Attachment B

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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.

The DNB design basis is as follows: there must be at least a 95 percent probability that the minimum DNBR of the limiting rod during steady-state operation, normal operational transients, and anticipated transients is greater than or equal to the DNBR limit of the DNB correlation being used (the WRB-1 for LOPAR fuel and the WRB-2 for VANTAGE 5 fuel in this application). The correlation DNBR limit is established based on the entire applicable experimental data set such that there is a 95 percent probability with a 95 percent confidence level that DNB will not occur when the minimum DNBR is at or greater than the DNBR limit (1.17 for both the WRB-1 and WRB-2 correlations).

In meeting this design basis, uncertainties in plant operating parameters, nuclear and thermal parameters, and fuel fabrication parameters are considered statistically such that there is at least a 95 percent probability with a 95 confidence level that the minimum DNBR for the limiting rod is greater than or equal to the DNBR limit. The uncertainties in the above plant parameters are used to determine the plant DNBR uncertainty. This DNBR uncertainty, combined with the correlation DNBR limit, establishes a design DNBR value which must be met in plant safety analyses using values of input parameters without uncertainties. For Diablo Canyon Units, the design DNBR values are 1.33 and 1.37 for thimble and typical cells, respectively, for LOPAR fuel, and 1.30 for thimble and 1.32 for typical cells for the VANTAGE 5 fuel. In addition, margin has been maintained in both designs by meeting safety analysis DNBR limits of 1.44 for thimble and 1.48 for typical cells for LOPAR fuel, and 1.68 and 1.71 for thimble and typical cells, respectively, for VANTAGE 5 fuel in performing safety analyses.

The curves in Figure 2.1-1 show the loci of points of THERMAL POWER, Reactor Coolant System pressure and average temperature below which the calculated DNBR is no less than the safety analysis DNBR limits, or the average enthalpy at the vessel exit is less than the enthalpy of saturated liquid.

The curves are based on limiting an enthalpy hot channel factor s, $F^N_{\Delta H}$ of 1.56 for LOPAR fuel and 1.59 for VANTAGE 5 fuel, 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 expressions:

$F^{RTP}_{\Delta H}$ $PF_{\Delta H}$

$$F^N_{\Delta H} = \cancel{1.56}\,[1 + \cancel{0.3}(1 - P)] \text{ for LOPAR fuel}$$

$$F^N_{\Delta H} = \cancel{1.59}\,[1 + \cancel{0.3}(1 - P)] \text{ for VANTAGE 5 fuel}$$

$F^{RTP}_{\Delta H}$ $PF_{\Delta H}$

where P is the fraction of RATED THERMAL POWER

, and $F^{RTP}_{\Delta H}$ are the limiting enthalpy hot channel factors at RATED THERMAL POWER specified in the CORE OPERATING LIMITS REPORT (COLR) for each fuel type, and where $PF_{\Delta H}$ are the power factor multipliers specified in the COLR.

The 4% measurement uncertainty associated with $F^N_{\Delta H}$ is accounted for in the DNBR design limit.

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.

POWER DISTRIBUTION LIMITS

3/4.2.2 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:

$$F_Q(Z) < \frac{\cancel{[2.45]}\ F_Q^{RTP}}{P}\ [K(Z)] \text{ for } P > 0.5$$

$$F_Q(Z) \le \cancel{[4.90]}\ \left[\frac{F_Q^{RTP}}{0.5}\right] [K(Z)] \text{ for } P \le 0.5$$

where F_Q^{RTP} = the F_Q limit at RATED THERMAL POWER (RTP) specified in the CORE OPERATING LIMITS REPORT (COLR),

~~Where~~ and P = $\frac{\text{THERMAL POWER}}{\text{RATED THERMAL POWER}}$, and

K(Z) = the ~~function obtained from Figure 3.2-2~~ normalized $F_Q(z)$ for a given core height ~~location.~~ specified in the COLR.

APPLICABILITY: MODE 1 ~~(Units 1 and 2 Cycle 4 and after).~~

ACTION:

With $F_Q(Z)$ exceeding its limit:

a. 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.

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

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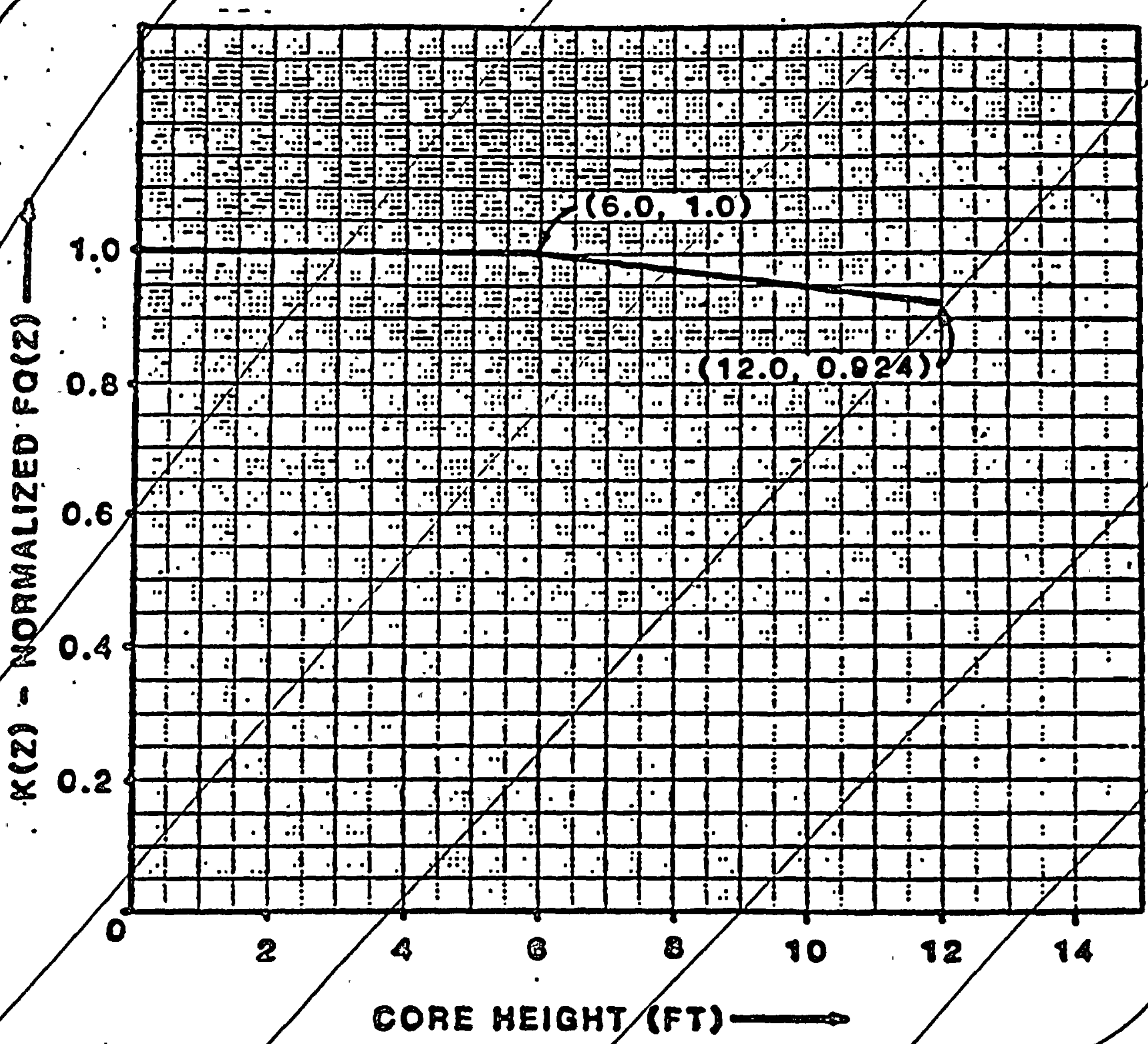

FIGURE 3.2-2
K(Z) - NORMALIZED $F_Q(Z)$ AS A FUNCTION OF CORE HEIGHT

SURVEILLANCE REQUIREMENTS

4.2.2.2.1 The provisions of Specification 4.0.4 are not applicable.

4.2.2.2.2 $F_Q(z)$ shall be evaluated to determine if $F_Q(z)$ is within its limits by:

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

b. Increasing the measured $F_Q(z)$ component of the power distribution map by 3% to account for manufacturing tolerances and further increasing the value by 5% to account for measurement uncertainties.

c. Satisfying the following relationship:

$$F_Q^M(z) \le \frac{F_Q^{RTP}\ \cancel{2.45} \times K(z)}{P \times W(z)} \quad \text{for } P > 0.5$$

$$F_Q^M(z) \le \frac{F_Q^{RTP}\ \cancel{2.45} \times K(z)}{W(z) \times 0.5} \quad \text{for } P \le 0.5$$

where $F_Q^M(z)$ is the measured $F_Q(z)$ increased by the allowances for manufacturing tolerances and measurement uncertainty, F_Q^{RTP} ~~2.45~~ is the F_Q limit, K(z) is ~~given in Figure 3.2-2~~ the normalized $F_Q(z)$ as a function of core height, P is the relative THERMAL POWER, and W(z) is the cycle dependent function that accounts for power distribution transients encountered during normal operation. F_Q^{RTP}, K(z), and W(z) are ~~This function is~~ specified in the ~~CORE OPERATING LIMITS REPORT as per Specification 6.9.1.8~~ COLR.

d. Measuring $F_Q^M(z)$ according to the following schedule:

1. Upon achieving equilibrium conditions after exceeding by 20% or more of RATED THERMAL POWER, the THERMAL POWER at which $F_Q(z)$ was last determined,* or
2. At least once per 31 Effective Full Power Days (EFPD), whichever occurs first.

e. With measurements indicating

$$\underset{\text{over } z}{\text{maximum}} \quad \frac{F_Q^M(z)}{K(z)}$$

has increased since the previous determination of $F_Q^M(z)$ either of the following actions shall be taken:

* During power escalation at the beginning of each cycle, power level may be increased until a power level for extended operation has been achieved and a power distribution map obtained.

[illegible]

SURVEILLANCE REQUIREMENTS (Continued)

1) $F_Q^M(z)$ shall be increased by 2% over that specified in Specification 4.2.2.1.2.c, or

2) $F_Q^M(z)$ shall be measured at least once per 7 EFPD until two successive maps indicate that

$$\underset{\text{over } z}{\text{maximum}} \left(\frac{F_Q^M(z)}{K(z)} \right) \text{ is not increasing.}$$

f. With the relationship specified in Specification 4.2.2.1.2.c above not being satisfied:

1) Calculate the percent $F_Q(z)$ exceeds it limit by the following expression:

$$\left\{ \left(\underset{\text{over } z}{\text{maximum}} \left[\frac{F_Q^M(z) \times W(z)}{\frac{\cancel{2.45}\; F_Q^{RTP}}{P} \times K(z)} \right] \right) - 1 \right\} \times 100 \text{ for } P \geq 0.5$$

$$\left\{ \left(\underset{\text{over } z}{\text{maximum}} \left[\frac{F_Q^M(z) \times W(z)}{\frac{\cancel{2.45}\; F_Q^{RTP}}{0.5} \times K(z)} \right] \right) - 1 \right\} \times 100 \text{ for } P < 0.5$$

2. Either one of the following actions shall be taken:

a) Place the core in an equilibrium condition where the limit in Specification 4.2.2.1.2.c is satisfied. Power level may then be increased provided the AFD limits of Specification 3.2.1 are reduced 1% AFD for each percent $F_Q(z)$ exceeds its limit, or

SURVEILLANCE REQUIREMENTS (Continued)

b) Comply with the requirements of Specification 3.2.2.1 for $F_Q(z)$ exceeding its limit by the percent calculated.

g. The limits specified in Specification 4.2.2.2.2.c, 4.2.2.2.2.e, and 4.2.2.2.2.f above are not applicable in the following core plane regions:

1. Lower core region from 0 to 15%, inclusive.
2. Upper core region from 85 to 100%, inclusive.

4.2.2.2.3 When $F_Q(Z)$ is measured pursuant to Specification 4.10.2.2, an overall measured $F_Q(Z)$ shall be obtained from power distribution map and increased by 3% to account for manufacturing tolerances and further increased by 5% to account for measurement uncertainty.

LIMITING CONDITION FOR OPERATION

3.2.2.2 $F_Q(Z)$ shall be limited by the following relationships:

$$F_Q(Z) \leq \frac{2.45}{P} [K(Z)] \text{ for } P > 0.5$$

$$F_Q(Z) \leq [4.64] [K(Z)] \text{ for } P \leq 0.5$$

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

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

APPLICABILITY: MODE 1 (Unit 2 Cycle 3).

ACTION:

With $F_Q(Z)$ exceeding its limit:

a. 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.

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

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SURVEILLANCE REQUIREMENTS

4.2.2.2.1 The provisions of Specification 4.0.4 are not applicable.

4.2.2.2.2 F_{xy} shall be evaluated to determine if $F_Q(Z)$ is within its limit by:

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

b. Increasing the measured F_{xy} component of the power distribution map by 3% to account for manufacturing tolerances and further increasing the value by 5% to account for measurement uncertainties,

c. Comparing the F_{xy} computed (F_{xy}^C) obtained in Specification 4.2.2.2.2b., above, to:

1. The F_{xy} limits for RATED THERMAL POWER (F_{xy}^{RTP}) for the appropriate measured core planes given in Specification 4.2.2.2.2e. and f. below, and

2. The relationship:

$$F_{xy}^L = (F_{xy}^{RTP}) \; [1+0.2(1-P)],$$

where F_{xy}^L is the limit for fractional THERMAL POWER operation expressed as a function of F_{xy}^{RTP} and P is the fraction of RATED THERMAL POWER at which F_{xy} was measured.

d. Remeasuring F_{xy} according to the following schedule:

1. When F_{xy}^C is greater than the F_{xy}^{RTP} limit for the appropriate measured core plane but less than the F_{xy}^L relationship, additional power distribution maps shall be taken and F_{xy}^C compared to F_{xy}^{RTP} and F_{xy}^L either:

a) Within 24 hours after exceeding by 20% of RATED THERMAL POWER or greater, the THERMAL POWER at which F_{xy}^C was last determined, or

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

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Effective at end of Unit 1 Cycle 3

SURVEILLANCE REQUIREMENTS (Continued)

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

e. The F_{xy} limit for RATED THERMAL POWER (F_{xy}^{RTP}) shall be provided for all core planes containing Bank "D" control rods and all unrodded core planes in the CORE OPERATING LIMITS REPORT per Specification 6.9.1.8.

f. The F_{xy} limits of Specification 4.2.2.2.2e., above, are not applicable in the following core plane regions as measured in percent of core height from the bottom of the fuel:

1. Lower core region from 0 to 15%, inclusive,
2. Upper core region from 85 to 100% inclusive,
3. Grid plane regions at 17.8 ± 2%, 32.1 ± 2%, 46.4 ± 2%, 60.6 ± 2% and 74.9 ± 2%, inclusive, and
4. Core plane regions within ± 2% of core height (± 2.88 inches) about the bank demand position of the Bank "D" control rods.

g. With F_{xy}^{C} exceeding F_{xy}^{L}, the effects of F_{xy} on $F_Q(Z)$ shall be evaluated to determine if $F_Q(Z)$ is within its limits.

