is unaffected by the uprate. Equipment qualification temperature and pressure in the Service Water Valve Pit and Turbine Building is limited by the Turbine Building siding pressure retaining capability. Any change to break effluent due to the uprate has no effect on this pressure or temperature, and therefore, on equipment qualification.

c. Auxiliary Steam Line Break

This break affects the Auxiliary Building and the Service Water Valve Pit. The releases are based on the Auxiliary Steam Line relief valve pressure setting which is unchanged by the uprate. Additionally, the Auxiliary Steam System pressure is controlled by a pressure reduction valve tied into the Main Steam header. The increased Main Steam Operating pressure will not affect the operation of this valve and therefore the Auxiliary Steam System pressure will remain unchanged.

d. Steam Generator Blowdown Line Break

This affects the Pipe Tunnel and the Auxiliary Building. The releases are based on the bounding condition of no-load steam generator pressure which is unchanged by the uprate.

Table 1

LOCA Containment Integrity Analysis Results

	Current	7.5°F Uprate	Required
Containment Peak Pressure (psig.)	40.27	40.62	<45.
Containment Depressurization Time (Sec.)	3300	3320	<3600
Containment Subatmospheric Peak (psig.)	-.12	-.11	<0.

Table 2

LOCA NPSH Analysis Results

	Current	7.5°F Uprate	Required NPSH*
Inside Recirc Spray Pumps (ft.)	11.7	11.7	8.8
Outside Recirc Spray Pumps (ft.)	17.0	17.0	9.7
Low Head Safety Injection Pumps (ft.)	15.6	15.2	13.4

*At design flow

D. TRANSIENT ANALYSIS

As a plant operational consideration, the 50% load rejection and turbine trip capability is being investigated. Any changes in load rejection capabilities will be evaluated prior to uprating implementation and be incorporated in a subsequent UFSAR amendment as necessary.

E. PIPE STRESS AND SUPPORTS

For all systems within the scope of the uprating program, the appropriate maximum stress levels and fatigue analysis results have been reviewed for the plant uprate conditions. This review indicates that in all cases the associated piping stress and fatigue limits will not be exceeded as a result of the uprate. The following systems were evaluated for the uprated conditions:

System	Safety Class
Main Steam	2
Feedwater	2
Reactor Coolant	1
RC Loop Letdown	1
RC Loop Excess Letdown	1
Pressurizer Spray	1
Residual Heat Removal	1,2
Low Head Safety Injection	1
High Head Safety Injection	1
CVCS Seal Water Inlet	1
CVCS Seal Water Outlet	1
RC Loop Fill	1
Resistance Temperature Detection	1

System	Safety Class
RC Loop Drain	1
RC Loop Charging	1
Steam Generator Blowdown	2
Steam Generator Wet Layup	2
Component Cooling to RC Pumps	2

The uprate also has minimal impact on the existing stress analysis of the unmodified Resistance Temperature Detector (RTD) lines in the Reactor Coolant System. Because the Reactor Vessel Level Indication System (RVLIS) now ties into the RTD lines, a review of portions of these lines will be necessary. This review will be performed prior to implementation of the uprate and it is expected that the usage factors and stresses resulting from the RVLIS tie-in will not exceed code allowables.

The above systems were reviewed with respect to pipe stress and the adequacy of pipe support designs. The maximum expected increase in the thermal expansion due to the uprate is approximately 2%. Considering that pipe support design loads include several loading components other than thermal, it is concluded that, with the exception of the Unit 2 Main Steam Mono-Ball supports identified during the 2.5°F uprate program review, all existing pipe support and equipment nozzle design loads will remain valid. The Mono-Ball supports will be modified to accommodate both the 2.5°F and 7.5°F uprating, prior to implementing the 2.5°F uprating.

A review of safety related piping for fatigue effects of thermal transients during the 2.5°F program review revealed that only the reactor coolant letdown line required reanalysis. The 2.5°F increase in Tavg subjected the letdown line to a more severe reinitiation of letdown flow transient than was originally used as a design basis. The results of the analysis showed that the fatigue effects on the piping would remain acceptable subsequent to both the proposed 2.5°F and 7.5°F uprates. The stress report will be revised to include this information

It has been determined that previously calculated break points in piping systems, used for the design of rupture restraints, jet shields and large pipe supports will remain unchanged as a result of the uprating. The points of maximum stress within any piping system remain unchanged as the stress increases due to the uprating are uniform.

In conclusion, all existing pipe stress and support analyses within the scope of this evaluation have been determined to remain valid under the conditions of the 7.5°F uprate in Reactor Coolant System Tavg.

F. MAJOR EQUIPMENT SUPPORTS AND PIPE RUPTURE RESTRAINTS

Design calculations for major equipment supports, seismic tanks, vessels, pipe rupture restraints and shields and miscellaneous mechanical equipment were reviewed with the uprated system parameters for 100% power and no-load conditions.

Approximately 450 calculations were reviewed. Of these calculations, a number did not use system pressures and temperatures which bounded the uprated conditions. Each one of those calculations was reviewed with respect to the uprated conditions. Only one of those calculations was found to have revised results. The remaining calculations were determined to be acceptable at the uprated conditions. The revised results consisted of an increased loading to a radial wall in the containment due to a postulated Main Feedwater pipe break. This loading is used as input to a structural calculation which was reviewed for the new loads due to the uprate. The results of the review are in Section G. below.

The data used to assess the effect of LOCA loading on major equipment supports remain applicable for both the 2.5°F and 7.5°F uprated conditions. Our evaluation concluded that an increase in Reactor Coolant System temperature results in a decrease in LOCA loadings with the frequency content of those loadings being unchanged. The amplitudes and time history of the LOCA loads data previously supplied by Westinghouse was verified as being applicable by Westinghouse.

G. STRUCTURES

The only structural loads subject to change as a result of the uprate were those resulting from system parameter changes such as pressure, temperature and flow and subsequent postulated pipe breaks as developed in the review in Section F above, those resulting from peak containment pressure subsequent to a LOCA and subcompartments pressurizations reviewed in Section C above. The new post LOCA peak containment pressure is less than containment design pressure and subcompartment pressures decrease.

Only one structural calculation (referenced in Section F above) was further reviewed because of increased loads from the uprating. This review was involved with the impact of a Main Feedwater pipe in the containment on a radial wall during a postulated break. The revised calculation showed that the structural integrity of the Reactor Containment would not be impacted by the postulated pipe break. A review of the expected spalling (concrete debris) zones was performed in order to determine if the spalling would have an adverse impact on any safety related components that would be required to operate to mitigate the consequences of that break. It was determined that the spalling would not adversely impact any safety related components in such a way that would preclude a safe shutdown of the plant subsequent to that break.

In summary, the uprating caused some structural loads to increase (due to a small increase in post accident containment peak pressure and feedwater pipe rupture discussed above), but those increased loads were found not to change the basis of the original plant design, or the structural integrity of any safety-related structures.

