CP&L

Carolina Power & Light Company
Robinson Nuclear Plant
3581 West Entrance Road
Hartsville SC 29550

Serial: RNP-RA/00-0120

JUL 20 2000

United States Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555

H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2
DOCKET NO. 50-261/LICENSE NO. DPR-23

SUBMITTAL OF CORE OPERATING LIMITS REPORT
AND REPORT OF ESTIMATES OF SIGNIFICANT ERRORS IN THE
APPLICATION OF THE LARGE BREAK LOSS OF COOLANT ACCIDENT MODEL

Ladies and Gentlemen:

This letter reports information to the NRC required in accordance with 10 CFR 50.46(a)(3)(ii) for the H. B. Robinson Steam Electric Plant (HBRSEP), Unit No. 2, regarding the estimated effect of errors and changes identified in the application of the Siemens Power Corporation (SPC) EXEM PWR Large Break Loss-of-Coolant Accident (LBLOCA) Evaluation Model[1] (EM) and transmits to the NRC the latest revisions to the Core Operating Limits Report (COLR). The sum of the absolute values of the estimated effects on calculated Peak Clad Temperature (PCT) of errors and changes reported by this letter is greater than 50 °F. Therefore, the estimated effect on PCT of these errors and changes are required to be reported to the NRC within 30 days, in accordance with 10 CFR 50.46(a)(3)(ii). Each error discovered and change made and its impact on PCT is discussed below.

Carolina Power & Light (CP&L) Company has identified an error in the flow of the low head safety injection lines in the LBLOCA analysis. The three loops for the Robinson plant are modeled as a broken loop and an intact loop that represents the two unbroken loops. The flow assumed for the two intact loops was incorrect, and this error resulted in an incorrect low head safety injection (LHSI) flow distribution between the broken loop and the two intact loops. The estimated effect of this application error on PCT is +17 °F.

CP&L also identified an error in the application of SPC's EM regarding simulations of the ECCS injection from the accumulators and the LHSI. The RELAP4 EM is used to calculate accumulator

[1] EXEM PWR LBLOCA Evaluation Model as accepted in NRC Letter, D. M. Crutchfield (NRC) to G. N. Ward, "Safety Evaluation of Exxon Nuclear Corporation's Large Break ECCS Evaluation Model EXEM/PWR and Acceptance for Referencing of Related Licensing Topical Reports," July 8, 1986.

A001

flow during blowdown, but because of code limitations, the code assumes that the accumulator flow is terminated just prior to the flow of the nitrogen gas into the lines. Recognizing that flow will not stop at this time, the model extends the accumulator flow to inject the water remaining in the accumulator, accumulator lines, and part of the cold legs into the reactor vessel. In the HBRSEP, Unit No. 2 analysis, the LHSI injects into the accumulator lines, and the model calculates LHSI flow into the accumulator line based on the backpressure at the injection points. The flow of LHSI is calculated to start at the same time the accumulator flow is stopped in the RELAP4 EM blowdown model. Since the accumulator line water is assumed to flow into the vessel, the LHSI should be delayed while the accumulators and lines finish emptying and until the calculated LHSI flow can be transported from the accumulator line injection point to the cold legs. The methodology is non-conservative by not modeling the delay. The estimated effect of this application error on PCT is +102 °F.

A conservative application of accumulator temperature has been identified in the SPC LBLOCA EM for the analysis of record. The accumulator water temperature was assumed as a limiting parameter and not as a nominal value. The SPC methodology permits the use of a nominal accumulator water temperature in the analysis. The estimated effect of this conservative application is a decrease in PCT of –62 °F.

Because of the correction of the above errors and changes in the SPC LBLOCA EM for the analysis of record, SPC has identified a change in the biasing of accumulator line loss coefficients. The coefficients used in the analysis of record were conservatively biased to represent the maximum measured loss coefficient data. SPC guidelines specify that the nominal value of the accumulator line loss coefficient should be used because this parameter can have either negative or positive effects on the peak cladding temperature. The use of the nominal accumulator line loss coefficient instead of the maximum coefficient resulted in an increase in the PCT of +3 °F.

SPC has previously stated to the NRC that, "Errors discovered while performing analyses to support the current plant licensing basis, which is based on the Current model [i.e., the EXEM PWR LBLOCA EM identified in Footnote 1], will be evaluated for reportability using the Revised model [i.e., the EXEM PWR LBLOCA EM identified in Footnote 2]until a new licensing basis analysis is established." The effects of these errors were estimated by SPC at the direction of CP&L using a LBLOCA EM based upon the HBRSEP, Unit No. 2 current licensing basis that has been corrected for existing code variability problems and uses the interim Fuel Cell Test Facility (FCTF) correlation. This model was chosen by CP&L because the timing of the PCT was significantly different between SEM/PWR-98 and the analysis of record. In the analysis of record, the PCT occurs approximately one minute into the transient, which is about 15 seconds after LHSI starts. In SEM/PWR-98 the peak occurs two minutes into the transient. Changes made that affect LHSI delivery rates will therefore have very different effects in the two models. These differences would result in estimates of the errors made with SEM/PWR-98 that are neither representative of nor applicable to the analysis of record. Analyses for the current operating cycle using the revised SPC LBLOCA EM[2] (i.e., SEM/PWR-98), with the above errors corrected, are currently in

[2] EMF-2087 (P)(A), "SEM/PWR-98: ECCS Evaluation Model for PWR LBLOCA Applications."

progress for HBRSEP, Unit No. 2. Preliminary results for the analyses indicate that the calculated limiting PCT is 2180°F.

During the period when these errors were identified but not quantified, the COLR was changed to reduce the Heat Flux Hot Channel Factor limit (i.e., F_q) to 2.32 to provide additional margin for PCT in the event that the quantification of errors and changes in application might result in exceeding the 10 CFR 50.46 limit. Accordingly, Revision 12 to the COLR was implemented on June 20, 2000. Subsequent to the completion of the estimates of the errors and changes in application, the cumulative effect remained acceptable, and Revision 13 to the COLR was implemented on July 17, 2000, restoring the Heat Flux Hot Channel Factor limit to 2.49. Attachments I and II transmit these recent revisions to the COLR.

The previously reported LBLOCA PCT values for HBRSEP, Unit No. 2 was 2125°F for the LBLOCA during the Emergency Core Cooling System (ECCS) Injection Mode. CP&L letter dated November 24, 1999, reported this value. The value did not include -42°F of PCT errors that had been reported in previous CP&L letters. The total effects of the errors reported above, which includes the previously reported but not accumulated -42°F, results in the currently reported LBLOCA Injection Mode PCT value of 2143°F.

The current PCTs associated with Loss-of-Coolant Accidents (LOCAs) are listed below.

Event	PCT (°F)
LBLOCA ECCS Injection Mode	2143
LBLOCA Transfer to Recirculation Mode	2102

Event	PCT (°F)
Small Break (SB) LOCA ECCS Injection Mode	2010.6
SB LOCA Transfer to Recirculation Mode	900

CP&L has requested NRC approval of an amendment to Technical Specifications Section 5.6.5, "Core Operating Limits Report," by letter dated June 14, 2000. HBRSEP, Unit No. 2 will implement a new analysis of record for operating cycle 20 using the SPC LBLOCA EM requested in the submittal, with the errors in the application of the EM corrected. This analysis is currently scheduled to be completed and implemented by January 31, 2001.

If you have any questions concerning this matter, please contact Mr. H. K. Chernoff.

Sincerely,

Harrison FOR

R. L. Warden
Manager - Regulatory Affairs

ALG/alg
Attachments

I. Core Operating Limits Report, Revision 12
II. Core Operating Limits Report, Revision 13

c: Mr. L. A. Reyes, USNRC, Region II
Mr. R. Subbaratnam, NRC, NRR
NRC Resident Inspector, HBRSEP

U. S. Nuclear Regulatory Commission
Attachment I to Serial: RNP-RA/00-0120
23 Pages

H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2

CORE OPERATING LIMITS REPORT, REVISION 12

CP&L

I
Information
Use

CAROLINA POWER & LIGHT COMPANY
H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2

PLANT OPERATING MANUAL

VOLUME 6
PART 5

FUEL MANAGEMENT PROCEDURE

FMP-001

CORE OPERATING LIMITS REPORT (COLR)

REVISION 12

SUMMARY OF CHANGES

REVISION #	REVISION COMMENTS
12	Revised to incorporate lower Fq peaking factor limit.

TABLE OF CONTENTS

1.0 PURPOSE

1.1 To present the cycle-specific Core Operating Limits Report (COLR) for HBRSEP Unit No. 2

1.2 To provide a means of incorporating the COLR into the Plant Operating Manual (POM). The COLR is placed in the POM to ensure that it resides in a controlled location, and that references are provided that ensure that the requirements specified in NRC Generic Letter 88-16 and Improved Technical Specification 5.6.5 are met.

2.0 REFERENCES

2.1 Improved Technical Specifications 1.1, 3.1.1, 3.1.3, 3.1.5, 3.1.6, 3.2.1, 3.2.2, 3.2.3, 3.4.5, 3.4.6, 3.9.1, and 5.6.5

2.2 PLP-100, Technical Requirements Manual (TRM)

2.3 NRC Generic Letter 88-16, "Removal of Cycle-Specific Parameter Limits from Technical Specifications," October 4, 1988.

2.4 License Amendment No. 141 - Regarding Removal of Cycle-Specific Parameter Limits to Core Operating Limits Report

2.5 ESR 98-00395, Cycle 20 Core Design and Safety Analysis

2.6 AP-022, Document Change Procedure

2.7 PLP-001, Plant Nuclear Safety Committee (PNSC)

2.8 REG-NGGC-0002, 10CFR50.59 and Other Regulatory Evaluations

2.9 Shearon Harris Nuclear Power Plant Procedure PLP-106, Technical Specification Equipment List Program

2.10 UFSAR Section 17.3, Quality Assurance Program

2.11 Calculation RNP-F/NFSA-0038, "Robinson Cycle 20 Core Operating Limits Report"

3.0 RESPONSIBILITIES

3.1 RESS Reactor Systems and/or the Nuclear Fuels Management and Safety Analysis Section (NFM&SA) is responsible for revising this procedure as changes to the COLR are required. At a minimum, revisions are required once per cycle, at Beginning of Cycle, to make the COLR cycle-specific.

