ENCLOSURE 1

RELOAD SAFETY EVALUATION
DIABLO CANYON POWER PLANT
UNIT 2 CYCLE 2

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Edited by
S. L. Davidson

Contributors:

M. W. Fecteau
R. E. Radcliffe
I. K. Paik
L. A. Wooten
P. J. Larouere
G. W. Stovar
J. J. Akers
R. Howard
J. Andrachek
C. Thompson

Approved: Edward A. Dzenis
E. A. Dzenis, Manager
Core Operations
Commercial Nuclear Fuel Division

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1.0 INTRODUCTION AND SUMMARY

1.1 INTRODUCTION

This report presents an evaluation for the Diablo Canyon Unit 2, Cycle 2, nuclear plant and demonstrates that the core reload will not adversely affect the safety of the plant. This evaluation was accomplished utilizing the methodology described in WCAP-9273-A, "Westinghouse Reload Safety Evaluation Methodology", Reference 1. For Cycle 2, the core design incorporates a change in the allowable $F^N_{\Delta H}$ as a function of power, a positive moderator temperature coefficient (PMTC), and Relaxed Axial Offset Control (RAOC). NRC approval has been received for the PMTC Licensing Amendment, Reference 10. In addition, NRC approval has been recently received for RAOC, Reference 12.

Based upon the above referenced methodology, only those incidents comprising the licensing basis, which could potentially be affected by this fuel reload have been reviewed for the Cycle 2 design described herein. The justification for the applicability of previous results is presented in Sections 3.1 and 3.2 of this report.

Evaluations/analysis for this RSE have considered plant operations with the Boron Injection Tank (BIT) operable with a concentration of 20,000 ppm as required by the Technical Specifications.

1.2 GENERAL DESCRIPTION

The Diablo Canyon Unit 2 reactor core consists of 193 fuel assemblies arranged in the core loading pattern configuration shown in Figure 1. During the Cycle 1/2 refueling, 68 fuel assemblies will be replaced with fresh Region 4A and 4B fuel. A summary of the Cycle 2 fuel inventory is given in Table 1.

Nominal core design parameters utilized for Cycle 2 are as follows:

Core Power (MWt)	3411
System Pressure (psia)	2250
Core Inlet Temperature (°F)	545.0
NSSS Thermal Design Flow (gpm)	354,000
Average Linear Power Density (kw/ft)	5.45*

The maximum Cycle 2 burnup at the end-of-full power capability (EOFPC)** is predicted to be 14,350 MWD/MTU for the anticipated Cycle 1 burnup of 14,200 MWD/MTU.

1.3 CONCLUSIONS

Evaluations based on the current Diablo Canyon Technical Specifications have concluded that the Cycle 2 core design does not satisfy the post-LOCA long term core cooling requirement, if Cycle 1 burnup is less than 15250 MWD/MTU. If the Cycle 1 burnup is less than 15250 MWD/MTU, then raising the minimum Technical Specification boron concentration for the RWST to 2300 ppm and the accumulators to 2200 ppm will meet the post-LOCA subcriticality requirement.

*Linear power density based on hot average fuel length (143.7 in.)

**Definition with control rods fully withdrawn and approximately 0-10 ppm of residual boron at the Cycle 2 3411 MWt rated reactor power conditions.

From the evaluation presented in this report, it is concluded that the Cycle 2 redesign does not cause the previously acceptable safety limits for any incident to be exceeded. This conclusion is based on the following:

1. Cycle 1 shutdown at a burnup between 13,500 MWD/MTU and 15,450 MWD/MTU.

2. Cycle 2 burnup is limited to a maximum burnup of 500 MWD/MTU beyond the end-of-full power capability.

3. There is adherence to plant operating limitations given in the Technical Specifications and the proposed changes given in Appendix A.

2.0 REACTOR DESIGN

2.1 MECHANICAL DESIGN

The mechanical design of the Region 4A and 4B fuel assemblies is the same as the Region 2 and 3 fuel assemblies, except for the use of (1) chamfered pellets, (2) 4g plenum spring (3) 304L stainless steel top and mid-grid sleeve material, (4) use of a fuel rod end plug with an internal gripper design, (5) grid straps with strap and corner modifications, and (6) a bottom nozzle reconstitution feature. These modifications are described below and do not impact the safe operation of the Region 4A and 4B fuel assemblies.

(1) The Region 4A and 4B pellets will have a small chamfer at the outer edge of the fuel pellet ends and a reduction in the dish diameter and depth compared to the previous unchamfered fuel pellets. The chamfer will improve pellet chip resistance during manufacturing and handling. All fuel rod design criteria are satisfied.

(2) Region 4 fuel has a smaller rod plenum spring than was used in previous fuel regions. This new spring design satisfies a change in the non-operational 6g loading design criterion to "4g axial and 6g lateral loading with dimensional stability." Notification of Westinghouse's plans to generically incorporate this criterion change and the justification of no unreviewed safety questions were previously transmitted to the NRC via Reference 11. The reduced spring force further reduces the already low potential for chamfered pellet chipping in the fuel rod.

(3) The change in top and mid-grid sleeve material from 304 stainless steel to 304L steel further reduces the already low potential for stress corrosion cracking of the grid sleeves.

(4) The bottom end plugs were modified with an internal gripper device to facilitate rod insertion during fabrication and for post irradiation rod removal.

(5) The grid straps were modified to prevent assembly hangup from grid strap interference during fuel assembly removal. This was accomplished by machining the grid strap corner geometry and the addition of an extra tab on the outer grid strap.

(6) The bottom nozzle design has a reconstitution (nozzle removal) feature that permits remote unlocking, removing and relocking of the thimble screws as a new or the same bottom nozzle is reattached without damaging the fuel assembly integrity.

Table 1 compares pertinent design parameters of the various fuel regions. The Region 4A and 4B fuel has been designed according to the fuel performance model in Reference 3. The fuel is designed to operate so that clad flattening will not occur, as predicted by the Westinghouse model, Reference 4. For all fuel regions, the fuel rod internal pressure design basis, which is discussed and shown acceptable in Reference 5, is satisfied.

Westinghouse has had considerable experience with Zircaloy clad fuel. This experience is extensively described in WCAP-8183, "Operational Experience with Westinghouse Cores," Reference 6.

2.2 NUCLEAR DESIGN

The Cycle 2 core loading is designed to meet the F_Q^T x P ECCS limit of $\leq$ 2.32 x K(Z) (Reference 15a) for a flux difference (ΔI) bandwidth during normal operation conditions of +6, -13 percent ΔI. Appendix A contains the K(Z) curve derived from the new small break LOCA analysis, Reference 15b, having the traditional third line segment removed. Removal of the third line segment allows greater flexibility in the design of cores.

Relaxed Axial Offset Control (RAOC) including enhanced load follow and reduced temperature return to power operating strategies will be employed in Cycle 2 to enhance operational flexibility. RAOC makes use of available margin by expanding the allowable ΔI band, particularly at reduced power. The RAOC methodology and application is fully described in Reference 7. The analysis for Cycle 2 indicates that no change to the safety parameters is required for RAOC operation. The new K(Z) curve, contained in Appendix A, was used in the determination that RAOC operation did not require a change to any safety parameters.

Table 2 provides a summary of changes in the Cycle 2 kinetics characteristics compared with the current limit based on previously submitted accident analyses, References 2 and 14.

Table 3 provides the control rod worths and requirements at the most limiting condition during the cycle. The required shutdown margin is based on the previously submitted accident analyses. The available shutdown margin exceeds the minimum required.

The loading contains a total of 352 fresh burnable absorber rods located in 40 Region 4 assemblies and 160 spent burnable absorber rods, 64 of which are located in 4 Region 3 assemblies, and the remaining 96 located in 8 Region 4B assemblies. The locations of the burnable absorber and source rods are shown in Figure 1.