4.2.2.2.3 When $F_Q(Z)$ is measured pursuant to Specification 4.10.2.2, an overall measured $F_Q(Z)$ shall be obtained from power distribution map and increased by 3% to account for manufacturing tolerances and further increased by 5% to account for measurement uncertainty.

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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: $F_{\Delta H}^{RTP}$ $PF_{\Delta H}$

a. $$R = \frac{F^N_{\Delta H}}{\cancel{1.56}\,F_{\Delta H}^{RTP}\,[1.0 + \cancel{0.3}\,PF_{\Delta H}\,(1.0 - P)]}$$ for LOPAR fuel

$$R = \frac{F^N_{\Delta H}}{\cancel{1.59}\,F_{\Delta H}^{RTP}\,[1.0 + \cancel{0.3}\,PF_{\Delta H}\,(1.0 - P)]}$$ For VANTAGE 5 fuel

b. $$P = \frac{\text{THERMAL POWER}}{\text{RATED THERMAL POWER}}$$, 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 2.4% for flow and 4% for incore measurement of $F^N_{\Delta H}$.

d. $F_{\Delta H}^{RTP}$ = The $F^N_{\Delta H}$ limits at Rated Thermal Power (RTP) specified in the Core Operating Limits Report (COLR).

e. $PF_{\Delta H}$ = The Power Factor Multipliers specified in the COLR.

APPLICABILITY: MODE 1.

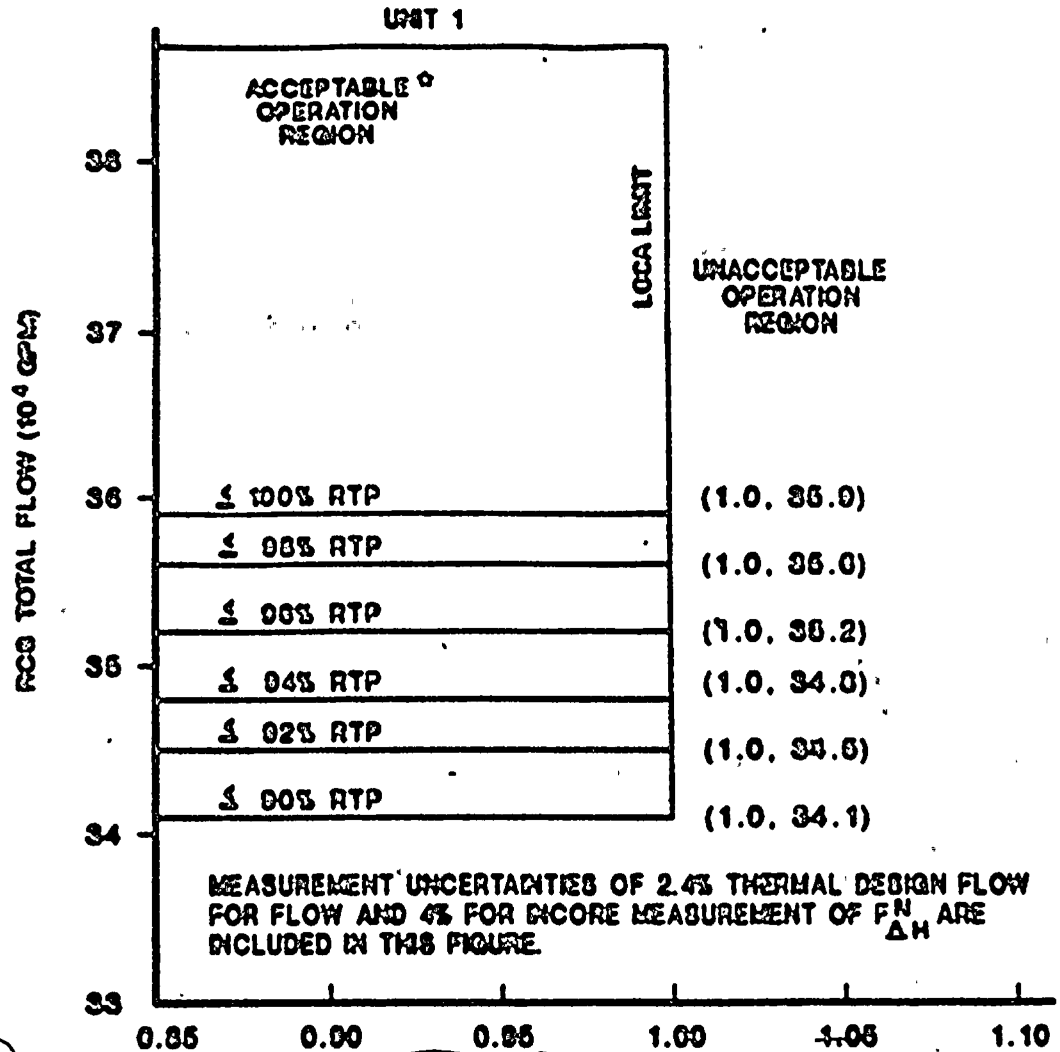
ACTION:

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.



$F^{RTP}_{\Delta H}$ $PF_{\Delta H}$

$R = F^N_{\Delta H} / (1.55(1.0 + 0.3(1.0 - P)))$ LOPAR FUEL

$PF_{\Delta H}$ $F^{RTP}_{\Delta H}$

$R = F^N_{\Delta H} / (1.59(1.0 + 0.3(1.0 - P)))$ VANTAGE 5 FUEL

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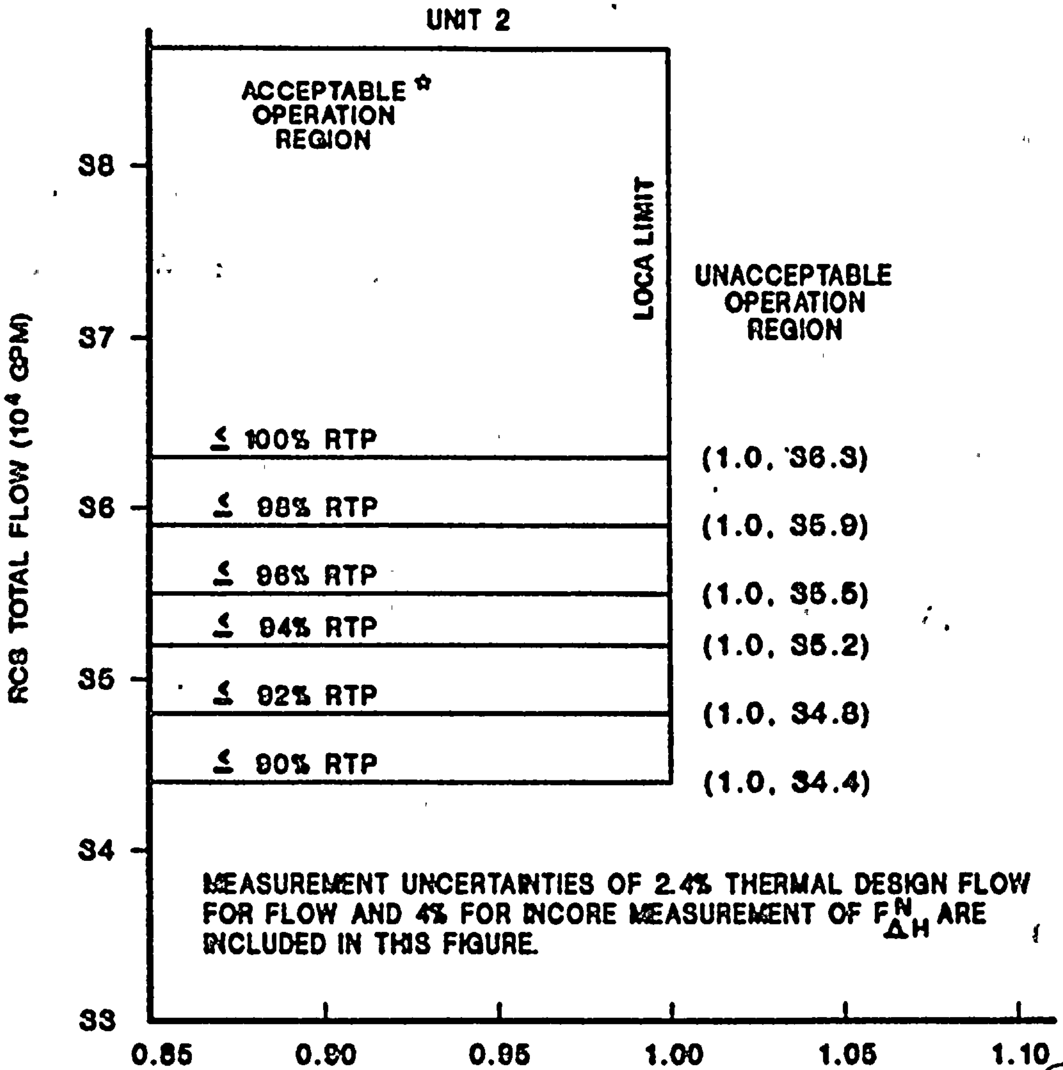
✩WHEN OPERATING IN THE RESTRICTED POWER REGION, THE RESTRICTED POWER LEVEL SHALL BE CONSIDERED 100% RTP FOR FIGURE 2.1-1

FIGURE 3.2-3a
RCS TOTAL FLOWRATE VERSUS R(UNIT 1)



$F^{RTP}_{\Delta H}$ $PF_{\Delta H}$ $PF_{\Delta H}$

$R = F^N_{\Delta H} / 1.56(1.0 + 0.3(1.0 - P))$ LOPAR FUEL

$R = F^N_{\Delta H} / 1.59(1.0 + 0.3(1.0 - P))$ VANTAGE 5 FUEL

$F^{RTP}_{\Delta H}$

*WHEN OPERATING IN THE RESTRICTED POWER REGION, THE RESTRICTED POWER LEVEL SHALL BE CONSIDERED 100% RTP FOR FIGURE 2.1-1

FIGURE 3.2-3b
RCS TOTAL FLOWRATE VERSUS R(UNIT 2)

SPECIAL TEST EXCEPTIONS

3/4.10.2 GROUP HEIGHT, INSERTION AND POWER DISTRIBUTION LIMITS

LIMITING CONDITION FOR OPERATION

3.10.2 The group height, insertion and power distribution limits of Specifications 3.1.3.1, 3.1.3.5, 3.1.3.6, 3.2.1, and 3.2.4 may be suspended during the performance of PHYSICS TESTS provided:

a. The THERMAL POWER is maintained less than or equal to 85% of RATED THERMAL POWER, and

b. The limits of Specifications 3.2.2~~.1 or 3.2.2.2~~ and 3.2.3 are maintained and determined at the frequencies specified in Specification 4.10.2.2 below.

APPLICABILITY: MODE 1.

ACTION:

With any of the limits of Specifications 3.2.2~~.1 or 3.2.2.2~~ or 3.2.3 being exceeded while the requirements of Specifications 3.1.3.1, 3.1.3.5, 3.1.3.6, 3.2.1 and 3.2.4 are suspended, either:

a. Reduce THERMAL POWER sufficient to satisfy the ACTION requirements of Specifications 3.2.2~~.1 or 3.2.2.2~~ and 3.2.3, or

b. Be in HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

4.10.2.1 The THERMAL POWER shall be determined to be less than or equal to 85% of RATED THERMAL POWER at least once per hour during PHYSICS TESTS.

4.10.2.2 The requirements of the below listed specifications shall be performed at least once per 12 hours during PHYSICS TESTS:

a. Specifications 4.2.2.~~1~~.2 ~~or 4.2.2.2.2~~ and 4.2.2.~~1~~.3 ~~or 4.2.2.2.3~~, and

b. Specification 4.2.3.2.

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POWER DISTRIBUTION LIMITS

BASES

HEAT FLUX HOT CHANNEL FACTOR, and RCS FLOWRATE AND NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR (Continued)

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

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

2. Control rod groups are sequenced with overlapping groups as described in Specification 3.1.3.6,

3. The control rod insertion limits of Specifications 3.1.3.5 and 3.1.3.6 are maintained, and

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

$F_{\Delta H}^{N}$ will be maintained within its limits provided Conditions 1. through 4., above, are maintained. The relaxation of $F_{\Delta H}^{N}$ as a function of THERMAL POWER allows changes in the radial power shape for all permissible rod insertion limits.

R, as calculated per Specification 3.2.3 and used in Figure 3.2-3a and Figure 3.2-3b accounts for $F_{\Delta H}^{N}$ less than or equal to ~~1.56~~ for LOPAR fuel and ~~1.59~~ for VANTAGE 5 fuel. These values are the values used in the various accident analyses where $F_{\Delta H}^{N}$ influences parameters other than DNBR, e.g., peak clad temperature, and thus are the maximum "as measured" values allowed.

