Wisconsin Electric
A WISCONSIN ENERGY COMPANY

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10 CFR 50.90

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Document Control Desk
U.S. NUCLEAR REGULATORY COMMISSION
Mail Station P1-137
Washington, DC 20555

Ladies and Gentlemen:

DOCKETS 50-266 AND 50-301
TECHNICAL SPECIFCATIONS CHANGE REQUEST 218
CORE OPERATING LIMITS REPORT IMPLEMENTATION
POINT BEACH NUCLEAR PLANT, UNITS 1 AND 2

In accordance with the requirements of 10 CFR 50.4 and 10 CFR 50.90, Wisconsin Electric Power Company, licensee, hereby requests amendments to Facility Operating Licenses DPR-24 and DPR-27 for Point Beach Nuclear Plant, Units 1 and 2, respectively. The purpose of the proposed amendments is to implement a Core Operating Limits Report (COLR) concurrent with implementation of Improved Standard Technical Specifications at the Point Beach Nuclear Plant (PBNP). An application to convert the PBNP custom Technical Specifications to Standard Technical Specifications based on NUREG 1431, "Standard Technical Specifications, Westinghouse Plants", Revision 1, was submitted on November 15, 1999.

NRC Generic Letter 88-16 provides guidance for licensees allowing relocation of cycle dependent variables from the Technical Specifications (TS), provided that the values of these variables are included in a COLR, are determined with NRC-approved methodologies referenced in the TS, and changes are reported to the NRC as they are made. The variables removed from the TS and relocated to the COLR can then be changed via the appropriate regulatory change mechanisms, principally 10 CFR 50.59, thereby avoiding the need to frequently change TS. The appropriate safety limits are maintained in the Specifications.

Cycle specific parameters previously approved for Westinghouse units for relocation to the COLR include: (1) moderator temperature coefficient, (2) shutdown bank insertion limits, (3) control bank insertion limits, (4) axial flux difference limits, (5) nuclear heat flux hot channel factor limit (F_Q), (6) nuclear enthalpy rise hot channel factor limit ($F^N_{\Delta H}$), (7) refueling boron concentration limit, and (8) shutdown margin. In addition, as described in WCAP-14483-A, "Generic Methodology For Expanded Core Operation Limits Report," the reactor core safety limit curves and cycle specific DNB related parameter limits (RCS flow, temperature and pressure) may also be relocated with the addition of the Departure From Nucleate Boiling Ratio (DNBR) design limit, the fuel centerline melt temperature limit, and the retention of the minimum reactor coolant flow design limit in the TS.

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Provided with this license amendment request is a description and justification of changes, safety evaluation, and no significant hazards determination. Also provided, for information only, is a sample COLR for the Point Beach units. The parameter limits contained within this sample COLR include limits applicable to present operation with standard and OFA fuel, as well as limits approved with Amendments 193 and 198 for Units 1 and 2 respectively, on February 8, 2000, which allow use of Westinghouse 422V+ fuel at PBNP.

Our application for conversion of the PBNP Technical Specifications to the Improved Standard Technical Specifications reflects the incorporation of a COLR into the Limiting Conditions of Operation (LCO) with the exception of those changes detailed in WCAP-14483-A, recently approved for inclusion in the COLR. This application includes those changes generically approved in WCAP-14483-A. Our conversion submittal will be supplemented to include the changes requested by this application.

We have determined that the proposed amendments do not involve a significant hazards consideration, authorize a significant change in the types or total amounts of effluent release, or result in any significant increase in individual or cumulative occupational radiation exposure. Therefore, we conclude that the proposed amendments meet the categorical exclusion requirements of 10 CFR 51.22(c)(9) and that an environmental impact appraisal need not be prepared.

We request that these amendments be reviewed and approved such that the COLR may be implemented with the improved TS at PBNP.

Sincerely,

Mark A. Reddemann

Mark A. Reddemann
Vice President
Point Beach Nuclear Plant

Subscribed and sworn before me on
this 2nd day of March, 2000.

Christine K. Pozorski Christine K. Pozorski
Notary Public, State of Wisconsin

My commission expires 8/25/2002.

cc: NRC Regional Administrator
NRC Resident Inspector
NRC Project Manager
PSCW

DOCKETS 50-266 AND 50-301
TECHNICAL SPECIFICATIONS CHANGE REQUEST 218
CORE OPERATING LIMITS REPORT IMPLEMENTATION
POINT BEACH NUCLEAR PLANT, UNITS 1 AND 2

DESCRIPTION AND REASON FOR CHANGES

Generic Letter (GL) 88-16, "Removal Of Cycle-Specific Parameter Limits From Technical Specifications," endorsed the removal of cycle specific parameter limits from the Technical Specifications (TS) provided the parameters are determined using NRC approved methodologies and changes to the parameters are reported. The alternative defined in GL 88-16 to specifying such limits in the Technical Specifications, controls the values of the cycle-specific parameters and assures conformance to 10 CFR 50.36 which calls for specifying the lowest functional performance levels acceptable for continued operation to be included in the TS, by specifying the calculational methodologies and appropriate acceptance criteria. This is accomplished within the Technical Specifications by defining a Core Operating Limits Report (COLR) in which the parameter limits are located, defining the approved methodologies used to determine such limits within the TS, and implementation of a reporting requirement to ensure changes to the cycle specific parameters are reported.

Cycle specific parameters previously approved for Westinghouse units for relocation to the COLR include: (1) moderator temperature coefficient, (2) shutdown bank insertion limits, (3) control bank insertion limits, (4) axial flux difference limits, (5) nuclear heat flux hot channel factor limit (F_Q), (6) nuclear enthalpy rise hot channel factor limit ($F^N_{\Delta H}$), (7) refueling boron concentration limit, and (8) shutdown margin. In addition, as described in WCAP-14483-A, "Generic Methodology For Expanded Core Operation Limits Report," the reactor core safety limit curves and cycle specific Departure from Nucleate Boiling (DNB) parameters limits (RCS flow, pressure and temperature) may also be relocated with the addition of the Departure From Nucleate Boiling Ration (DNBR) design limit, the fuel centerline melt temperature limit, and the retention of the minimum reactor coolant flow design limit in the TS.

The changes proposed to implement the use of the COLR for Point Beach Nuclear Plant, Units 1 and 2 are described below.

Description of change:

The definition of a Core Operating Limits Report (COLR) is proposed for addition to the Technical Specifications. This definition corresponds to that proposed in our November 15, 1999, submittal for conversion to the Standard Technical Specifications as described in NUREG 1431. The COLR is defined as follows (the TS number in brackets in the definition will be replaced with the plant specific with the approval of the TS conversion):

Core Operating Limits Report

The COLR is the unit specific document that provides cycle specific parameter limits for the current reload cycle. These cycle specific parameter limits shall be determined for each reload cycle in accordance with Specification [5.6.5]. Plant operation within these limits is addressed in individual Specifications.

Basis for change:

The addition of the COLR definition is in accordance with the guidance provided in GL 88-16. The definition is as provided in NUREG 1431, "Standard Technical Specifications, Westinghouse Plants," Revision 1.

This change is administrative only, as it defines the COLR and its use. The definition of the COLR does not, in itself, impose new requirements on the operation of the Point Beach Nuclear Plant. Therefore, this change is administrative.

Description of change:

The addition of an administrative reporting requirement which defines the approved methodologies to be used for Point Beach is proposed for addition to the TS. This proposed reporting requirement includes changes approved with Amendments 193 and 198 for Units 1 and 2, respectively on February 8, 2000, allowing use of the Westinghouse 422V+ fuel at PBNP and the application of the best-estimate LOCA analysis methodology described in WCAP-14449-P-A. The proposed reporting requirement, which will be included in a supplement to our November 15, 1999, ITS conversion submittal is as follows (the LCO corresponds to applicable proposed converted TS and may change pending review and approval of amendments to implement the converted TS):

CORE OPERATING REPORT (COLR) LIMITS

a. Core operating limits shall be established prior to each reload cycle, or prior to any remaining portion of a reload cycle, and shall be documented in the COLR for the following:

    (1) LCO 2.1.1, "Reactor Core Safety Limits (SLs)"

    (2) LCO 3.1.1, "Shutdown Margin"

    (3) LCO 3.1.3, "Moderator Temperature Coefficient"

    (4) LCO 3.1.5, "Shutdown Bank Insertion Limits"

    (5) LCO 3.1.6, "Control Bank Insertion Limits"

(6) LCO 3.2.1, "Nuclear Heat Flux Hot Channel Factor ($F_Q(Z)$)"

(7) LCO 3.2.2, "Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$)"

(8) LCO 3.2.3, "Axial Flux Difference"

(9) LCO 3.3.1, Overtemperature ΔT

(10) LCO 3.3.1, Overpower ΔT

(11) LCO 3.4.1, "DNB Limits"

(12) LCO 3.9.1, "Refueling Boron Concentration"

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

(1) WCAP-14449-P-A, "Application of Best Estimate Large Break LOCA Methodology to Westinghouse PWR's with Upper Plenum Injection," Revision 1, October 1999. (cores containing 422V+ fuel)

(2) WCAP-9272-P-A, "Westinghouse Reload Safety Evaluation Methodology," July 1985.

