Nuclear Operating Company
South Texas Project Electric Generating Station P.O. Box 289 Wadsworth, Texas 77483

August 30, 2005
NOC-AE-05001921
10 CFR 50.90

U. S. Nuclear Regulatory Commission
Attention: Document Control Desk
One White Flint North
11555 Rockville Pike
Rockville, MD 20852

South Texas Project
Units 1 and 2
Docket No. STN 50-498 and STN 50-499
License Amendment Request -
Proposed Amendment to Technical Specifications
to Incorporate Best Estimate Analyzer for Core Operations - Nuclear

STP Nuclear Operating Company (STPNOC) proposes to amend South Texas Project Operating Licenses NPF-76 and NPF-80 by incorporating the attached changes into the Technical Specifications. The proposed changes incorporate the Westinghouse Best Estimate Analyzer for Core Operations - Nuclear (BEACON) power distribution monitoring system (PDMS).

The BEACON PDMS utilizes an NRC-approved Westinghouse proprietary computer code and plant data fed to the plant process computer from the incore thermocouples and excore nuclear instruments. BEACON serves as a three-dimensional core monitor, operational analysis tool, and operational support package. The NRC review of Westinghouse Topical Report WCAP-12472-P concluded that BEACON provides a greatly improved continuous online power distribution measurement and operation prediction information system for Westinghouse reactors.

Attachment 1 to this letter provides the No Significant Hazards Determination and Attachment 2 provides the TS pages marked with the proposed changes. The associated changes to the TS Bases in Attachment 3 (for information only) will be made as a part of the implementation of this license amendment. Attachment 4 provides a sample Core Operating Limits Report revision for information only.

The Plant Operations Review Committee has recommended approval of the proposed change. STPNOC has notified the State of Texas in accordance with 10 CFR 50.91(b).

There are no commitments in this letter.

STPNOC requests approval of the proposed amendment by January 30, 2006. Once approved, the amendment shall be implemented within 90 days.

A001

STI: 31911842

If there are any questions regarding this proposed amendment, please contact Jim Morris at (361) 972-8652 or me at (361) 972-7902.

I declare under penalty of perjury that the foregoing is true and correct.

Executed on August 30, 2005

T. J. Jordan
Vice President, Engineering

jrm/

Attachments:

1. Licensee's Evaluation
2. Proposed Technical Specification Changes (Mark-up)
3. Proposed Changes to the Technical Specification Bases (For Information Only)
4. Sample Core Operating Limits Report (For Information Only)

cc:

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Attachment 1

Licensee's Evaluation

LICENSEE'S EVALUATION

1.0 DESCRIPTION

This letter is a request to amend Operating Licenses NPF-76 and NPF-80 for South Texas Project (STP) Units 1 and 2. The proposed changes would revise the Technical Specifications (TS) to allow use of a power distribution monitoring system (PDMS).

The PDMS utilizes an NRC-approved Westinghouse proprietary computer code, the Best Estimate Analyzer for Core Operations – Nuclear (BEACON), and plant data fed to the plant process computer from the incore thermocouples and excore nuclear instruments. BEACON serves as a three-dimensional core monitor, operational analysis tool, and operational support package.

Westinghouse submitted the BEACON topical report (Reference 1) to the NRC for review and the NRC issued a Safety Evaluation Report approving the topical report on February 16, 1994. The NRC concluded that BEACON provides a greatly improved continuous online power distribution measurement and operation prediction information system for Westinghouse reactors.

STP Nuclear Operating Company (STPNOC) proposes to use BEACON to augment the functional capability of the flux mapping system for the purpose of power distribution surveillances. WCAP-12472-P-A discusses an application of BEACON in which the TS and core power distribution limits are changed to take credit for continuous monitoring by plant operators. STPNOC proposes to use a more conservative application of BEACON in which the core power distribution limits are unchanged. This limited application of BEACON is referred to as the BEACON Technical Specification Monitor (TSM). STPNOC intends to use the BEACON PDMS as the primary method for power distribution measurements and the flux mapping system, if required, when thermal power is greater than 25% rated thermal power (RTP). At thermal power levels less than or equal to 25% RTP, or when the PDMS is inoperable, the movable incore detector system will be used.