Margin between the safety analysis limit DNBRs (1.44 and 1.48 for the LOPAR fuel thimble and typical cells, respectively, and 1.68 and 1.71 for the VANTAGE 5 thimble and typical cells) and the design limit DNBRs (1.33 and 1.37 for the LOPAR fuel thimble and typical cells and 1.30 and 1.32 for the VANTAGE 5 fuel thimble and typical cells, respectively) is maintained. A fraction of this margin is utilized to accommodate the transition core DNBR penalty of maximum 12.5 percent and the appropriate fuel rod bow DNBR penalty (less than 1.5 percent for both fuel types per WCAP-8691, Revision 1). The rest of the margin between design and safety analysis DNBR limits can be used for plant design flexibility.

the $F_{\Delta H}^{RTP}$ limits specified in the COLR

MONTHLY OPERATING REPORT

6.9.1.7 Routine reports of operating statistics and shutdown experience, including documentation of all challenges and failures to the PORVs or safety valves, shall be submitted on a monthly basis to the NRC in accordance with 10 CFR 50.4, no later than the 15th of each month following the calendar month covered by the report.

CORE OPERATING LIMITS REPORT

6.9.1.8.a Core operating limits shall be established and documented in the CORE OPERATING LIMITS REPORT before each reload cycle or any remaining part of a reload cycle for the following:

1. Shutdown Rod Insertion Limits for Specification 3/4.1.3.5,
2. Control Rod Insertion Limits for Specification 3/4.1.3.6,
3. Axial Flux Difference for Specification 3/4.2.1,
4. Heat Flux Hot Channel Factor, $K(z)$ and $W(z)$ - $F_Q(z)$ (F_Q^{RTP} for Specification 3/4.2.2). - $F_Q(z)$ ~~(Surveillance Requirement W(z) of Specification 3/4.2.2), and~~
5. ~~Heat Flux Hot Channel Factor - $F_Q(z)$ (Surveillance Requirement F_{xy} of Specification 3/4.2.2.2).~~

5. RCS Flow Rate and Nuclear Enthalpy Rise Hot Channel Factor - $F^N_{\Delta H}$ ($F^{RTP}_{\Delta H}$ and $PF_{\Delta H}$ for Specification 3/4.2.3).

b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC in:

1. WCAP-10216-P-A, Relaxation of Constant Axial Offset Control F_Q Surveillance Technical Specification, June 1983 (Westinghouse Proprietary),
2. WCAP-9272-P-A, Westinghouse Reload Safety Evaluation Methodology, July 1985 (Westinghouse Proprietary), and
3. WCAP-8385, Power Distribution Control and Load Following Procedures, September 1974 (Westinghouse Proprietary).

c. The core operating limits shall be determined so that all applicable limits (e.g., fuel thermal-mechanical limits, core thermal-hydraulic limits, ECCS limits, nuclear limits such as SHUTDOWN MARGIN, and transient and accident analysis limits) of the safety analysis are met.

d. The CORE OPERATING LIMITS REPORT, including any mid-cycle revisions or supplements thereto, shall be provided, upon issuance for each reload cycle, to the NRC Document Control Desk, with copies to the Regional Administrator and Resident Inspector.

SPECIAL REPORTS

6.9.2 Special reports shall be submitted to the NRC in accordance with 10 CFR 50.4 within the time period specified for each report..

6.10 RECORD RETENTION

In Addition to the applicable record retention requirements of Title 10, Code of Federal Regulations, the following records shall be retained for at least the minimum period indicated.

6.10.1 The following records shall be retained for at least 5 years:

a. Records and logs of unit operation covering time interval at each power level;
b. Records and logs of principal maintenance activities, inspections, repair and replacement of principal items of equipment related to nuclear safety;
c. ALL REPORTABLE EVENTS;

[illegible]

Attachment C

SAMPLE CORE OPERATING LIMITS REPORTS

PACIFIC GAS AND ELECTRIC COMPANY

DEPARTMENT OF NUCLEAR POWER GENERATION

DIABLO CANYON POWER PLANT

NUMBER COLR 1-5

REVISION ~~0~~ 1

UNIT 1

TITLE: CORE OPERATING LIMITS REPORT
COLR FOR DIABLO CANYON UNIT 1 CYCLE 5

APPROVED: John Townsend 3/27/91 DATE 3/27/91 EFFECTIVE DATE

** PROCEDURE CLASSIFICATION - QUALITY RELATED **

1.0 CORE OPERATING LIMITS REPORT

This Core Operating Limits Report (COLR) for Diablo Canyon Unit 1 Cycle 5 has been prepared in accordance with the requirements of Technical Specification (TS) 6.9.1.8.

The Technical Specifications affected by this report are listed below:

3/4.1.3.5 - Shutdown Rod Insertion Limit
3/4.1.3.6 - Control Rod Insertion Limits
3/4.2.1 - Axial Flux Difference
3/4.2.2 - Heat Flux Hot Channel Factor - $F_Q(z)$ ~~(Surveillance Requirements only)~~
3/4.2.3 RCS Flow Rate and Nuclear Enthalpy Rise Hot Channel Factor - $F_{\Delta H}^N$

2.0 OPERATING LIMITS

The cycle-specific parameter limits for the specifications listed in Section 1.0 are presented in the following subsections. These limits have been developed using the NRC-approved methodologies specified in Technical Specification 6.9.1.8.

2.1 Shutdown Rod Insertion Limit (TS 3/4.1.3.5)

2.1.1 The shutdown rods shall be withdrawn to at least 225 steps.

2.2 Control Rod Insertion Limits (TS 3/4.1.3.6)

2.2.1 The control rod banks shall be limited in physical insertion as shown in Figure 1.

2.3 Axial Flux Difference (TS 3/4.2.1)

2.3.1 The AXIAL FLUX DIFFERENCE (AFD) Limits are provided in Figure 2.

2.4 Heat Flux Hot Channel Factor - $F_Q(z)$ (TS 3/4.2.2)

INSERT A →

~~2.4.1 The W(z) curves for RAOC operation provided in Figures 3 through 5 are sufficient to determine the RAOC W(z) versus core height for Cycle 5 burnups through the end of full power reactivity plus a power coastdown of up to 1000 MWD/MTU.~~

3.0 FIGURES

3.1 Figure 1 - Rod Bank Insertion Limits Versus Rated Thermal Power

3.2 Figure 2 - AFD Limits as a Function of Rated Thermal Power

3.3 Figure 3 - W(Z) at 150 MWD/MTU as a Function of Core Height

3.4 Figure 4 - W(Z) at 6,000 MWD/MTU as a Function of Core Height

3.5 Figure 5 - W(Z) at 16,000 MWD/MTU as a Function of Core Height

3.6 Figure 6 - Normalized $F_Q(z)$ as a Function of Core Height

4.0 REFERENCES

Westinghouse Reload Safety Evaluation for Diablo Canyon Power Plant Unit 1 Cycle 5, Revision 1, dated February 1991.

5.0 RECORD OF REVIEWS

5.1 Prepared by: C.A. Hieb

5.2 Sponsored by: J.R. Hinds

5.3 Independent Technical Review by: T. Grebel

6.0 RECORDS

None

**********INSERT A**********

2.4.1

$$F_Q(Z) \leq \frac{F_Q^{RTP}}{P} * K(Z) \qquad \text{for } P > 0.5$$

$$F_Q(Z) \leq \frac{F_Q^{RTP}}{0.5} * K(Z) \qquad \text{for } P \leq 0.5$$

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

$F_Q^{RTP} = 2.45$

K(Z) is provided in Figure 6.

2.4.2 The W(Z) curves for Specification 4.2.2.2.c Relaxed Axial Offset Control (RAOC) operation, provided in figures 3 through 5, are sufficient to determine the RAOC W(Z) versus core height for Cycle burnups through the end of full power reactivity plus a power coastdown of up to 1000 MWD/MTU.

2.5 <u>RCS Flow Rate and Nuclear Enthalpy Rise Hot Channel Factor</u> - $F^N_{\Delta H}$ (Specification 3/4.2.3)

$$R = \frac{F^N_{\Delta H}}{F^{RTP}_{\Delta H} * (1 + PF_{\Delta H} * (1-P))}$$

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

$F^{RTP}_{\Delta H}$ = 1.56 (LOPAR fuel)
= 1.59 (VANTAGE 5 fuel)