(3) WCAP-11397-P-A, "Revised Thermal Design Procedure," April 1989.

(4) WCAP-14787-P, "Westinghouse Revised Thermal Design Procedures Instrument Uncertainty Methodology, Wisconsin Electric Power Company, Point Beach Unit 1 and 2," April 1999 (approved by NRC Safety Evaluation, February 8, 2000).

(5) WCAP-10054-P-A, "Westinghouse Small Break ECCS Evaluation Model Using The NOTRUMP Code," August 1985.

(6) WCAP-10054-P-A, "Addendum to the Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code: Safety Injection into the Broken Loop and COSI Condensation Model," Addendum 2, Revision 1, July 1997.

(7) WCAP-8745-P-A, "Design Bases for the Thermal Overpower ΔT and Thermal Overtemperature ΔT Trip Functions," September 1986.

(8) WCAP-10216-P-A, "Relaxation of Constant Axial Offset Control," Revision 1A, February 1994.

(9) WCAP-10924-P-A, "Large Break LOCA Best Estimate Methodology, Volume 2: Application to Two-Loop PWRs Equipped with Upper Plenum Injection," and Addenda, December 1988 (cores not containing 422 V+ fuel).

(10) WCAP-10924-P-A, "LBLOCA Best Estimate Methodology: Model Description and Validation: Model Revisions," Volume 1, Addendum 4, August 1990 (cores not containing 422 V+ fuel).

c. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling Systems (ECCS) limits, nuclear limits such as SDM, transient analysis limits, and accident analysis limits) of the safety analysis are met.

d. The COLR, including any midcycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC.

Basis for change:

The reporting condition proposed for addition to the TS is in accordance with the guidance in GL 88-16 and NUREG-1431. The addition of this requirement ensures that the cycle-specific operating parameters are determined in accordance with NRC approved methodologies and continue to meet the acceptance criteria of the safety analyses as described in the updated Final Safety Analysis Report. Thus, reactor operation will continue to meet all regulatory and design basis requirements.

Application of each of the above, approved methodologies and the corresponding cycle specific parameter is as follows:

Parameter	NRC Approved Methodology
Reactor Core Safety Limits	WCAP-9272-P-A, "Westinghouse Reload Safety Evaluation Methodology," July 1985
Shutdown Margin	WCAP-9272-P-A, "Westinghouse Reload Safety Evaluation Methodology," July 1985
Moderator Temperature Coefficient	WCAP-9272-P-A, "Westinghouse Reload Safety Evaluation Methodology," July 1985
Shutdown Bank Insertion Limits	WCAP-9272-P-A, "Westinghouse Reload Safety Evaluation Methodology," July 1985
Control Bank Insertion Limits	WCAP-9272-P-A, "Westinghouse Reload Safety Evaluation Methodology," July 1985

Parameter	NRC Approved Methodology
Height Dependent Heat Flux Hot Channel Factor (F_Q)	WCAP-10216-P-A, Revision 1A, "Relaxation Of Constant Axial Offset Control F_Q Surveillance Technical Specification," February 1994. WCAP-14449-P-A, "Application Of Best Estimate Large Break LOCA Methodology To Westinghouse PWRs With Upper Plenum Injection," Revision 1, October 1999. WCAP-10054-P-A, "Westinghouse Small Break ECCS Evaluation Model Using The NOTRUMP Code," August 1985. WCAP-10924-P-A, "Large Break LOCA Best Estimate Methodology, Volume 2: Application to Two-Loop PWRs Equipped with Upper Plenum Injection," and Addenda, December 1988. (cores not containing 422 V+ fuel) WCAP-10924-P-A, "LBLOCA Best Estimate Methodology: Model Description and Validation: Model Revisions," Volume 1, Addendum 4, August 1990. (cores not containing 422 V+ fuel)
Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$)	WCAP-9272-P-A, "Westinghouse Reload Safety Evaluation Methodology," July 1985
Axial Flux Difference	WCAP-10216-P-A, Revision 1A, "Relaxation Of Constant Axial Offset Control F_Q Surveillance Technical Specification," February 1994.
Overtemperature ΔT	WCAP-8745-P-A, "Design Bases For The Thermal Overpower ΔT and Thermal Overtemperature ΔT Trip Functions," September 1986
Overpower ΔT	WCAP-8745-P-A, "Design Bases For The Thermal Overpower ΔT and Thermal Overtemperature ΔT Trip Functions," September 1986
DNB Limits	WCAP-11397-P-A, "Revised Thermal Design Procedure," April 1989 (for those

Parameter	NRC Approved Methodology
	events analyzed using RTDP). WCAP 9272-P-A, "Westinghouse Reload Safety Evaluation Methodology," July 1985 (for those events not utilizing RTDP).
Refueling Boron Concentration	WCAP-9272-P-A, "Westinghouse Reload Safety Evaluation Methodology," July 1985

These methodologies were used in the analyses supporting amendments approved on February 8, 2000 and/or for the safety analyses for PBNP applicable to standard and OFA fuel as documented in the updated Final Safety Analysis Report. In addition, the referenced methodologies for the DNB Limits and Reactor Core Safety limits are as approved for referencing in WCAP-14483-A, "Generic Methodology for Expanded Core Operating Limits Report," January 1999.

These methodologies have been approved for the use at PBNP. The addition of these methodologies to the Technical Specifications therefore, does not impose any new requirements on the operation of PBNP and is administrative.

<u>Description of change</u>:

Relocation of the Reactor Core Safety Limit curves was approved and is subject to the conditions defined in WCAP-14483-P-A. In accordance with the conditions of approval for the WCAP, it is proposed to add the DNBR design limits and fuel centerline melt temperature limits to the Technical Specifications and relocate the reactor core safety limit curves to the COLR. Operation within the acceptable region of the reactor core safety limit curves ensures these safety limits are met. The following DNBR limits will be added to the Technical Specifications in proposed section 2.0 by supplement to our November 15, 1999, conversion submittal:

≥ 1.22/1.21 (typical/thimble) for the WRB-1 correlation – cores not containing 422V+ fuel.

≥ 1.24/1.23 (typical/thimble) for the WRB-1 correlation – cores containing 422V+ fuel.

<u>OR</u>

≥ 1.30 for the W-3 correlation when system pressure is > 1000 psia

≥ 1.45 for the W-3 correlation when system pressure is ≥ 500 psia and ≤ 1000 psia

In addition, the following peak fuel centerline temperature limit will be added:

< 5080°F, decreasing by 58°F per 10,000 MWD/MTU of burnup.

<u>Bases for change</u>:

The addition of the above limits to the Technical Specifications and relocation of the reactor core safety limit curves to the COLR was approved as described in WCAP-14483-A.

The DNBR limits proposed for addition to the Technical Specification are as approved with the February 8, 2000 amendments and in the FSAR. The fuel centerline melt temperature limit is applicable to PBNP as described in WCAP- 9272-P-A.

PBNP analyses utilize two different DNBR correlations dependent on the use of the Revised Thermal Design Procedure (RTDP). RTDP, described in WCAP-11397-P-A, is used in several analyses for PBNP. For those analyses where one or more parameters fall outside the bounds of the RTDP, the W-3 correlation along with the Standard Thermal Design Procedure is used. Use and application of these correlations are as described in Chapters 3 and 14 of the PBNP updated FSAR and as approved in the February 8, 2000 amendments.

Description of change:

In accordance WCAP-14483-A and associated conditions of approval, the cycle specific DNB parameters related to reactor coolant system temperature, pressure and flow will be relocated to the COLR. The following reactor coolant system design minimum flow limits will be retained in the Technical Specifications by supplement to our November 15, 1999 submittal:

$\geq$ 181,800 gpm for cores not containing 422V+ fuel assemblies

$\geq$ 182,400 gpm for cores containing 422V+fuel assemblies

Bases for change:

Relocation of the DNB parameter limits to the COLR is in accordance with WCAP-14483-A. The methodology accepted for reference is as described in the table previously. The flow limits are as approved with Amendment 193 and 198 for Units 1 and 2, respectively, on February 8, 2000.

DOCKETS 50-266 AND 50-301
TECHNICAL SPECIFICATIONS CHANGE REQUEST 218
CORE OPERATING LIMITS REPORT IMPLEMENTATION
POINT BEACH NUCLEAR PLANT, UNITS 1 AND 2

SAFETY EVALUATION

NRC Generic Letter 88-16 provides guidance for licensees, allowing relocation of cycle dependent variables from the Technical Specifications (TS) provided that the values of these variables are included in a Core Operating Limits Report (COLR) and are determined with NRC-approved methodologies contained in a reporting requirement in the TS. The variables removed from the TS and relocated to the COLR can then be changed via the appropriate regulatory change mechanisms, principally 10 CFR 50.59, thereby avoiding the need to frequently change TS. Appropriate safety limits are maintained in the Specifications. Changes to the COLR parameters are reported to the NRC so that they can be monitored and trended.