H. BOP SYSTEMS AND NSSS INTERFACES

1. Main Steam System

The Main Steam System piping is designed for 560°F and 1100 psia. These conditions bound the uprated Main Steam conditions of 547°F/1020 psia at no-load and 536°F/930 psia at 100% power.

The Main Steam Safety valves have a total relieving capacity of 12,826,260 lb/hr. Based on Heat Balance 13075.01-HM-7A the total Main Steam flow rate will be 12,251,367 lb/hr. The Main Steam Safety Valves must be capable of passing the entire flow without taking credit for operation of the Power Operated Relief Valves or the Main Steam Dump System. Using this criterion, it has been shown that the Main Steam Safety Valves are adequately designed for use at the uprated conditions.

The Main Steam Trip and Non-Return Valves were evaluated for rapid closure impact loads applied subsequent to Main Steam System Pipe rupture at uprated conditions. The results of computer runs that modeled the transient's effect on the valves showed that they would close as required during a Main Steam Pipe break without jeopardizing the integrity of the pressure boundary.

2. Auxiliary Feedwater System

The Auxiliary Feedwater pumps are designed to deliver rated flow to the Steam Generator at a static discharge head equivalent to the set pressure of the lowest set Main Steam Safety Valve, 1085 psig. Because this set point will not change and the Main Steam Safety Valves have been shown to be acceptable at the uprated conditions, it is concluded that the resistance parameters associated with the Auxiliary Feedwater System are unchanged.

The Westinghouse NSSS safety evaluation states that the existing Auxiliary Feedwater flow requirements (based on 2910 MW_t core power plus 2%) for North Anna are unchanged.

Our conclusion is that the existing Auxiliary Feedwater System is adequate at the uprated condition as the flow requirements and system resistance parameters are unchanged as a result of the uprating.

3. Extraction Steam System, Feedwater Heaters and Flash Evaporator

A heat balance was developed for the proposed 7.5°F increase in Tavg at 2787 MW_t/930 psia steam pressure. From this heat balance (13075.01-HM-7A) the values were taken for the temperatures and pressures of the Extraction Steam lines.

We have determined that the uprated operating pressures fall within the design conditions for all extraction lines. It was observed that the uprated operating temperatures of the third point extraction lines on both units were above the design temperature. An analysis was performed to determine the thermal stresses produced in the third point extraction lines at the uprated condition. The maximum thermal stress in the third point extraction lines were found to be below the code allowable stress, confirming that all Extraction Steam lines (including the third point extractions) are adequate to operate at the uprated conditions.

We have also verified that the uprated extraction pressures are within the shell side design pressures for all of the Feedwater Heaters and the Flash Evaporator. It was conservatively assumed that the pressure at the turbine extraction nozzle was equivalent to the operating pressure at the Feedwater Heater shell without considering pressure drop in the extraction lines.

4. Condensate and Feedwater Systems

To determine if the Condensate and Feedwater System piping would be adequate at the uprated conditions, a comparison was made between the current North Anna heat balance (13075.01-HM-1) and the uprated heat balance (13075.01-HM-7A). It was shown that the entire Condensate/

Feedwater piping train uprated temperatures were within one degree of the current condition at all locations. The existing feedwater temperature entering the Steam Generator is approximately 1°F higher than at the uprated condition. This same relationship exists throughout the Condensate and Feedwater systems at each Feedwater Heater inlet and outlet location. Discharge pressure at the Condensate pump discharge increased by approximately 1 psia and discharge pressure at the Feedwater pump discharge decreased by approximately 2 psia.

Therefore the predicted change in Condensate and Feedwater System temperature and pressure parameters due to the uprating is insignificant. These small changes are within the capability of the current system.

The total Condensate and Feedwater System resistance was evaluated for the new flow rates and Steam Generator pressure pertaining to the 7.5°F increase in Tavg. The overall system resistance increases slightly due to the higher Steam Generator pressure and greater friction losses due to increased flow rate. However, it has been determined from analyzing the existing Condensate Feedwater System calculation that the existing pumps have sufficient head to overcome the slightly increased total system resistance with two Condensate and two Steam Generator Feed Pumps in operation at the uprated condition. The NPSHA at the suction of the Condensate and Feedwater pumps was evaluated at the uprated conditions. It was determined that sufficient NPSHA exists to allow acceptable operation at the uprated flows.

5. <u>Feedwater Regulating Valves</u>

A review of the operation of the Main Feedwater Regulating and Bypass Valves has shown that increasing Tavg by 7.5°F will result in a significant decrease in flexibility of the Feedwater Regulating Valve. To ensure the Feedwater Regulating Valves can pass the required flow, the Bypass Valves may need to be maintained partially open or a change in Feedwater Regulating Valve trim may be required.

6. Low Pressure and High Pressure Heater Drain System

As with the Condensate and Feedwater systems, the uprated temperature and pressure conditions associated with the Low and High Pressure Heater Drain Systems are within one degree of temperature and one psia of pressure from current conditions. The Low and High Pressure Heater Drain Pumps have been shown to be adequate at the uprated flow conditions. Uprated NPSHA has been evaluated at the pump suctions and has been determined to be acceptable for pump operation.

7. Steam Generator Blowdown System

A review of the Steam Generator Blowdown System has indicated that uprating Reactor Coolant System Tavg by 7.5°F will not affect the present safety aspects or operability of the system.

The design of the excess flow high energy line break isolation valves was for an inlet pressure of 1100 psig which is higher than the lowest Main Steam Safety Valve setpoint and is therefore acceptable with regard to the uprate.

All remaining portions of the Steam Generator Blowdown System including flow control valves, safety valves, tanks and pressure control valves were reviewed for any expected temperature and pressure changes and are unaffected by the uprate.

8. Condensate Polishing System

The design conditions for the Condensate Polishing System were evaluated to determine the ability of the polishers to operate at the uprated conditions. The proposed temperature and pressure of the condensate at the polisher inlet (470 psig and 103°D) remain below the polishing system design conditions of 700 psig and 180°F.

From heat balance 13075.01-HM-7A, the condensate flow rate is given as 8,001,103 lb/hr or 2.68 gpm per sq. ft. of Condensate Polishing System filtering surface area. This flow rate is less than the vendors guaranteed filtering capacity of 4.0 gpm per sq. ft. of surface area at design pressure drop. Therefore it has been determined that the condensate polishing system is adequate to operate at the 7.5°F uprated conditions, although at increased flow rates, filter differential pressure will increase at a faster rate and the backwash frequency will be slightly higher.

9. Auxiliary Steam PCV

The Auxiliary Steam Pressure Control Valve PCV-AS-105 has been reviewed for the uprated pressure conditions. The valve is designed for a maximum inlet pressure of 1200 psig. This design pressure is higher than the maximum possible upstream pressure of 1020 psia at no-load conditions. It is therefore concluded that the Auxiliary Steam PCV is adequate for the 7.5° uprated conditions and will reduce uprated Main Steam pressure as originally designed to 150 psig.