$PF_{\Delta H} = 0.3$

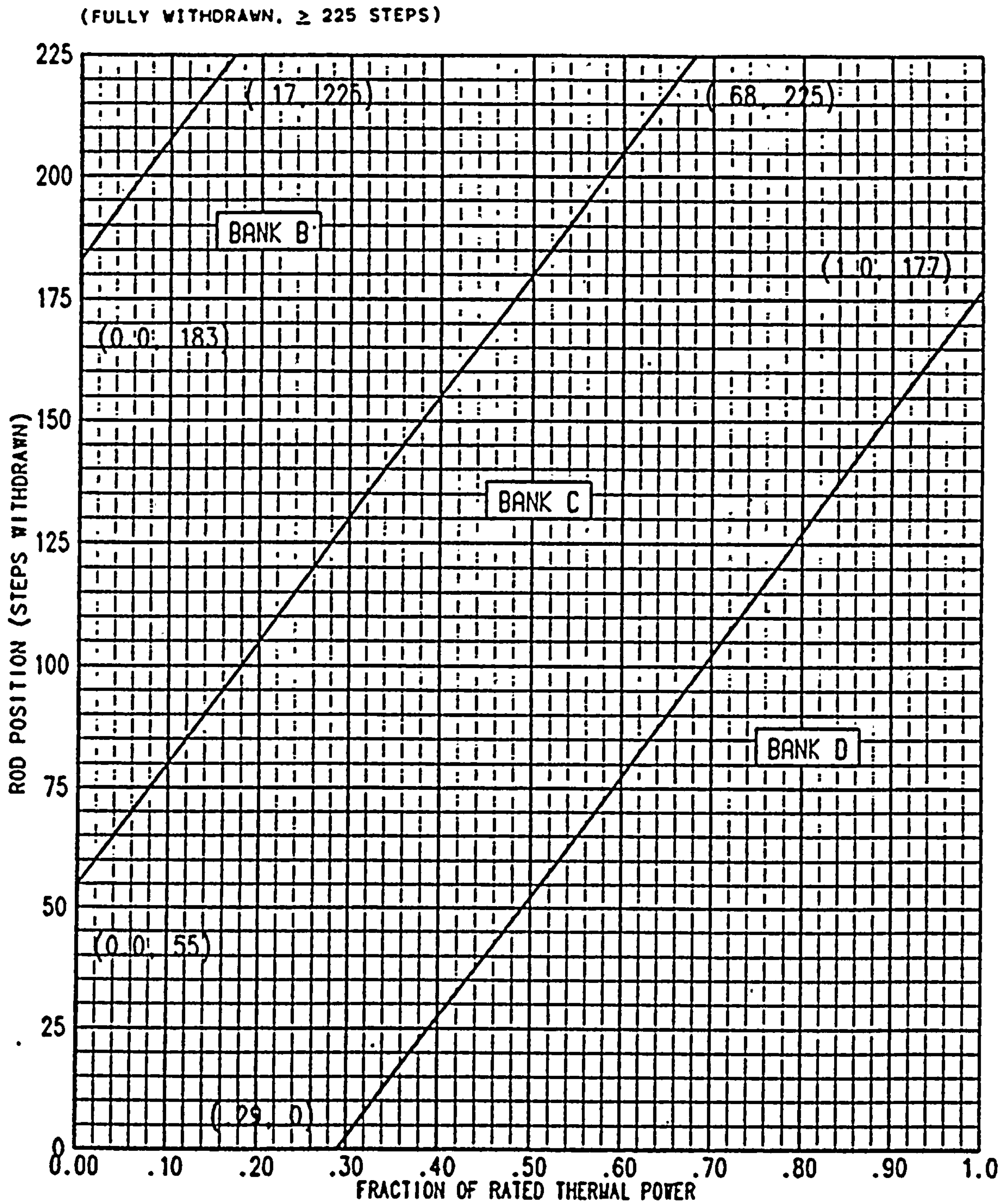


Figure 1

Rod Bank Insertion Limits Versus Rated Thermal Power

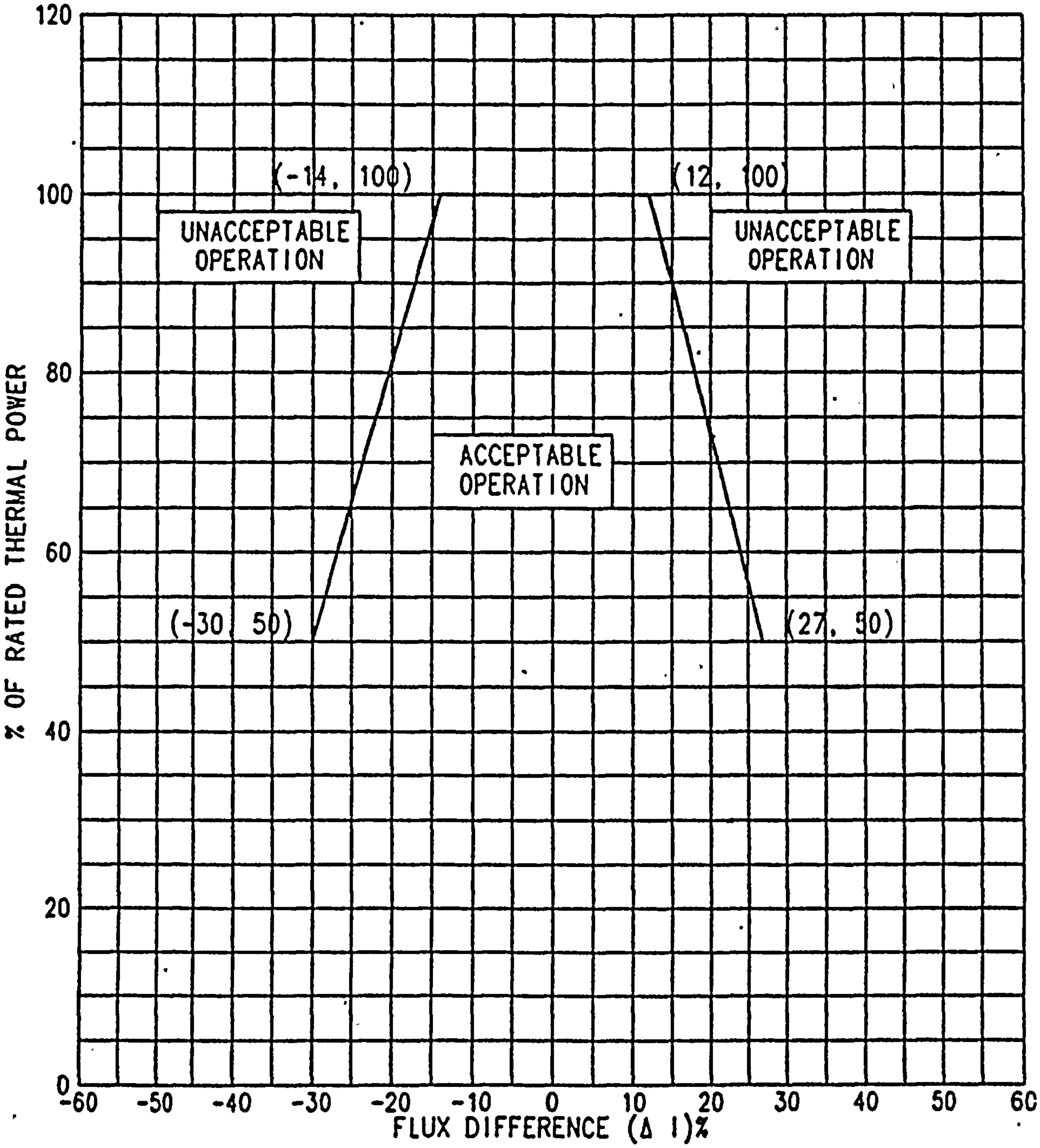
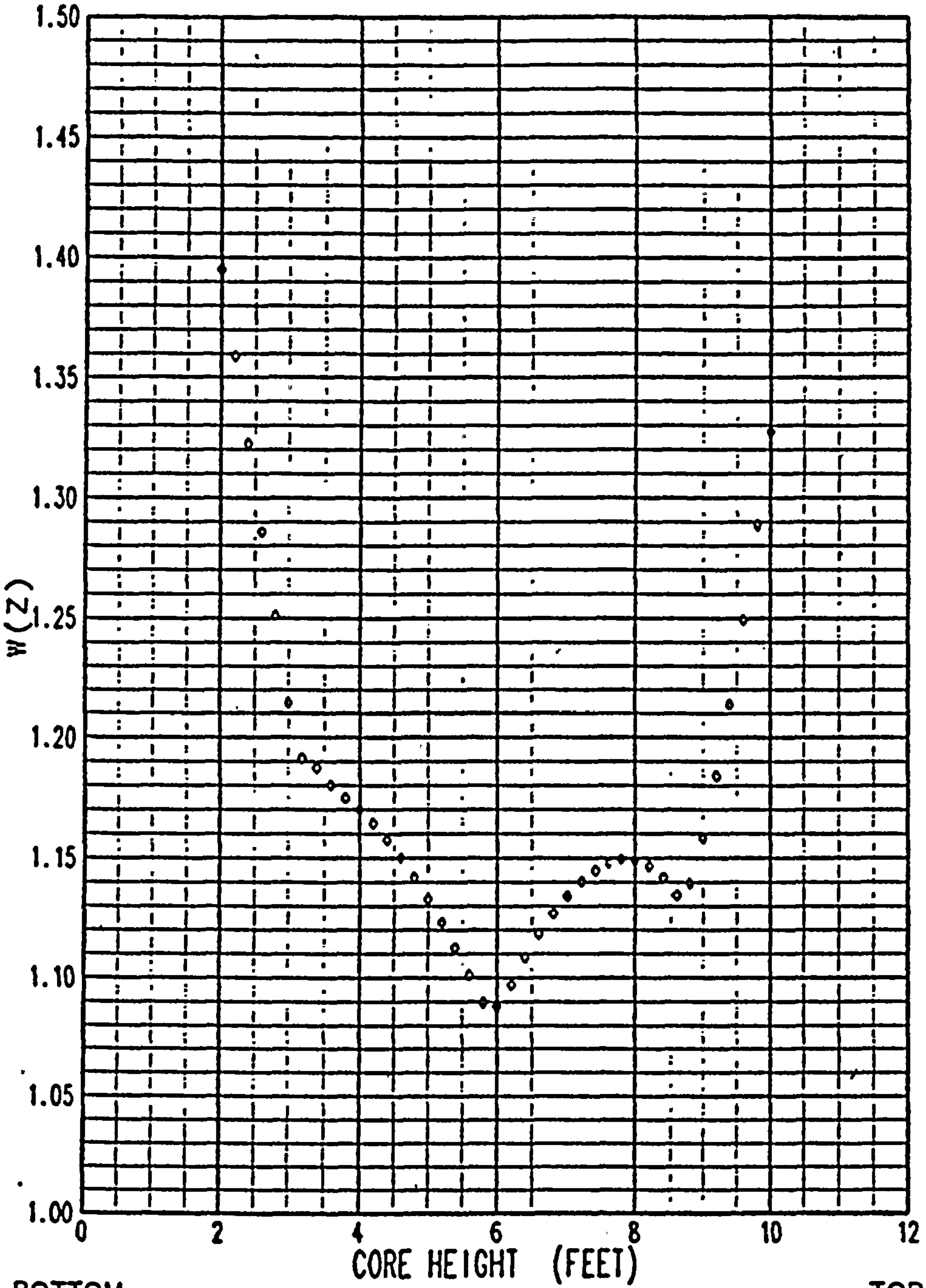


Figure 2

AFD Limits as a Function of Rated Thermal Power

TITLE: COLR FOR DIABLO CANYON UNIT 1 CYCLE 5



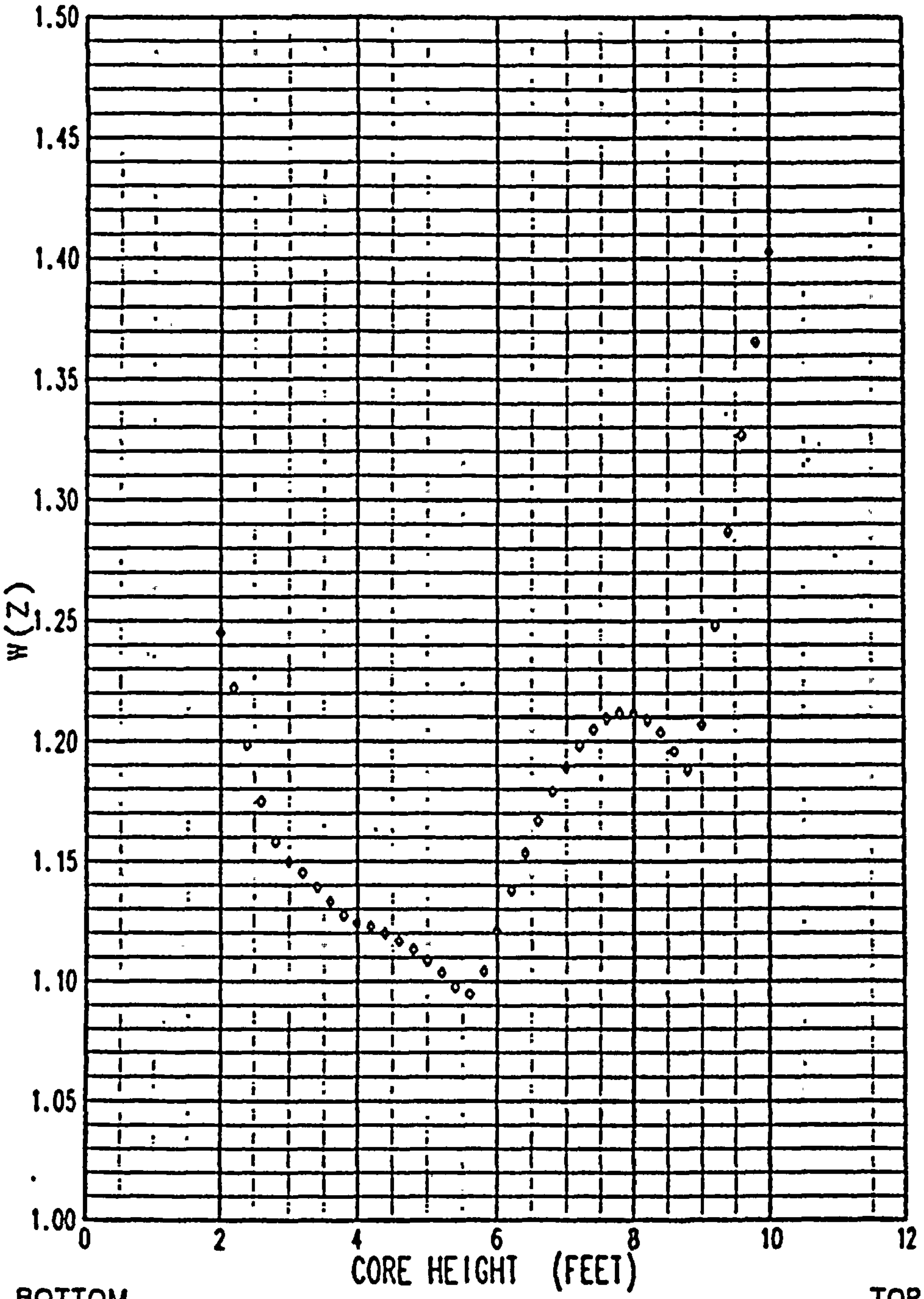
HEIGHT (FEET)	BOL W(Z)
• 0.0000	1.0000
• 0.2000	1.0000
• 0.4000	1.0000
• 0.6000	1.0000
• 0.8000	1.0000
• 1.0000	1.0000
• 1.2000	1.0000
• 1.4000	1.0000
• 1.6000	1.0000
• 1.8000	1.0000
2.0000	1.3951
2.2000	1.3588
2.4000	1.3223
2.6000	1.2859
2.8000	1.2510
3.0000	1.2144
3.2000	1.1914
3.4000	1.1876
3.6000	1.1803
3.8000	1.1748
4.0000	1.1700
4.2000	1.1642
4.4000	1.1576
4.6000	1.1502
4.8000	1.1420
5.0000	1.1329
5.2000	1.1230
5.4000	1.1124
5.6000	1.1009
5.8000	1.0895
6.0000	1.0880
6.2000	1.0970
6.4000	1.1084
6.6000	1.1183
6.8000	1.1269
7.0000	1.1342
7.2000	1.1402
7.4000	1.1449
7.6000	1.1481
7.8000	1.1495
8.0000	1.1489
8.2000	1.1468
8.4000	1.1420
8.6000	1.1344
8.8000	1.1395
9.0000	1.1582
9.2000	1.1839
9.4000	1.2135
9.6000	1.2494
9.8000	1.2887
10.000	1.3275
• 10.200	1.0000
• 10.400	1.0000
• 10.600	1.0000
• 10.800	1.0000
• 11.000	1.0000
• 11.200	1.000C
• 11.400	1.0000
• 11.600	1.0000
• 11.800	1.0000
• 12.000	1.0000

Figure 3

W(Z) at 150 MWD/MTU

* Top and Bottom 15% excluded as per TS 4.2.2.~~1~~ 9 2.g

TITLE: COLR FOR DIABLO CANYON UNIT 1 CYCLE 5



HEIGHT (FEET)	MOL W(Z)
• 0.0000	1.0000
• 0.2000	1.0000
• 0.4000	1.0000
• 0.6000	1.0000
• 0.8000	1.0000
• 1.0000	1.0000
• 1.2000	1.0000
• 1.4000	1.0000
• 1.6000	1.0000
• 1.8000	1.0000
2.0000	1.2451
2.2000	1.2221
2.4000	1.1984
2.6000	1.1749
2.8000	1.1581
3.0000	1.1499
3.2000	1.1451
3.4000	1.1392
3.6000	1.1331
3.8000	1.1273
4.0000	1.1242
4.2000	1.1226
4.4000	1.1199
4.6000	1.1168
4.8000	1.1131
5.0000	1.1087
5.2000	1.1035
5.4000	1.0976
5.6000	1.0946
5.8000	1.1041
6.0000	1.1209
6.2000	1.1379
6.4000	1.1532
6.6000	1.1670
6.8000	1.1792
7.0000	1.1895
7.2000	1.1981
7.4000	1.2050
7.6000	1.2095
7.8000	1.2118
8.0000	1.2116
8.2000	1.2089
8.4000	1.2037
8.6000	1.1960
8.8000	1.1881
9.0000	1.2070
9.2000	1.2480
9.4000	1.2871
9.6000	1.3268
9.8000	1.3659
10.000	1.4028
• 10.200	1.0000
• 10.400	1.0000
• 10.600	1.0000
• 10.800	1.0000
• 11.000	1.0000
• 11.200	1.0000
• 11.400	1.0000
• 11.600	1.0000
• 11.800	1.0000
• 12.000	1.0000