Cycle specific parameters previously approved for Westinghouse units for relocation to the COLR include: (1) moderator temperature coefficient, (2) shutdown bank insertion limits, (3) control bank insertion limits, (4) axial flux difference limits, (5) nuclear heat flux hot channel factor limit (F_Q), (6) nuclear enthalpy rise hot channel factor limit ($F^N_{\Delta H}$), (7) refueling boron concentration limit, and (8) shutdown margin. In addition, as described in WCAP-14483-A, "Generic Methodology For Expanded Core Operation Limits Report," the reactor core safety limit curves and specific DNB related parameters may also be relocated with the addition of the Departure From Nucleate Boiling Ration (DNBR) design limit, the fuel centerline melt temperature limit, and the retention of the minimum reactor coolant flow design limit in the TS.

Amendments are proposed to allow implementation of a COLR with implementation of the improved Standard Technical Specifications at the Point Beach Nuclear Plant (PBNP). An amendment request was submitted converting the PBNP custom TS to the standard TS on November 15, 1999. That submittal assumed implementation of the COLR for parameters (1) through (8) above. The November 15, 1999, submittal will be supplemented to relocate the DNB parameters and Reactor Core Safety Limit Curves in accordance with approved WCAP-14483-A, and approved Industry/Standard Technical Specification Change Traveler (TSTF) 339, Revision 1.

This amendment request adds the definition of the COLR, and the reporting requirement as recommended by NUREG-1431, meeting the guidance in Generic Letter 88-16. The cycle specific parameters that are relocated will be determined by analyses performed in accordance with NRC approved methodologies. The methodologies are defined by the added reporting requirement and are approved for use at PBNP as described in the NRC Safety Evaluation, dated February 8, 2000, supporting Amendments193 and 198 for Units 1 and 2, respectively, and/or the PBNP updated Final Safety Analysis Report (FSAR).

Relocation of the specific DNB parameter limits and Reactor Core Safety Limit Curves is in accordance with approved WCAP-14483-A. Specific safety limits related to DNBR, fuel centerline melt temperature and reactor coolant system flow are added or retained in the Specifications with this change.

Therefore, the Specifications continue to meet the criteria of 10 CFR 50.36, Criterion 2, in that appropriate safety limits are maintained in the Technical Specifications. The reporting requirement ensures that the limits on the safety analyses continue to be met and that all changes to the design are performed in accordance with approved methodologies and reviewed under existing regulatory change mechanisms, principally 10 CFR 50.59. As such, these changes are essentially administrative and continue to ensure that PBNP is operated within the limits of its design and analyses.

Operation of PBNP in accordance with these proposed changes continue to ensure safe operation of the plant and will not be inimical to the health and safety of the public.

<u>DOCKETS 50-266 AND 50-301</u>
<u>TECHNICAL SPECIFICATIONS CHANGE REQUEST 218</u>
<u>CORE OPERATING LIMITS REPORT IMPLEMENTATION</u>
<u>POINT BEACH NUCLEAR PLANT, UNITS 1 AND 2</u>

<u>NO SIGNIFICANT HAZARDS DETERMINATION</u>

In accordance with the requirements of 10 CFR 50.4 and 10 CFR 50.90, Wisconsin Electric Power Company, licensee, requests amendments to Facility Operating Licenses DPR-24 and DPR-27 for Point Beach Nuclear Plant, Units 1 and 2, respectively. The purpose of the proposed amendments is to implement a Core Operating Limits Report (COLR) concurrent with implementation of standardized Technical Specifications at the Point Beach Nuclear Plant (PBNP).

In accordance with the requirements of 10 CFR 50.91, Wisconsin Electric has evaluated operation of the Point Beach Nuclear Plant in accordance with the proposed changes against the standards of 10 CFR 50.92. Operation of the Point Beach Nuclear Plant in accordance with the proposed changes results in no significant hazards consideration. Our evaluation and basis for this conclusion follows.

1. **Operation of the Point Beach Nuclear Plant in accordance with the proposed amendments does not result in a significant increase in the probability or consequences of any accident previously evaluated.**

   The proposed changes relocate certain cycle specific parameters from the Technical Specifications to a Core Operating Limits Report (COLR). Appropriate design and safety limits are retained or added to the Specifications thereby meeting the requirements of 10 CFR 50.36. Specific, approved methodologies used to determine and evaluate the parameter requirements are added to the Specifications and a reporting requirement is added to ensure the NRC is apprised of all changes. As approved methodologies are required to be used to evaluate and change parameters, and appropriate safety and design limits maintained in the Technical Changes, operation of PBNP will continue to meet all design and safety analysis requirements. Therefore, neither the probability nor consequences of an accident previously evaluated can be increased.

2. **Operation of the Point Beach Nuclear Plant in accordance with the proposed amendment does not create a new or different kind of accident from any accident previously evaluated.**

   Operation of PBNP, in accordance with the proposed changes, will continue to meet all design and safety limits. Appropriate design and safety limits continue to be controlled within the Technical Specifications as they are presently. These changes will not result in a change to the design and safety limits under which PBNP operation has been determined to be acceptable, these changes cannot result in a new or different kind of accident from any accident previously evaluated.

3. Operation of the Point Beach Nuclear Plant in accordance with the proposed amendment does not result in a significant reduction in a margin of safety.

Appropriate safety limits continue to be controlled by the Specifications. Changes to cycle specific parameters related to these limits will be accomplished using NRC approved methodologies, thereby ensuring operation will continue within the bounds of the existing safety analyses including all applicable margins of safety. Therefore, operation in accordance with the proposed changes cannot result in a significant reduction in a margin of safety.

Operation of the Point Beach Nuclear Plant in accordance with the proposed amendments does not result in a significant increase in the probability or consequences of any accident previously evaluated; does not create a new or different kind of accident from any accident previously evaluated; and, does not result in a significant reduction in a margin of the safety. Therefore, operation in accordance with the proposed amendments involves no significant hazards consideration.

Point Beach Nuclear Plant

Core Operating Limits Report

Unit 1, Cycle 26
Unit 2, Cycle 25

Note: This report is not part of the PBNP Technical Specifications. This report is referenced in the PBNP Technical Specifications.

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TABLE OF CONTENTS

TABLE OF CONTENTS (continued)

1.0 CORE OPERATING LIMITS REPORT

This Core Operating Limits Report (COLR) for Point Beach Nuclear Plant has been prepared in accordance with the requirements of Technical Specification (TS) 5.6.4.

A cross-reference between the COLR sections and the PBNP Technical Specifications affected by this report is given below:

COLR Section	PBNP ITS	Description
2.1	2.1.1	Reactor Core Safety Limits
2.2	3.1.1	Shutdown Margin
2.3	3.1.3	Moderator Temperature Coefficient
2.4	3.1.5	Shutdown Bank Insertion Limit
2.5	3.1.6	Control Bank Insertion Limits
2.6	3.2.1	Height Dependent Heat Flux Hot Channel Factor (F_Q)
2.7	3.2.2	Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$)
2.8	3.2.3	Axial Flux Difference
2.9	3.3.1	Overtemperature ΔT Setpoint
2.10	3.3.1	Overpower ΔT Setpoint
2.11	3.4.1	RCS Pressure, Temperature, and Flow Departure From Nucleate Boiling (DNB) Limits
2.12	3.9.1	Refueling Boron Concentration
Figure 1a	2.1.1	Reactor Core Safety Limits – cores not containing 422V+ fuel
Figure 1b	2.1.1	Reactor Core Safety Limits – cores containing 422V+ fuel
Figure 2	3.1.1	Required Shutdown Margin
Figure 3	3.1.6	Control Bank Insertion Limits
Figure 4a	3.2.1	Hot Channel Factor Normalized Operating Envelope - cores not containing 422V+ fuel (K(Z))
Figure 4b	3.2.1	Hot Channel Factor Normalized Operating Envelope – cores containing 422V+ fuel (K(Z))
Figure 5a	3.2.1	W(Z)
Figure 5b	3.2.1	W(Z)
Figure 6	3.2.3	Flux Difference Operating Envelope
Figure 7a	3.4.1	Reactor Coolant System Average Temperature Limits – cores not containing 422V+ fuel
Figure 7b	3.4.1	Reactor Coolant System Average Temperature Limits – cores containing 422V+ fuel

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 5.5.2.

2.1 Reactor Core Safety Limits (ITS 2.1.1)

2.1.1 The combination of THERMAL POWER, Reactor Coolant System (RCS) highest loop average temperature, and pressurizer pressure shall not exceed the limits specified in:

a. Figure 1a for cores not containing 422V+ fuel.

b. Figure 1b for cores containing 422V+ fuel.

Applicability: MODES 1 and 2

2.2 Shutdown Margin (ITS 3.1.1)

2.2.1 SDM shall be within the limits provided in Figure 2.

Applicability: MODES 1, 2, and 3

2.2.2 SDM shall be ≥ 1% Δk/k.

Applicability: MODES 4 and 5

2.3 Moderator Temperature Coefficient (ITS 3.1.3)

2.3.1 The upper MTC limits shall be maintained within the limits.

The maximum upper MTC limits shall be:

≤ 5 pcm/°F for power levels ≤70% RTP
≤ 0 pcm/°F for power levels > 70% RTP.