10. Component Cooling System

The increased RCS cold leg temperature increases the heat loadings on the Component Cooling Water System due to the Chemical and Volume Control System heat exchangers which are designed to remove heat from the letdown flow stream. The letdown line is tied into one RCS cold leg.

The affected heat exchangers are the Non-Regenerative, Excess Letdown and Seal Water Return heat exchangers. The cumulative heat loading to the component Cooling Water System at the uprated normal operating condition remains below the design value used for the original plant design. Consequently, the Service Water System will not be impacted as a result of the uprating.

11. Spent Fuel Pool Cooling System

There is no impact on the Spent Fuel Pit Heat Loads as a result of the uprating since core power and associated decay heat levels remain unchanged.

12. Containment Air Recirculation System

The heat input into the containment has been estimated to increase approximately 2% because of the increased average Reactor Coolant System temperature. This increased heat load is well below the installed design capacity of cooling units in the containment air recirculation system. Since Technical Specifications limit the ambient temperature inside the containment to 105°F, no change in the operating procedures are necessary and containment temperatures will not exceed that limit.

I. HEAT BALANCE CALCULATIONS

A heat balance diagram was developed for analytical and operational use at the uprated conditions of 2787 MW_t (2775 MW_t core power plus 12 MW_t pump heat) and 930 psia steam pressure. This document, 13075.01-HM-7A, was transmitted to Vepco in September 1982 and was used to evaluate secondary system adequacy at the uprated conditions.

J. CONTROLS SYSTEMS AND INSTRUMENTATION

The 7.5°F uprate Heat Balance Diagram (13075.01-HM-7A) has been reviewed to determine the effect of this uprate on BOP instrumentation and control valves in the Feedwater, Condensate, Main Steam and Heater Drains Systems. The changes in pressures, temperatures and flows were reviewed against the design parameters of the existing equipment. It was determined that the presently installed equipment is adequate for the 7.5°F uprate. No setpoint changes are required as a result of this uprating in addition to those specified by Westinghouse.

K. ELECTRICAL SYSTEMS

An uprated generator output of 954,698 KW (from Heat Balance 13075.01-HM-7A)
at .9 PF yields 27,847 Amps to the Iso-Phase Bus Duct. The existing bus duct is rated at 30,500 Amps continuous and is adequate for the uprated conditions.

A review has been performed to evaluate the increased loading on the Feedwater, Condensate and Heater Drain Pump motors resulting from increases in fluid flow rate. As a result of this review, it has been determined that the motors and their associated power feeds are adequate for the uprated conditions.

Because the increased loadings on the above mentioned pumps does not exceed the rated horsepower for the respective motors, there is no impact on the Station Service Transformers, Normal Buses or connecting cables due to the uprate.

The Emergency Buses are not affected because the uprating causes none of the emergency loads to increase.

The GDC-17 Confirmatory Analysis Studies have been evaluated with respect to the uprating and it has been determined that no changes in the results of that study have been evidenced.

L. REVIEW OF THE TECHNICAL SPECIFICATIONS

The Technical Specifications have been reviewed to determine if any sections could be affected by the proposed 7.5 Tavg increased from 580.3°F to 587.8° F. With the exception of the Technical Specification revisions in Enclosure 3, no additional sections are affected.

ENCLOSURE 3

North Anna

Technical Specifications Changes

for

Tavg of 587.8°F

Note: Changes denoted by a double bar, ||, are identical to those in the most recent LOCA analysis NRC submittal, Vepco letter Serial No. 490, dated August 16, 1982.