3.2 The Plant Nuclear Safety Committee (PNSC) is responsible for reviewing revisions to the COLR and providing concurrence prior to implementation of COLR revisions (UFSAR Section 17.3, Quality Assurance Program, Appendix A Item A.1.6.6.j).

3.3 RESS Reactor Systems and Operations are responsible for monitoring plant conditions to ensure the Core Operating Limits specified in this procedure are met.

3.4 Licensing/Regulatory Programs is responsible for providing prompt notification of COLR revisions to the NRC in accordance with ITS 5.6.5.d.

4.0 DEFINITIONS/ABBREVIATIONS

4.1 Definitions

4.1.1 $F_Q^V(Z)$ - the Heat Flux Hot Channel Factor is the maximum local heat flux on the surface of a fuel rod divided by the average fuel rod heat flux and including the V (z) penalty and measurement uncertainties.

4.1.2 CFQ = F_Q^{RTP} - the cycle-specific F_Q limit at Rated Thermal Power (RTP).

4.1.3 K(Z) - the normalized axial dependence factor for F_Q versus core elevation.

4.1.4 $F_{\Delta H}^N$ - the Nuclear Enthalpy Rise Hot Channel Factor is the integral of linear power along the rod with the highest integrated power divided by the average rod power

4.1.5 $F_{\Delta H}^{RTP}$ - the cycle-specific $F_{\Delta H}$ limit at Rated Thermal Power (RTP).

4.1.6 $PF_{\Delta H}$ - the Power Factor Multiplier for $F_{\Delta H}$

4.1.7 AFD - the Axial Flux Difference is the difference in signals between the top and bottom halves of a two-section excore detector which is proportional to the difference in power between the top and bottom halves of the core.

4.1.8 V(Z) - the ratio of the maximum $F_Q(Z)$ produced during and following transient maneuvers to the equilibrium $F_Q(Z)$ value at target axial offset conditions.

4.1.9 P - the fraction of rated power (2300 Mwt) at which the core is operating

4.1.10 RTP - Rated Thermal Power, 2300 Mwt

4.2 Abbreviations

4.2.1 POM - Plant Operating Manual

4.2.2 PNSC - Plant Nuclear Safety Committee

4.2.3 COLR - Core Operating Limits Report

4.2.4 MTC - Moderator Temperature Coefficient

4.2.5 ITS - Improved Technical Specifications

4.2.6 RIL - Rod Insertion Limits

4.2.7 EFPD - Effective Full Power Day

5.0 **GENERAL**

5.1 Background Information

5.1.1 HBRSEP Unit No. 2, like all other commercial nuclear power plants, is required to operate within the specific core operating limits and restrictions as specified in the Technical Specifications. Examples of these limits/restrictions include power dependent rod insertion limits, and limits of $F_Q(Z)$ and $F_{\Delta H}$, among others. Technical Specification changes and NRC approval were required as specific numerical values for these limits/restrictions were revised. If these changes were frequent, e.g. on a cycle-specific basis, or if they were needed on accelerated schedules, considerable administrative burdens were placed on both the NRC and on utility personnel.

5.1.2 To reduce this burden, the CORE OPERATING LIMITS REPORT (COLR) concept was developed in which specific numerical values for certain core operating limits and/or restrictions would be removed from the Technical Specifications and relocated to a COLR document. Using NRC approved methodologies, numerical values for these operating limits and/or restrictions can be updated on an as-needed basis (e.g. each cycle) by simply revising the COLR with appropriate review and notification to the NRC, hence, revisions to the Technical Specifications are not required.

5.1.3 The NRC endorsed the COLR concept by encouraging licensees to develop such a document in Generic Letter 88-16 which provided guidance for relocation of specific numerical values for various core operating limits and/or restrictions to a COLR and indicated that these values could be changed without prior NRC approval so long as an NRC-approved methodology is followed. Future changes and updates would be allowable provided an Unreviewed Safety Question Determination is performed in accordance with the provisions of 10CFR 50.59, the COLR is suitably revised, and the NRC is promptly informed of the revision.

5.1.4 The use of a COLR at H. B. Robinson was accepted by the NRC per License Amendment 141. The amendment established requirements for a cycle-specific COLR and for notification of the NRC (ITS 5.6.5.d) when revisions are made. Since the COLR is cycle-specific, the COLR will be revised at least once per cycle, that is, at the beginning of the cycle.

5.2 Contents of the H.B. Robinson Unit 2 COLR

5.2.1 Technical Specification ITS 5.6.5.a requires the following cycle-specific core operating limits be established and documented in the Core Operating Limits Reports

1. Moderator Temperature Coefficient (MTC) Limits
2. Shutdown Bank Insertion Limits
3. Control Bank Insertion Limits
4. Heat Flux Hot Channel Factor ($F_Q(Z)$) Limit, CFQ
5. K(Z) Curve
6. Nuclear Enthalpy Rise Hot Channel Factor ($F_{\Delta H}{}^{N}$) Limit, $F_{\Delta H}{}^{RTP}$
7. $F_{\Delta H}$ Power Factor Multiplier ($PF_{\Delta H}$)
8. Axial Flux Difference (AFD) Limits
9. V(Z) Curve
10. Shutdown Margin
11. Refueling Boron Concentration

5.2.2 The COLR will also contain a listing of the specific methodologies used to support the core operating limits.

5.3 Requirements for Revision of the COLR

5.3.1 Since the COLR is cycle-specific, this procedure will be revised at least once per cycle, that is, at the beginning of the cycle. The methods and requirements established by this procedure for revision of the COLR supplement those of AP-022. Changes will require a 10CFR 50.59 Unreviewed Safety Question Determination as well as PNSC concurrence and notification of the NRC as part of the revision process.

5.4 Core Operating Limits Report (COLR)

5.4.1 The current cycle-specific Core Operating Limits Report is provided in ATTACHMENT 7.1.

6.0 PROCEDURE

6.1 Nuclear Fuels Management & Safety Analysis Section (NFM&SA) shall review and recommend for implementation any changes to the COLR. The review is normally documented in an ESR including any required Owner's Reviews, calculations and other reviews. The use of NRC approved methodologies is also confirmed in the ESR. Changes recommended by NFM&SA are normally transmitted to the plant via a memo recommending the revision of the COLR.

6.2 Once NFM&SA recommends a revision to the COLR, a Reactor Engineer shall prepare a revision to FMP-001 in accordance with the requirements of AP-022.

6.3 Other plant procedures shall be reviewed to determine if they require revision in order to implement the revised COLR. At a minimum, the procedures listed in ATTACHMENT 7.2 shall be reviewed.

6.4 Any required procedure revisions or new procedures necessary to incorporate the change to the COLR shall be completed by the effective date of the COLR change.

6.5 The proposed revision of the COLR shall be submitted to the PNSC for review.

6.6 The PNSC shall review the proposed revision to the COLR and concur with the changes prior to their implementation in accordance with UFSAR Section 17.3 Appendix A Item A.1.6.6.j.

6.7 Upon PNSC concurrence with the revision to the COLR, Licensing/Regulatory Programs shall notify the NRC per ITS 5.6.5.d.

7.0 ATTACHMENTS

7.1 HBRSEP Unit No. 2 Cycle 20 Core Operating Limits Report, Revision 1

7.2 Procedures Potentially Affected By COLR Revisions

HBRSEP UNIT NO. 2, CYCLE 20
CORE OPERATING LIMITS REPORT
REVISION 1

1.0 OPERATING LIMITS REPORT

This Core Operating Limits Report (COLR) for HBRSEP Unit No. 2, Cycle 20 has been prepared per ESR 98-00395 in accordance with the requirements of ITS 5.6.5. CFQ has been reduced from the value in reference 2.11 to provide additional margin.

The Improved Technical Specifications affected by this report and the methodologies used for the various parameters are listed below.

Parameter	ITS Reference	Applicable Methodology (Section 3.0 Number)
MTC	3.1.3	1, 4, 18, 19, 22
Shutdown Bank RILs	3.1.5	1, 4, 8, 18, 19, 22
Control Bank RILs	3.1.6	1, 4, 8, 18, 19, 22
$F_Q^V(Z)$	3.2.1, 3.2.3	1, 5, 6, 7, 8, 11, 17, 18, 19, 21, 22
$F_{\Delta H}$	3.2.2, 3.2.3	1, 3, 4, 5, 6, 7, 9, 11, 17, 18, 19, 20, 21, 22
AFD	3.2.1, 3.2.3	1, 6, 7, 16, 18, 19, 21, 22
Shutdown Margin Requirements	3.4.5, 3.1.1	1, 4, 8, 18, 19, 22
Refueling Boron Requirements	3.9.1	1, 4, 8, 18, 19, 22
COLR	5.6.5	None

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 ITS 5.6.5 and the COLR Section 3.0.

2.1 Moderator Temperature Coefficient (ITS 3.1.3)

2.1.1 The Moderator Temperature Coefficient (MTC) limits are:

a) The Positive MTC (ARO) shall be less positive than +5.0 pcm/°F for power levels up to 50% RTP, and

b) The Positive MTC (ARO) shall be less than or equal to 0.0 pcm/°F at 50% RTP and above.

c) The Negative MTC (ARO/RTP) shall be less negative than -40.0 pcm/°F.

2.1.2 The 300 ppm Surveillance limit is:

At an equilibrium boron concentration of 300 ppm the MTC shall be less negative than or equal to -32.6 pcm/°F.

2.1.3 The 60 ppm Surveillance limit is:

At an equilibrium boron concentration of 60 ppm the MTC shall be less negative than or equal to -36.6 pcm/°F.