Applicability: MODE 1,
MODE 2 with $k_{eff} \geq 1.0$.

2.4 Shutdown Bank Insertion Limit (ITS 3.1.5)

----------------------------NOTE------------------------------
This limit is not applicable while performing SR 3.1.4.2.

2.4.1 Each shutdown bank shall be fully withdrawn.

Fully withdrawn is defined as ≥ 225 steps

Applicability: MODE 1,
MODE 2 with $K_{eff} \geq 1.0$

2.5 Control Bank Insertion Limits (ITS 3.1.6)

----------------------------NOTE------------------------------
This limit is not applicable while performing SR 3.1.4.2.

2.5.1 The control banks shall be within the insertion, sequence and overlap limits specified in Figure 3.

Fully withdrawn is defined as a bank demand position ≥ 225 steps.

Applicability: MODE 1,
MODE 2 with $K_{eff} \geq 1.0$

2.6 Nuclear Heat Flux Hot Channel Factor (F_Q) (ITS 3.2.1)

2.6.1 The Heat Flux Hot Channel Factor be within the following limits:

a. for cores not containing 422V+ fuel:

$CF_Q = 2.50$

K(Z) is the function in Figure 4a

W(Z) is the function in Figure 5a

b. for cores containing 422V+ fuel:

$CF_Q = 2.60$

K(Z) is the function in Figure 4b

W(Z) is the function in Figure 5b

Applicability: MODE 1

2.7 Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$) (ITS 3.2.2)

2.7.1 The Nuclear Enthalpy Rise Hot Channel Factor shall be within the following limit:

a. for cores not containing 422V+ fuel:

$$F^N_{\Delta H} < 1.70 \times [1 + 0.3(1 - P)]$$

b. for cores containing 422V+ fuel:

$$F^N_{\Delta H} < 1.77 \times [1 + 0.3(1 - P)]$$

where: P is the fraction of Rated Power at which the core is operating.

Applicability: MODE 1

2.8 Axial Flux Difference (ITS 3.2.3)

---NOTE---
The AFD shall be considered outside limits when two or more OPERABLE excore channels indicate AFD to be outside limits.

2.8.1 The indicated axial flux difference, in % flux difference units, shall be maintained within the allowed operational space defined by Figure 5.

Applicability: MODE 1 with THERMAL POWER ≥ 50% RTP

2.9 Overtemperature ΔT Setpoint (ITS 3.3.1)

Overtemperature ΔT setpoint parameter values:

ΔT_o	=	indicated ΔT at Rated Power, °F
T	=	average temperature, °F
T′	≤	569.0°F (for cores containing 422V+ fuel assemblies)
T′	≤	572.9°F (for cores not containing 422V+ fuel assemblies)
P′	=	2235 psig (2250 psia operation only)
P′	=	1985 psig (2000 psia operation only and cores not containing 422V+ fuel assemblies)
K_1	≤	1.16 (for 2250 psia operation and cores containing 422V+ fuel assemblies)
K_1	≤	1.19 (for 2250 psia operation and cores not containing 422V+ fuel assemblies)
K_1	≤	1.14 (for 2000 psia operation and cores not containing 422V+ fuel assemblies)
K_2	=	0.0149 (for 2250 psia operation and cores containing 422V+ fuel assemblies
K_2	=	0.025 (for 2250 psia operation and cores not containing 422V+ fuel assemblies)
K_2	=	0.022 (for 2000 psia operation and cores not containing 422 V+ fuel assemblies)
K_3	=	0.00072 (for 2250 psia operation and cores containing 422V+ fuel assemblies)
K_3	=	0.0013 (2250 psia operation and cores not containing 422V+ fuel assemblies)
K_3	=	0.001 (2000 psia operation and cores not containing 422V+ fuel assemblies)
τ_1	=	25 sec
τ_2	=	3 sec
τ_3	=	2 sec for Rosemont or equivalent RTD
	=	0 sec for Sostman or equivalent RTD
τ_4	=	2 sec for Rosemont or equivalent RTD
	=	0 sec for Sostman or equivalent RTD

f(ΔI) is an even function of the indicated difference between top and bottom detectors of the power-range nuclear ion chambers; with gains to be selected based on measured instrument response during plant startup tests, where q_t and q_b are the percent power in the top and bottom halves of the core respectively, and $q_t + q_b$ is total core power in percent of Rated Power, such that:

(a) for q_t - q_b within -17, +5 percent, f(ΔI) = 0 for cores not containing 422V+ fuel assemblies.

for q_t - q_b within –12, +5 percent, f(ΔI) = 0 for cores containing 422V+ fuel assemblies

(b) for each percent that the magnitude of q_t - q_b exceeds +5 percent, the ΔT trip setpoint shall be automatically reduced by an equivalent of 2.0 percent of Rated Power for cores not containing 422V+ fuel assemblies.

for each percent that the magnitude of q_t - q_b exceeds +5 percent, the ΔT trip setpoint shall be automatically reduced by an equivalent of 2.12 percent of Rated Power for cores containing 422V+ fuel assemblies

(c) for each percent that the magnitude of q_t - q_b exceeds -17 percent, the ΔT trip setpoint shall be automatically reduced by an equivalent of 2.0 percent of Rated Power for cores not containing 422V+ fuel assemblies.

for each percent that the magnitude of q_t - q_b exceeds -12 percent, the ΔT trip setpoint shall be automatically reduced by an equivalent of 2.0 percent of Rated Power for cores containing 422V+ fuel assemblies.

2.10 Overpower ΔT Setpoint (ITS 3.3.1)

Overpower ΔT setpoint parameter values:

ΔT_o	=	indicated ΔT at Rated Power, °F
T	=	average temperature, °F
T′	≤	569.0°F (for cores containing 422V+ fuel assemblies)
T′	≤	572.9°F (for cores not containing 422V+ fuel assemblies)
K_4	≤	1.10 of Rated Power (for cores containing 422V+ fuel assemblies)
K_4	≤	1.09 of Rated Power (for cores not containing 422V+ fuel assemblies)
K_5	=	0.0262 for increasing T
K_5	=	0.0 for decreasing T
K_6	=	0.00103 for $T \geq T'$ (for cores containing 422V+ fuel assemblies)
K_6	=	0.00123 for $T \geq T'$ (for cores not containing 422V+ fuel assemblies)
K_6	=	0.0 for $T < T'$
τ_5	=	10 sec
τ_3	=	2 sec for Rosemont or equivalent RTD
	=	0 sec for Sostman or equivalent RTD
τ_4	=	2 sec for Rosemont or equivalent RTD
	=	0 sec for Sostman or equivalent RTD

2.11 RCS Pressure, Temperature, and Flow Departure From Nucleate Boiling (DNB) Limits (ITS 3.4.1)

2.11.1 T_{avg} shall be maintained within the limits of :

a. Figure 7a for cores not containing 422V+ fuel.

b. Figure 7b for cores containing 422V+ fuel.

--NOTE--

Pressurizer pressure limit does not apply during:

a. THERMAL POWER ramp > 5% RTP per minute; or

b. THERMAL POWER step > 10% RTP.

2.11.2 Pressurizer pressure shall be maintained:

a. ≥ 1955 psig during operation at 2000 psia for cores not containing 422V+ fuel assemblies.

b. ≥ 2205 psig during operation at 2250 psia.

2.11.3 Reactor Coolant System raw measured Total Flow Rate shall be maintained

a. ≥ 181,800 gpm for cores not containing 422V+ fuel assemblies

b. ≥ 182,400 gpm for cores containing 422V+fuel assemblies

Applicability: MODE 1

2.12 Refueling Boron Concentration (ITS 3.9.1)

2.12.1 Boron concentrations of the Reactor Coolant System, the refueling canal, and the refueling cavity shall be maintained ≥ 2100 ppm.