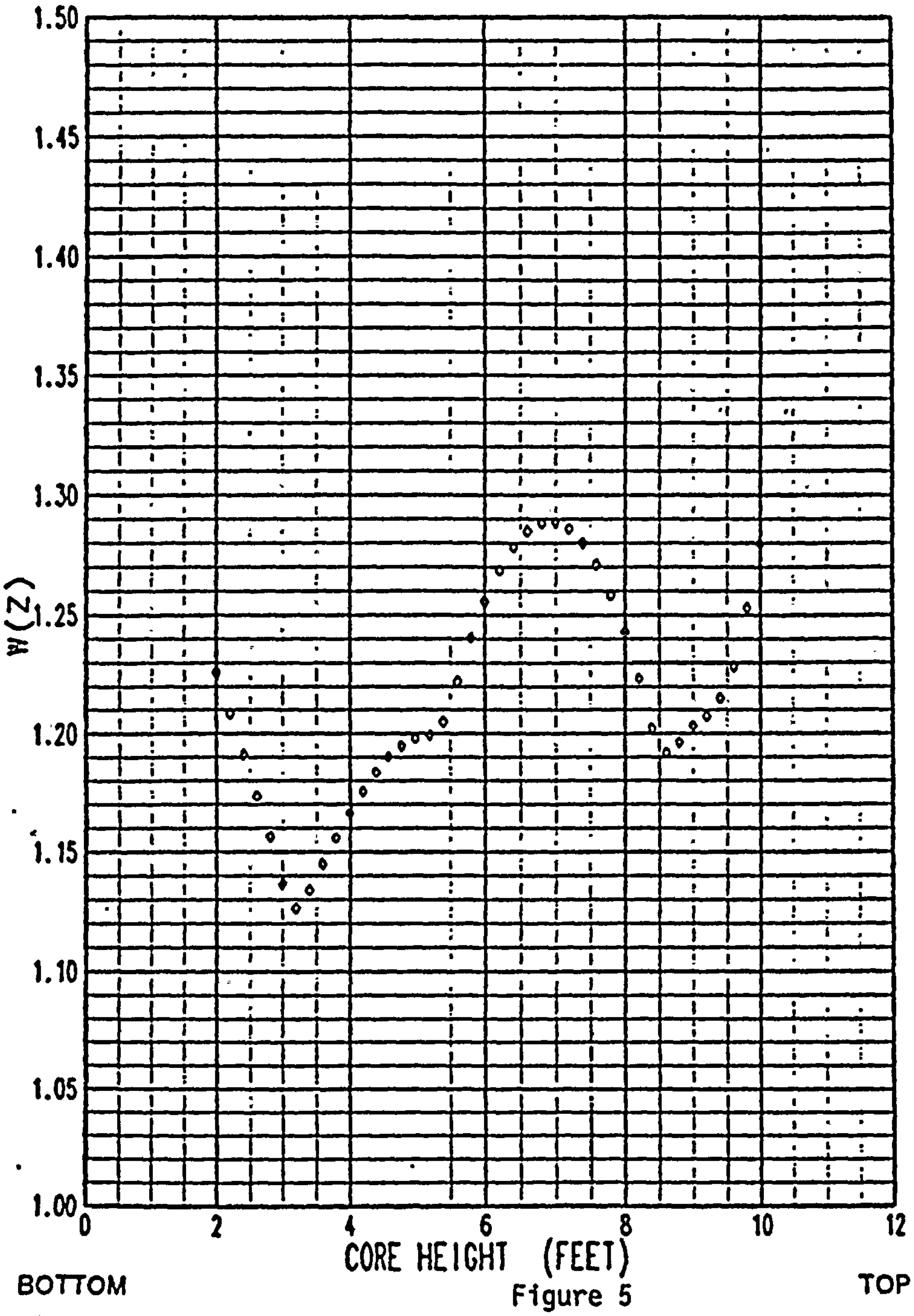
Figure 4

W(Z) at 6,000 MWD/MTU

9

* Top and Bottom 15% excluded as per TS 4.2.2.~~1~~2.g

TITLE: COLR FOR DIABLO CANYON UNIT 1 CYCLE 5



HEIGHT (FEET)	EOL W(Z)
• 0.0000	1.0000
• 0.2000	1.0000
• 0.4000	1.0000
• 0.6000	1.0000
• 0.8000	1.0000
• 1.0000	1.0000
• 1.2000	1.0000
• 1.4000	1.0000
• 1.6000	1.0000
• 1.8000	1.0000
2.0000	1.2257
2.2000	1.2085
2.4000	1.1910
2.6000	1.1734
2.8000	1.1565
3.0000	1.1368
3.2000	1.1264
3.4000	1.1338
3.6000	1.1448
3.8000	1.1559
4.0000	1.1663
4.2000	1.1756
4.4000	1.1836
4.6000	1.1902
4.8000	1.1950
5.0000	1.1980
5.2000	1.1995
5.4000	1.2050
5.6000	1.2221
5.8000	1.2403
6.0000	1.2558
6.2000	1.2685
6.4000	1.2782
6.6000	1.2849
6.8000	1.2882
7.0000	1.2886
7.2000	1.2860
7.4000	1.2802
7.6000	1.2709
7.8000	1.2583
8.0000	1.2428
8.2000	1.2232
8.4000	1.2023
8.6000	1.1919
8.8000	1.1965
9.0000	1.2032
9.2000	1.2074
9.4000	1.2150
9.6000	1.2284
9.8000	1.2528
10.000	1.2798
• 10.200	1.0000
• 10.400	1.0000
• 10.600	1.0000
• 10.800	1.0000
• 11.000	1.0000
• 11.200	1.0000
• 11.400	1.0000
• 11.600	1.0000
• 11.800	1.0000
• 12.000	1.0000

Figure 5

W(Z) at 16,000 MWD/MTU

* Top and Bottom 15% excluded as per TS 4.2.2.~~1~~ 9 2.g

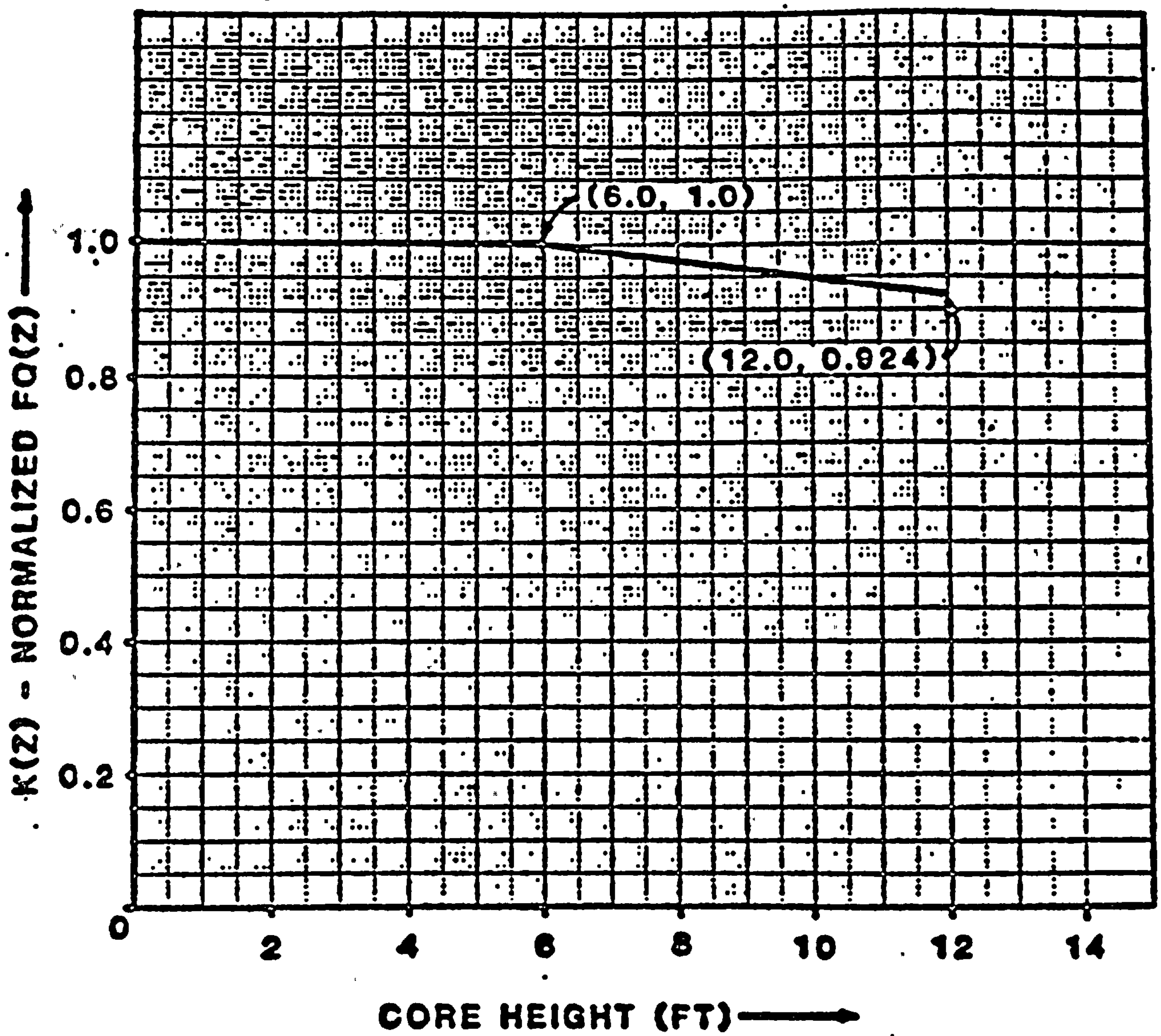


Figure 6

K(z) - Normalized FQ(z) as a Function of Core Height

FOR INFORMATION ONLY

PACIFIC GAS AND ELECTRIC COMPANY

DEPARTMENT OF NUCLEAR POWER GENERATION

DIABLO CANYON POWER PLANT

NUMBER COLR 2-4
REVISION ~~0~~ 1
PAGE 1 OF ~~7~~ 8
UNIT

2

TITLE: CORE OPERATING LIMITS REPORT
COLR FOR DIABLO CANYON UNIT 2 CYCLE 4

APPROVED: XXXXXXXXXXXXXXXXXXXXXXXXXXX 04/24/90 DATE 04/25/90 EFFECTIVE DATE

** PROCEDURE CLASSIFICATION - QUALITY RELATED **
** THIS PROCEDURE CONTAINS GRAPHICS. REFER TO CONTROLLED HARD COPY. **

1.0 CORE OPERATING LIMITS REPORT

This Core Operating Limits Report (COLR) for Diablo Canyon Unit 2 Cycle 4 has been prepared in accordance with the requirements of Technical Specification 6.9.1.8.

The Technical Specifications affected by this report are listed below:

3/4.1.3.5	Shutdown Rod Insertion Limit
3/4.1.3.6	Control Rod Insertion Limits
3/4.2.1	Axial Flux Difference
3/4.2.2	Heat Flux Hot Channel Factor - $F_Q(z)$ ~~(Surveillance Requirements only)~~
3/4.2.3	RCS Flow Rate and Nuclear Enthalpy Rise Hot Channel Factor - $F^N_{\Delta H}$

2.0 OPERATING LIMITS

The cycle-specific parameter limits for the specifications listed in Section 1.0 are presented in the following subsections. These limits have been developed using the NRC-approved methodologies specified in Technical Specification 6.9.1.8.

2.1 Shutdown Rod Insertion Limit (Specification 3/4.1.3.5)

2.1.1 The shutdown rods shall be withdrawn to at least 225 steps.

2.2 Control Rod Insertion Limits (Specification 3/4.1.3.6)

2.2.1 The control rod banks shall be limited in physical insertion as shown in Figure 1.

2.3 Axial Flux Difference (Specification 3/4.2.1)

2.3.1 The AXIAL FLUX DIFFERENCE (AFD) Limits for Unit 2 Cycle 4 are provided in Figure 2.

2.4 Heat Flux Hot Channel Factor - $F_Q(z)$ (Specification 3/4.2.2)

INSERT A →

~~2.4.1 The W(z) curves for Relaxed Axial Offset Control (RAOC) operation, provided in Figures 3 through 5 for Unit 2, Cycle 4, are sufficient to determine the RAOC W(z) versus core height for Cycle 4 burnups through the end of full power reactivity plus a power coastdown of up to 1000 MWD/MTU.~~

3.0 FIGURES

3.1 Figure 1 - Rod Bank Insertion Limits Versus Thermal Power

3.2 Figure 2 - AFD Limits as a Function of Rated Thermal Power

3.3 Figure 3 - W(z) at 150 MWD//MTU as a Function of Core Height

3.4 Figure 4 - W(z) at 8000 MWD/MTU as a Function of Core Height

3.5 Figure 5 - W(z) at 18000 MWD/MTU as a Function of Core Height

3.6 Figure 6 - Normalized $F_Q(z)$ as a Function of Core Height

4.0 REFERENCE

Westinghouse Reload Safety Evaluation for Diablo Canyon Power Plant Unit 2 Cycle 4, Rev. 1, dated April 1990

~~4~~ 5.0 RECORD OF REVIEWS

5 ~~4~~.1 Prepared by: ~~Michael L. O'Connell~~ C. A. Hieb

5 ~~4~~.2 Sponsored by: ~~Michael L. O'Connell~~ J. R. HINDS

5 ~~4~~.3 Independent Technical Review by: ~~Jacqueline R. Hinds~~ T. L. GREBEL

5 ~~4~~.4 Cross Discipline Review by: ~~Peter G. Sarafian~~

6.0 RECORDS

NONE

**********INSERT A***********

2.4.1

$$F_Q(Z) \leq \frac{F_Q^{RTP}}{P} * K(Z) \qquad \text{for } P > 0.5$$

$$F_Q(Z) \leq \frac{F_Q^{RTP}}{0.5} * K(Z) \qquad \text{for } P \leq 0.5$$

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

$F_Q^{RTP} = 2.45$

K(Z) is provided in Figure 6.

2.4.2 The W(Z) curves for Specification 4.2.2.2.c Relaxed Axial Offset Control (RAOC) operation, provided in figures 3 through 5, are sufficient to determine the RAOC W(Z) versus core height for Cycle burnups through the end of full power reactivity plus a power coastdown of up to 1000 MWD/MTU.