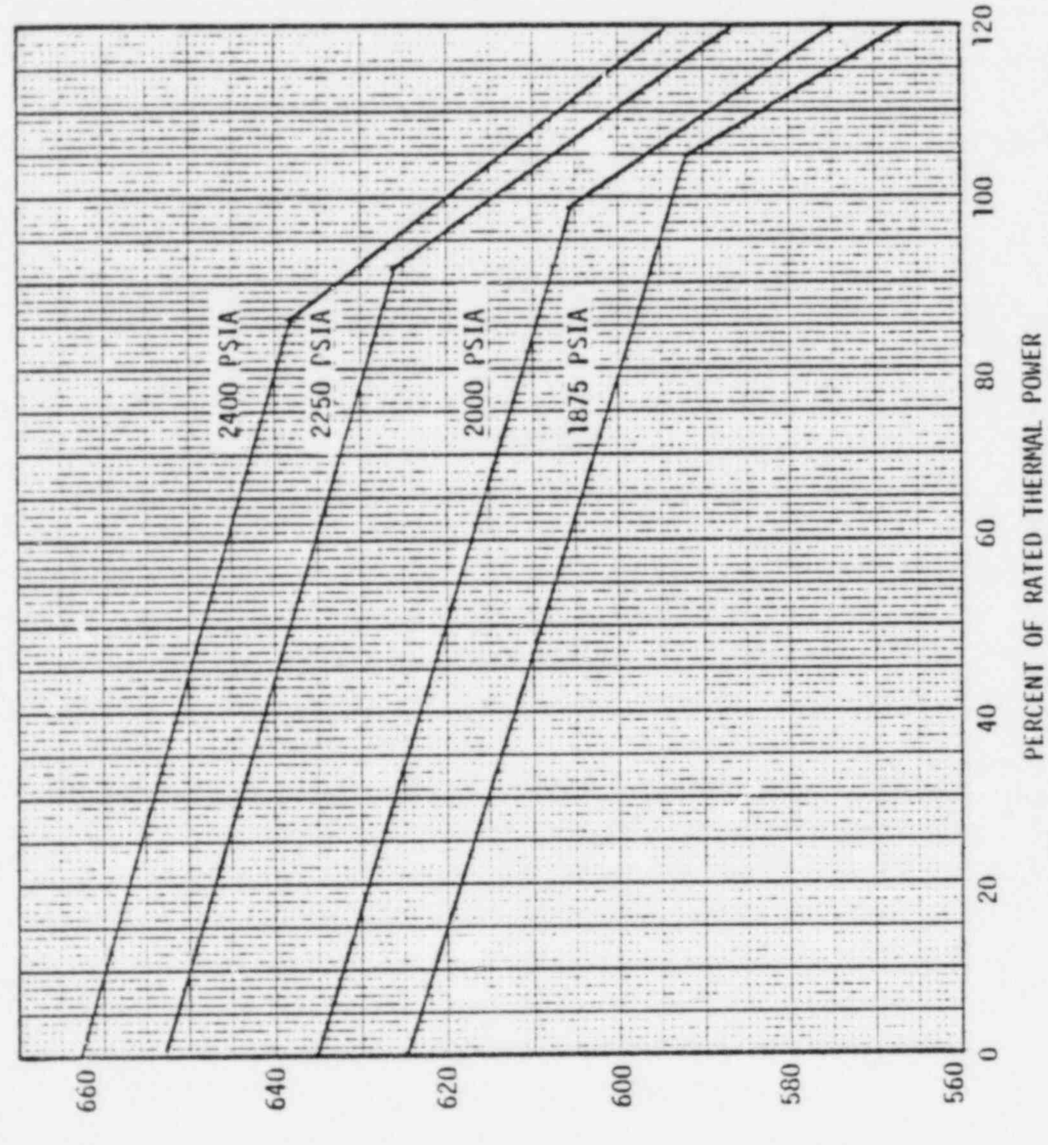


FIGURE 2.1-1 REACTOR CORE SAFETY LIMITS FOR THREE LOOP OPERATION, 100% FLOW

TABLE 2.2-1

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
1. Manual Reactor Trip	Not Applicable	Not Applicable
2. Power Range, Neutron Flux	Low Setpoint - $\leq$ 25% of RATED THERMAL POWER	Low Setpoint - $\leq$ 26% of RATED THERMAL POWER
	High Setpoint - $\leq$ 109% of RATED THERMAL POWER	High Setpoint - $\leq$ 110% of RATED THERMAL POWER
3. Power Range, Neutron Flux, High Positive Rate	$\leq$ 5% of RATED THERMAL POWER with a time constant $\geq$ 2 seconds	$\leq$ 5.5% of RATED THERMAL POWER with a time constant $\geq$ 2 seconds
4. Power Range, Neutron Flux, High Negative Rate	$\leq$ 5% of RATED THERMAL POWER with a time constant $\geq$ 2 seconds	$\leq$ 5.5% of RATED THERMAL POWER with a time constant $\geq$ 2 seconds
5. Intermediate Range, Neutron Flux	$\leq$ 25% of RATED THERMAL POWER	$\leq$ 30% of RATED THERMAL POWER
6. Source Range, Neutron Flux	$\leq 10^5$ counts per second	$\leq 1.3 \times 10^5$ counts per second
7. Overtemperature ΔT	See Note 1	See Note 3
8. Overpower ΔT	See Note 2	See Note 3
9. Pressurizer Pressure--Low	$\geq$ 1870 psig	$\geq$ 1860 psig
10. Pressurizer Pressure--High	$\leq$ 2385 psig	$\leq$ 2395 psig
11. Pressurizer Water Level--High	$\leq$ 92% of instrument span	$\leq$ 93% of instrument span
12. Loss of Flow	$\geq$ 90% of design flow per loop*	$\geq$ 89% of design flow per loop*

*Design flow is 95,000 gpm per loop.

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

NOTATION

NOTE 1: Overtemperature $\Delta T \leq \Delta T_0 \left[K_1 - K_2\left(\frac{1+\tau_1 S}{1+\tau_2 S}\right)(T-T') + K_3(P-P') - f_1(\Delta I)\right]$

where: ΔT_0 = Indicated ΔT at RATED THERMAL POWER

T = Average temperature, °F

T' = Indicated T_{avg} at RATED THERMAL POWER $\leq$ 587.8°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 = 25 secs, τ_2 = 4 secs.

S = Laplace transform operator (sec^{-1})

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

NOTATION (Continued)

Operation with 3 Loops	Operation with 2 Loops (no loops isolated)*	Operation with 2 Loops (1 loop isolated)*
K_1 = 1.078	K_1 = ( )	K_1 = ( )
K_2 = 0.0143	K_2 = ( )	K_2 = ( )
K_3 = 0.000674	K_3 = ( )	K_3 = ( )

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 - 36 percent and + 11 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 - 36 percent, the ΔT trip setpoint shall be automatically reduced by 1.2 percent of its value at RATED THERMAL POWER.

(iii) for each percent that the magnitude of ($q_t - q_b$) exceeds + 11 percent, the ΔT trip setpoint shall be automatically reduced by 1.6 percent of its value at RATED THERMAL POWER.

*Values dependent on NRC approval of ECCS evaluation for these operating conditions.

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 THERMAL POWER

T = Average temperature, °F

T'' = Indicated T_{avg} at RATED THERMAL POWER $\leq$ 587.8°F

K_4 = 1.091

K_5 = 0.02/°F for increasing average temperature

K_5 = 0 for decreasing average temperatures

K_6 = 0.00126 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 (sec^{-1})

$f_2(\Delta I)$ = 0 for all ΔI

Note 3: The channel's maximum trip point shall not exceed its computed trip point by more than 2 percent span.

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:

$$F_Q(Z) \leq \left[\frac{2.20}{P}\right] [K(Z)] \text{ for } P > 0.5$$

$$F_Q(Z) \leq [4.40] \; [K(Z)] \text{ for } P \leq 0.5$$

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

and K(Z) is the function obtained from Figure 3.2-2 for a given core height location.

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 similarly 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 $\bar{R}$.

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.