Applicability: MODE 6

Figure 1a: Reactor Core Safety Limits
(cores not containing 422V+ fuel)

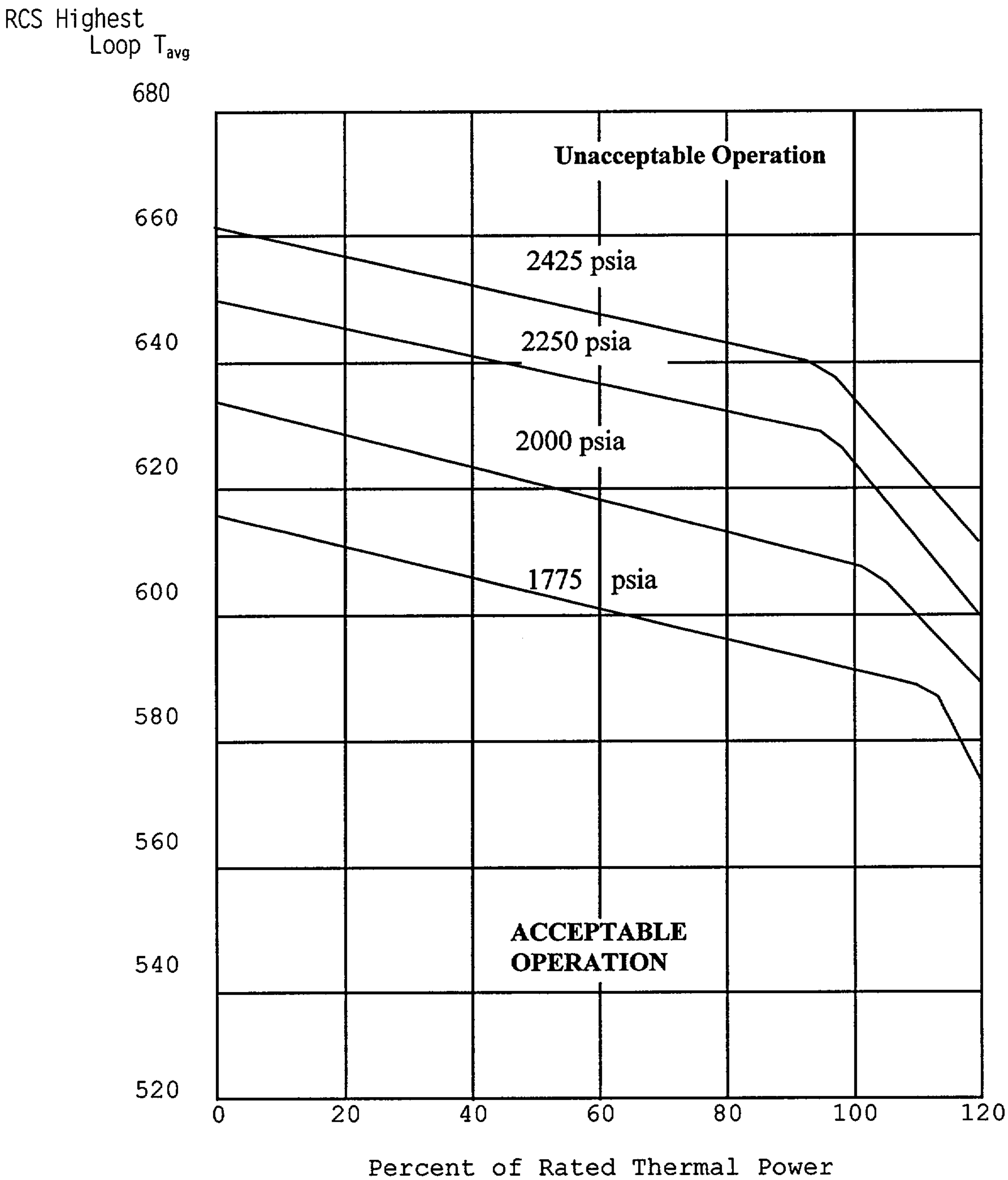

Figure 1b: Reactor Core Safety Limits Curve
(cores containing 422V+ fuel)