Appendix B contains the Radial Peaking Factor Limit Report in accordance with paragraph 6.9.1.8 of the Diablo Canyon Technical Specifications.

2.3 THERMAL AND HYDRAULIC DESIGN

No significant variations in thermal margins will result from the Cycle 2 reload. The DNB core limits and the safety analyses used for Cycle 2 are based on the conditions given in Sections 1.0 and 4.0. Since the $F^N_{\Delta H}$ multiplier was changed from 0.2 to 0.3, new core limits were established. Fuel temperatures were calculated using the revised thermal safety model, described in Reference 13, and include the effects of chamfered pellets. Steady-state DNBR calculations are not affected by the revised fuel temperatures.

3.0 POWER CAPABILITY AND ACCIDENT EVALUATION

3.1 POWER CAPABILITY

The plant power capability has been evaluated considering the consequences of those incidents examined in the FSAR (Reference 2), and the consequences of the incidents considering the PMTC change approved in Reference 10. It is concluded that the core reload will not adversely affect the ability to safely operate at 100 percent of 3411 MWt rated reactor power during Cycle 2. The fuel centerline temperature limit of 4700°F can be accommodated with margin in the Cycle 2 core using the methodology described in Reference 1. The time dependent densification model in Reference 8, was used for these fuel temperature evaluations. The LOCA limit at rated power can be met by maintaining F_Q at or below 2.32. The impact of using chamfered fuel pellets has been considered and found to be acceptable. Therefore, the current analyses of record, References 2 and 14, remain applicable when using chamfered fuel pellets.

3.2 ACCIDENT EVALUATION

The effects of the reload on the design bases and postulated incidents were examined. In all cases it was found that the effects were accommodated within the conservatism of the initial assumptions used in the previous applicable safety analyses or in the safety analyses performed in support of the PMTC and the revised $F^N_{\Delta H}$. The fission product inventory resulting from the use of a nominal 4.5 w/o U-235 fuel is not significantly different, Reference 16, from the isotopic inventory which was provided by Westinghouse for the first cycle of operation under normal and accident conditions. Any deviations from the original isotopic inventory are insignificant and are within the uncertainty level of the accident calculation. Therefore, there is no need to recalculate the radiological consequences of any accident due to the increase in fuel enrichment.

The effects of the reload on the design basis and postulated incidents analyzed in the FSAR (Reference 2) were examined with the BIT. With the exception of the post-LOCA subcritical cooling requirement, it was found that the effects were accommodated within the conservatism of the assumptions used in the previous applicable safety analyses.

A safety criterion that the reactor core remain subcritical on the soluble boron provided by the ECCS following a hypothetical large break LOCA has been evaluated for the Cycle 2 design. This criterion is met assuming Cycle 1 burnup extends past 15,250 MWD/MTU, but is not satisfied for end-of-cycle (EOC) burnups less than 15,250 MWD/MTU. This conclusion is based on using minimum Technical Specifications values for the boron concentrations of the RWST and accumulators, RCS boron concentration based on HFP all-rods-out with peak xenon, and boron concentration in the Boron Injection Tank (BIT) assuming the BIT is active.

Evaluations based upon the current minimum Technical Specifications value for the RWST (2000 ppm), the accumulators (1900 ppm) and upon the core reactivity assuming an end of Cycle 1 burnup of less than 15,250 MWD/MTU, have concluded that the Cycle 2 design does not satisfy the long term core cooling requirement that the reactor remain subcritical on the soluble boron provided by the ECCS. Should the Cycle 1 burnup be less than 15,250 MWD/MTU, then raising the boron concentration minimum Technical Specification limits to 2300 ppm for the RWST and 2200 ppm for the accumulators will provide sufficient soluble boron to satisfy the post-LOCA subcriticality requirement.

A core reload can typically affect accident analysis input parameters in the following areas: core kinetic characteristics, control rod worths, and core peaking factors. Cycle 2 parameters in each of these three areas were examined as discussed below to ascertain whether new accident analyses were required.

3.2.1 KINETICS PARAMETERS

A comparison of Cycle 2 kinetic parameters with current limits is given in Table 2. The current limits reflect parameters used in the previously applicable safety analyses (FSAR) Reference 2 or in the approved PMTC Licensing Amendment No. 8, Reference 10. The exception to the moderator temperature coefficient limit has already been demonstrated to be acceptable in Reference 14. Thus, no further reanalysis was necessary. The most negative Doppler temperature coefficient is -2.9 pcm/°F for Cycle 2 compared to the current limit of -2.0 pcm/°F. This difference was evaluated and was found to result in a negligible effect on all of the transient analyses.

3.2.2 CONTROL ROD WORTHS

Changes in control rod worths may affect differential rod worths, shutdown margin, ejected rod worths, and trip reactivity. Table 2 shows that the maximum differential rod worth of two RCCA control banks moving together in their highest worth region for Cycle 2 meets the current limit. Table 3 shows that the Cycle 2 shutdown margin requirements are satisfied.

Cycle 2 has a normalized trip reactivity insertion rate which is slightly different than the current limit, Reference 2. The effects of this reduced normalized trip reactivity rate have been evaluated for those accidents affected and compared to previous analyses. Fast transients are evaluated to confirm that the limiting transient conditions are unchanged. Slow transients are relatively insensitive to trip reactivity insertion rate and are investigated only for increases in total energy release from the fuel to the coolant after the trip. The reload rod worth versus position is less than the current limit rod worth versus position between 0 to approximately 15 percent of rod insertion and also after 60 percent of rod insertion. An investigation of the affected transients has shown that these effects will not change the conclusions of either the FSAR or Reference 14; therefore, no reanalysis was performed.

3.2.3 CORE PEAKING FACTORS

Peaking factors for the dropped RCCA incidents were evaluated based on the approved dropped rod methodology described in Reference 9. Evaluation of peaking factors for dropped RCCA shows that the DNBR limit value is not violated.

3.3 INCIDENTS REANALYZED

Because of a change in the part power multiplier from 0.2 to 0.3 in the $F^N_{\Delta H}$ equation, the core limits changed for this reload. This resulted in a need to change the overtemperature ΔT trip setpoint equation. The rod withdrawal at power and the loss of Load/Turbine Trip analyses were reanalyzed to determine the effect of these changes on the conclusions presented in the FSAR, Reference 2 and the PMTC submittal, Reference 14. The conclusions of the FSAR and Reference 14 for these accidents remain valid for the reload.

The boron dilution accident at hot full power was reanalyzed because of a change in the boron reactivity worth times concentration for the reload. The reload value increased to a maximum of 16,000 pcm when compared to the previous analysis limit of 14,156 pcm. The analysis was performed using a shutdown margin of 1.6% ΔK. The analysis has shown that the effect of the new reload value will not change the conclusions of either the FSAR or Reference 14.

Rod ejection was reanalyzed for this reload because of a change to the least negative doppler only power defect at the beginning of Cycle 2 and the ejected rod worth and power peaking factor for the end of life hot zero power case. The investigation determined that the conclusions reached in the FSAR and Reference 14 for this accident do not change for this reload.

4.0 TECHNICAL SPECIFICATION CHANGES

To insure that plant operation is consistent with the design and safety evaluation conclusion statements made in this report and to ensure that these conclusions remain valid, several Technical Specification Changes are required for Cycle 2 operation. These changes are given in Appendix A.

If the Cycle 1 burnup is less than 15,250 MWD/MTU, then the RWST and the Accumulator boron concentration Technical Specifications lower limits must be raised in order to satisfy the post-LOCA subcriticality requirement. Proposed Technical Specifications to raise the boron concentrations of the RWST and Accumulator to ensure that post-LOCA subcritical requirements will be satisfied are includec in Appendix A.