2.2 Shutdown Rod Insertion Limits (ITS 3.1.5)

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

2.3 Control Rod Insertion Limits (ITS 3.1.6)

2.3.1 The control rods shall be limited in physical insertion as shown in Figure 1.0

2.4 **Heat Flux Hot Channel Factor - F_Q^V (Z) (ITS 3.2.1, 3.2.3)**

$F_Q^V(Z) \leq (CFQ/P) \times K(Z)$ for $P > 0.5$

$F_Q^V(Z) < (CFQ/0.5) \times K(Z)$ for $P \leq 0.5$

Where: P = (Thermal Power / Rated Thermal Power)

2.4.1 CFQ = 2.32 for ROB-13, ROB-15, ROB-16, and ROB-17 reload batches

2.4.2 K(Z) is specified in Figure 2.0

2.5 **Nuclear Enthalpy Rise Hot Channel Factor - $F_{\Delta H}$ (ITS 3.2.2, 3.2.3)**

$F_{\Delta H} < F_{\Delta H}^{RTP} (1 + PF_{\Delta H} (1-P))$

Where: P = (Thermal Power / Rated Thermal Power)

2.5.1 $F_{\Delta H}$ is the measured $F_{\Delta H}^N$ multiplied by the measurement uncertainty (1.04)

2.5.2 $F_{\Delta H}^{RTP} = 1.80$ for ROB-13, ROB-15, ROB-16, and ROB-17 reload batches

2.5.3 $PF_{\Delta H} = 0.2$

2.6 **Axial Flux Difference (ITS 3.2.1, 3.2.3)**

2.6.1 The axial flux difference target bands are ±3% and ±5% about the target AFD.

2.6.2 V(Z) values for the ±3% and ±5% target bands are specified in Figure 3.0

2.6.3 The AFD Acceptable Operation Limits are specified in Figure 4.0

2.7 Shutdown Margin Requirements (SDM) (ITS 3.1.1, 3.4.5, 3.4.6)

2.7.1 The Mode 1 and Mode 2 required SDM versus RCS boron concentration is presented in Figure 5.0.

2.7.2 The Mode 3 SDM requirements are as follows:

a) With at least 2 reactor coolant pumps in operation, the SDM shall be greater than or equal to that specified in Figure 5.0.

b) With less than 2 reactor coolant pumps in operation and the rod control system capable of rod withdrawal, the SDM shall be greater than or equal to 4% Δk/k.

c) With less than 2 reactor coolant pumps in operation and with the rod control system not capable of rod withdrawal, the SDM shall be greater than or equal to that specified in Figure 5.0.

2.7.3 The Mode 4 SDM requirements are as follows:

a) With at least 2 reactor coolant pumps in operation, the SDM shall be greater than or equal to that specified in Figure 5.0.

b) With less than 2 reactor coolant pumps in operation and the rod control system capable of rod withdrawal, the SDM shall be greater than or equal to 4% Δk/k.

c) With less than 2 reactor coolant pumps in operation and with the rod control system not capable of rod withdrawal, the SDM shall be greater than or equal to that specified in Figure 5.0.

2.7.4 The minimum required SDM for Mode 5 is 1% Δk/k.

2.7.5 The minimum required SDM for Mode 6 is 6% Δk/k.

2.8 Refueling Boron Concentration (ITS 3.9.1)

2.8.1 In Mode 6 the minimum boron concentration shall be 1950 ppm.

3.0 METHODOLOGY REFERENCES

1) XN-75-27(A), and Supplements 1, 2, 3, and 4 "Exxon Nuclear Neutronics Design Methods for Pressurized Water Reactors", Exxon Nuclear Company.

2) Not Used For Cycle 20

3) XN-NF-82-21(A), Revision 1, "Application of Exxon Nuclear Company PWR Thermal Margin Methodology to Mixed Core Configurations," Exxon Nuclear Company.

4) XN-NF-84-093(A), and Supplement 1, "Steamline Break Methodology for PWRs," Advanced Nuclear Fuels Corporation

5) XN-75-32(A) Supplements 1, 2, 3, and 4, "Computational Procedure for Evaluating Fuel Rod Bow," Exxon Nuclear Company.

6) XN-NF-82-49(A), Revision 1 and Supplement 1, "Exxon Nuclear Company Evaluation Model Revised EXEM PWR Small Break Model," Siemens Power Corporation.

7) XN-NF-82-20(A), Revision 1 and Supplements 1, 2, 3, and 4, "Exxon Nuclear Company Evaluation Model EXEM/PWR ECCS Model Updates", Exxon Nuclear Company.

   XN-NF-82-07(A), Revision 1, Exxon Nuclear Company ECCS Cladding Swelling and Rupture Model," Exxon Nuclear Company.

   XN-NF-81-58(A), Revision 2, and Supplements 1, 2, 3, and 4, "RODEX2 Fuel Rod Thermal-Mechanical Response Evaluation Model," Exxon Nuclear Company.

   XN-NF-85-16(A), Volume 1 and Supplements 1, 2, and 3, Volume 2, Revision 1, and Supplement 1, "PWR 17x17 Fuel Cooling Test Program,: Exxon Nuclear Company.

   XN-NF-85-105(A), and Supplement 1, "Scaling of FCTF Based Reflood Heat Transfer Correlation for Other Bundle Designs," Exxon Nuclear Company.

8) XN-NF-78-44(A)," A Generic Analysis of the Control Rod Ejection Transient for Pressurized Water Reactors," Exxon Nuclear Company.

9) Not Used For Cycle 20

10) Not Used For Cycle 20

11) XN-NF-82-06(A), Revision 1 and Supplements 2, 4, and 5, "Qualification of Exxon Nuclear Fuel for Extended Burnup (PWR)," Exxon Nuclear Company.

12) Not Used For Cycle 20

13) Not Used For Cycle 20

14) Not Used For Cycle 20

15) Not Used For Cycle 20

16) ANF-88-054(A), "PDC-3: Advanced Nuclear Fuels Corporation Power Distribution Control for Pressurized Water Reactors and Application of PDC-3 to H.B. Robinson Unit 2," Advanced Nuclear Fuels Corporation.

17) ANF-88-133(A), and Supplement 1, "Qualification of Advanced Nuclear Fuels PWR Design Methodology for Rod Burnups of 62 GWd/MTU," Advanced Nuclear Fuels Corporation.

18) ANF-89-151 (P) (A), and correspondence "ANF-RELAP Methodology for Pressurized Water Reactors: Analysis of Non-LOCA Chapter 15 Events," Advanced Nuclear Fuels Corporation.

19) EMF-92-081 (P) (A), and Supplement 1, "Statistical Setpoint/Transient Methodology for Westinghouse Type Reactors," Siemens Power Corporation

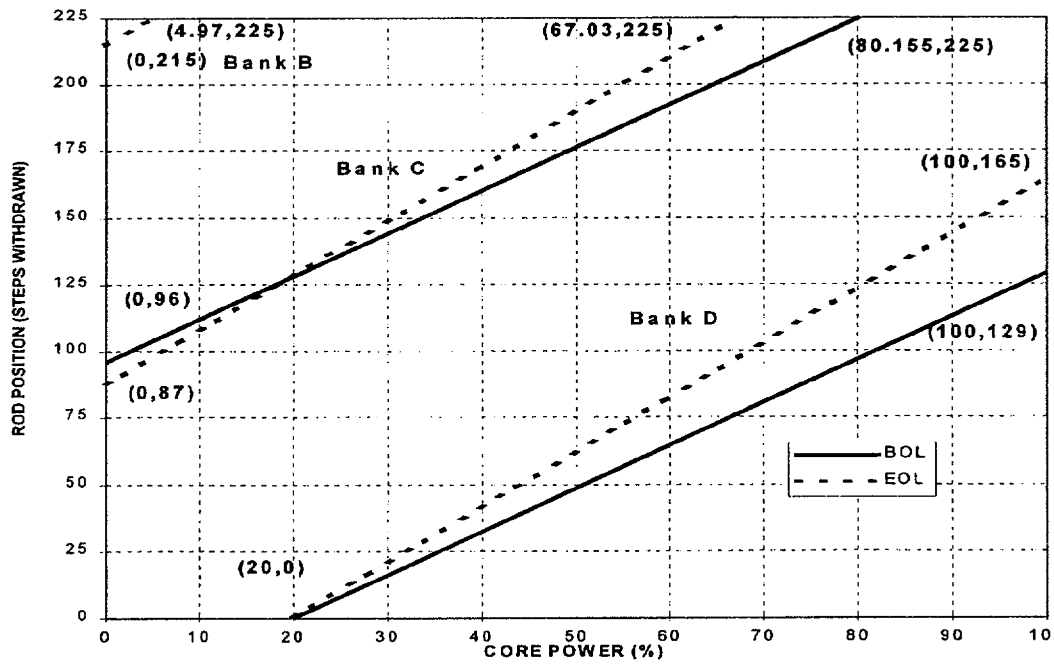
20) EMF-92-153 (P) (A), Revision 0 and Supplement 1, "HTP: Departure from Nucleate Boiling Correlation for High Thermal Performance Fuel," Siemens Power Corporation.

21) XN-NF-85-92 (P)(A), "Exxon Nuclear Uranium Dioxide/Gadolinia Irradiation Examination and Thermal Conductivity Results," Exxon Nuclear Company, November 1986.

22) EMF-96-029 (P)(A), Volume 1, Volume 2 and Attachment, "Reactor Analysis System for PWRs," Siemens Power Corporation, January 1997.

HBRSEP UNIT NO. 2, CYCLE 20
CORE OPERATING LIMITS REPORT
REVISION 1

Figure 1.0, Control Group Insertion Limits for Three Loop Operation



NOTE: The breakpoint between BOL and EOL RIL occurs at 50% of the cycle as defined by burnup. For Cycle 20, this burnup occurs at 246 EFPDs.

Control rod banks shall always be withdrawn and inserted in the prescribed sequence. For withdrawal, the sequence is Shutdown "A", Shutdown "B", Control "A", Control "B", Control "C", and Control "D". The insertion sequence is the reverse of the withdrawal sequence.