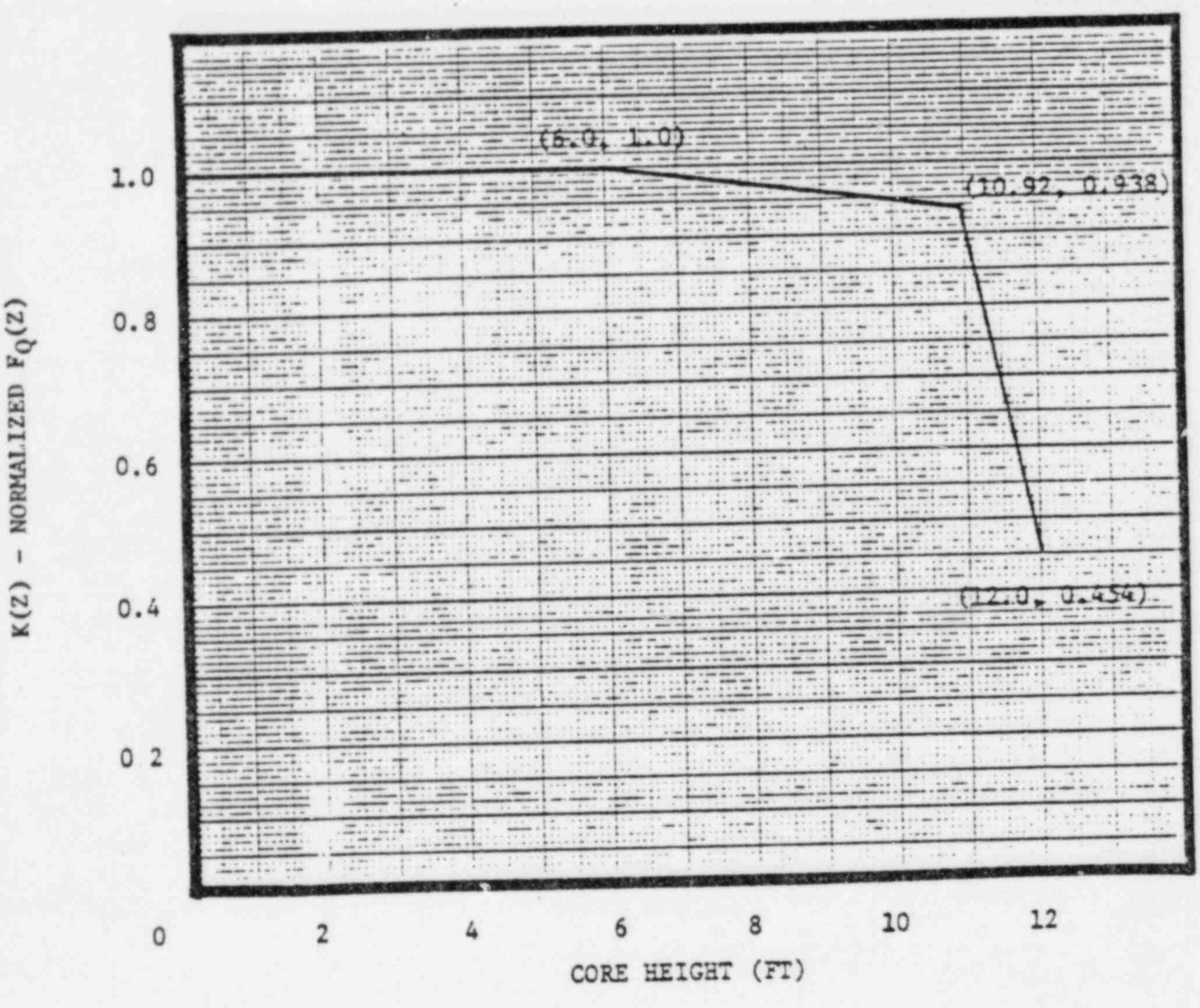


Figure 3.2-2 K(Z) - Normalized F_Q(Z) as a Function of Core Height

TABLE 3.2-1

DNB PARAMETERS

PARAMETER	LIMITS: 3 Loops In Operation	LIMITS: 2 Loops In Operation** & Loop Stop Valves Open	LIMITS: 2 Loops In Operation ** & Isolated Locp Stop Valves Closed
Reactor Coolant System T_{avg}	$\leq 592°F$		
Pressurizer Pressure	≥ 2205 psig*		
Reactor Coolant System Total Flow Rate	$\geq 285,000$ gpm		

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

**Values dependent on NRC approval of ECCS evaluation for these conditions.

POWER DISTRIBUTION LIMITS

AXIAL POWER DISTRIBUTION

LIMITING CONDITION FOR OPERATION

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

$$[F_j(Z)]_S = \frac{[2.20] [K(Z)]}{(\bar{R}_j)(P_L)(1.03)(1 + \sigma_j)(1.07)}$$

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 a given core height location.

d. $\bar{R}_j$, for thimble j, is determined from at least n=6 incore flux maps covering the full configuration of permissible rod patterns above P_m% of RATED THERMAL POWER in accordance with:

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

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 had a measured peaking factor without uncertainties or densification allowance of F_Q^{Meas}.

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 minimum DNBR in the core greater than or equal to 1.30 during normal operation and in short term transients, and (b) limiting the fission gas release, fuel pellet temperature & 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 powe to the average rod power.

$F_{xy}(Z)$ Radial Peaking Factor, is defined as the ratio of peak power density to average power density in the horizontal plane at core elevation Z.

3/4.2.1 AXIAL FLUX DIFFERENCE (AFD)

The limits on AXIAL FLUX DIFFERENCE assure that the $F_Q(Z)$ upper bound envelope, as given in Specification 3.2.2, is not exceeded during either normal operation or in the event of xenon redistribution following power changes.

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 mutiplying 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.