2.5 <u>RCS Flow Rate and Nuclear Enthalpy Rise Hot Channel Factor</u> - $F_{\Delta H}^N$ (Specification 3/4.2.3)

$$R = \frac{F_{\Delta H}^N}{F_{\Delta H}^{RTP} * (1 + PF_{\Delta H} * (1-P))}$$

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

$F_{\Delta H}^{RTP}$ = 1.56 (LOPAR fuel)
= 1.59 (VANTAGE 5 fuel)

$PF_{\Delta H} = 0.3$

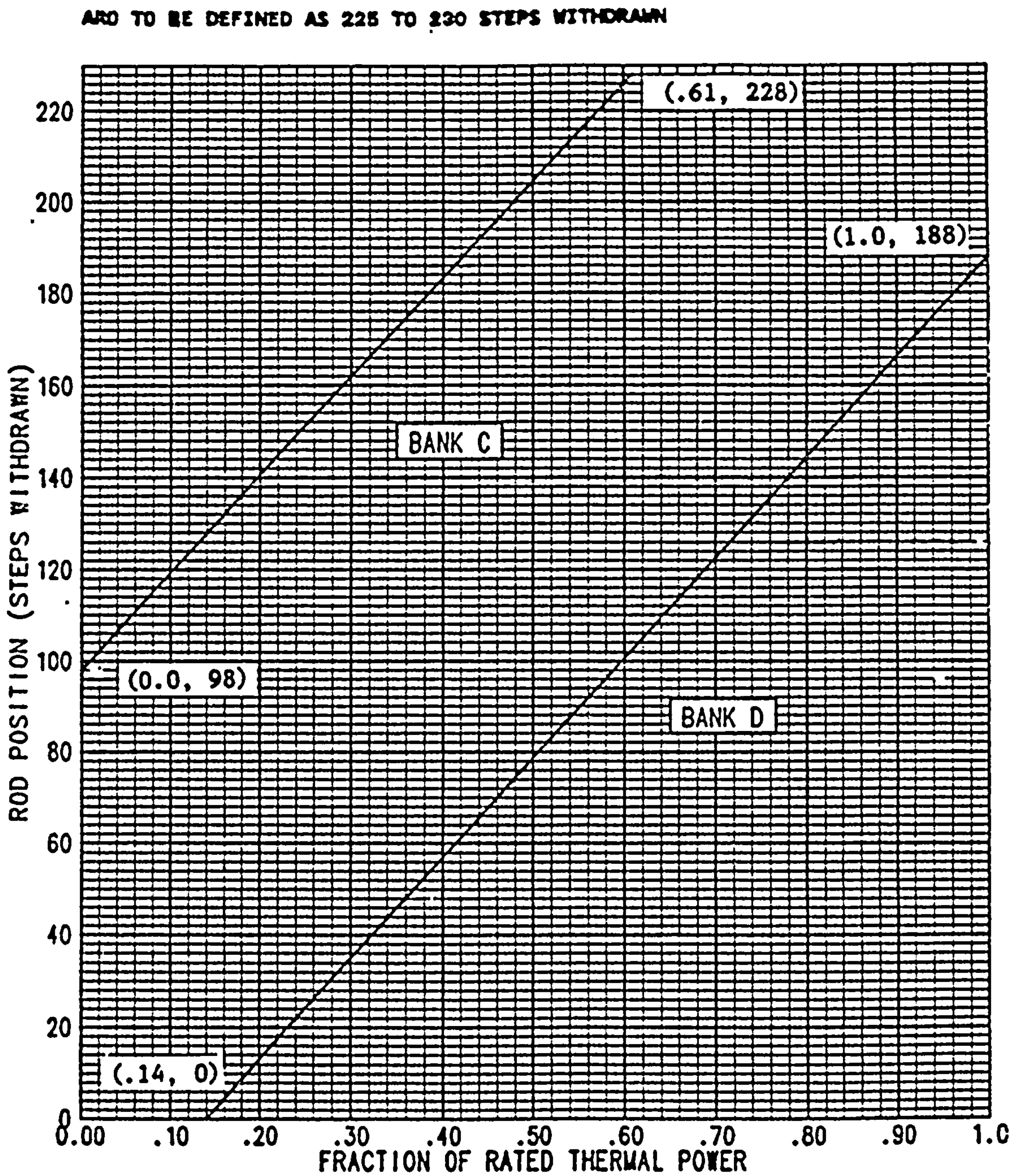


Figure 1

Rod Bank Insertion Limits Versus Thermal Power

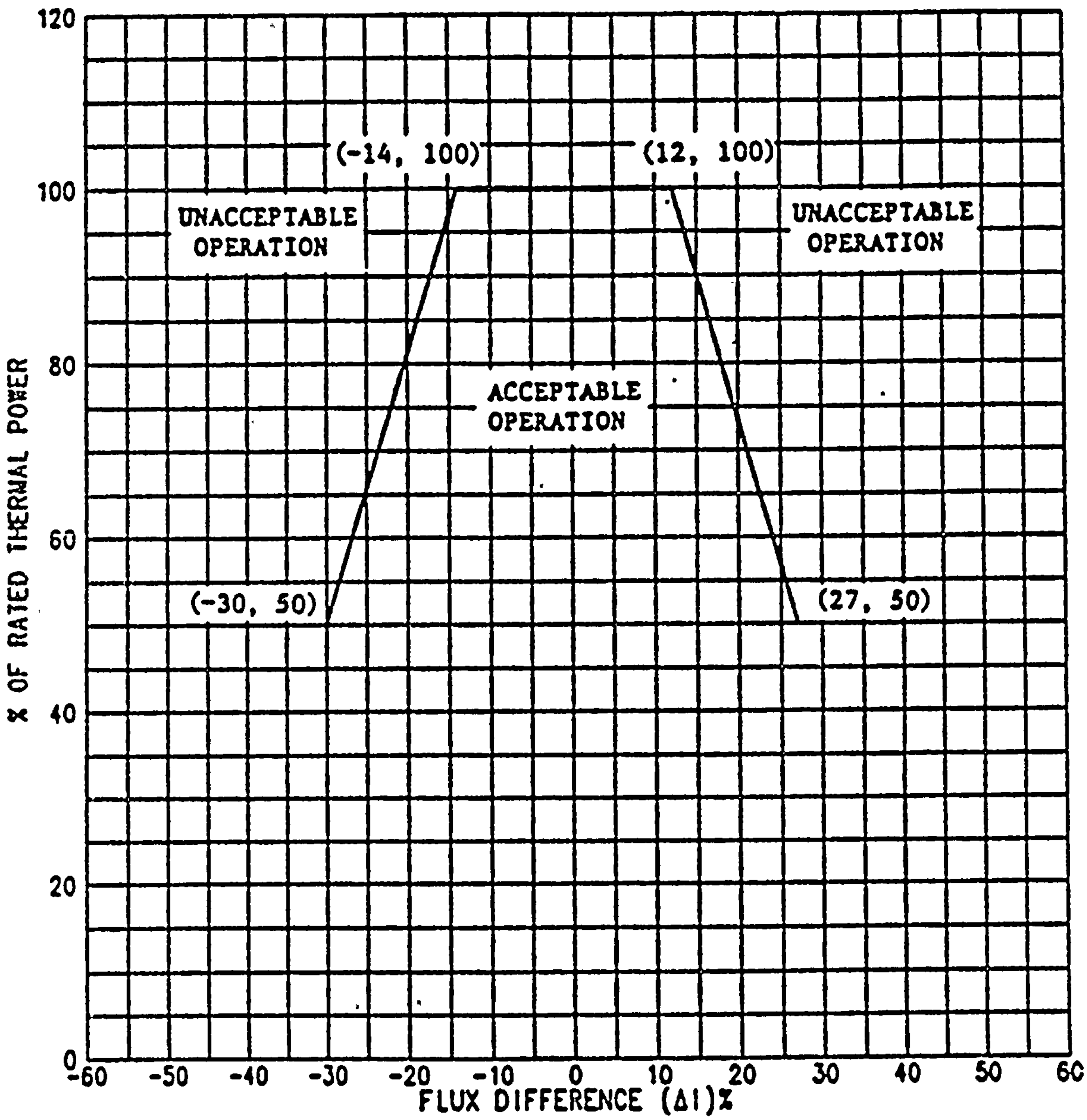
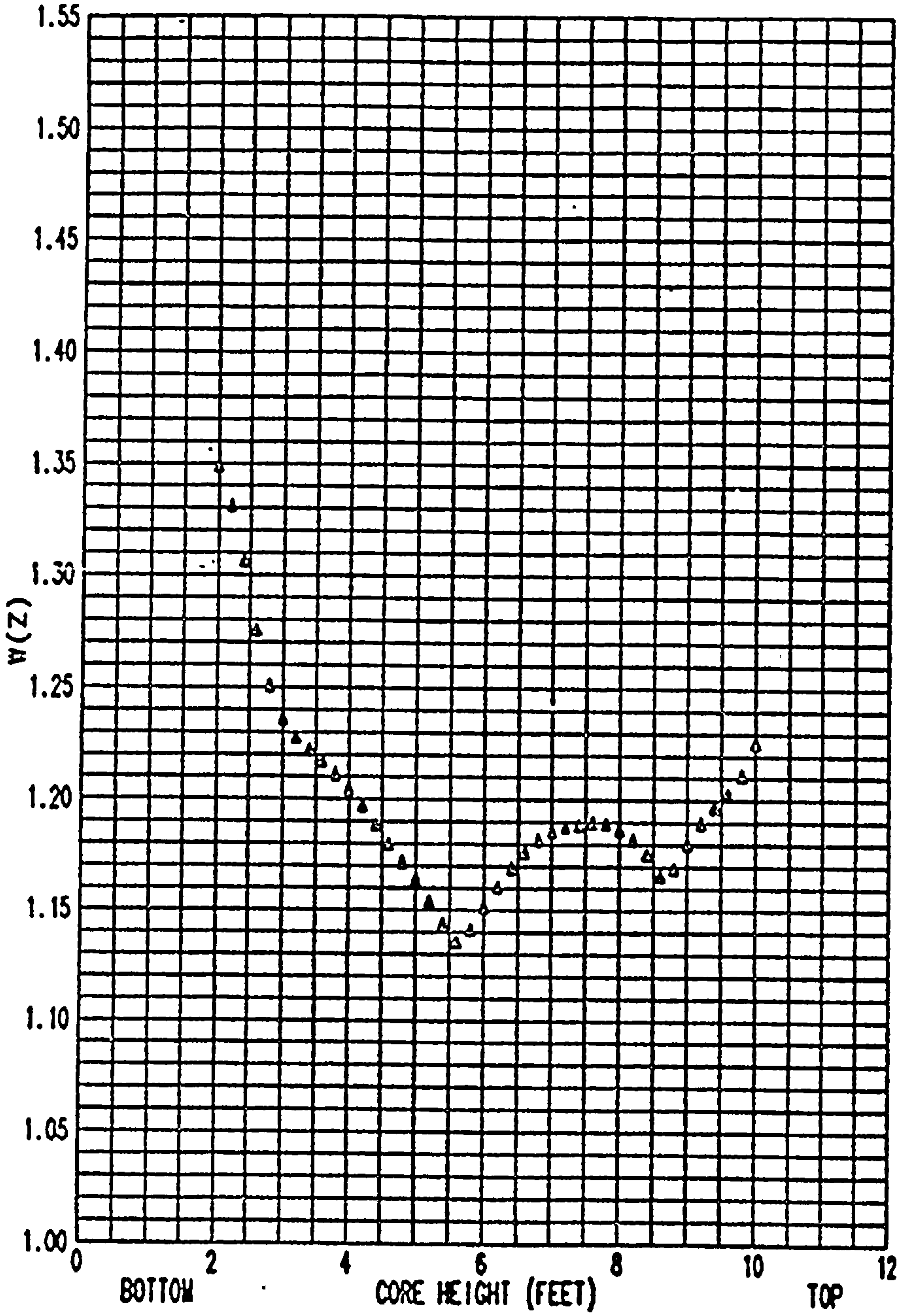


Figure 2

AFD Limits as a function of Rated Thermal Power



Height (Feet)	DOL W(Z)
* 0.0000	1.0000
* 0.2000	1.0000
* 0.4000	1.0000
* 0.6000	1.0000
* 0.8000	1.0000
* 1.0000	1.0000
* 1.2000	1.0000
* 1.4000	1.0000
* 1.6000	1.0000
* 1.8000	1.0000
2.0000	1.3488
2.2000	1.3312
2.4000	1.3034
2.6000	1.2787
2.8000	1.2507
3.0000	1.2389
3.2000	1.2279
3.4000	1.2221
3.6000	1.2171
3.8000	1.2113
4.0000	1.2048
4.2000	1.1972
4.4000	1.1888
4.6000	1.1800
4.8000	1.1719
5.0000	1.1635
5.2000	1.1540
5.4000	1.1438
5.6000	1.1357
5.8000	1.1409
6.0000	1.1515
6.2000	1.1606
6.4000	1.1692
6.6000	1.1764
6.8000	1.1820
7.0000	1.1861
7.2000	1.1881
7.4000	1.1887
7.6000	1.1899
7.8000	1.1885
8.0000	1.1868
8.2000	1.1825
8.4000	1.1758
8.6000	1.1660
8.8000	1.1684
9.0000	1.1804
9.2000	1.1899
9.4000	1.1973
9.6000	1.2028
9.8000	1.2113
10.000	1.2258
* 10.200	1.0000
* 10.400	1.0000
* 10.600	1.0000
* 10.800	1.0000
* 11.000	1.0000
* 11.200	1.0000
* 11.400	1.0000
* 11.600	1.0000
* 11.800	1.0000
* 12.000	1.0000

* Top and Bottom 15% excluded as per Technical Specification 4.2.2.2.2.g

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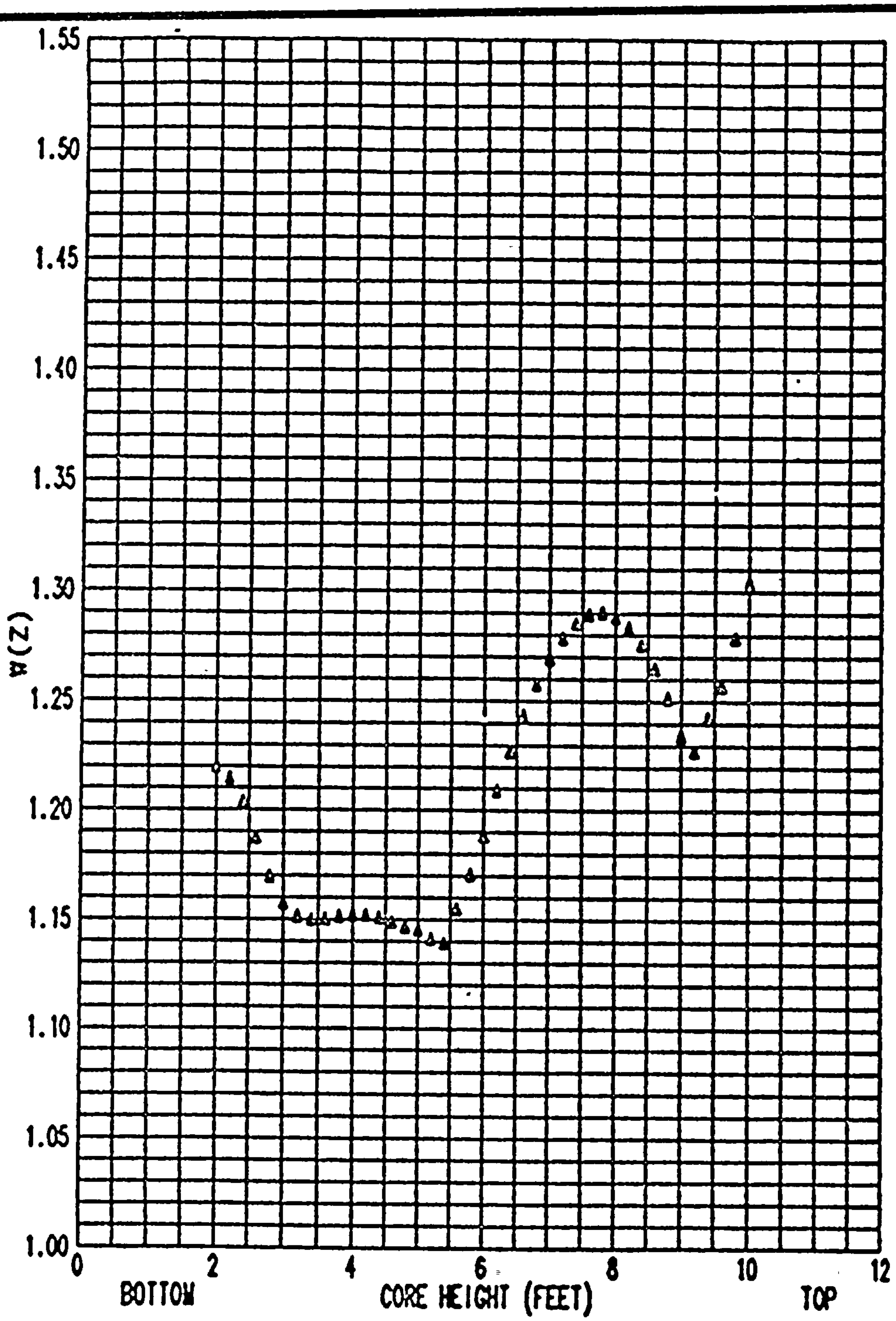
Figure 3