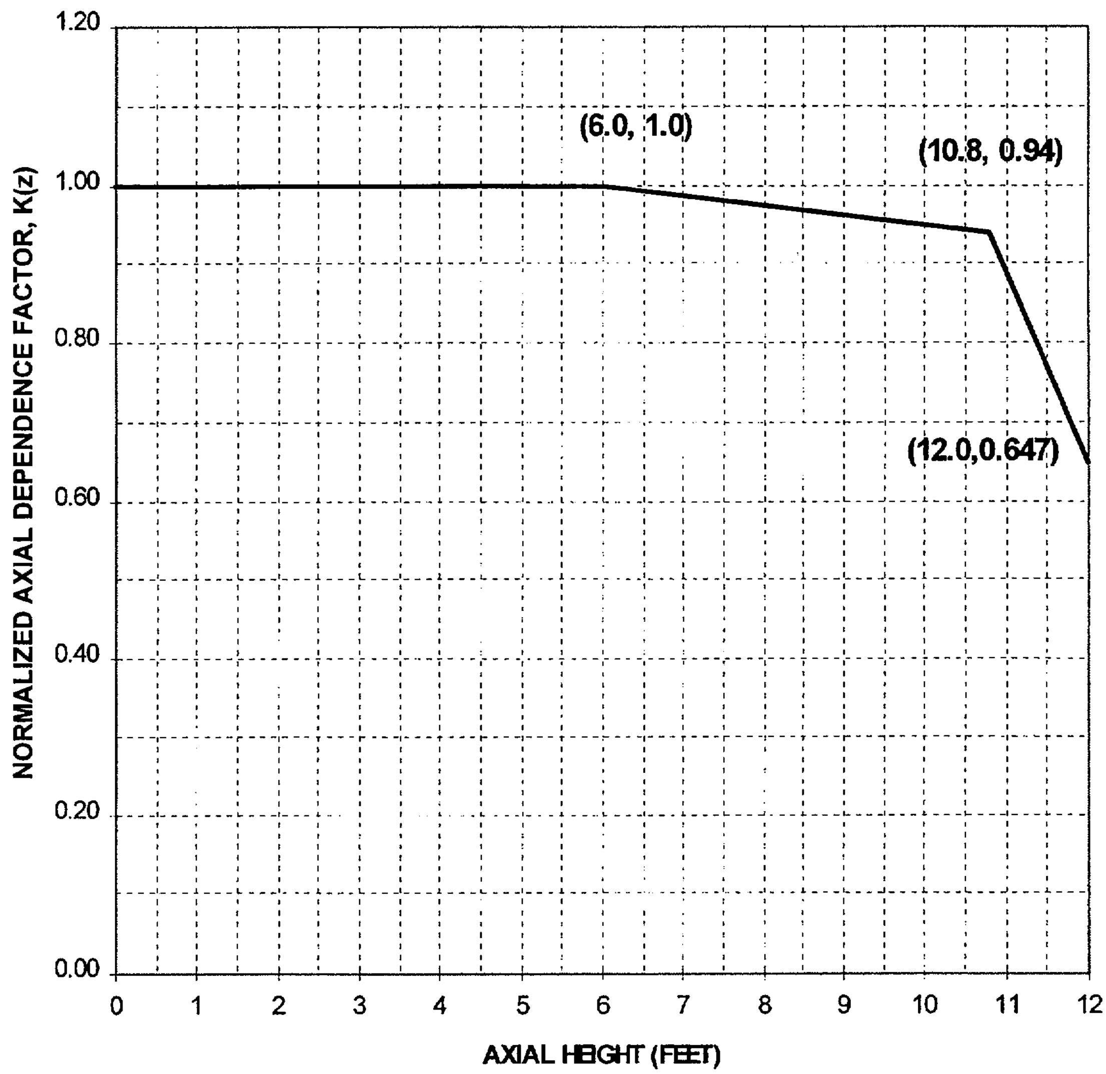
Overlap of consecutive Control Banks shall not exceed the prescribed setpoint for automatic overlap. The setpoint is 97 steps.

Control Bank A must be withdrawn from the core prior to power operation.

At BOL and 0% core power, Control Bank B will be at or above step 224.

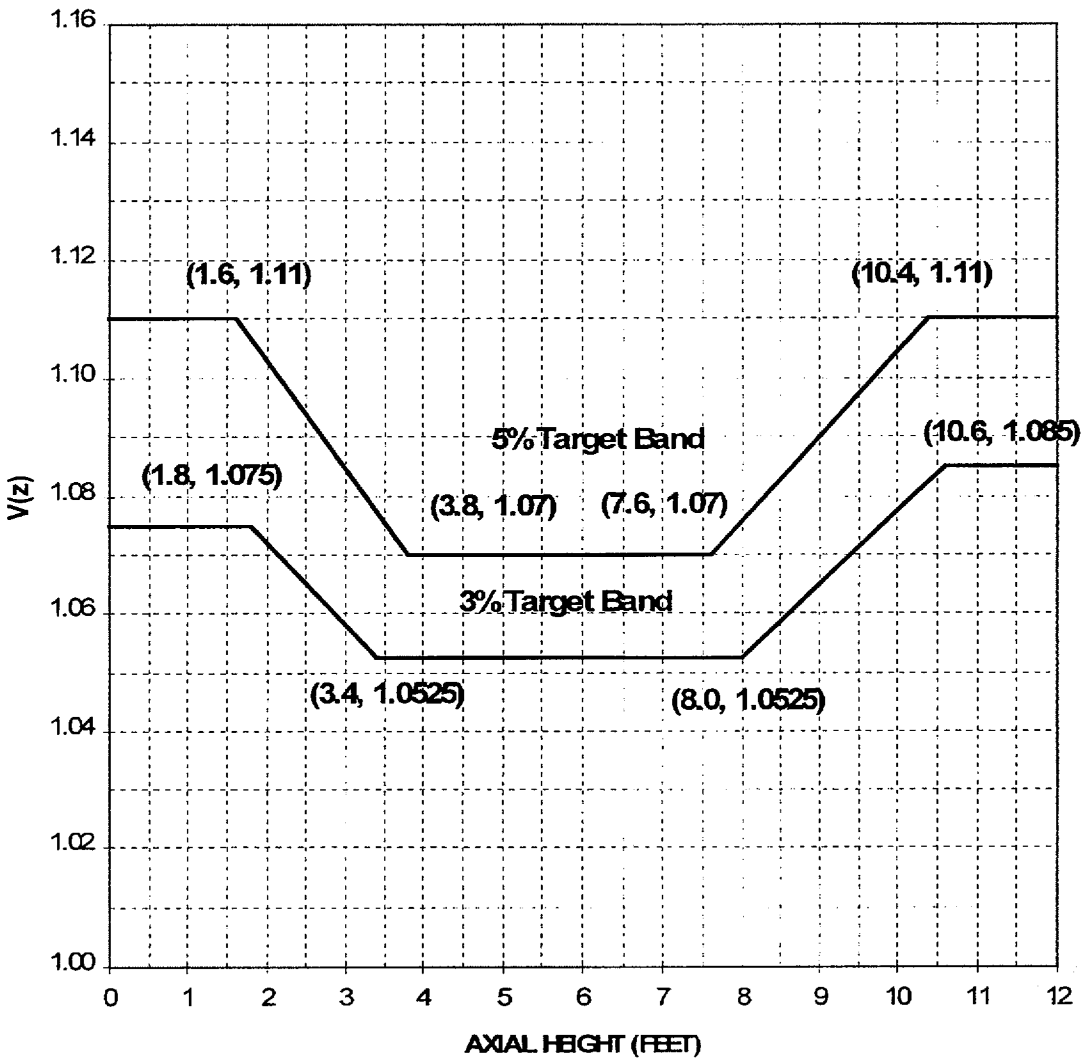
HBRSEP UNIT NO. 2, CYCLE 20
CORE OPERATING LIMITS REPORT
REVISION 1

Figure 2.0, Normalized Axial Dependence Factor K(z) for Fq Versus Elevation



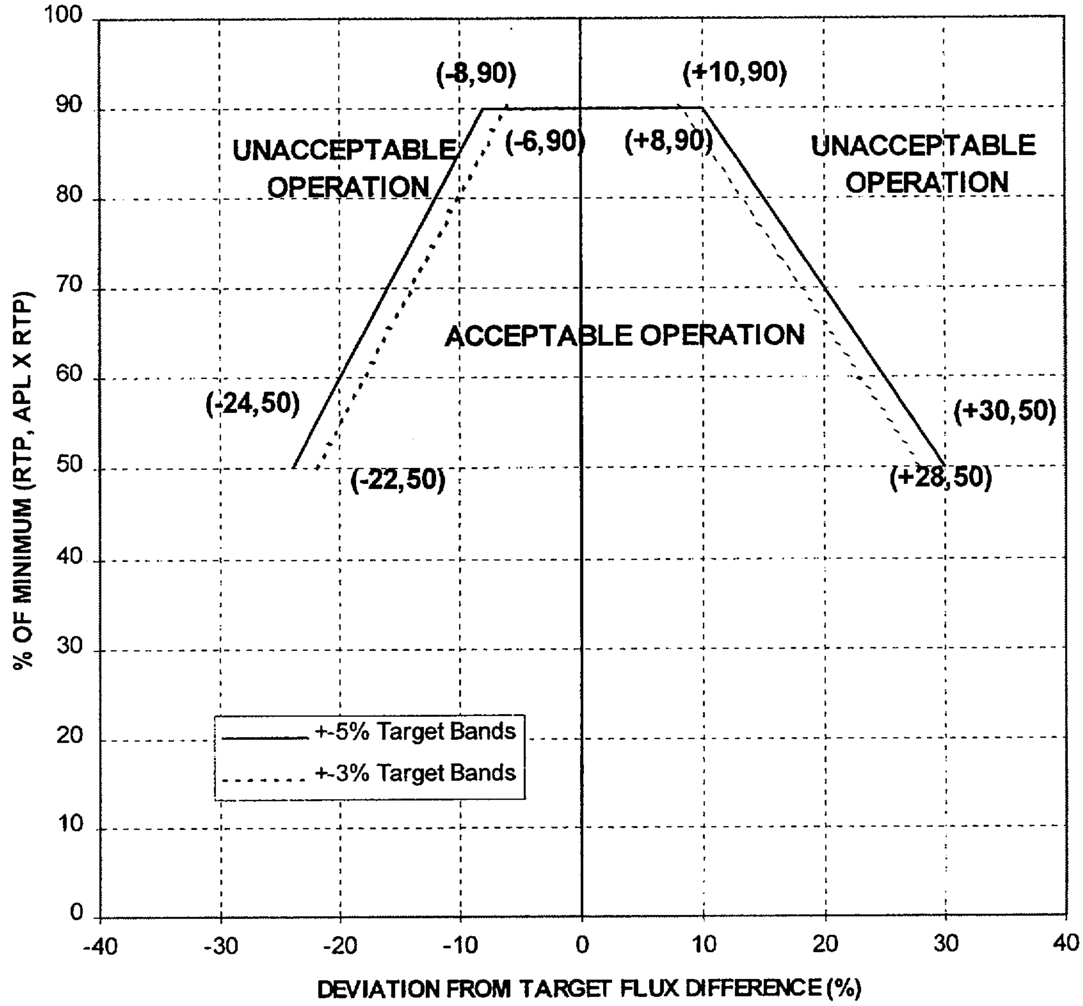
HBRSEP UNIT NO. 2, CYCLE 20
CORE OPERATING LIMITS REPORT
REVISION 1