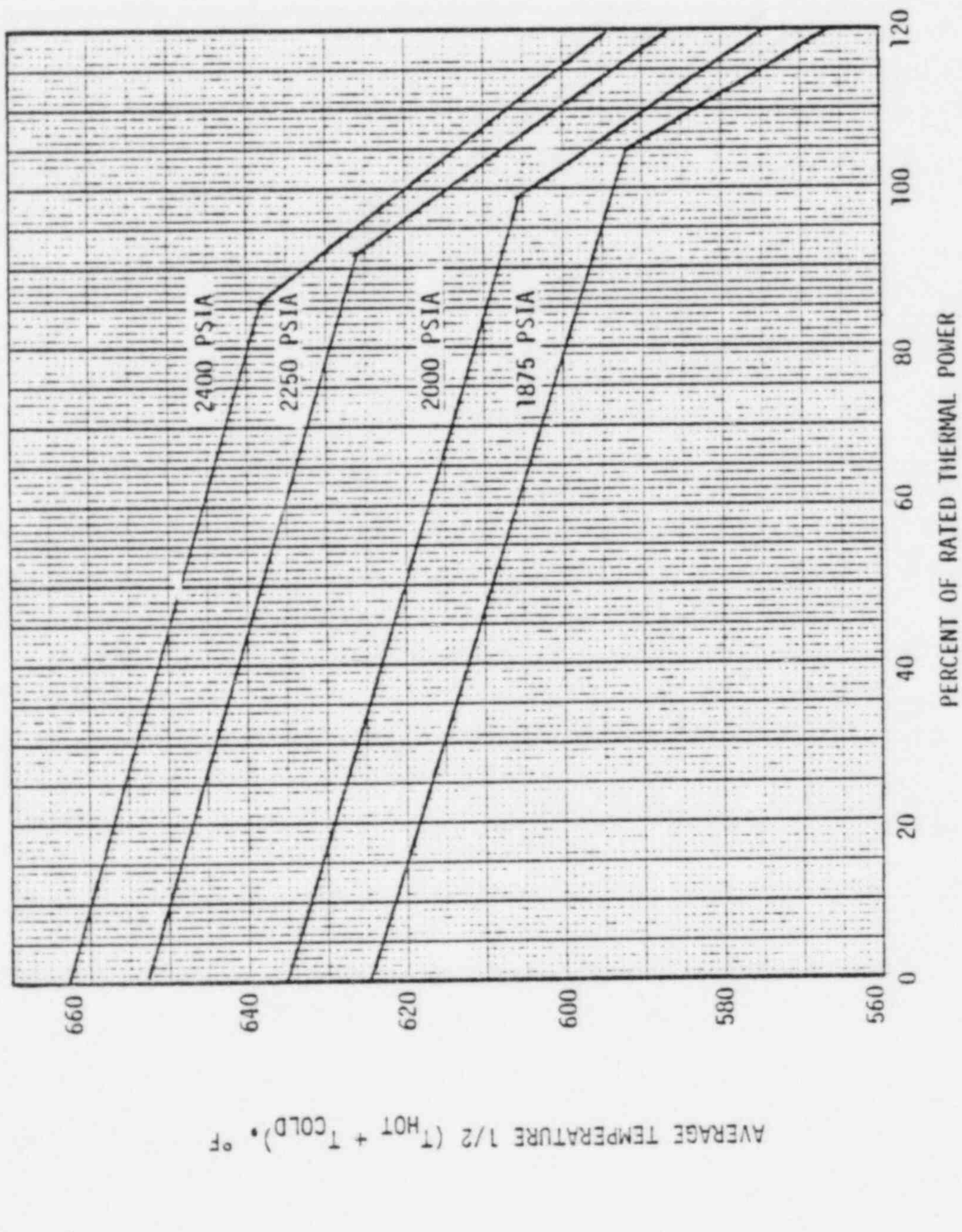


FIGURE 2.1-1 REACTOR CORE SAFETY LIMITS FOR THREE LOOP OPERATION, 100% FLOW

TABLE 2.2-1

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
1. Manual Reactor Trip	Not Applicable	Not Applicable
2. Power Range, Neutron Flux	Low Setpoint - $\leq$ 25% of RATED THERMAL POWER High Setpoint - $\leq$ 109% of RATED THERMAL POWER	Low Setpoint - $\leq$ 26% of RATED THERMAL POWER High Setpoint - $\leq$ 110% of RATED THERMAL POWER
3. Power Range, Neutron Flux, High Positive Rate	$\leq$ 5% of RATED THERMAL POWER with a time constant $\geq$ 2 seconds	$\leq$ 5.5% of RATED THERMAL POWER with a time constant $\geq$ 2 seconds
4. Power Range, Neutron Flux, High Negative Rate	$\leq$ 5% of RATED THERMAL POWER with a time constant $\geq$ 2 seconds	$\leq$ 5.5% of RATED THERMAL POWER with a time constant $\geq$ 2 seconds
5. Intermediate Range, Neutron Flux	$\leq$ 25% of RATED THERMAL POWER	$\leq$ 30% of RATED THERMAL POWER
6. Source Range, Neutron Flux	$\leq 10^5$ counts per second	$\leq 1.3 \times 10^5$ counts per second
7. Overtemperature ΔT	See Note 1	See Note 3
8. Overpower ΔT	See Note 2	See Note 3
9. Pressurizer Pressure--Low	$\geq$ 1870 psig	$\geq$ 1860 psig
10. Pressurizer Pressure--High	$\leq$ 2385 psig	$\leq$ 2395 psig
11. Pressurizer Water Level--High	$\leq$ 92% of instrument span	$\leq$ 93% of instrument span
12. Loss of Flow	$\geq$ 90% of design flow per loop*	$\geq$ 89% of design flow per loop*

*Design flow is 95,000 gpm per loop.

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

NOTATION

NOTE 1: Overtemperature $\Delta T \leq \Delta T_o \left[K_1 - K_2 \left(\frac{1+\tau_1 S}{1+\tau_2 S}\right)(T-T') + K_3(P-P') - f_1(\Delta I)\right]$

where: ΔT_o = Indicated ΔT at RATED THERMAL POWER

T = Av rage temperature, °F

T' = Indicated T_{avg} at RATED THERMAL POWER $\leq$ 587.8°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 = 25 secs, τ_2 = 4 secs.

S = Laplace transform operator (sec^{-1})

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

NOTATION (Continued)

Operation with 3 Loops	Operation with 2 Loops (no loops isolated)*	Operation with 2 Loops (1 loop isolated)*
K_1 = 1.078	K_1 = ( )	K_1 = ( )
K_2 = 0.0143	K_2 = ( )	K_2 = ( )
K_3 = 0.000674	K_3 = ( )	K_3 = ( )

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 - 36 percent and + 11 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 - 36 percent, the ΔT trip setpoint shall be automatically reduced by 1.2 percent of its value at RATED THERMAL POWER.

(iii) for each percent that the magnitude of ($q_t - q_b$) exceeds + 11 percent, the ΔT trip setpoint shall be automatically reduced by 1.6 percent of its value at RATED THERMAL POWER.

*Values dependent on NRC approval of ECCS evaluation for these operating conditions.

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 THERMAL POWER

T = Average temperature, °F

$T"$ = Indicated T_{avg} at RATED THERMAL POWER $\leq$ 587.8°F

K_4 = 1.091

K_5 = 0.02/°F for increasing average temperature

K_5 = 0 for decreasing average temperatures

K_6 = 0.00126 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 (sec^{-1})

$f_2(\Delta I)$ = 0 for all ΔI

Note 3: The channel's maximum trip point shall not exceed its computed trip point by more than 2 percent span.

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:

$$F_Q(Z) \leq \left[\frac{2.20}{P}\right] K(Z) \text{ for } P > 0.5$$

$$F_Q(Z) \leq [4.40] \; K(Z) \text{ for } P \leq 0.5$$

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

and K(Z) is the function obtained from Figure 3.2-2 for a given core height location.

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 similarly 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 $\bar{R}$.

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.