W(z) at 150 MWD/MTU as a Function of Core Height

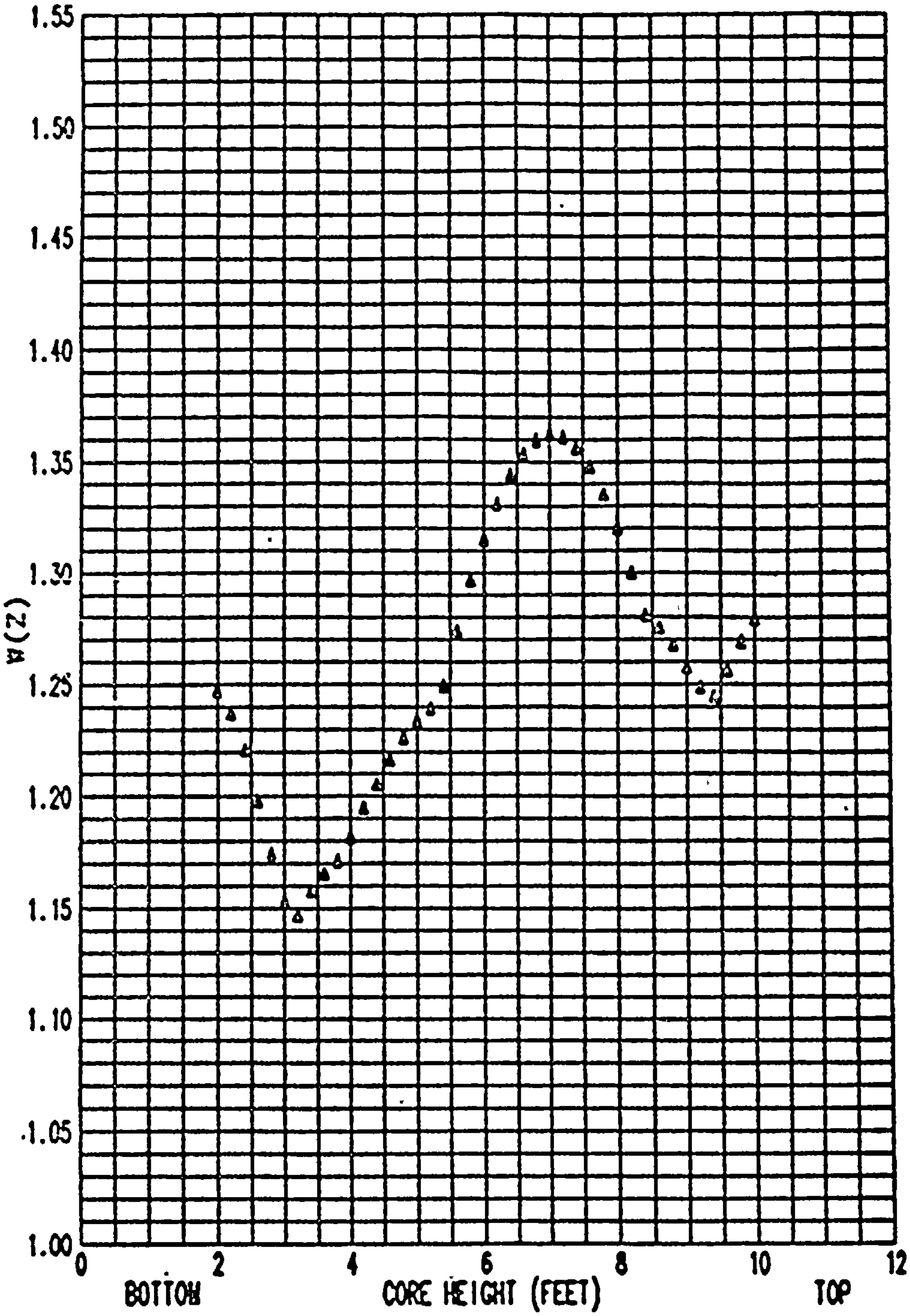


Height (Feet)	MOL W(z)
* 0.0000	1.0000
* 0.2000	1.0000
* 0.4000	1.0000
* 0.6000	1.0000
* 0.8000	1.0000
* 1.0000	1.0000
* 1.2000	1.0000
* 1.4000	1.0000
* 1.6000	1.0000
* 1.8000	1.0000
2.0000	1.2188
2.2000	1.2147
2.4000	1.2037
2.6000	1.1875
2.8000	1.1699
3.0000	1.1574
3.2000	1.1515
3.4000	1.1490
3.6000	1.1504
3.8000	1.1515
4.0000	1.1521
4.2000	1.1520
4.4000	1.1510
4.6000	1.1490
4.8000	1.1467
5.0000	1.1453
5.2000	1.1413
5.4000	1.1384
5.6000	1.1552
5.8000	1.1707
6.0000	1.1884
6.2000	1.2083
6.4000	1.2272
6.6000	1.2424
6.8000	1.2[illegible]
7.0000	[illegible]
7.2000	[illegible]
7.4000	1.2[illegible]4
7.6000	1.2883
7.8000	1.2903
8.0000	1.2884
8.2000	1.2835
8.4000	1.2757
8.6000	1.2648
8.8000	1.2515
9.0000	1.2342
9.2000	1.2275
9.4000	1.2424
9.6000	1.2571
9.8000	1.2788
10.000	1.3041
* 10.200	1.0000
* 10.400	1.0000
* 10.600	1.0000
* 10.800	1.0000
* 11.000	1.0000
* 11.200	1.0000
* 11.400	1.0000
* 11.600	1.0000
* 11.800	1.0000
* 12.000	1.0000

* Top and Bottom 10% excluded as per Technical Specification 4.2.2.2.g

Figure 4

W(z) at 8000 MWD/MTU as a Function of Core Height



Height (Feet)	EOL W(z)
* 0.0000	1.0000
* 0.2000	1.0000
* 0.4000	1.0000
* 0.6000	1.0000
* 0.8000	1.0000
* 1.0000	1.0000
* 1.2000	1.0000
* 1.4000	1.0000
* 1.6000	1.0000
* 1.8000	1.0000
2.0000	1.2474
2.2000	1.2374
2.4000	1.2208
2.6000	1.1983
2.8000	1.1744
3.0000	1.1530
3.2000	1.1489
3.4000	1.1578
3.6000	1.1653
3.8000	1.1712
4.0000	1.1818
4.2000	1.1948
4.4000	1.2064
4.6000	1.2168
4.8000	1.2263
5.0000	1.2339
5.2000	1.2394
5.4000	1.2484
5.6000	1.2730
5.8000	1.2962
6.0000	1.3152
6.2000	1.3313
6.4000	1.3441
6.6000	1.3536
6.8000	1.3597
7.0000	1.3622
7.2000	1.3611
7.4000	1.3563
7.6000	1.3478
7.8000	1.3355
8.0000	1.3200
8.2000	1.3001
8.4000	1.2809
8.6000	1.2764
8.8000	1.2676
9.0000	1.2576
9.2000	1.2498
9.4000	1.2447
9.6000	1.2564
9.8000	1.2691
10.000	1.2795
* 10.200	1.0000
* 10.400	1.0000
* 10.600	1.0000
* 10.800	1.0000
* 11.000	1.0000
* 11.200	1.0000
* 11.400	1.0000
* 11.600	1.0000
* 11.800	1.0000
* 12.000	1.0000

* Top and Bottom 15% excluded as per Technical Specification 4.2.2.~~4~~6.2.g

Figure 5

W(z) at 18000 MWD/MTU as a Function of Core Height

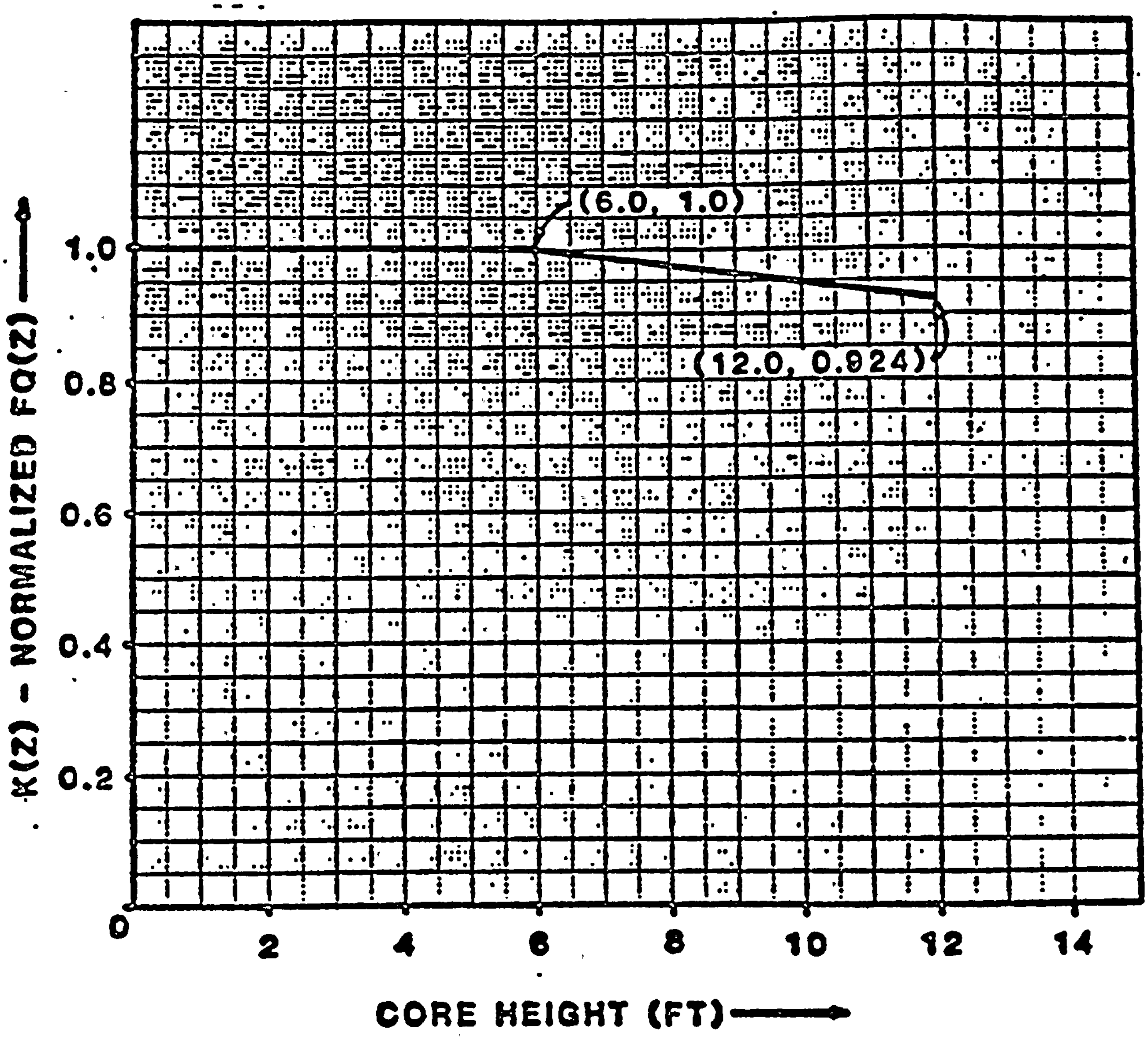


Figure 6

K(z) - Normalized $F_Q(z)$ as a Function of Core Height

TABLE 3.6-1

CONTAINMENT ISOLATION VALVES

VALVE NO.	FUNCTION	ISOLATION TIME (Seconds)
1. Phase "A" Isolation Valves		
FCV-151#	Steam Generator No. 1 Blowdown OC	$\leq$ 10
FCV-154#	Steam Generator No. 2 Blowdown OC	$\leq$ 10
FCV-157#	Steam Generator No. 3 Blowdown OC	$\leq$ 10
FCV-160#	Steam Generator No. 4 Blowdown OC	$\leq$ 10
FCV-244#	Steam Generator No. 4 Sample OC	$\leq$ 10
FCV-246#	Steam Generator No. 3 Sample OC	$\leq$ 10
FCV-248#	Steam Generator No. 2 Sample OC	$\leq$ 10
FCV-250#	Steam Generator No. 1 Sample OC	$\leq$ 10
FCV-253	Reactor Coolant Dr. Tk. PP Disch. Isol. IC	$\leq$ 10
FCV-254	Reactor Coolant Dr. Tk. PP Disch. OC	$\leq$ 10
FCV-255	Reactor Coolant Dr. Tk. Vent. Isol. IC	$\leq$ 10
FCV-256	Reactor Coolant Dr. Tk. Vent. Isol. OC	$\leq$ 10
FCV-257	Reactor Coolant Dr. Tk. Sample to GA OC	$\leq$ 10
FCV-258	Reactor Coolant Dr. Tk. Sample to GA IC	$\leq$ 10
FCV-260	Reactor Coolant Dr. Tk. N_2 Supply OC	$\leq$ 10
FCV-361	CCW Return from Excess Letdown HX OC	$\leq$ 10
FCV-500	Containment Sump Discharge Isolation IC	$\leq$ 10
FCV-501	Containment Sump Discharge Isolation OC	$\leq$ 10
FCV-584	Containment Instrument Air Supply OC	$\leq$ 10
FCV-633	Containment Fire Water Isolation OC	$\leq$ 10
FCV-654	Incore Cooler Chilled H_2O Supply OC	$\leq$ 10
FCV-655	Incore Cooler Chilled H_2O Supply IC	$\leq$ 10