Figure 3.0, V(z) as a Function of Core Height



HBRSEP UNIT NO. 2, CYCLE 20
CORE OPERATING LIMITS REPORT
REVISION 1

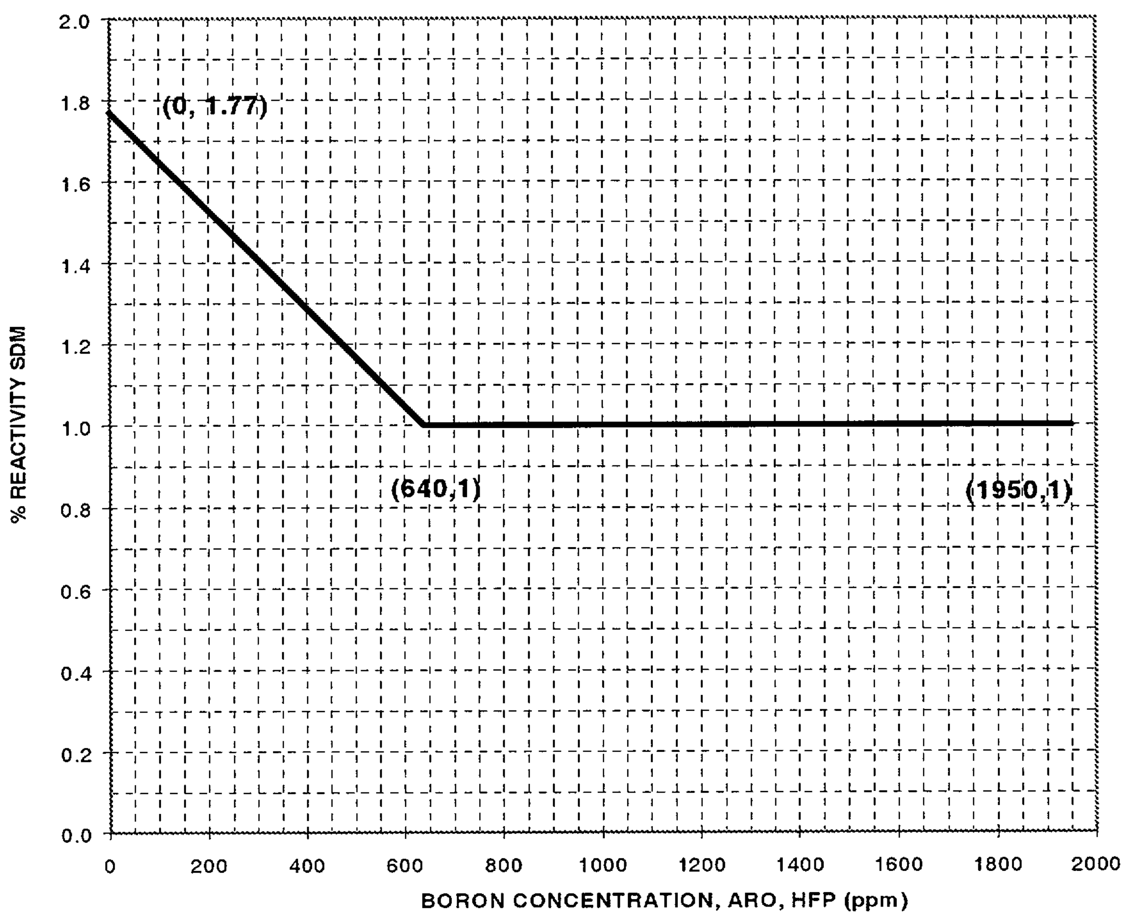
Figure 4.0, Allowable Deviation from Target Flux Difference



NOTE: For power levels above 90%, power operation is allowed within the target bands (±3% and ±5%).

HBRSEP UNIT NO. 2, CYCLE 20
CORE OPERATING LIMITS REPORT
REVISION 1

Figure 5.0, Shutdown Margin Versus Boron Concentration



ATTACHMENT 7.2
Page 1 of 1

PROCEDURES POTENTIALLY AFFECTED BY COLR REVISIONS

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CP-010	GP-002
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Station Curve Book

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The procedures listed above are those that are typically affected by COLR revisions; however, other procedures may also be affected.

U. S. Nuclear Regulatory Commission
Attachment II to Serial: RNP-RA/00-0120
23 Pages

H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2

CORE OPERATING LIMITS REPORT, REVISION 13

CP&L

I
Information Use

CAROLINA POWER & LIGHT COMPANY
H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2

PLANT OPERATING MANUAL

VOLUME 6
PART 5

FUEL MANAGEMENT PROCEDURE

FMP-001

CORE OPERATING LIMITS REPORT (COLR)

REVISION 13

SUMMARY OF CHANGES

REVISION #	REVISION COMMENTS
13	Revised to incorporate an Fq peaking factor limit of 2.49.

TABLE OF CONTENTS

1.0 PURPOSE

1.1 To present the cycle-specific Core Operating Limits Report (COLR) for HBRSEP Unit No. 2

1.2 To provide a means of incorporating the COLR into the Plant Operating Manual (POM). The COLR is placed in the POM to ensure that it resides in a controlled location, and that references are provided that ensure that the requirements specified in NRC Generic Letter 88-16 and Improved Technical Specification 5.6.5 are met.

2.0 REFERENCES

2.1 Improved Technical Specifications 1.1, 3.1.1, 3.1.3, 3.1.5, 3.1.6, 3.2.1, 3.2.2, 3.2.3, 3.4.5, 3.4.6, 3.9.1, and 5.6.5

2.2 PLP-100, Technical Requirements Manual (TRM)

2.3 NRC Generic Letter 88-16, "Removal of Cycle-Specific Parameter Limits from Technical Specifications," October 4, 1988.

2.4 License Amendment No. 141 - Regarding Removal of Cycle-Specific Parameter Limits to Core Operating Limits Report

2.5 ESR 98-00395, Cycle 20 Core Design and Safety Analysis

2.6 AP-022, Document Change Procedure

2.7 PLP-001, Plant Nuclear Safety Committee (PNSC)

2.8 REG-NGGC-0002, 10CFR50.59 and Other Regulatory Evaluations

2.9 Shearon Harris Nuclear Power Plant Procedure PLP-106, Technical Specification Equipment List Program

2.10 UFSAR Section 17.3, Quality Assurance Program

2.11 Calculation RNP-F/NFSA-0038, "Robinson Cycle 20 Core Operating Limits Report"

3.0 RESPONSIBILITIES

3.1 RESS Reactor Systems and/or the Nuclear Fuels Management and Safety Analysis Section (NFM&SA) is responsible for revising this procedure as changes to the COLR are required. At a minimum, revisions are required once per cycle, at Beginning of Cycle, to make the COLR cycle-specific.

3.2 The Plant Nuclear Safety Committee (PNSC) is responsible for reviewing revisions to the COLR and providing concurrence prior to implementation of COLR revisions (UFSAR Section 17.3, Quality Assurance Program, Appendix A Item A.1.6.6.j).

3.3 RESS Reactor Systems and Operations are responsible for monitoring plant conditions to ensure the Core Operating Limits specified in this procedure are met.

3.4 Licensing/Regulatory Programs is responsible for providing prompt notification of COLR revisions to the NRC in accordance with ITS 5.6.5.d.

4.0 DEFINITIONS/ABBREVIATIONS

4.1 Definitions

4.1.1 $F_Q^V(Z)$ - the Heat Flux Hot Channel Factor is the maximum local heat flux on the surface of a fuel rod divided by the average fuel rod heat flux and including the V (z) penalty and measurement uncertainties.

4.1.2 CFQ = F_Q^{RTP} - the cycle-specific F_Q limit at Rated Thermal Power (RTP).

4.1.3 K(Z) - the normalized axial dependence factor for F_Q versus core elevation.

4.1.4 $F_{\Delta H}^N$ - the Nuclear Enthalpy Rise Hot Channel Factor is the integral of linear power along the rod with the highest integrated power divided by the average rod power

4.1.5 $F_{\Delta H}^{RTP}$ - the cycle-specific $F_{\Delta H}$ limit at Rated Thermal Power (RTP).

4.1.6 $PF_{\Delta H}$ - the Power Factor Multiplier for $F_{\Delta H}$

4.1.7 AFD - the Axial Flux Difference is the difference in signals between the top and bottom halves of a two-section excore detector which is proportional to the difference in power between the top and bottom halves of the core.

4.1.8 V(Z) - the ratio of the maximum $F_Q(Z)$ produced during and following transient maneuvers to the equilibrium $F_Q(Z)$ value at target axial offset conditions.

4.1.9 P - the fraction of rated power (2300 Mwt) at which the core is operating

4.1.10 RTP - Rated Thermal Power, 2300 Mwt

4.2 Abbreviations

4.2.1 POM - Plant Operating Manual

4.2.2 PNSC - Plant Nuclear Safety Committee

4.2.3 COLR - Core Operating Limits Report

4.2.4 MTC - Moderator Temperature Coefficient

4.2.5 ITS - Improved Technical Specifications

4.2.6 RIL - Rod Insertion Limits

4.2.7 EFPD - Effective Full Power Day

5.0 **GENERAL**

5.1 Background Information

5.1.1 HBRSEP Unit No. 2, like all other commercial nuclear power plants, is required to operate within the specific core operating limits and restrictions as specified in the Technical Specifications. Examples of these limits/restrictions include power dependent rod insertion limits, and limits of $F_Q(Z)$ and $F_{\Delta H,}$ among others. Technical Specification changes and NRC approval were required as specific numerical values for these limits/restrictions were revised. If these changes were frequent, e.g. on a cycle-specific basis, or if they were needed on accelerated schedules, considerable administrative burdens were placed on both the NRC and on utility personnel.

5.1.2 To reduce this burden, the CORE OPERATING LIMITS REPORT (COLR) concept was developed in which specific numerical values for certain core operating limits and/or restrictions would be removed from the Technical Specifications and relocated to a COLR document. Using NRC approved methodologies, numerical values for these operating limits and/or restrictions can be updated on an as-needed basis (e.g. each cycle) by simply revising the COLR with appropriate review and notification to the NRC, hence, revisions to the Technical Specifications are not required.

5.1.3 The NRC endorsed the COLR concept by encouraging licensees to develop such a document in Generic Letter 88-16 which provided guidance for relocation of specific numerical values for various core operating limits and/or restrictions to a COLR and indicated that these values could be changed without prior NRC approval so long as an NRC-approved methodology is followed. Future changes and updates would be allowable provided an Unreviewed Safety Question Determination is performed in accordance with the provisions of 10CFR 50.59, the COLR is suitably revised, and the NRC is promptly informed of the revision.