Figure 15.2.1-2*

POINT BEACH NUCLEAR PLANT UNITS 1 AND 2
REACTOR CORE SAFETY LIMITS

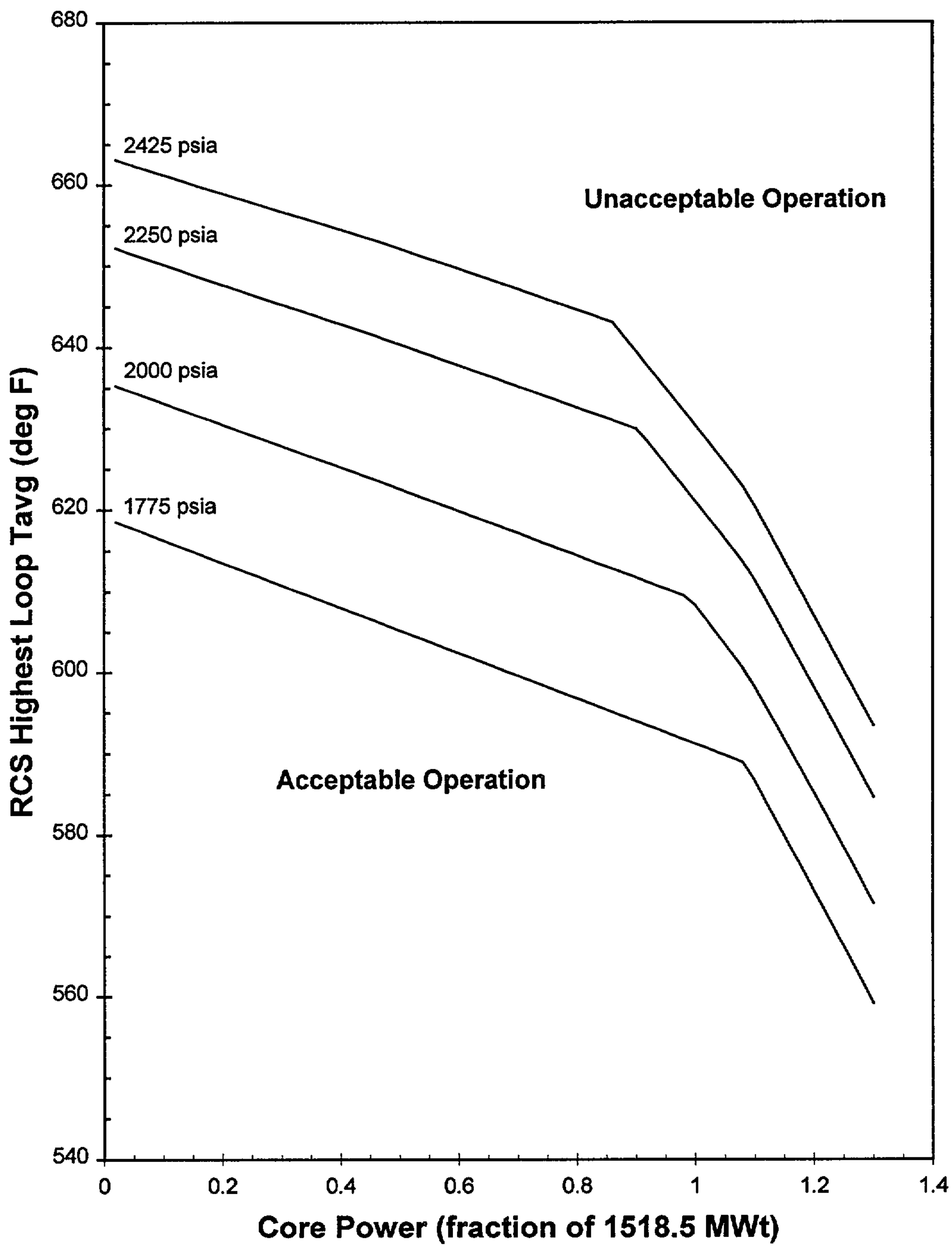

Figure 2: Required Shutdown Margin

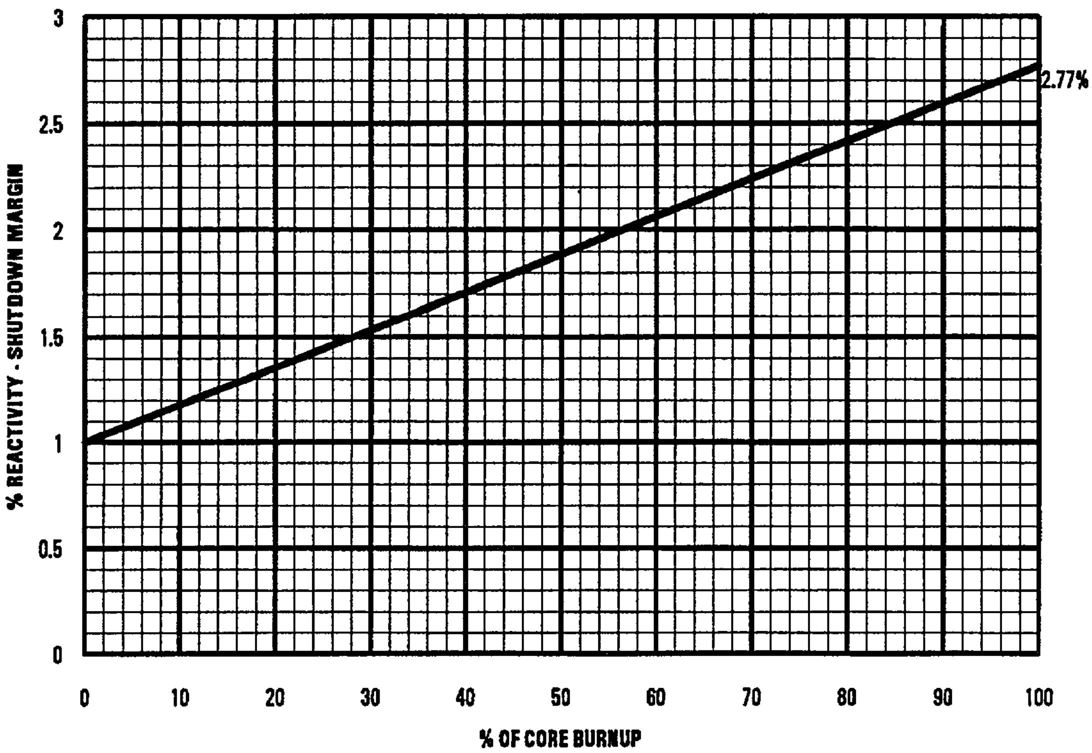

Figure 3: Control Bank Insertion Limits

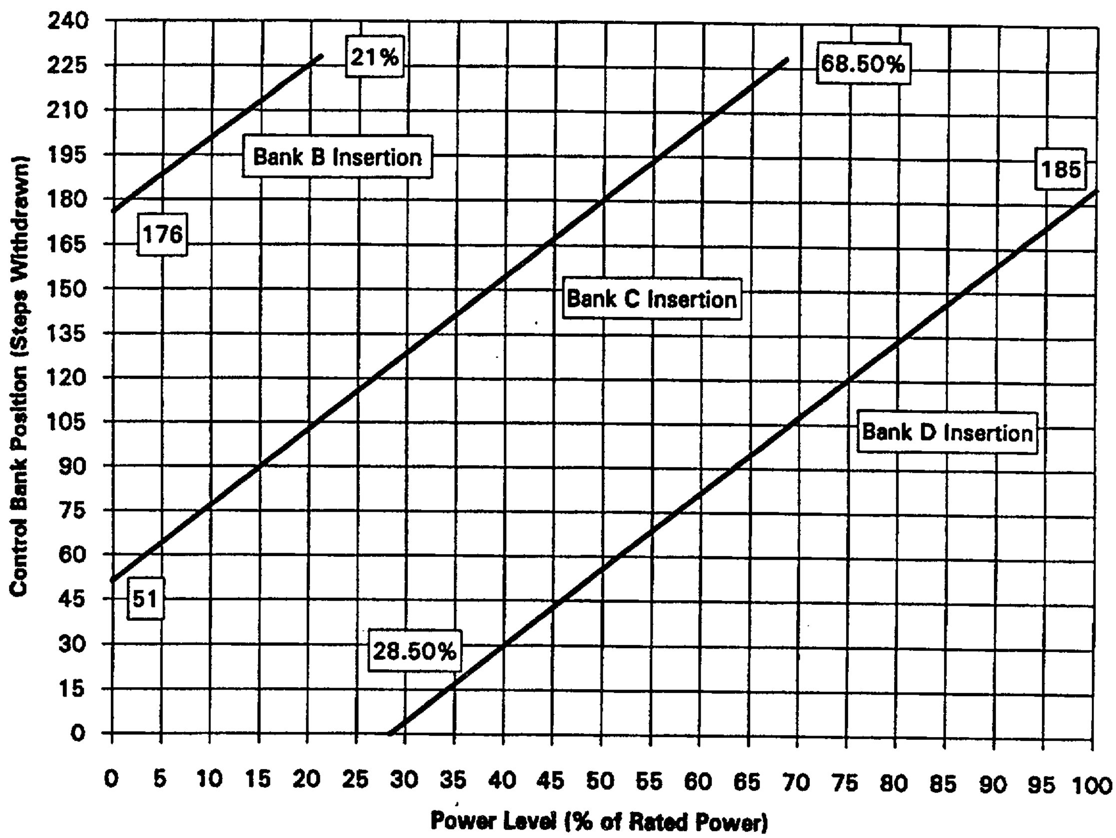

Figure 4a: Hot Channel Factor Normalized Operating Envelope (K(Z))
(cores not containing 422V+ fuel)

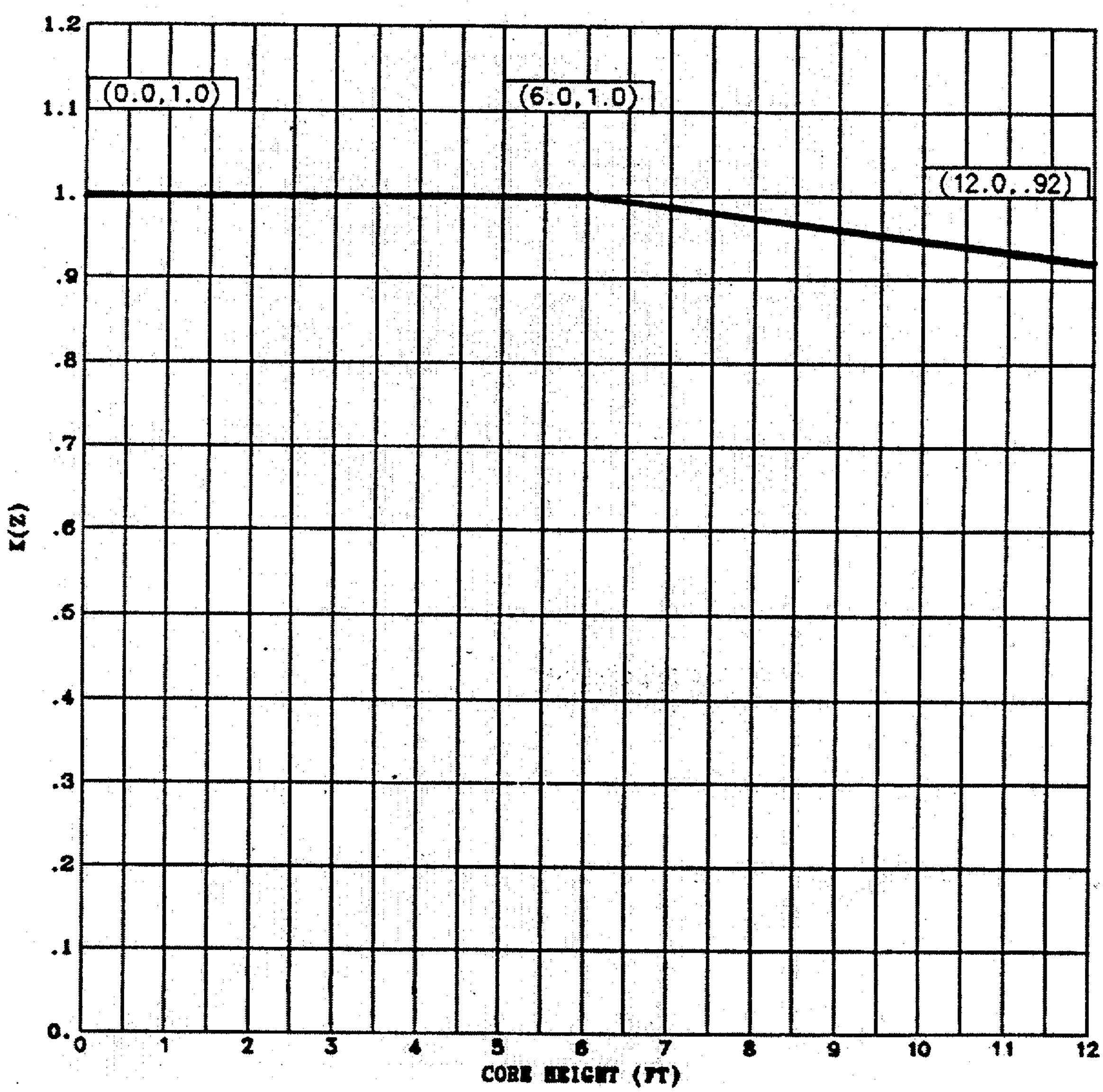

Figure 4b: Hot Channel Factor Normalized Operating Envelope (K(Z))
(cores containing 422V+ fuel)