DELETE

TABLE 3.6-1 (Continued)

VALVE NO.	FUNCTION	ISOLATION TIME (Seconds)
1. Phase "A" Isolation Valves (Continued)		
FCV-656	Incore Cooler Chilled H_2O Return OC	$\leq$ 10
FCV-657	Incore Cooler Chilled H_2O Return IC	$\leq$ 10
8029	Primary H_2O Supply to Pressurizer Relief Tk. OC	$\leq$ 10
8034A	Pressurizer Relief Tk. to GA IC	$\leq$ 10
8034B	Pressurizer Relief Tk. to GA OC	$\leq$ 10
8045	Pressurizer Relief Tk. N_2 Supply OC	$\leq$ 10
8149A	Letdown Orifice RO-27 Outlet IC	$\leq$ 10
8149B	Letdown Orifice RO-28 Outlet IC	$\leq$ 10
8149C	Letdown Orifice RO-29 Outlet IC	$\leq$ 10
8152	Letdown Line Isolation OC	$\leq$ 10
8871	ECCS Check Valve Test Line IC	$\leq$ 10
8880	Accumulator N_2 Fill OC	$\leq$ 10
8883	ECCS Check Valve Test Line OC	$\leq$ 10
8961	ECCS Check Valve Test Line OC	$\leq$ 10
9354A	Pressurizer Steam Space Sample IC	$\leq$ 10
9354B	Pressurizer Steam Space Sample OC	$\leq$ 10
9355A	Pressurizer Liquid Space Sample IC	$\leq$ 10
9355B	Pressurizer Liquid Space Sample OC	$\leq$ 10
9356A	RCS Hot Leg Sample IC	$\leq$ 10
9356B	RCS Hot Leg Sample OC	$\leq$ 10
9357A	Accumulator Sample IC	$\leq$ 10

DELETE

TABLE 3.6-1 (Continued)

VALVE NO.	FUNCTION	ISOLATION TIME (Seconds)
1. Phase "A" Isolation Valves (Continued)		
9357B	Accumulator Sample OC	≤ 10
8100	RCP Seal Water Return OC	≤ 10
8112	RCP Seal Water Return IC	≤ 10
2. Phase "B" Isolation Valves		
FCV-356	CCW Supply to RCP's and Support Coolers OC	N.A.
FCV-357	RCP Thermal Barrier CCW Return OC	N.A.
FCV-363	RCP Oil Cooler/Support Cooler CCW Return OC	N.A.
FCV-749	RCP Oil Cooler/Support Cooler CCW Return IC	N.A.
FCV-750	RCP Thermal Barrier CCW Return IC	N.A.
3. Containment Ventilation Isolation Valves		
FCV-660##	Containment Purge Supply IC	≤ 2
FCV-661##	Containment Purge Supply OC	≤ 2
FCV-662	Containment Vacuum/Pressure Relief IC	≤ 5
FCV-663	Containment Pressure Relief OC	≤ 5
FCV-664	Containment Vacuum Relief OC	≤ 5
FCV-678	Containment Air Sample Supply IC	≤ 10
FCV-679	Containment Air Sample Supply OC	≤ 10
FCV-681	Containment Air Sample Return OC	≤ 10
RCV-11##	Containment Purge Exhaust IC	≤ 2
RCV-12##	Containment Purge Exhaust OC	≤ 2

DELETE

TABLE 3.6-1 (Continued)

VALVE NO.	FUNCTION	ISOLATION TIME (Seconds)
4. Manual Valves		
AIR-I-585*	Instrument Air Supply to Containment (FCV-584 Bypass) OC	N.A.
AIR-S-200*	Service Air Supply to Containment OC	N.A.
AXS-26*	Aux. Steam Supply to Containment OC	N.A.
CS-31	Containment Spray to Misc. Equipment Drain Tank OC	N.A.
CS-32	Containment Spray to Misc. Equipment Drain Tank OC	N.A.
FW-140#	Auxiliary Feedwater to Stm. Gen. No. 1 OC	N.A.
FW-147#	Auxiliary Feedwater to Stm. Gen. No. 2 OC	N.A.
FW-153#	Auxiliary Feedwater to Stm. Gen. No. 3 OC	N.A.
FW-157#	Auxiliary Feedwater to Stm. Gen. No. 4 OC	N.A.
NS-902#	Nitrogen to Steam Generators OC	N.A.
RCS-512*	Miscellaneous Equipment Drain Tank Isolation Valve OC	N.A.
SI-161*	Isolating Valve FI-927 OC	N.A.
VAC-1*	Containment Hydrogen Purge Supply Fan No. 1 and External H_2 Recombiner to Containment OC	N.A.
VAC-2*	Containment Hydrogen Purge Supply Fan No. 2 and External H_2 Recombiner to Containment OC	N.A.
8767	Refueling Cavity to Refueling Water Purification Pump OC	N.A.
8787	Refueling Water Purification Pump to Refueling Cavity OC	N.A.

DELETE

TABLE 3.6-1 (Continued)

VALVE NO.	FUNCTION	ISOLATION TIME (Seconds)
4. Manual Valves (Continued)		
8795	Refueling Cavity to Refueling Water Purification Pump IC	N.A.
8796	Refueling Water Purification Pump to Refueling Cavity IC	N.A.
8969#	Charging Pump to S.I. Test Line OC	N.A.
PEN-65A*#	Main Airlock Equalizing Valve to Atmosphere	N.A.
PEN-65B*#	Main Airlock Equalizing Valve to Containment	N.A.
PEN-66A*#	Emergency Airlock Equalizing Valve to Atmosphere	N.A.
PEN-66B*#	Emergency Airlock Equalizing Valve to Containment	N.A.
5. Power-Operated Valves		
FCV-22#	No. 4 Stm. Gen. Mn. Steam Isol. Valve Bypass OC	N.A.
FCV-23#	No. 3 Stm. Gen. Mn. Steam Isol. Valve Bypass OC	N.A.
FCV-24#	No. 2 Stm. Gen. Mn. Steam Isol. Valve Bypass OC	N.A.
FCV-25#	No. 1 Stm. Gen. Mn. Steam Isol. Valve Bypass OC	N.A.
FCV-37#	Auxiliary FWP Turb. Steam Supply S/G No. 2 OC	N.A.
FCV-38#	Auxiliary FWP Turb. Steam Supply S/G No. 3 OC	N.A.
FCV-41#	No. 1 Stm. Generator Mn. Steam Isol. OC	≤ 5
FCV-42#	No. 2 Stm. Generator Mn. Steam Isol. OC	≤ 5
FCV-43#	No. 3 Stm. Generator Mn. Steam Isol. OC	≤ 5
FCV-44#	No. 4 Stm. Generator Mn. Steam Isol. OC	≤ 5

DELETE

TABLE 3.6-1 (Continued)

VALVE NO.	FUNCTION	ISOLATION TIME (Seconds)
5. Power-Operated Valves (Continued)		
FCV-235*	Containment H_2 Sample Supply IC	N.A.
FCV-236*	Containment H_2 Sample Supply OC	N.A.
FCV-237*	Containment H_2 Sample Return OC	N.A.
FCV-238*	Containment H_2 Sample Supply IC	N.A.
FCV-239*	Containment H_2 Sample Supply OC	N.A.
FCV-240*	Containment H_2 Sample Return OC	N.A.
FCV-658	Containment Purge to Aux. Bldg. Filters/ Ext. H_2 Recombiners Supply IC	N.A.
FCV-668	Containment Purge to Aux. Bldg. Filters/Ext. H_2 Recombiner Supply OC	N.A.
FCV-659	Containment Purge to Purge System Filters/Ext. H_2 Recombiners Supply IC	N.A.
FCV-669	Containment Purge to Purge System Filters/Ext. H_2 Recombiners Supply OC	N.A.
FCV-760#	Steam Generator No. 1 Blowdown IC	N.A.
FCV-761#	Steam Generator No. 2 Blowdown IC	N.A.
FCV-762#	Steam Generator No. 3 Blowdown IC	N.A.
FCV-763#	Steam Generator No. 4 Blowdown IC	N.A.
FCV-696*	Reactor Cavity Sump Sample (Post LOCA) Supply IC	N.A.
FCV-697*	Reactor Cavity Sump Sample (Post LOCA) Supply OC	N.A.

DELETE

TABLE 3.6-1 (Continued)

VALVE NO.	FUNCTION	ISOLATION TIME (Seconds)
5. Power-Operated Valves (Continued)		
FCV-698*	Containment Air Sample (Post LOCA) Supply IC	N.A.
FCV-699*	Containment Air Sample (Post LOCA) Supply OC	N.A.
FCV-700*	Containment Air Sample (Post LOCA) Return OC	N.A.
PCV-19#	Steam Generator No. 1 10% Atmosphere Steam Dump OC	N.A.
PCV-20#	Steam Generator No. 2 10% Atmosphere Steam Dump OC	N.A.
PCV-21#	Steam Generator No. 3 10% Atmosphere Steam Dump OC	N.A.
PCV-22#	Steam Generator No. 4 10% Atmosphere Steam Dump OC	N.A.
8107#	Charging Line Isolation OC	N.A.
8700A#	RCS Hot Leg to RHR Pump 1 OC	N.A.
8700B#	RCS Hot Leg to RHR Pump 2 OC	N.A.
8701#	RCS Loop 4 Hot Leg to RHR IC	N.A.
8703#	RHR to RCS Hot Legs 1 and 2 IC	N.A.
8716A#	RHR to RCS Hot Legs OC	N.A.
8716B#	RHR to RCS Hot Legs OC	N.A.
8801A#	Charging Injection OC	N.A.
8801B#	Charging Injection OC	N.A.
8802A#	Safety Injection to RCS Hot Legs OC	N.A.
8802B#	Safety Injection to RCS Hot Legs OC	N.A.
8809A#	Residual Heat Removal to RCS Cold Legs 1 and 2	N.A.
8809B#	Residual Heat Removal to RCS Cold Legs 3 and 4	N.A.
8823#	Safety Injection Check Valve Test Line IC	N.A.

DELETE

TABLE 3.6-1 (Continued)

VALVE NO.	FUNCTION	ISOLATION TIME (Seconds)
5. Power-Operated Valves (Continued)		
8824#	Safety Injection Check Valve Test Line IC	N.A.
8843#	Charging Injection IC	N.A.
8835#	Safety Injection to RCS Cold Legs OC	N.A.
8885A#	RHR to Cold Leg Test Line IC	N.A.
8885B#	RHR to Cold Leg Test Line IC	N.A.
8982A#	Containment Sump to Residual Heat Removal Train 1 OC	N.A.
8982B#	Containment Sump to Residual Heat Removal Train 2 OC	N.A.
8980#	Refueling Water Storage Tank to RHR OC	N.A.
9001A	Containment Spray Pump No. 1 Isolation OC	N.A.
9001B	Containment Spray Pump No. 2 Isolation OC	N.A.
9003A#	Residual Heat Removal to Containment Spray OC	N.A.
9003B#	Residual Heat Removal to Containment Spray OC	N.A.
6. Check Valves		
8028	Relief Valve Outlets to Pressurizer Relief Tank IC	N.A.
8046	Primary Water to Pressurizer Relief Tank IC	N.A.
8047	Nitrogen to Pressurizer Relief Tank IC	N.A.
8109	Seal Water Return IC	N.A.
8368A thru 8368D	Seal Water to Reactor Coolant Pumps IC	N.A.
8916	Nitrogen Supply to Accumulators IC	N.A.

DELETE

TABLE 3.6-1 (Continued)

VALVE NO.	FUNCTION	ISOLATION TIME (Seconds)
6. Check Valves (Continued)		
9011A	Containment Spray IC	N.A.
9011B	Containment Spray IC	N.A.
CCW-585	CCW Supply to RCP IC	N.A.
CCW-581	CCW Return from RCP (FCV-749 Bypass) IC	N.A.
CCW-670	CCW Return from RCP (FCV-750 Bypass) IC	N.A.
MS-5200#	Nitrogen Supply to Stm. Gen. IC	N.A.
CCW-695	CCW Supply to Excess Letdown Heat Exchanger OC	N.A.
VAC-200	Containment Hydrogen Purge Supply IC	N.A.
VAC-201	Containment Hydrogen Purge Supply IC	N.A.
VAC-116	Containment Air Sample (Post LOCA) Return IC	N.A.
LWS-60	Nitrogen Supply to Reactor Coolant Drain Tank IC	N.A.
AIR-I-587	Instrument Air Supply IC	N.A.
AIR-S-114	Service Air Supply IC	N.A.
VAC-21	Containment Air Sample Return IC	N.A.
AXS-208	Auxiliary Stm. Supply to Containment IC	N.A.
FP-180	Containment Fire Water IC - Unit 1 only	N.A.
FP-867	Containment Fire Water IC - Unit 2 only	N.A.
VAC-252	Containment H_2 Sample Return IC	N.A.
VAC-253	Containment H_2 Sample Return IC	N.A.

*May be opened on an intermittent basis under administrative control (Normally closed manual or remotely OPERABLE valves only)
#Not subject to Type C leakage tests
##The provisions of Specification 3.0.4 are not applicable.

DELETE