5.1.4 The use of a COLR at H. B. Robinson was accepted by the NRC per License Amendment 141. The amendment established requirements for a cycle-specific COLR and for notification of the NRC (ITS 5.6.5.d) when revisions are made. Since the COLR is cycle-specific, the COLR will be revised at least once per cycle, that is, at the beginning of the cycle.

5.2 Contents of the H.B. Robinson Unit 2 COLR

5.2.1 Technical Specification ITS 5.6.5.a requires the following cycle-specific core operating limits be established and documented in the Core Operating Limits Reports

1. Moderator Temperature Coefficient (MTC) Limits
2. Shutdown Bank Insertion Limits
3. Control Bank Insertion Limits
4. Heat Flux Hot Channel Factor ($F_Q(Z)$) Limit, CFQ
5. K(Z) Curve
6. Nuclear Enthalpy Rise Hot Channel Factor ($F_{\Delta H}{}^N$) Limit, $F_{\Delta H}{}^{RTP}$
7. $F_{\Delta H}$ Power Factor Multiplier ($PF_{\Delta H}$)
8. Axial Flux Difference (AFD) Limits
9. V(Z) Curve
10. Shutdown Margin
11. Refueling Boron Concentration

5.2.2 The COLR will also contain a listing of the specific methodologies used to support the core operating limits.

5.3 Requirements for Revision of the COLR

5.3.1 Since the COLR is cycle-specific, this procedure will be revised at least once per cycle, that is, at the beginning of the cycle. The methods and requirements established by this procedure for revision of the COLR supplement those of AP-022. Changes will require a 10CFR 50.59 Unreviewed Safety Question Determination as well as PNSC concurrence and notification of the NRC as part of the revision process.

5.4 Core Operating Limits Report (COLR)

5.4.1 The current cycle-specific Core Operating Limits Report is provided in ATTACHMENT 7.1.

6.0 PROCEDURE

6.1 Nuclear Fuels Management & Safety Analysis Section (NFM&SA) shall review and recommend for implementation any changes to the COLR. The review is normally documented in an ESR including any required Owner's Reviews, calculations and other reviews. The use of NRC approved methodologies is also confirmed in the ESR. Changes recommended by NFM&SA are normally transmitted to the plant via a memo recommending the revision of the COLR.

6.2 Once NFM&SA recommends a revision to the COLR, a Reactor Engineer shall prepare a revision to FMP-001 in accordance with the requirements of AP-022.

6.3 Other plant procedures shall be reviewed to determine if they require revision in order to implement the revised COLR. At a minimum, the procedures listed in ATTACHMENT 7.2 shall be reviewed.

6.4 Any required procedure revisions or new procedures necessary to incorporate the change to the COLR shall be completed by the effective date of the COLR change.

6.5 The proposed revision of the COLR shall be submitted to the PNSC for review.

6.6 The PNSC shall review the proposed revision to the COLR and concur with the changes prior to their implementation in accordance with UFSAR Section 17.3 Appendix A Item A.1.6.6.j.

6.7 Upon PNSC concurrence with the revision to the COLR, Licensing/Regulatory Programs shall notify the NRC per ITS 5.6.5.d.

7.0 ATTACHMENTS

7.1 HBRSEP Unit No. 2 Cycle 20 Core Operating Limits Report, Revision 2

7.2 Procedures Potentially Affected By COLR Revisions

HBRSEP UNIT NO. 2, CYCLE 20
CORE OPERATING LIMITS REPORT
REVISION 2

1.0 OPERATING LIMITS REPORT

This Core Operating Limits Report (COLR) for HBRSEP Unit No. 2, Cycle 20 has been prepared per ESR 98-00395 in accordance with the requirements of ITS 5.6.5.

The Improved Technical Specifications affected by this report and the methodologies used for the various parameters are listed below.

Parameter	ITS Reference	Applicable Methodology (Section 3.0 Number)
MTC	3.1.3	1, 4, 18, 19, 22
Shutdown Bank RILs	3.1.5	1, 4, 8, 18, 19, 22
Control Bank RILs	3.1.6	1, 4, 8, 18, 19, 22
$F_Q^V(Z)$	3.2.1, 3.2.3	1, 5, 6, 7, 8, 11, 17, 18, 19, 21, 22
$F_{\Delta H}$	3.2.2, 3.2.3	1, 3, 4, 5, 6, 7, 9, 11, 17, 18, 19, 20, 21, 22
AFD	3.2.1, 3.2.3	1, 6, 7, 16, 18, 19, 21, 22
Shutdown Margin Requirements	3.4.5, 3.1.1	1, 4, 8, 18, 19, 22
Refueling Boron Requirements	3.9.1	1, 4, 8, 18, 19, 22
COLR	5.6.5	None

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 ITS 5.6.5 and the COLR Section 3.0.

2.1 Moderator Temperature Coefficient (ITS 3.1.3)

2.1.1 The Moderator Temperature Coefficient (MTC) limits are:

a) The Positive MTC (ARO) shall be less positive than +5.0 pcm/°F for power levels up to 50% RTP, and

b) The Positive MTC (ARO) shall be less than or equal to 0.0 pcm/°F at 50% RTP and above.

c) The Negative MTC (ARO/RTP) shall be less negative than -40.0 pcm/°F.

2.1.2 The 300 ppm Surveillance limit is:

At an equilibrium boron concentration of 300 ppm the MTC shall be less negative than or equal to -32.6 pcm/°F.

2.1.3 The 60 ppm Surveillance limit is:

At an equilibrium boron concentration of 60 ppm the MTC shall be less negative than or equal to -36.6 pcm/°F.

2.2 Shutdown Rod Insertion Limits (ITS 3.1.5)

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

2.3 Control Rod Insertion Limits (ITS 3.1.6)

2.3.1 The control rods shall be limited in physical insertion as shown in Figure 1.0

2.4 **Heat Flux Hot Channel Factor - F_Q^V (Z) (ITS 3.2.1, 3.2.3)**

$F_Q^V(Z) \leq (CFQ/P) \times K(Z)$ for $P > 0.5$

$F_Q^V(Z) < (CFQ/0.5) \times K(Z)$ for $P \leq 0.5$

Where: P = (Thermal Power / Rated Thermal Power)

2.4.1 CFQ = 2.49 for ROB-13, ROB-15, ROB-16, and ROB-17 reload batches

2.4.2 K(Z) is specified in Figure 2.0

2.5 **Nuclear Enthalpy Rise Hot Channel Factor - $F_{\Delta H}$ (ITS 3.2.2, 3.2.3)**

$F_{\Delta H} < F_{\Delta H}^{RTP} (1 + PF_{\Delta H} (1-P))$

Where: P = (Thermal Power / Rated Thermal Power)

2.5.1 $F_{\Delta H}$ is the measured $F_{\Delta H}^N$ multiplied by the measurement uncertainty (1.04)

2.5.2 $F_{\Delta H}^{RTP} = 1.80$ for ROB-13, ROB-15, ROB-16, and ROB-17 reload batches

2.5.3 $PF_{\Delta H} = 0.2$

2.6 **Axial Flux Difference (ITS 3.2.1, 3.2.3)**

2.6.1 The axial flux difference target bands are ±3% and ±5% about the target AFD.

2.6.2 V(Z) values for the ±3% and ±5% target bands are specified in Figure 3.0

2.6.3 The AFD Acceptable Operation Limits are specified in Figure 4.0

2.7 Shutdown Margin Requirements (SDM) (ITS 3.1.1, 3.4.5, 3.4.6)

2.7.1 The Mode 1 and Mode 2 required SDM versus RCS boron concentration is presented in Figure 5.0.

2.7.2 The Mode 3 SDM requirements are as follows:

a) With at least 2 reactor coolant pumps in operation, the SDM shall be greater than or equal to that specified in Figure 5.0.

b) With less than 2 reactor coolant pumps in operation and the rod control system capable of rod withdrawal, the SDM shall be greater than or equal to 4% $\Delta k/k$.

c) With less than 2 reactor coolant pumps in operation and with the rod control system not capable of rod withdrawal, the SDM shall be greater than or equal to that specified in Figure 5.0.

2.7.3 The Mode 4 SDM requirements are as follows:

a) With at least 2 reactor coolant pumps in operation, the SDM shall be greater than or equal to that specified in Figure 5.0.

b) With less than 2 reactor coolant pumps in operation and the rod control system capable of rod withdrawal, the SDM shall be greater than or equal to 4% $\Delta k/k$.

c) With less than 2 reactor coolant pumps in operation and with the rod control system not capable of rod withdrawal, the SDM shall be greater than or equal to that specified in Figure 5.0.

2.7.4 The minimum required SDM for Mode 5 is 1% $\Delta k/k$.

2.7.5 The minimum required SDM for Mode 6 is 6% $\Delta k/k$.

2.8 Refueling Boron Concentration (ITS 3.9.1)

2.8.1 In Mode 6 the minimum boron concentration shall be 1950 ppm.

3.0 METHODOLOGY REFERENCES

1) XN-75-27(A), and Supplements 1, 2, 3, and 4 "Exxon Nuclear Neutronics Design Methods for Pressurized Water Reactors", Exxon Nuclear Company.

2) Not Used For Cycle 20

3) XN-NF-82-21(A), Revision 1, "Application of Exxon Nuclear Company PWR Thermal Margin Methodology to Mixed Core Configurations," Exxon Nuclear Company.

4) XN-NF-84-093(A), and Supplement 1, "Steamline Break Methodology for PWRs," Advanced Nuclear Fuels Corporation

5) XN-75-32(A) Supplements 1, 2, 3, and 4, "Computational Procedure for Evaluating Fuel Rod Bow," Exxon Nuclear Company.

6) XN-NF-82-49(A), Revision 1 and Supplement 1, "Exxon Nuclear Company Evaluation Model Revised EXEM PWR Small Break Model," Siemens Power Corporation.

7) XN-NF-82-20(A), Revision 1 and Supplements 1, 2, 3, and 4, "Exxon Nuclear Company Evaluation Model EXEM/PWR ECCS Model Updates", Exxon Nuclear Company.

   XN-NF-82-07(A), Revision 1, Exxon Nuclear Company ECCS Cladding Swelling and Rupture Model," Exxon Nuclear Company.