5.0 REFERENCES

1. Davidson, S. L. (Ed.), et al., "Westinghouse Reload Safety Evaluation Methodology," WCAP-9273-A, July 1985.

2. Diablo Canyon Updated Final Safety Analysis Report, USNRC Docket Number 50-275.

3. Miller, J. V. (Ed.), "Improved Analytical Model used in Westinghouse Fuel Rod Design Computations," WCAP-8785, October 1976.

4. George, R. A., et al., "Revised Clad Flattening Model," WCAP-8377 (Proprietary) and WCAP-8381, (Non-Proprietary) July 1974.

5. Risher, D. H., et al., "Safety Analysis for the Revised Fuel Rod Internal Pressure Design Basis," WCAP-8964-A (Non-Proprietary), August 1978.

6. Skaritka, J., "Operational Experience with Westinghouse Cores, (through December 31, 1985)" issued September 1986, to NRC (NS-NRC-86-3814).

7 Miller, R. W., et al., "Relaxation of Constant Axial Offset Control, F_Q Surveillance Technical Specification," WCAP-10216-P-A, June 1983.

8. Hellman, J. M., (Ed.), "Fuel Densification Experimental Results and Model for Reactor Operation," WCAP-8219-A, March 1975.

9. Morita, T., Osborne, M. P., et al., "Dropped Rod Methodology for Negative Flux Rate Trip Plants," WCAP-10297-P-A (Proprietary) and WCAP-10298-A (Non-Proprietary), June 1983.

10. Diablo Unit 2, Licensing Amendment No. 8 (PMTC) to DPR-82, October 1986.

11. Letter from E. P. Rahe, Jr. (Westinghouse) to L. E. Phillips (NRC) dated April 12, 1984, NS-EPR-2893, Subject: Fuel Handling Load Criteria (6g vs 4g).

12. Diablo Unit 2, Licensing Amendment No. 10, (RAOC) to DPR-82, January 1987.

13. Leech, W. J., et. al., "Revised PAD Code Thermal Safety Model," WCAP-8720, Addenda 2, October 1982.

14. Diablo Unit 1, Cycle 2 Reload and Associated Technical Specification Changes, License Amendment Request 86-07, dated May 1986.

15a. NRC Letter from S. A. Varga to J. D. Shiffer, "Lifting of Exemption from Requirement of 10CFR50; Section 50.46, Allowing Limit to go back to 2.32 Fq," dated January 22, 1987.

15b. Westinghouse Letter to J. D. Shiffer from J. C. Hoebel, "Small Break LOCA Analysis with NOTRUMP," PGE-86-6819, dated November 24, 1986.

16. Davidson, S. L. (Ed.) et. al., "Extended Burnup Evaluation of Westinghouse Fuel," WCAP-10125-P-A, December 1985.

TABLE 1

FUEL ASSEMBLY DESIGN PARAMETERS
DIABLO CANYON 2 - CYCLE 2

Region	2	3	4A	4B
Enrichment (w/o U235)*	2.61	3.09	3.40	3.80
Geometric Density (percent theoretical)*	94.44	94.54	95.0	95.0
Number of Assemblies	61	64	44	24
Approximate Burnup at Beginning of Cycle (MWD/MTU)+	16000	11000	0	0

\+ Based on EOC Cycle 1 of 14,200 MWD/MTU.

* All values are as built except for those Regions 4A and 4B which are nominal values.

TABLE 2

KINETICS CHARACTERISTICS
DIABLO CANYON UNIT 2 - CYCLE 2

	Current Limit Reference (2) (14)	Cycle 2
Moderator Temperature Coefficient (pcm/°F*)	-39.0 to +5.0	-39.0 to +5.0
Doppler Temperature Coefficient (pcm/°F*)	-2.0 to -1.4	-2.9 to -1.4
Least Negative Doppler - Only Power Coefficient, Zero to Full Power (pcm/%power)*	-10.18 to -6.68	-10.18 to -6.68
Most Negative Doppler - Only Power Coefficient, Zero to Full Power (pcm/%power)*	-19.4 to -12.6	-19.4 to -12.6
Delayed Neutron Fraction β_{eff}, (percent)	0.44 to 0.7337	0.44 to 0.7337
β_{eff}, (percent) minimum (BOL Rod ejection only)	0.52	>0.55
Maximum Differential Rod Worth of Two Banks Moving Together at HZP with 100% overlap (pcm/in.)*	89	< 66.6

* pcm = $10^{-5}\ \Delta\rho$

TABLE 3

SHUTDOWN REQUIREMENTS AND MARGINS
END-OF-CYCLE
DIABLO CANYON UNIT 2 - CYCLES 1 AND 2

	Cycle 1	Cycle 2
Control Rod Worth (%Δρ)		
All Rods Inserted	7.54	6.69
All Rods Inserted Less Worst Stuck Rod	6.38	5.90
(1) Less 10%	5.74	5.31
Control Rod Requirements (%Δρ)		
Reactivity Defects (Doppler, T_{avg} Void, Redistribution)	2.93	2.95
Rod Insertion Allowance	0.50	0.50
(2) Total Requirements	3.43	3.45
Shutdown Margin [(1)-(2)] (%Δρ)	2.31	1.86
Required Shutdown Margin (%Δρ)	1.60	1.60

FIGURE 1
Diablo Canyon Unit 2 Cycle 2
Core Loading Pattern

	R	P	N	M	L	K	J	H	G	F	E	D	C	B	A
1					2	3	2	3	2	3	2				
2			2	4A	4B +12	3	4B	3	4B	3	4B +12	4A	2		
3		2	4A	3	3	4B 12	2	3 •16	2	4B 12	3	3	4A	2	
4		4A	3	3	4A 8	2	4A 8	3	4A 8	2	4A 8	3	3	4A	
5	2	4B +12	3	4A 8	2	4A 8	2	4A 8	2	4A 8	2	4A 8	3	4B +12	2
6	3	3	4B 12	2	4A 8	2	3	2	3	2	4A 8	2	4B 12	3	3
7	2	4B	2	4A 8	2	3	3	4A 8	3	3	2	4A 8	2	4B	2
8	3	3	3 +16	3	4A 8	2	4A 8	2	4A 8	2	4A 8	3	3 +16	3	3
9	2	4B	2	4A 8	2	3	3	4A 8	3	3	2	4A 8	2	4B	2
10	3	3	4B 12	2	4A 8	2	3	2	3	2	4A 8	2	4B 12	3	3
11	2	4B +12	3	4A 8	2	4A 8	2	4A 8	2	4A 8	2	4A 8	3	4B +12	2
12		4A	3	3	4A 8	2	4A 8	3	4A 8	2	4A 8	3	3	4A	
13		2	4A	3	3	4B 12	2	3 •16	2	4B 12	3	3	4A	2	
14			2	4A	4B +12	3	4B	3	4B	3	4B +12	4A	2		
15					2	3	2	3	2	3	2				

XX	Region
YY	Number of Burnable Absorber Rods

• Assemblies contain 16 spent burnable absorber rods and four once irradiated secondary source rods

\+ Assemblies contain spent burnable absorber rods

APPENDIX A

TECHNICAL SPECIFICATIONS CHANGE PAGES

The following technical specification changes are required for Diablo Canyon Unit 2, Cycle 2.