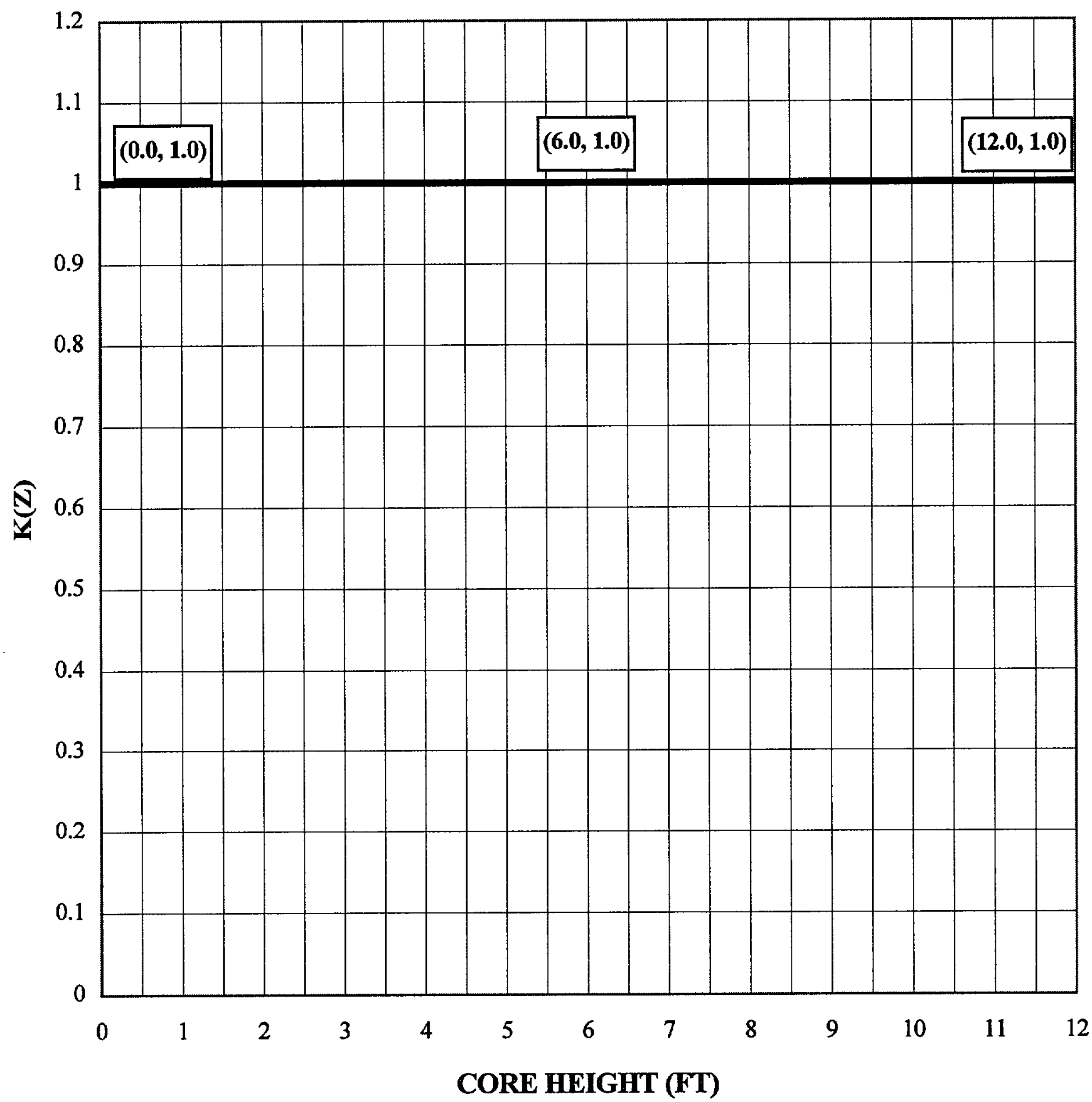

Figure 5a: W(Z) (cores not containing 422V+ Fuel)

To Be Provided

Figure 5b: W(Z) (cores containing 422V+ fuel)

To Be Provided

Figure 6: Flux Difference Operating Envelope

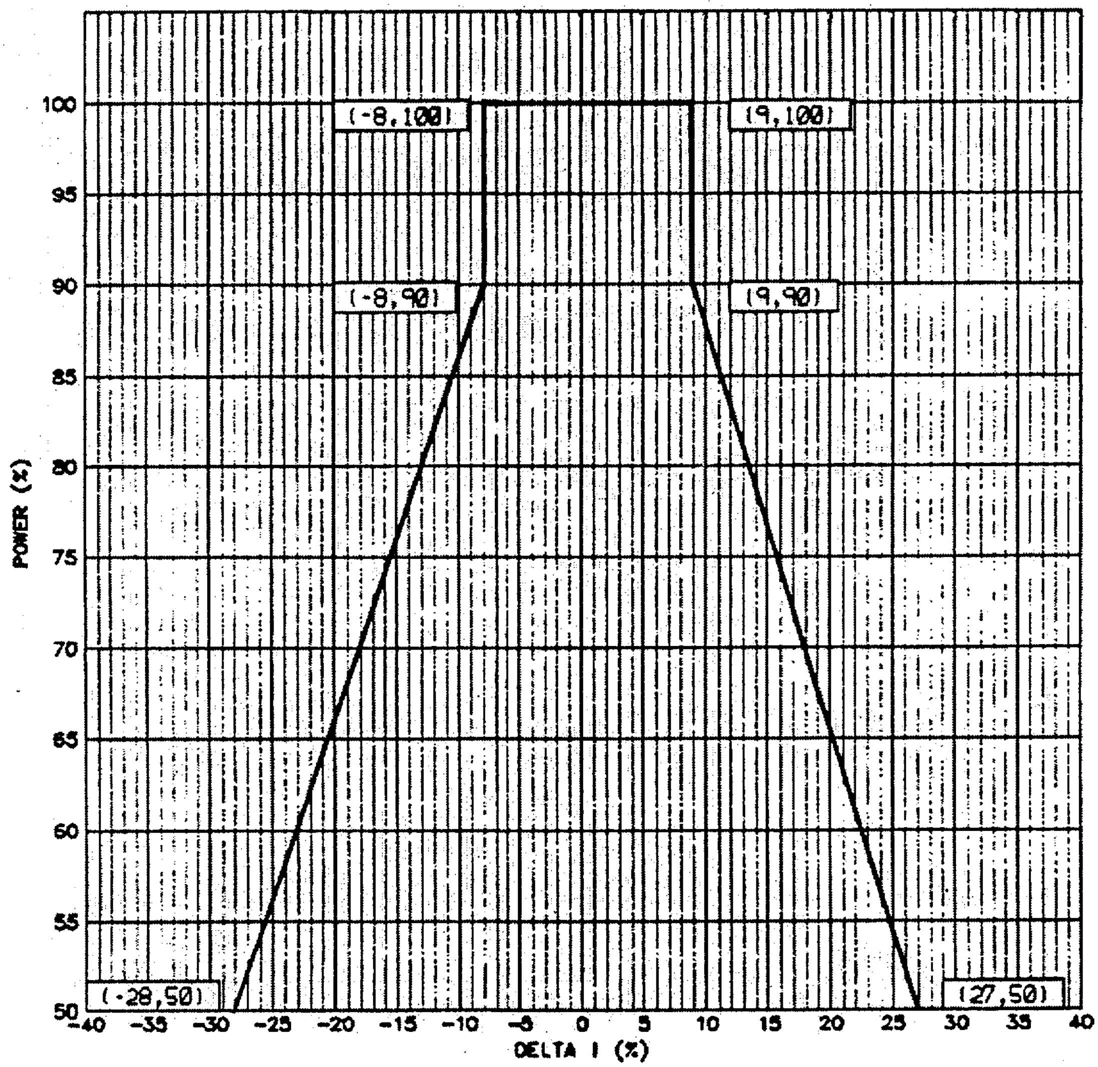

Figure 7a: Reactor Coolant System Average Temperature Limits
(Cores not containing 422V+ fuel)