   XN-NF-81-58(A), Revision 2, and Supplements 1, 2, 3, and 4, "RODEX2 Fuel Rod Thermal-Mechanical Response Evaluation Model," Exxon Nuclear Company.

   XN-NF-85-16(A), Volume 1 and Supplements 1, 2, and 3, Volume 2, Revision 1, and Supplement 1, "PWR 17x17 Fuel Cooling Test Program,: Exxon Nuclear Company.

   XN-NF-85-105(A), and Supplement 1, "Scaling of FCTF Based Reflood Heat Transfer Correlation for Other Bundle Designs," Exxon Nuclear Company.

8) XN-NF-78-44(A)," A Generic Analysis of the Control Rod Ejection Transient for Pressurized Water Reactors," Exxon Nuclear Company.

9) Not Used For Cycle 20

10) Not Used For Cycle 20

11) XN-NF-82-06(A), Revision 1 and Supplements 2, 4, and 5, "Qualification of Exxon Nuclear Fuel for Extended Burnup (PWR)," Exxon Nuclear Company.

12) Not Used For Cycle 20

13) Not Used For Cycle 20

14) Not Used For Cycle 20

15) Not Used For Cycle 20

16) ANF-88-054(A), "PDC-3: Advanced Nuclear Fuels Corporation Power Distribution Control for Pressurized Water Reactors and Application of PDC-3 to H.B. Robinson Unit 2," Advanced Nuclear Fuels Corporation.

17) ANF-88-133(A), and Supplement 1, "Qualification of Advanced Nuclear Fuels PWR Design Methodology for Rod Burnups of 62 GWd/MTU," Advanced Nuclear Fuels Corporation.

18) ANF-89-151 (P) (A), and correspondence "ANF-RELAP Methodology for Pressurized Water Reactors: Analysis of Non-LOCA Chapter 15 Events," Advanced Nuclear Fuels Corporation.

19) EMF-92-081 (P) (A), and Supplement 1, "Statistical Setpoint/Transient Methodology for Westinghouse Type Reactors," Siemens Power Corporation

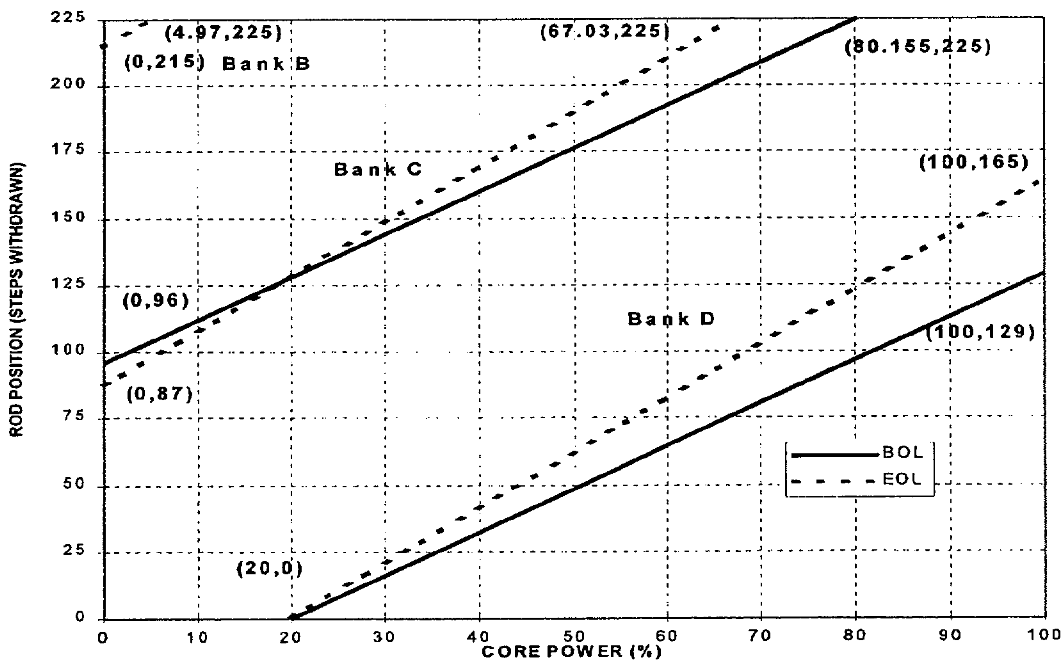
20) EMF-92-153 (P) (A), Revision 0 and Supplement 1, "HTP: Departure from Nucleate Boiling Correlation for High Thermal Performance Fuel," Siemens Power Corporation.

21) XN-NF-85-92 (P)(A), "Exxon Nuclear Uranium Dioxide/Gadolinia Irradiation Examination and Thermal Conductivity Results," Exxon Nuclear Company, November 1986.

22) EMF-96-029 (P)(A), Volume 1, Volume 2 and Attachment, "Reactor Analysis System for PWRs," Siemens Power Corporation, January 1997.

HBRSEP UNIT NO. 2, CYCLE 20
CORE OPERATING LIMITS REPORT
REVISION 2

Figure 1.0, Control Group Insertion Limits for Three Loop Operation



NOTE: The breakpoint between BOL and EOL RIL occurs at 50% of the cycle as defined by burnup. For Cycle 20, this burnup occurs at 246 EFPDs.

Control rod banks shall always be withdrawn and inserted in the prescribed sequence. For withdrawal, the sequence is Shutdown "A", Shutdown "B", Control "A", Control "B", Control "C", and Control "D". The insertion sequence is the reverse of the withdrawal sequence.

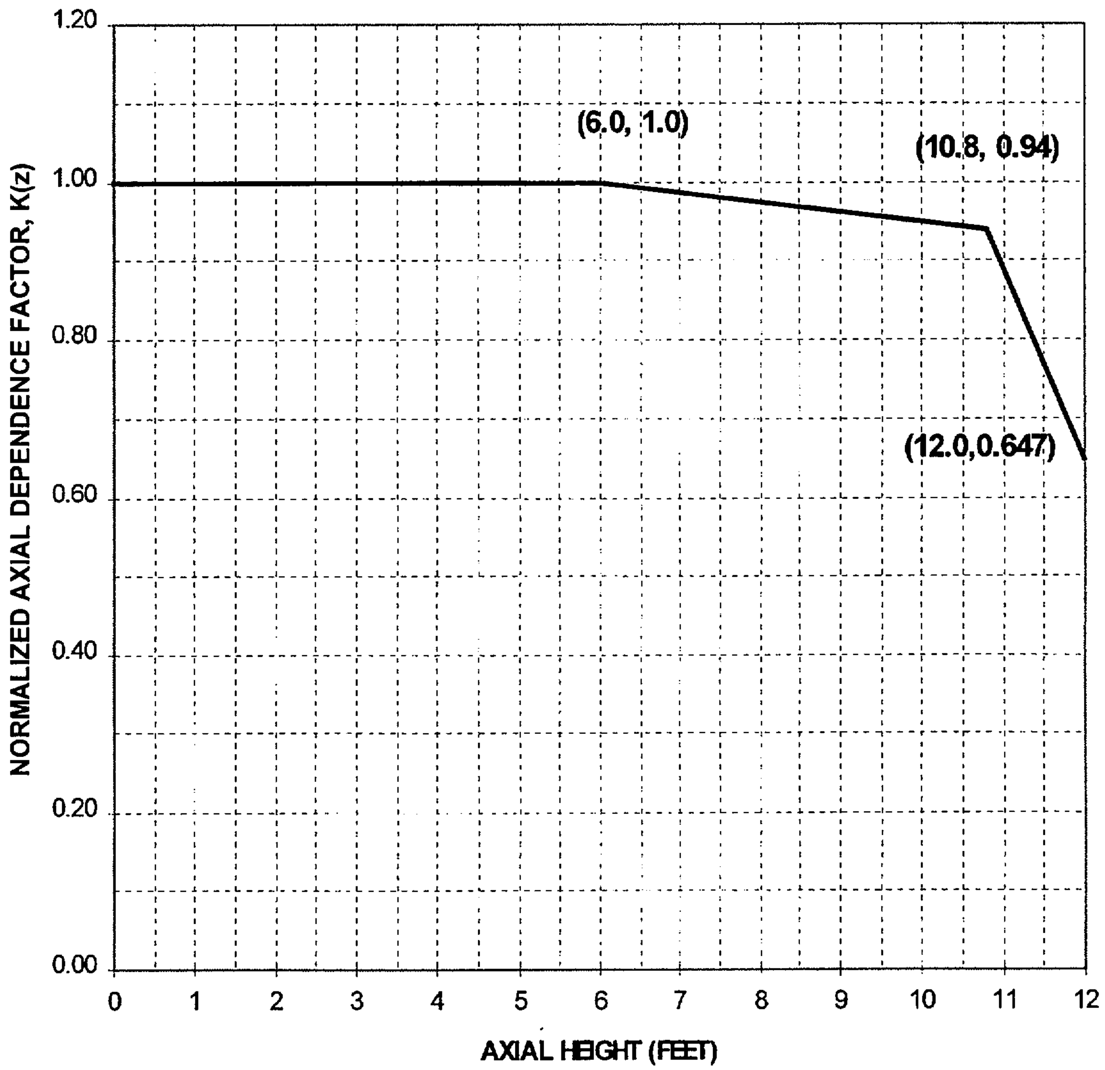
Overlap of consecutive Control Banks shall not exceed the prescribed setpoint for automatic overlap. The setpoint is 97 steps.

Control Bank A must be withdrawn from the core prior to power operation.

At BOL and 0% core power, Control Bank B will be at or above step 224.