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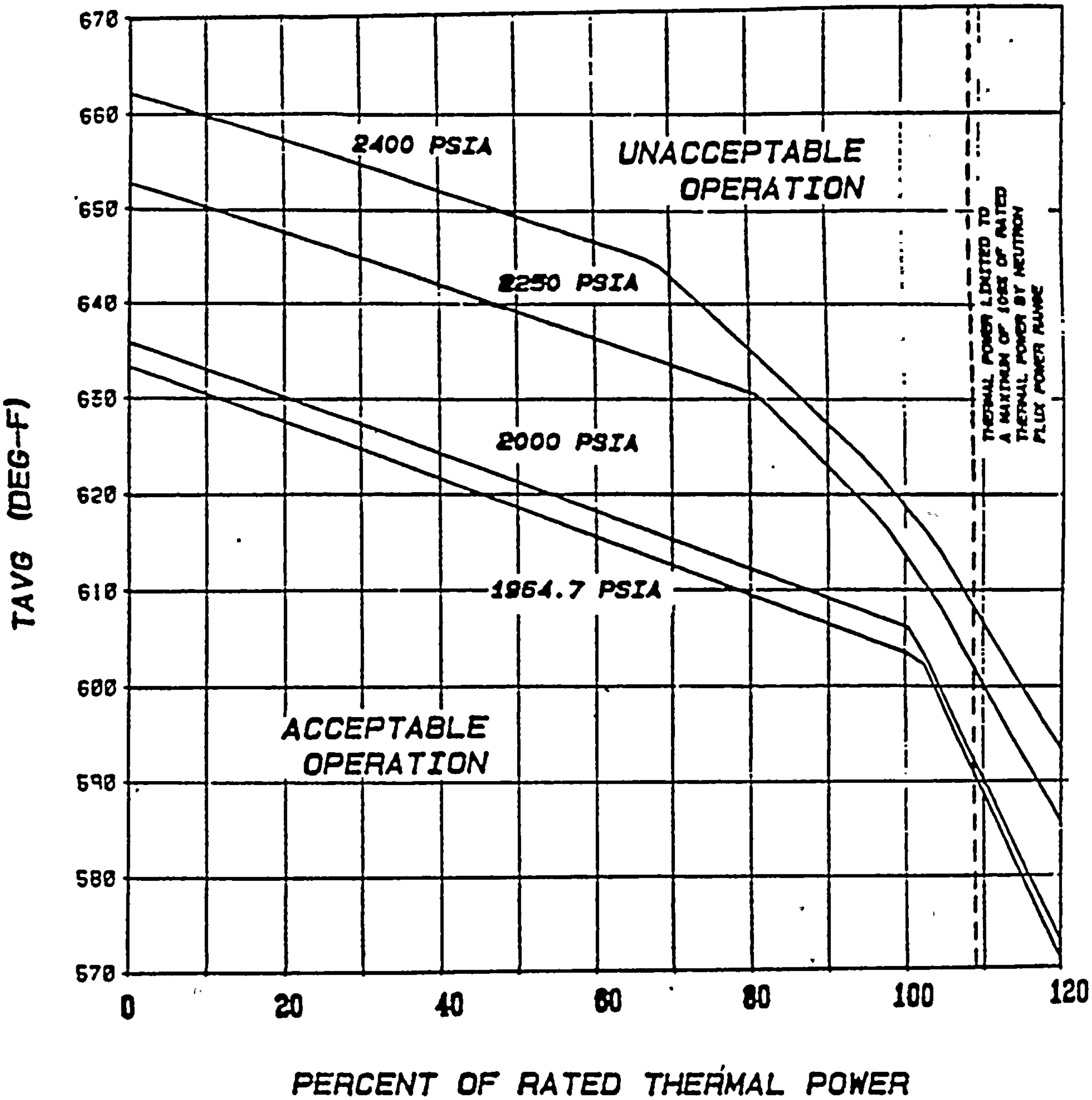

FIGURE 2.1-1b
REACTOR CORE SAFETY LIMIT

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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 through the R-Grid correlation. The R-Grid DNB correlation has been developed to predict the DNB flux and the location of DNB for axially uniform and non-uniform heat flux distributions. The local DNB heat flux ratio (DNBR) is defined as the ratio of the heat flux that would cause DNB at a particular core location to the local heat flux and is indicative of the margin to DNB.

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

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

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

$$F^N_{\Delta H} = 1.55\ [1+ 0.3\ (1-P)]\ \cancel{\text{(Unit 1)}}$$

$$\cancel{F^N_{\Delta H} = 1.55\ [1+ 0.2\ (1-P)]\ \text{(Unit 2)}}$$

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 trip will reduce the Setpoints to provide protection consistent with core Safety Limits.

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

TABLE NOTATIONS

NOTE 1: OVERTEMPERATURE ΔT

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

Where: ΔT_o = Indicated ΔT at RATED THERMAL POWER;

T = Average temperature, °F;

T' = $\leq$ 576.6°F for Unit 1 and $\leq$ 577.6°F for Unit 2 Reference T_{avg} at RATED THERMAL POWER;

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 = 30 s, τ_2 = 4 s;

S = Laplace transform operator, s^{-1};

K_1 = ~~1.174~~; 1.166

K_2 = ~~0.01350/°F~~; 0.011493/°F

K_3 = ~~0.000685/psig~~; 0.00050159 /psig

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

TABLE NOTATIONS (Continued)

NOTE 1 (Continued)

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 - 32% and + ~~10%~~ 9%, 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 - 32%, the ΔT Trip Setpoint shall be automatically reduced by ~~2.11%~~ 2.02% of its value at RATED THERMAL POWER.

(iii) for each percent that the magnitude of ($q_t - q_b$) exceeds + ~~10%~~ 9%, the ΔT Trip Setpoint shall be automatically reduced by ~~1.45%~~ 1.454% of its value at RATED THERMAL POWER.

NOTE 2: The channel's maximum Trip Setpoint shall not exceed its computed Trip Setpoint by more than 4%.

BORATED WATER SOURCE - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.2.5 As a minimum, one of the following borated water sources shall be OPERABLE:

a. A Boric Acid Storage System and at least one associated heat tracing channel with:

   1) A minimum contained borated water volume of 835 gallons,

   2) A boron concentration between 20,000 and 22,500 ppm, and

   3) A minimum solution temperature of 145°F.

b. The Refueling Water Storage Tank (RWST) with:

   1) A minimum contained borated water volume of 50,000 gallons,

   2) A minimum boron concentration of 2000 ppm, and 2300

   3) A minimum solution temperature of 35°F.

APPLICABILITY: MODES 5 and 6.

ACTION:

With no borated water source OPERABLE, suspend all operations involving CORE ALTERATIONS or positive reactivity changes.

SURVEILLANCE REQUIREMENTS

4.1.2.5 The above required borated water source shall be demonstrated OPERABLE:

a. At least once per 7 days by:

   1) Verifying the boron concentration of the water,

   2) Verifying the contained borated water volume, and

   3) Verifying the boric acid storage tank solution temperature when it is the source of borated water.

b. At least once per 24 hours by verifying the RWST temperature when it is the source of borated water and the outside ambient air temperature is less than 35°F.

BORATED WATER SOURCES - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.6 Each of the following borated water source(s) shall be OPERABLE:

a. A Boric Acid Storage System and at least one associated heat tracing channel with:

1) A minimum contained borated water volume of 5106 gallons,

2) A boron concentration between 20,000 and 22,500 ppm, and

3) A minimum solution temperature of 145°F.

b. The Refueling Water Storage Tank (RWST) with:

1) A contained borated water volume of greater than or equal to 400,000 gallons,

2) A boron concentration between ~~2000~~ 2300 and ~~2200~~ 2500 ppm, and

3) A minimum solution temperature of 35°F.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With the Boric Acid Storage System inoperable, restore the system to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and borated to a SHUTDOWN MARGIN equivalent to at least 1% $\Delta k/k$ at 200°F; restore the Boric Acid Storage System to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

b. With the RWST inoperable, restore the tank to OPERABLE status within 1 hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

replace

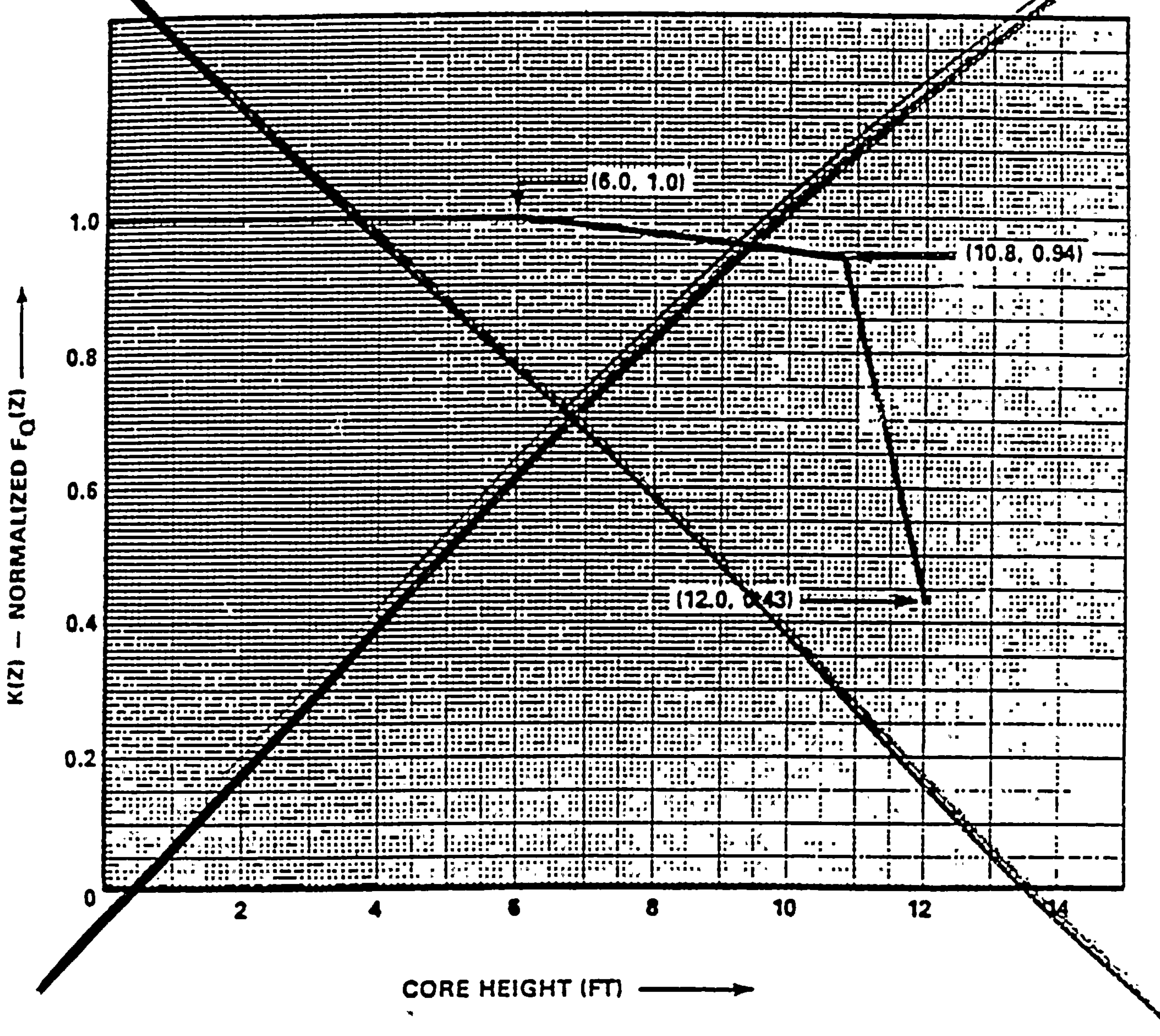
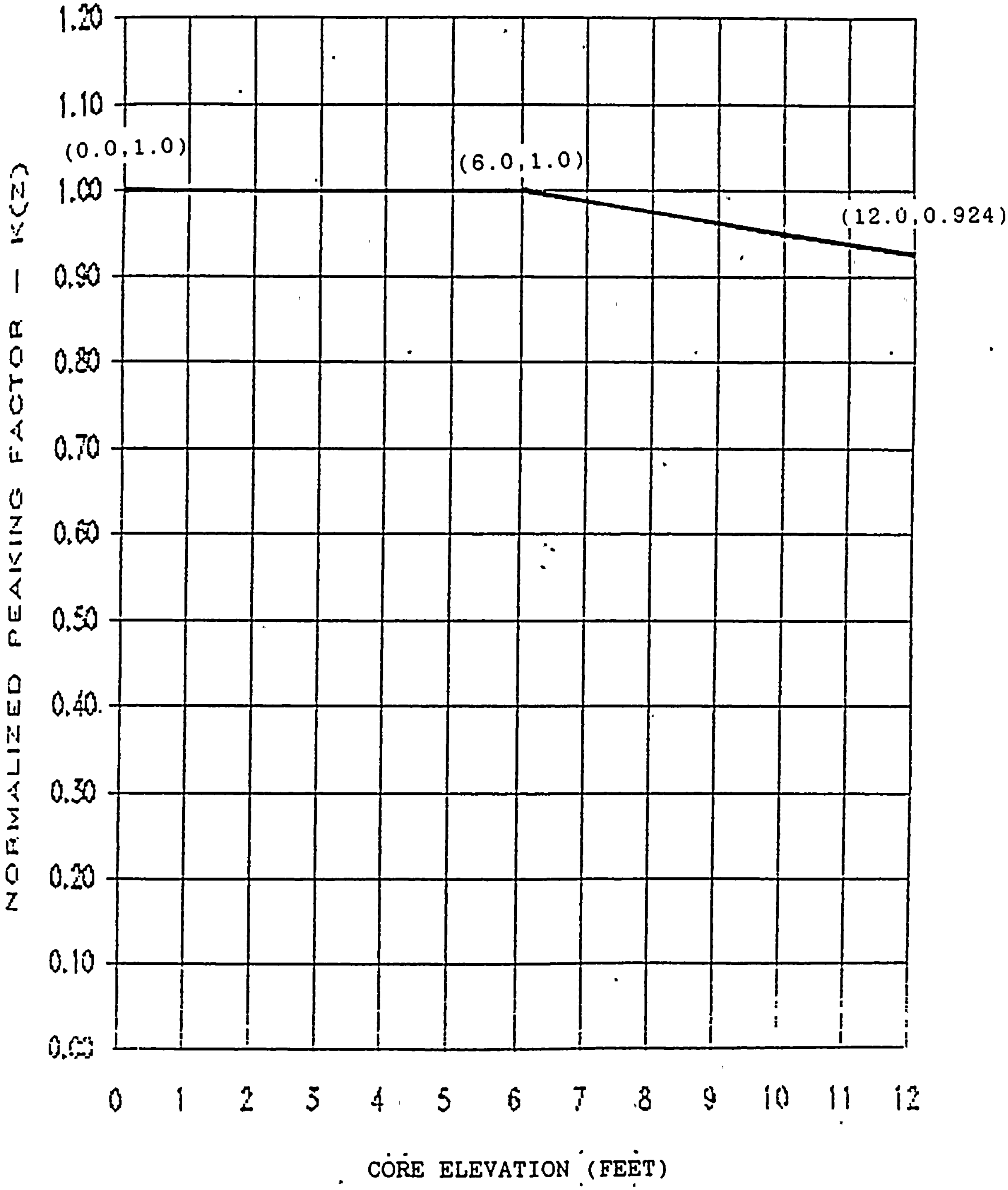


FIGURE 3.2-2
K(Z) - NORMALIZED $F_Q(Z)$ AS A FUNCTION OF CORE HEIGHT

DIABLO CANYON UNIT NO.1 AND NO.2

NORMALIZED FQ(Z) AS A FUNCTION OF CORE HEIGHT

FIGURE 3.2-2



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POWER DISTRIBUTION LIMITS

3/4.2.3 RCS FLOW RATE AND NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR

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 Figure 3.2-3a for Unit 1 and Figure 3.2-3b for Unit 2 for four loop operation.