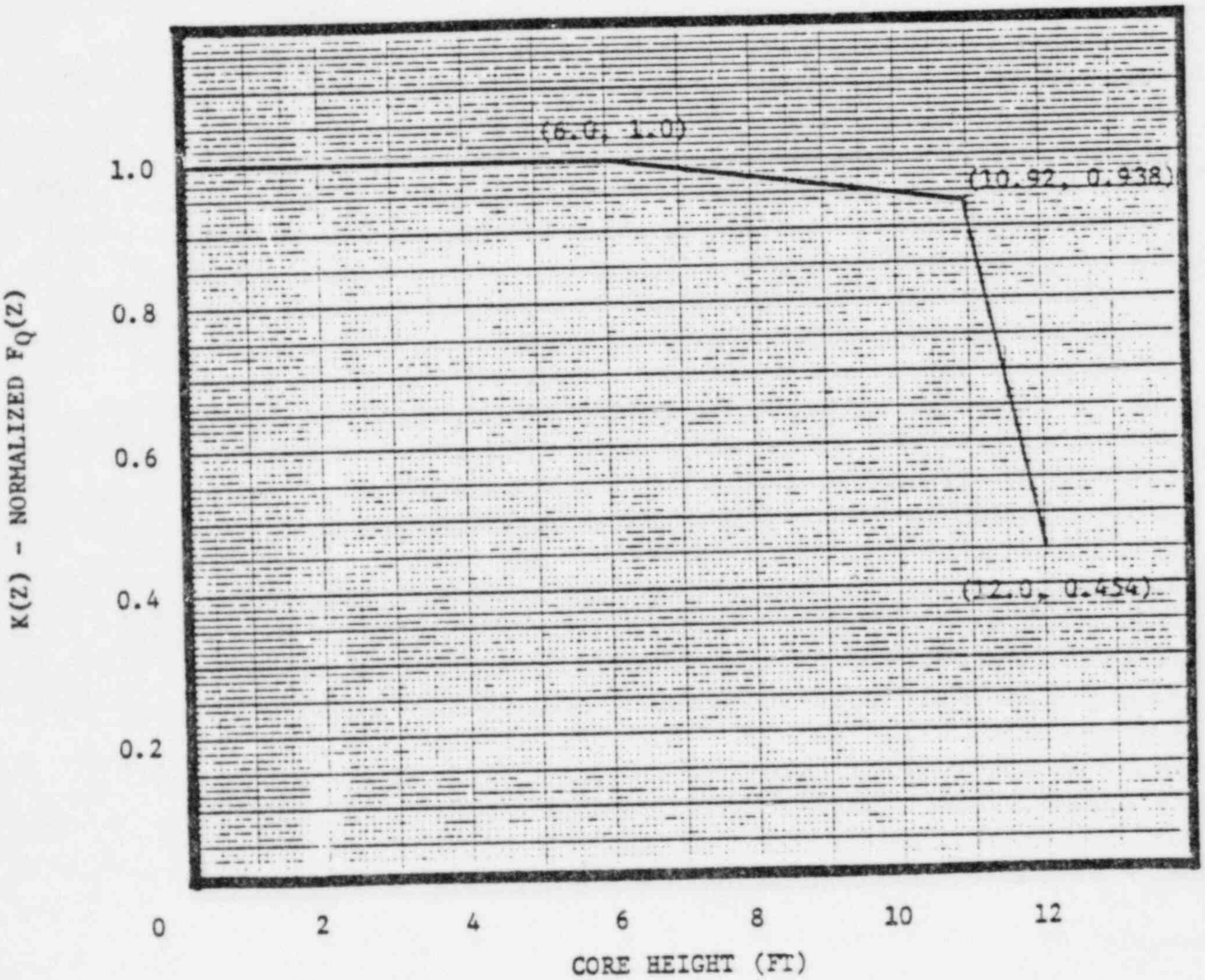


Figure 3.2-2 K(Z) - Normalized $F_Q(Z)$ as a Function of Core Height

TABLE 3.2-1

DNB PARAMETERS

PARAMETER	LIMITS: 3 Loops In Operation	LIMITS: 2 Loops In Operation** & Loop Stop Valves Open	LIMITS: 2 Loops In Operation ** & Isolated Loop Stop Valves Closed
Reactor Coolant System T_{avg}	$\leq$ 592°F		
Pressurizer Pressure	$\geq$ 2205 psig*		
Reactor Coolant System Total Flow Rate	$\geq$ 285,000 gpm		

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

**Values dependent on NRC approval of ECCS evaluation for these conditions.

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

$$[F_j(Z)]_S = \frac{[2.20]\ [K(Z)]}{(\overline{R}_j)(P_L)(1.03)(1+\sigma_j)(1.07)}$$

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 a given core height location.

d. $\overline{R}_j$, for thimble j, is determined from at least n=6 incore flux maps covering the full configuration of permissible rod patterns above P_m% 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

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 minimum DNBR in the core greater than or equal to 1.30 during normal operation and in short term transients, and (b) limiting the fission gas release, fuel pellet temperature & 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.

$F_{xy}(Z)$ Radial Peaking Factor, is defined as the ratio of peak power density to average power density in the horizontal plane at core elevation Z.

3/4.2.1 AXIAL FLUX DIFFERENCE (AFD)

The limits on AXIAL FLUX DIFFERENCE assure that the $F_Q(Z)$ upper bound envelope, as given in Specification 3.2.2, is not exceeded during either normal operation or in the event of xenon redistribution following power changes.