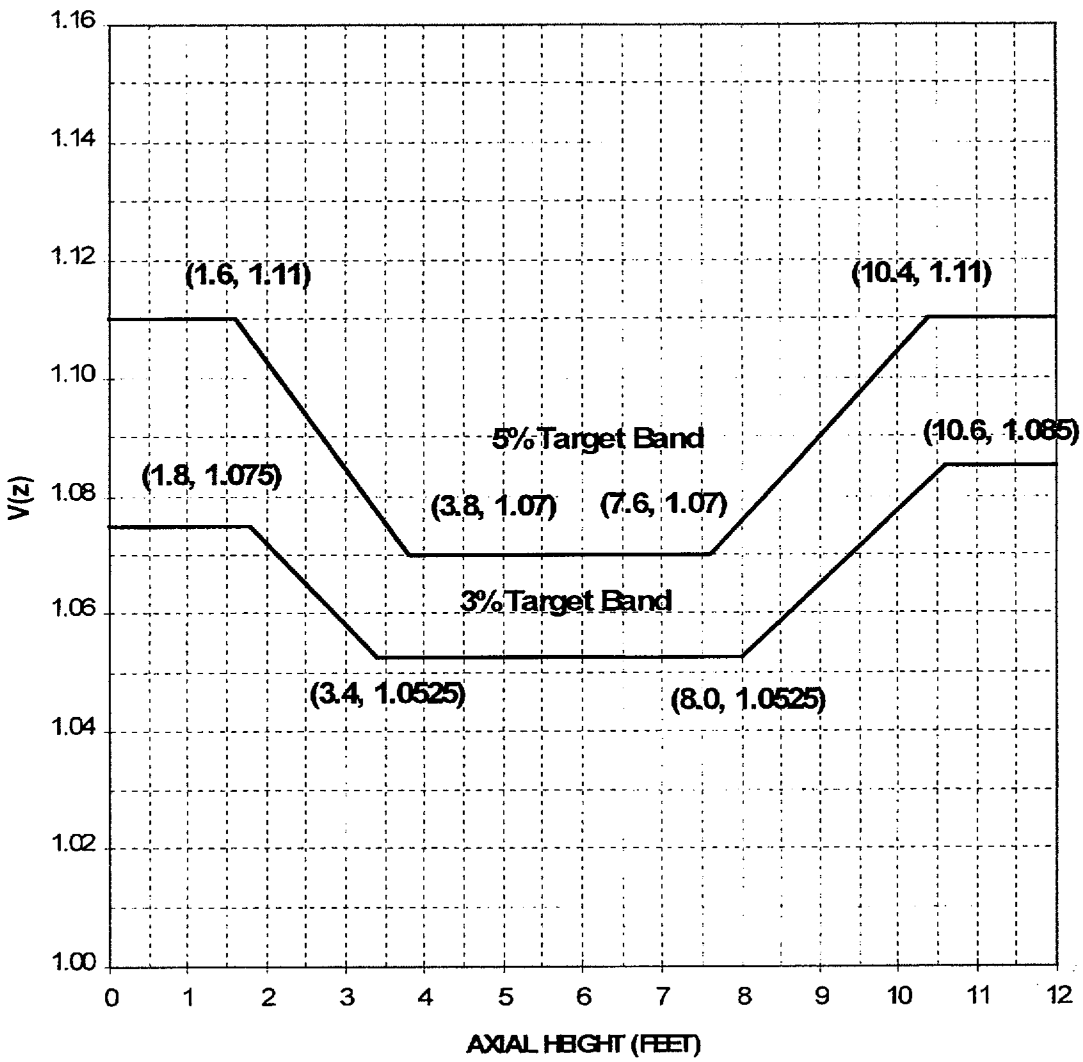
HBRSEP UNIT NO. 2, CYCLE 20
CORE OPERATING LIMITS REPORT
REVISION 2

Figure 2.0, Normalized Axial Dependence Factor K(z) for Fq Versus Elevation



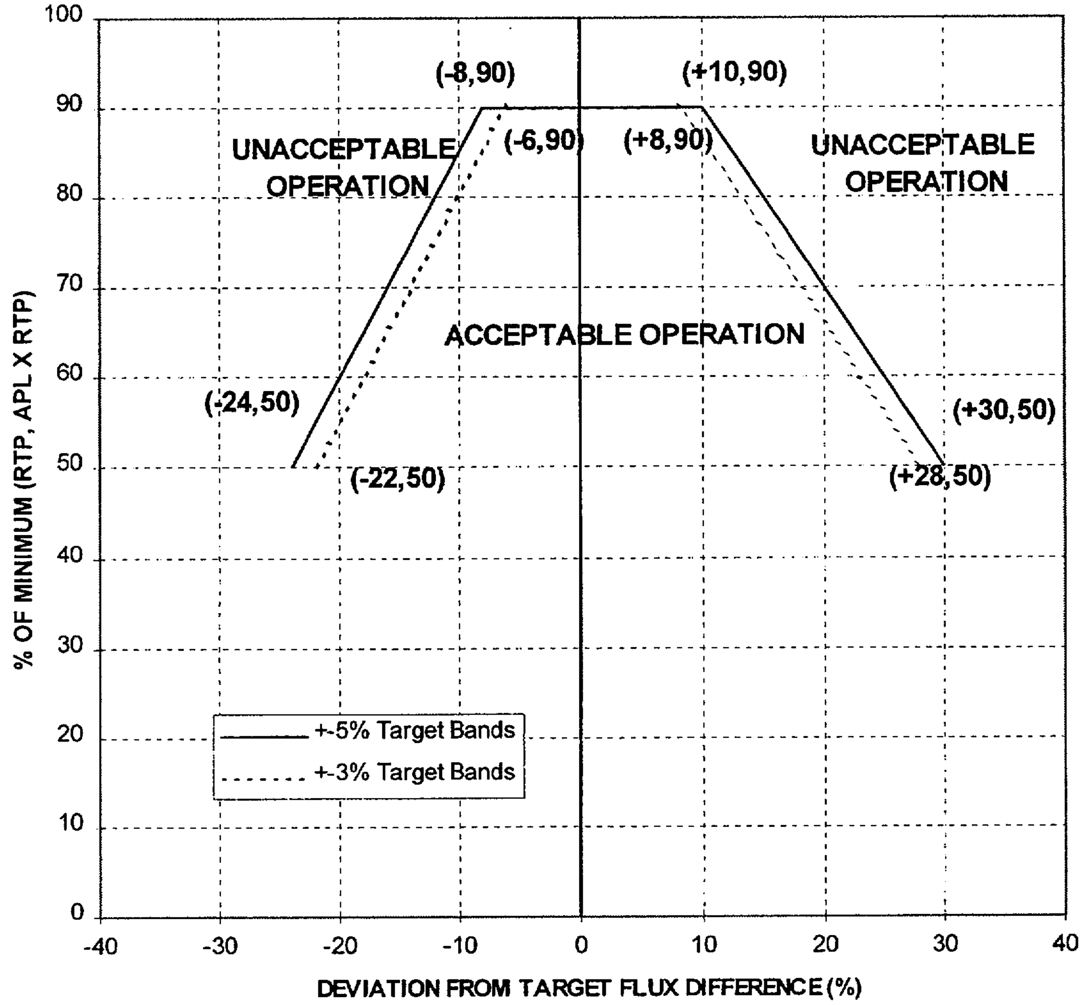
HBRSEP UNIT NO. 2, CYCLE 20
CORE OPERATING LIMITS REPORT
REVISION 2

Figure 3.0, V(z) as a Function of Core Height



HBRSEP UNIT NO. 2, CYCLE 20
CORE OPERATING LIMITS REPORT
REVISION 2

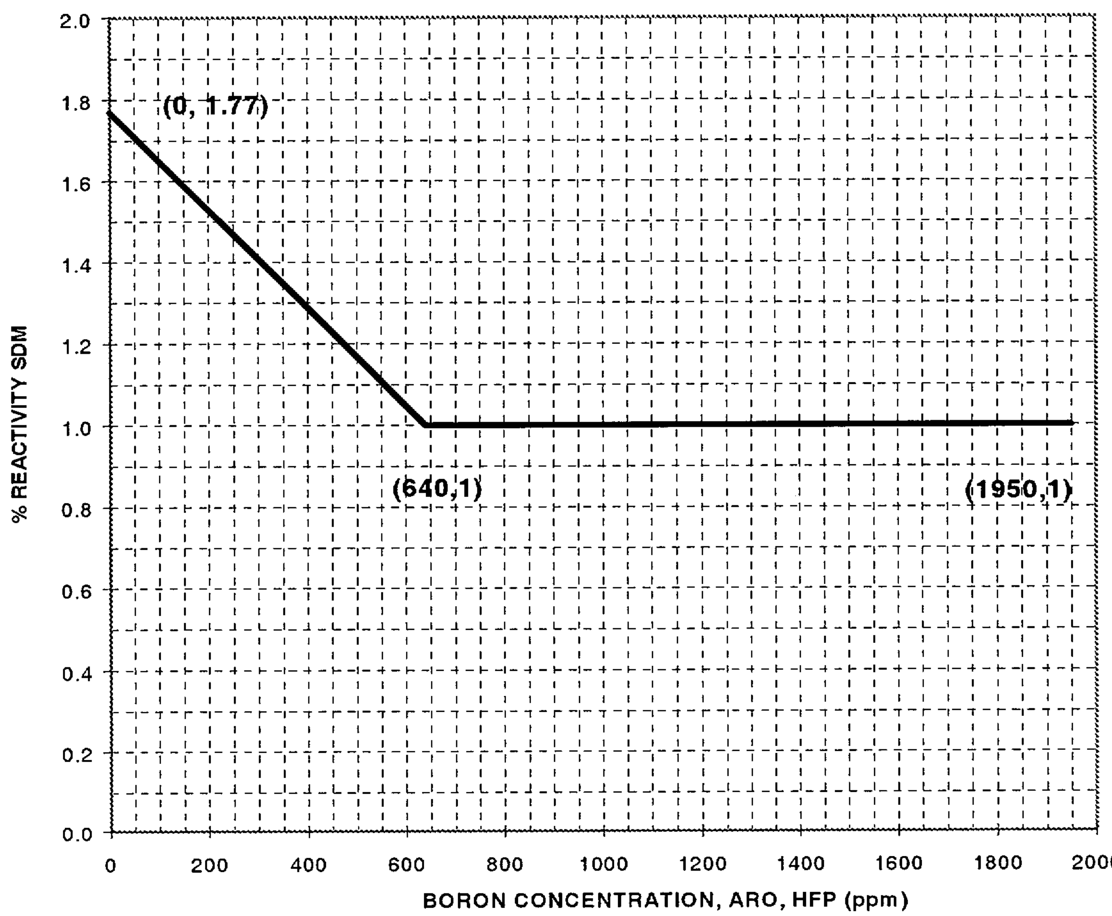
Figure 4.0, Allowable Deviation from Target Flux Difference



NOTE: For power levels above 90%, power operation is allowed within the target bands (±3% and ±5%).

HBRSEP UNIT NO. 2, CYCLE 20
CORE OPERATING LIMITS REPORT
REVISION 2

Figure 5.0, Shutdown Margin Versus Boron Concentration



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