Where:

a. $R = \dfrac{F^N_{\Delta H}}{1.49\ [1.0 + 0.3\ (1.0 - P)]}$, ~~(Unit 1),~~

~~$R = \dfrac{F^N_{\Delta H}}{1.49\ [1.0 + 0.2\ (1.0 - P)]}$, (Unit 2)~~

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

c. $F^N_{\Delta H}$ = 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 Figure 3.2-3a for Unit 1 and Figure 3.2-3b for Unit 2 include measurement uncertainties of 3.5% for flow and 4% for incore measurement of $F^N_{\Delta H}$.

APPLICABILITY: MODE 1.

ACTION:

With the combination of RCS total flow rate and R outside the region of acceptable operation shown on Figure 3.2-3a for Unit 1 and Figure 3.2-3b for Unit 2:

a. Within 2 hours either:

1. 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 less than or equal to 55% of RATED THERMAL POWER within the next 4 hours.

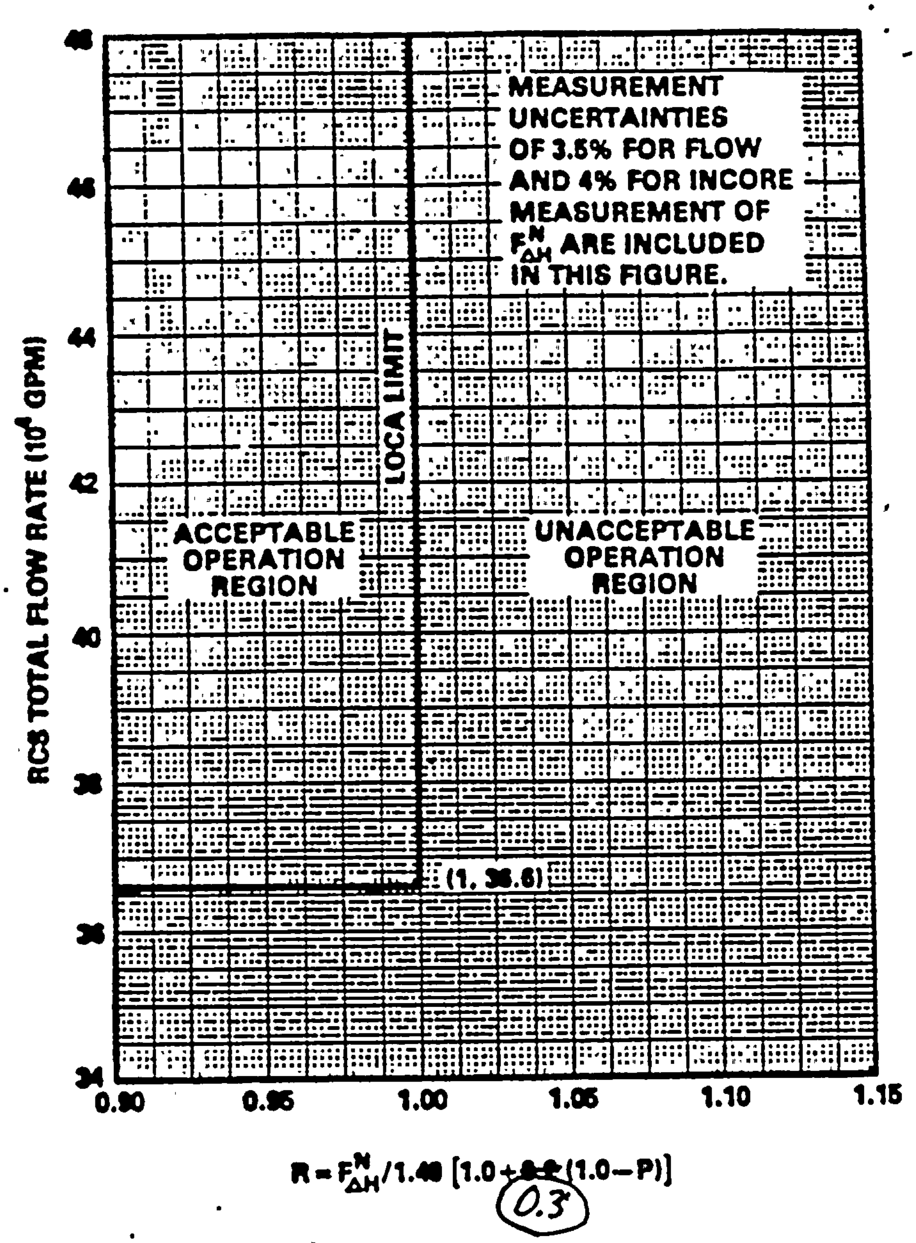

FIGURE 3.2-3b

RCS TOTAL FLOWRATE VERSUS R (UNIT 2)

3/4.5 EMERGENCY CORE COOLING SYSTEMS

3/4.5.1 ACCUMULATORS

LIMITING CONDITION FOR OPERATION

3.5.1 Each Reactor Coolant System accumulator shall be OPERABLE with:

a. The isolation valve open and power removed,

b. A contained borated water volume of between 836 and 864 cubic feet of borated water,

c. A boron concentration of between 2200 ~~1900~~ and 2500 ~~2200~~ ppm, and

d. A nitrogen cover-pressure of between 595.5 and 647.5 psig.

APPLICABILITY: MODES 1, 2 and 3.*

ACTION:

a. With one accumulator inoperable, except as a result of a closed isolation valve, restore the inoperable accumulator to OPERABLE status within 1 hour or be in at least HOT STANDBY within the next 6 hours and in at least HOT SHUTDOWN within the following 6 hours.

b. With one accumulator inoperable due to the isolation valve being closed, either immediately open the isolation valve or be in HOT STANDBY within 6 hours and in HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.5.1.1 Each accumulator shall be demonstrated OPERABLE:

a. At least once per 12 hours by:

1) Verifying the contained borated water volume and nitrogen cover-pressure in the tanks, and

2) Verifying that each accumulator isolation valve is open.

*Pressurizer pressure above 1000 psig.

EMERGENCY CORE COOLING SYSTEMS

3/4.5.5 REFUELING WATER STORAGE TANK

LIMITING CONDITION FOR OPERATION

3.5.5 The Refueling Water Storage Tank (RWST) shall be OPERABLE with:

a. A minimum contained borated water volume of 400,000 gallons,

b. A boron concentration of between ~~2000~~ 2300 and ~~2200~~ 2500 ppm, and

c. A minimum solution temperature of 35°F.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With the RWST inoperable, restore the tank to OPERABLE status within 1 hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.5.5 The RWST shall be demonstrated OPERABLE:

a. At least once per 7 days by:

1) Verifying the contained borated water volume in the tank, and

2) Verifying the boron concentration of the water.

b. At least once per 24 hours by verifying the RWST temperature when the outside ambient air temperature is less than 35°F.

3/4.1.1.4 MINIMUM TEMPERATURE FOR CRITICALITY

This specification ensures that the reactor will not be made critical with the Reactor Coolant System average temperature less than 541°F. This limitation is required to ensure: (1) the moderator temperature coefficient is within its analyzed temperature range, (2) the protective instrumentation is within its normal operating range, (3) the pressurizer is capable of being in an OPERABLE status with a steam bubble, and (4) the reactor vessel is above its minimum RT_{NDT} temperature.

3/4.1.2 BORATION SYSTEMS

The boron injection system ensures that negative reactivity control is available during each mode of facility operation. The components required to perform this function include: (1) borated water sources, (2) charging pumps, (3) separate flow paths, (4) boric acid transfer pumps, (5) associated heat tracing systems, and (6) an emergency power supply from OPERABLE diesel generators.

With the RCS average temperature above 200°F, a minimum of two boron injection flow paths are required to ensure single functional capability in the event an assumed failure renders one of the flow paths inoperable. The boration capability of either flow path is sufficient to provide a SHUTDOWN MARGIN from expected operating conditions of 1.6% Δk/k after xenon decay and cooldown to 200°F. The maximum expected boration capability requirement occurs at EOL from full power equilibrium xenon conditions and requires 5106 gallons of 20,000 ppm borated water from the boric acid storage tanks or 75,000 gallons of ~~2000~~ ppm borated water from the refueling water storage tank.