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

ENCLOSURE 4

Core Surveillance Reports

for

North Anna 1 Cycle 4

and

North Anna 2 Cycle 2

TABLE 1

NORTH ANNA UNIT 1, CYCLE 4 CORE SURVEILLANCE LIMITS, FQ = 2.20

I. The F-XY limits for RATED THERMAL POWER within specific core planes shall be:

1. Fxy-RTP $\leq$ 1.71 for all core planes containing bank "D" control rods, and

2. Fxy-RTP $\leq$ 1.65 for all unrodded core planes between 15 % and 25 % of core height, or

3. Fxy-RTP $\leq$ 1.70 for all unrodded core planes between 25 % and 55 % of core height, or

4. Fxy-RTP $\leq$ 1.65 for all unrodded core planes between 55 % and 85 % of core height.

II. The axial power distribution surveillance threshold power level shall be:

1. Pm = 100% of RATED THERMAL POWER.

NORTH ANNA UNIT 1 CYCLE 4

MAXIMUM FQ-TOTAL*P VS. AXIAL CORE HEIGHT
DURING NORMAL CORE OPERATION

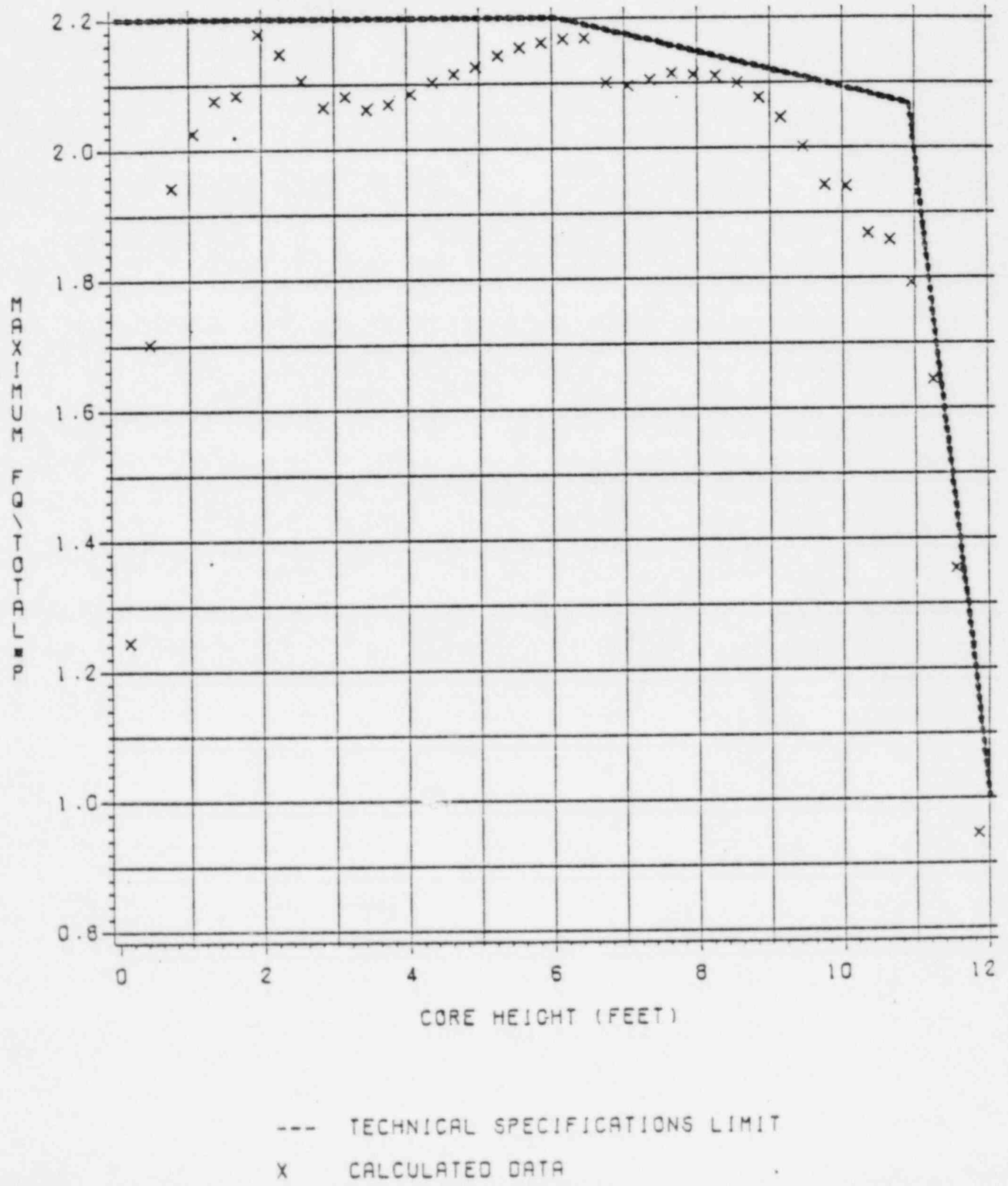


TABLE 1

NORTH ANNA UNIT 2, CYCLE 2 CORE SURVEILLANCE LIMITS, FQ = 2.20

I. The F-XY limits for RATED THERMAL POWER within specific core planes shall be:

1. Fxy-RTP $\leq$ 1.71 for all core planes containing bank "D" control rods, and

2. Fxy-RTP $\leq$ 1.65 for all unrodded core planes between 15 % and 25 % of core height, or

3. Fxy-RTP $\leq$ 1.70 for all unrodded core planes between 25 % and 55 % of core height, or

4. Fxy-RTP $\leq$ 1.65 for all unrodded core planes between 55 % and 85 % of core height.

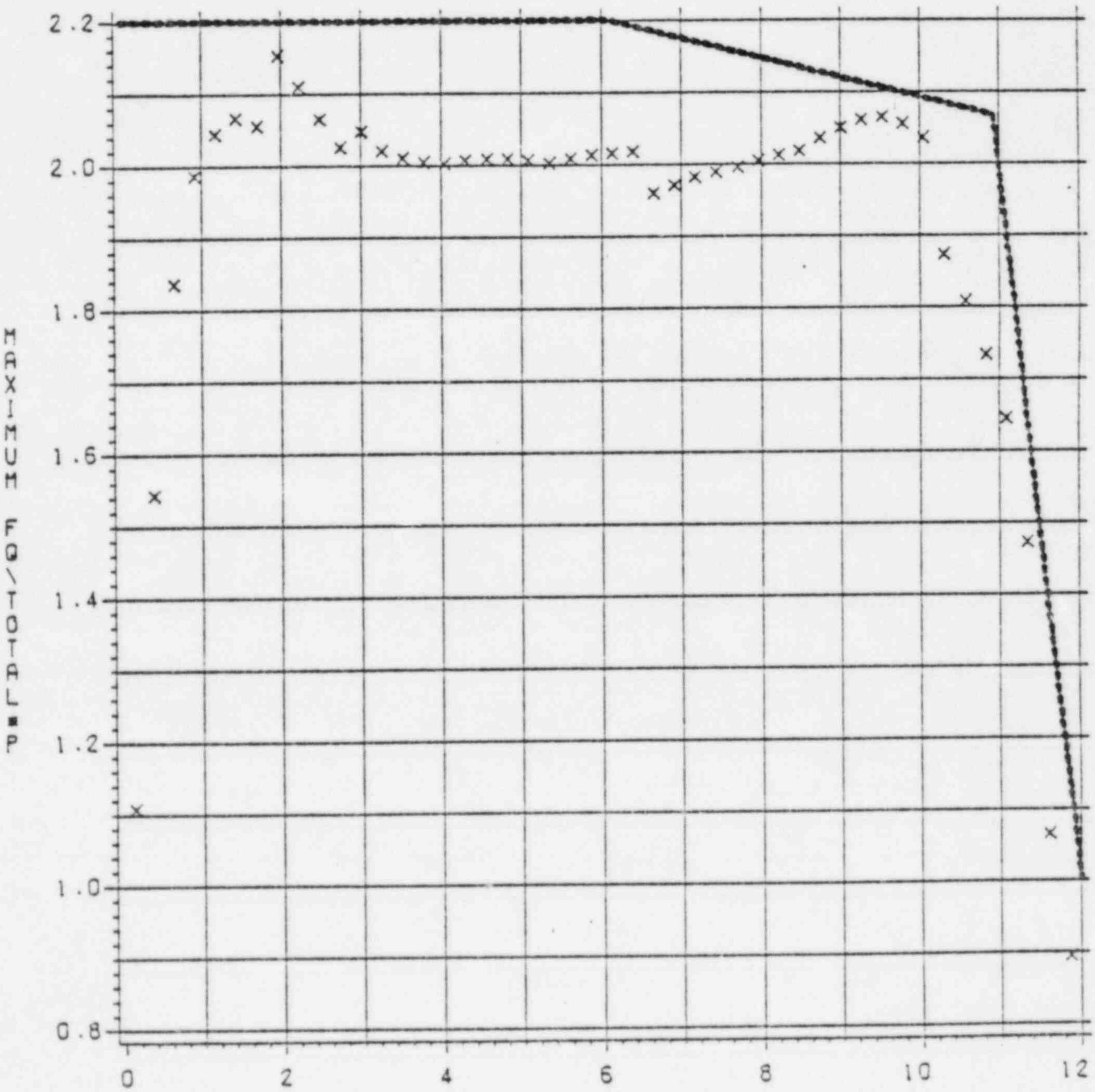
II. The axial power distribution surveillance threshold power level shall be:

1. Pm = 100% of RATED THERMAL POWER.

NORTH ANNA UNIT 2 CYCLE 2

MAXIMUM FQ-TOTAL*P VS. AXIAL CORE HEIGHT

DURING NORMAL CORE OPERATION



--- TECHNICAL SPECIFICATIONS LIMIT

X CALCULATED DATA