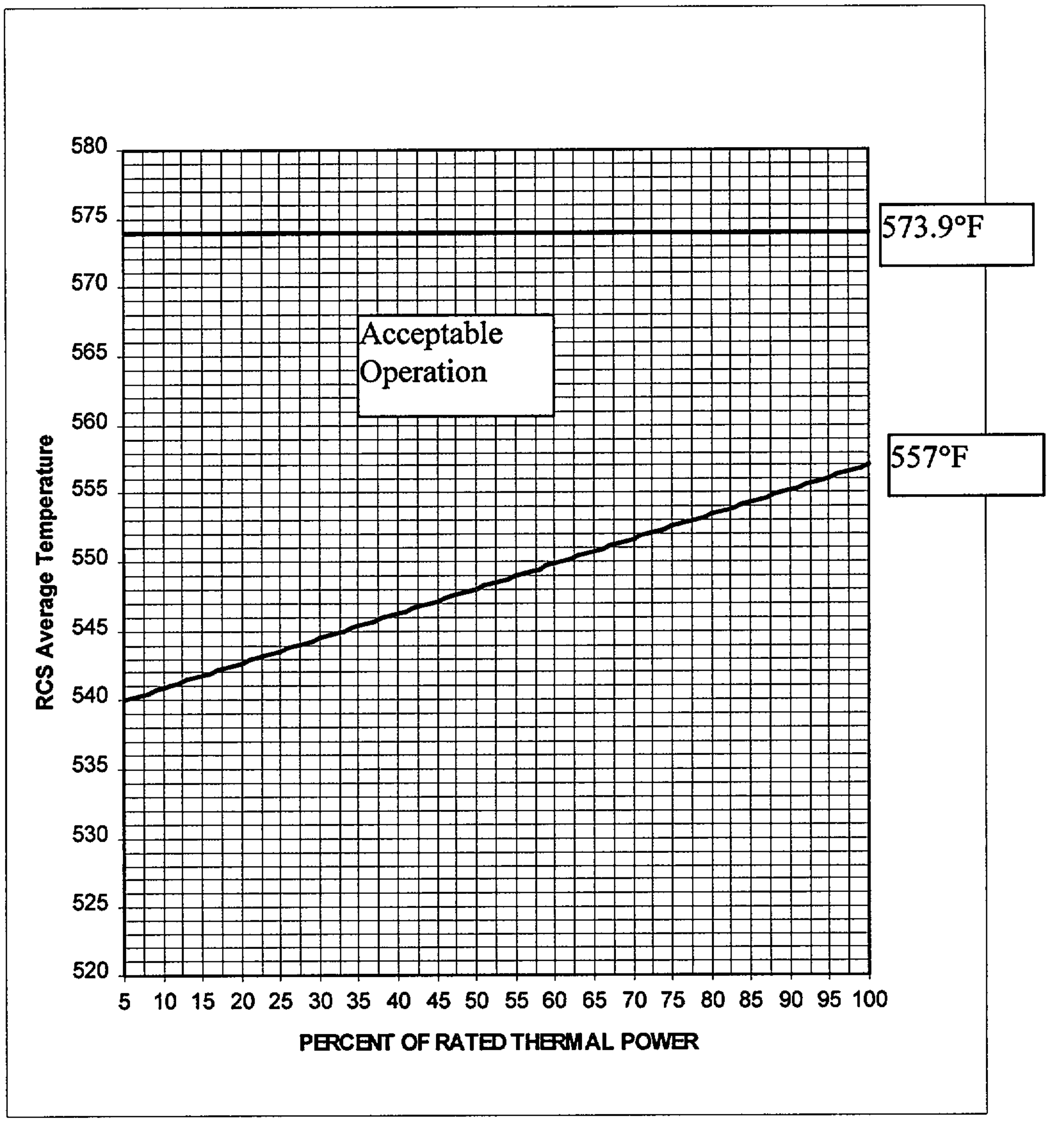

Figure 7b: Reactor Coolant System Average Temperature Limits
(Cores containing 422V+ fuel)

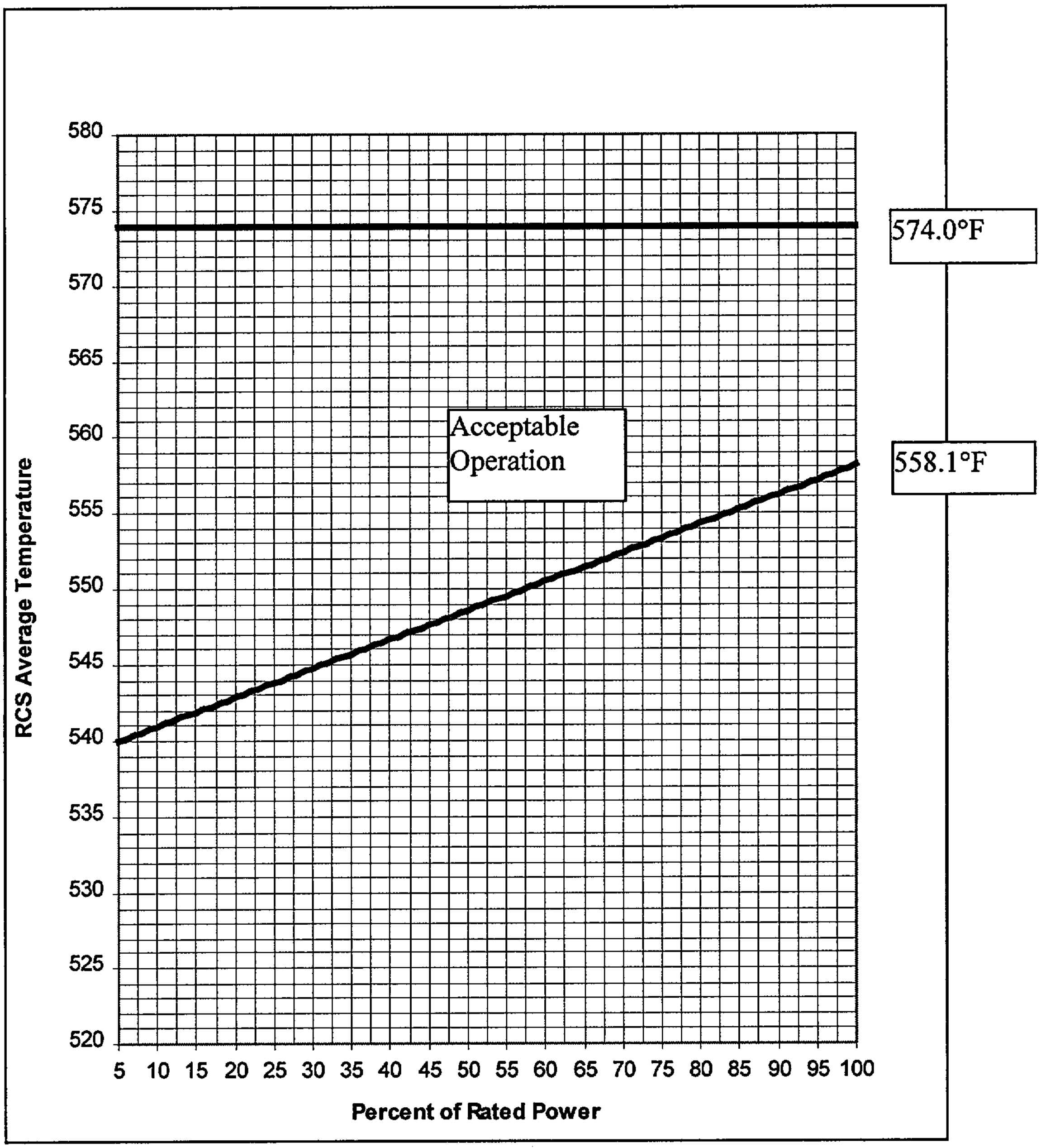

Table 1: NRC Approved Methodologies for COLR Parameters

COLR Section	Parameter	NRC Approved Methodology
2.1	Reactor Core Safety Limits	WCAP-9272-P-A, "Westinghouse Reload Safety Evaluation Methodology," July 1985
2.2	Shutdown Margin	WCAP-9272-P-A, "Westinghouse Reload Safety Evaluation Methodology," July 1985
2.3	Moderator Temperature Coefficient	WCAP-9272-P-A, "Westinghouse Reload Safety Evaluation Methodology," July 1985
2.4	Shutdown Bank Insertion Limit	WCAP-9272-P-A, "Westinghouse Reload Safety Evaluation Methodology," July 1985
2.5	Control Bank Insertion Limits	WCAP-9272-P-A, "Westinghouse Reload Safety Evaluation Methodology," July 1985
2.6	Height Dependent Heat Flux Hot Channel Factor (F_Q)	WCAP-10216-P-A, Revision 1A, "Relaxation Of Constant Axial Offset Control F_Q Surveillance Technical Specification," February 1994. WCAP-14449-P-A, "Application Of Best Estimate Large Break LOCA Methodology To Westinghouse PWRs With Upper Plenum Injection," Revision 1, October 1999. (cores containing 422V+ fuel) WCAP-10924-P-A, "Large Break LOCA Best Estimate Methodology, Volume 2: Application to Two-Loop PWRs Equipped with Upper Plenum Injection," and Addenda, December 1988. (cores not containing 422 V+ fuel) WCAP-10924-P-A, "LBLOCA Best Estimate Methodology: Model Description and Validation: Model Revisions," Volume 1, Addendum 4, August 1990. (cores not containing 422 V+ fuel)

COLR Section	Parameter	NRC Approved Methodology
		WCAP-10054-P-A, "Westinghouse Small Break ECCS Evaluation Model Using The NOTRUMP Code," August 1985. WCAP-10054-P-A, Addendum 2, Revision 1, "Addendum to the Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code: Safety Injection into the Broken Loop and COSI Condensation Model," July 1997.
2.7	Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$)	WCAP-9272-P-A, "Westinghouse Reload Safety Evaluation Methodology," July 1985
2.8	Axial Flux Difference	WCAP-10216-P-A, Revision 1A, "Relaxation Of Constant Axial Offset Control F_Q Surveillance Technical Specification," February 1994.
2.9	Overtemperature ΔT Setpoint	WCAP-8745-P-A, "Design Bases For The Thermal Overpower ΔT and Thermal Overtemperature ΔT Trip Functions," September 1986
2.10	Overpower ΔT Setpoint	WCAP-8745-P-A, "Design Bases For The Thermal Overpower ΔT and Thermal Overtemperature ΔT Trip Functions," September 1986
2.11	RCS Pressure, Temperature, and Flow Departure From Nucleate Boiling (DNB) Limits	WCAP-11397-P-A, "Revised Thermal Design Procedure," April 1989, for those events analyzed using RTDP. WCAP-14787-P, "Westinghouse Revised Thermal Design Procedures Instrument Uncertainty Methodology, Wisconsin Electric Power Company, Point Beach Unit 1 and 2," April 1999. WCAP 9272-P-A, "Westinghouse Reload Safety Evaluation Methodology," for those events not utilizing RTDP.
2.12	Refueling Boron Concentration	WCAP-9272-P-A, "Westinghouse Reload Safety Evaluation Methodology," July

COLR Section	Parameter	NRC Approved Methodology
		1985