2300

With the RCS temperature below 200°F, one Boron Injection System is acceptable without single failure consideration on the basis of the stable reactivity condition of the reactor and the additional restrictions prohibiting CORE ALTERATIONS and positive reactivity change in the event the single injection system becomes inoperable.

Add + insert B 3/4.1.2 Here

The boron capability required below 200°F is sufficient to provide a SHUTDOWN MARGIN of 1% Δk/k after xenon decay and cooldown from 200°F to 140°F. This condition requires either 835 gallons of 20,000 ppm borated water from the boric acid storage tanks or 9690 gallons of ~~2000~~ ppm borated water from the refueling water storage tank.

2300

New ¶

The contained water volume limits include allowance for water not available because of discharge line location and other physical characteristics.

The OPERABILITY of one Boron Injection System during REFUELING ensures that this system is available for reactivity control while in MODE 6.

REACTIVITY CONTROL SYSTEMS

BASES

BORATION SYSTEMS (Continued)

MARGIN from expected operating conditions of 1.6% Δk/k after xenon decay and cooldown to 200°F. The maximum expected boration capability requirement occurs at EOL from full power equilibrium xenon conditions and requires [5106] gallons of [7000] ppm borated water from the boric acid storage tanks or [52,622] gallons of 2000 ppm borated water from the refueling water storage tank (RWST).

55/010
Pending

With the RCS temperature below 200°F, one Boron Injection System is acceptable without single failure consideration on the basis of the stable reactivity condition of the reactor and the additional restrictions prohibiting CORE ALTERATIONS and positive reactivity changes in the event the single Boron Injection System becomes inoperable.

CSDA PROVIDES

The limitation for a maximum of one centrifugal charging pump to be OPERABLE and the Surveillance Requirement to verify all charging pumps except the required OPERABLE pump to be inoperable below [275]°F provides assurance that a mass addition pressure transient can be relieved by the operation of a single PORV.

The boron capability required below 200°F is sufficient to provide a SHUTDOWN MARGIN of 1% Δk/k after xenon decay and cooldown from 200°F to 140°F. This condition requires either ___ gallons of [7000] ppm borated water from the boric acid storage tanks or ___ gallons of 2000 ppm borated water from the RWST.

Usable Volume

PROVIDES
SS/CW

The contained water volume limits include allowance for water not available because of discharge line location and other physical characteristics.

Insert in B 3/4.1.2 BORATION SYSTEMS

The limits on contained water volume and boron concentration of the RWST also ensure a pH value of between [8.5] and [11.0] for the solution recirculated within containment after a LOCA. This pH band minimizes the evolution of iodine and minimizes the effect of chloride and caustic stress corrosion on mechanical systems and components. 8.0 9.5

INSERT B 3/4.6.2

The OPERABILITY of one Boron Injection System during REFUELING ensures that this system is available for reactivity control while in MODE 6.

3/4.1.3 MOVABLE CONTROL ASSEMBLIES

The specifications of this section ensure that: (1) acceptable power distribution limits are maintained, (2) the minimum SHUTDOWN MARGIN is maintained, and (3) the potential effects of rod misalignment on associated accident analyses are limited. OPERABILITY of the control rod position indicators is required to determine control rod positions and thereby ensure compliance with the control rod alignment and insertion limits. Verification that the Digital Rod Position Indicator agrees with the demanded position within ± 12 steps at 24, 48, 120,

INSERT B 3/4.5.5 HERE

3/4.5.5 REFUELING WATER STORAGE TANK

~~The OPERABILITY of the Refueling Water Storage Tank (RWST) as part of the ECCS ensures that a sufficient supply of borated water is available for injection by the ECCS in the event of a LOCA. The limits on RWST minimum volume and boron concentration ensure that: (1) sufficient water is available within containment to permit recirculation cooling flow to the core, and (2) the reactor will remain subcritical in the cold condition following mixing of the RWST and the RCS water volumes with all control rods inserted except for the most reactive control assembly. These assumptions are consistent with the LOCA analyses.~~

The contained water volume limit includes an allowance for water not usable because of tank discharge line location or other physical characteristics.

INSERT ~~ATTACHMENT~~ B B 3/4.5.5

The OPERABILITY of the Refueling Water Storage Tank (RWST) as part of the ECCS ensures that a sufficient supply of borated water is available for injection by the ECCS in the event of either a LOCA or a steamline break. The limits on RWST minimum volume and boron concentration ensure that: 1) sufficient water is available within containment to permit recirculation cooling flow to the core, 2) the reactor will remain subcritical in the cold condition (68 to 212 degrees-F) following a small break LOCA assuming complete mixing of the RWST, RCS, SAT, containment spray system piping and ECCS water volumes with all control rods inserted except the most reactive control rod assembly (ARI-1), 3) the reactor will remain subcritical in the cold condition following a large break LOCA (break flow area > 3.0 ft^2) assuming complete mixing of the RWST, RCS, ECCS water and other sources of water that may eventually reside in the sump Post-LOCA with all control rods assumed to be out (ARO), 4) long term subcriticality following a steamline break assuming ARI-1 and preclude fuel failure.

The maximum allowable value for the RWST boron concentration forms the basis for determining the time (post-LOCA) at which operator action is required to switch over the ECCS to hot leg recirculation in order to avoid precipitation of the soluble boron.

3/4.6.2 DEPRESSURIZATION AND COOLING SYSTEMS

3/4.6.2.1 CONTAINMENT SPRAY SYSTEM

The OPERABILITY of the Containment Spray System ensures that containment depressurization and cooling capability will be available in the event of a LOCA. The pressure reduction and resultant lower containment leakage rate are consistent with the assumptions used in the safety analyses.

The Containment Spray System and the Containment Cooling System are redundant to each other in providing post accident cooling of the containment atmosphere. However, the Containment Spray System also provides a mechanism for removing iodine from the containment atmosphere and therefore the time requirements for restoring an inoperable Spray System to OPERABLE status have been maintained consistent with that assigned other inoperable ESF equipment.

3/4.6.2.2 SPRAY ADDITIVE SYSTEM

INSERT B3/4.6.2.2 Here

~~The OPERABILITY of the Spray Additive System ensures that sufficient NaOH is added to the containment spray in the event of a LOCA. The limits on NaOH minimum volume and concentration ensure that: (1) the iodine removal efficiency of the spray water is maintained because of the increase in pH value, and (2) corrosion effects on components within containment are minimized. The contained water volume limit includes an allowance for water not usable because of tank discharge line location or other physical characteristics. These assumptions are consistent with the iodine removal efficiency assumed in the safety analyses.~~

3/4.6.2.3 CONTAINMENT COOLING SYSTEM

The OPERABILITY of the containment fan cooler units ensures that: (1) the containment air temperature will be maintained within limits during normal operation, (2) adequate heat removal capacity is available when operated in conjunction with the containment spray systems during post LOCA conditions, and (3) adequate mixing of the containment atmosphere following a LOCA to prevent localized accumulations of hydrogen from exceeding the flammable limit.

The Containment Cooling System and the Containment Spray System are redundant to each other in providing post accident cooling of the containment atmosphere. As a result of this redundancy in cooling capability, the allowable out of service time requirements for the Containment Cooling System have been appropriately adjusted. However, the allowable out of service time requirements for the Containment Spray System have been maintained consistent with that assigned other inoperable ESF equipment since the Containment Spray System also provides a mechanism for removing iodine from the containment atmosphere.

CONTAINMENT SPRAY SYSTEM (Continued)

[Credit taken for iodine removal]

The Containment Spray System and the Containment Cooling System are redundant to each other in providing post-accident cooling of the containment atmosphere. However, the Containment Spray System also provides a mechanism for removing iodine from the containment atmosphere and therefore the time requirements for restoring an inoperable Spray System to OPERABLE status have been maintained consistent with that assigned other inoperable ESF equipment.

[No credit taken for iodine removal]

The Containment Spray System and the Containment Cooling System are redundant to each other in providing post-accident cooling of the containment atmosphere. Since no credit has been taken for iodine removal by the Containment Spray System, the allowable out-of-service time requirements for the Containment Spray System and Containment Cooling System have been interrelated and adjusted to reflect this additional redundancy in cooling capability.

INSERT B 3/4.6.2.2

3/4.6.2.2 SPRAY ADDITIVE SYSTEM [OPTIONAL]

The OPERABILITY of the Spray Additive System ensures that sufficient NaOH is added to the containment spray in the event of a LOCA. The limits on NaOH volume and concentration ensure a pH value of between ~~[8.5]~~ 8.0 and ~~[11.0]~~ 9.5 for the solution recirculated within containment after a LOCA. This pH band minimizes the evolution of iodine and minimizes the effect of chloride and caustic stress corrosion on mechanical systems and components. The contained solution volume limit includes an allowance for solution not usable because of tank discharge line location or other physical characteristics. These assumptions are consistent with the iodine removal efficiency assumed in the safety analyses.

3/4.6.2.3 CONTAINMENT COOLING SYSTEM [OPTIONAL]

The OPERABILITY of the Containment Cooling System ensures that: (1) the containment air temperature will be maintained within limits during normal operation, and (2) adequate heat removal capacity is available when operated in conjunction with the Containment Spray Systems during post-LOCA conditions.

[Credit taken for iodine removal by spray systems]

The Containment Cooling System and the Containment Spray System are redundant to each other in providing post-accident cooling of the containment atmosphere. As a result of this redundancy in cooling capability, the allowable out-of-service time requirements for the Containment Cooling System have been appropriately adjusted. However, the allowable out-of-service time requirements for the Containment Spray System have been maintained consistent with that assigned other inoperable ESF equipment since the Containment Spray System also provides a mechanism for removing iodine from the containment atmosphere.

[No credit taken for iodine removal by spray systems]

The Containment Cooling System and the Containment Spray System are redundant to each other in providing post-accident cooling of the containment atmosphere. Since no credit has been taken for iodine removal by the Containment Spray System, the allowable out-of-service time requirements for the Containment Cooling System and Containment Spray System have been interrelated and adjusted to reflect this additional redundancy in cooling capacity.

APPENDIX B

RADIAL PEAKING FACTOR LIMIT REPORT

This Radial Peaking Factor Limit Report is provided in accordance with Paragraph 6.9.1.8 of the Diablo Canyon Unit 2 Nuclear Plant Technical Specifications.

The F_{xy} limits for RATED THERMAL POWER within specific core planes shall be:

1. F_{xy}^{RTP} less than or equal to 1.836 for all core planes containing bank "D" control rods, and

2. F_{xy}^{RTP} less than or equal to 1.687 for all unrodded core planes.

These $F_{xy}(z)$ limits were used to confirm that the heat flux hot channel factor $F_Q(z)$ will be limited to the Technical Specification values of:

$F_Q(z) \leq \frac{[2.32]}{P}[K(z)]$ for $P > 0.5$ and,

$F_Q(z) \leq [4.64]\ [K(z)]$ for $P \leq 0.5$

assuming the most limiting axial power distributions expected to result from the insertion and removal of Control Banks C and D during operation, including the accompanying variations in the axial xenon and power distributions as described in the "Power Distribution Control and Load Following Procedures," WCAP-8403, September, 1974. Therefore, these F_{xy} limits provide assurance that the initial conditions assumed in the LOCA analyses are met, along with the ECCS acceptance criteria of 10CFR50.46.

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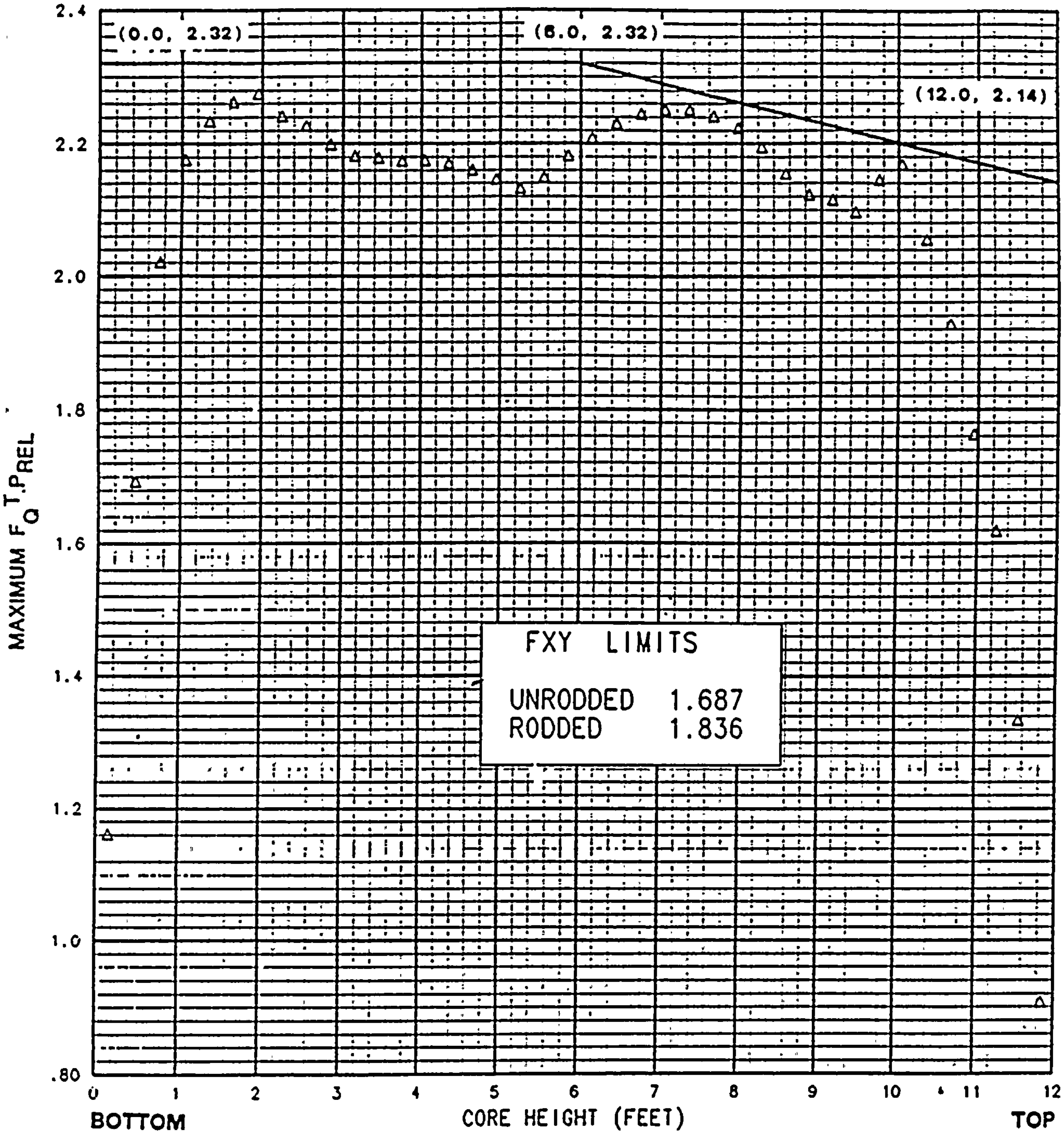

FIGURE 1B

MAXIMUM $F_Q^T \cdot P_{REL}$ vs. AXIAL CORE HEIGHT DURING NORMAL OPERATION

[illegible]