The TS affected by implementation of BEACON are:

3.1.3.1	Movable Control Assemblies - Group Height
3.1.3.2	Position Indicating Systems - Operating
3/4.2.2	Heat Flux Hot Channel Factor - $F_Q(Z)$
3/4.2.3	Nuclear Enthalpy Rise Hot Channel Factor
4.2.4.2	Quadrant Power Tilt Ratio
6.9.1.6a	Core Operating Limits Report

In addition, corresponding sections will be added to the Core Operating Limits Report (COLR) for each unit. These sections will define the equations and constants used to determine the applicable measurement uncertainties applied to the core peaking factors when determined by either the PDMS or the flux mapping system. The constants found in this section of the COLR are used as coefficients in the uncertainty calculations and are determined using the methodology approved by the NRC in its review of the Westinghouse BEACON topical report. The constants may be revised periodically as appropriate to reflect cycle-specific variables.

Movable incore detectors will continue to be used for the monthly flux map surveillances prior to issuance and implementation of the license amendment.

In addition to the BEACON changes, STPNOC proposes three administrative changes correcting editorial errors:

3.4.1.4.2c	Delete references to LCO 3.1.1. and LCO 3.1.1.2
4.4.9.3.4	Correct a footnote reference
6.2.2b.	Correct a reference to TS 6.2.2g

STPNOC requests approval of the proposed amendment by January 30, 2006. Once approved, the amendment shall be implemented within 90 days.

2.0 PROPOSED CHANGES

The changes proposed in this license amendment request are summarized below:

Page #	Section	Proposed Change	Reason for Change
3/4 1-17	3.1.3.1 Action b.3.c	Change "...power distribution map is obtained from the movable incore detectors..." to read "...core power distribution measurement is obtained..."	Above 25% RTP, the PDMS may be used to obtain power distribution information instead of the movable incore detector system (MIDS). The generic phrase is appropriate for either method.
3/4 1-19	3.1.3.2 Action a.1	Add "... or a core power distribution measurement..."	This will allow use of either MIDS or PDMS to determine the position of non-indicating rod(s). The use of PDMS is consistent with WCAP-12472-P-A.

Page #	Section	Proposed Change	Reason for Change
3/4 2-5	3.2.2 Action b	Change "…incore mapping…" to read "…core power distribution measurement…"	Above 25% RTP, the PDMS may be used to obtain power distribution information instead of the MIDS. The generic phrase is appropriate for either method.
3/4 2-7	4.2.2.2a.	Replace directions to use the moveable incore detectors with directions to obtain a core power distribution measurement for evaluating F_{xy}.	The phrase "core power distribution measurements" is appropriate for the use of either the PDMS or MIDS.
3/4 2-7	4.2.2.2b.	Replace instructions specifying corrections to be applied to the measured F_{xy} to account for manufacturing tolerances and measurement uncertainties with instructions refering to those parameters specified in the COLR.	The specific corrections are being moved to the COLR consistent with Westinghouse Beacon PDMS and incore measurement methods.
3/4 2-7	4.2.2.2d.1	Change wording from "power distribution maps" to "core power distribution measurements".	The phrase "core power distribution measurements" is appropriate for the use of either the PDMS or MIDS.
3/4 2-8	4.2.2.2d.2	Change wording from "power distribution maps" to "core power distribution measurements".	The phrase "core power distribution measurements" is appropriate for the use of either the PDMS or MIDS.
3/4 2-8	4.2.2.3	Change wording from "power distribution maps" to "core power distribution measurements". Revise wording to refer to COLR for applicable manufacturing tolerances and measurement uncertainty values.	The phrase "core power distribution measurements" is appropriate for the use of either the PDMS or MIDS. The revision of the wording regarding uncertainties is a result of the addition of the requirement to use the PDMS for power distribution measurements above 25% and the PDMS-related uncertainties that are different than when the MIDS is used. The change is generic to either PDMS- or a MIDS-generated flux map.

Page #	Section	Proposed Change	Reason for Change
3/4 2-9	3.2.3 Action b.	Change wording from "incore mapping" to "core power distribution measurement."	The phrase "power distribution measurement" is appropriate for the use of either the PDMS or MIDS.
3/4 2-9	4.2.3.2a.	Replace directions to use the moveable incore detectors with directions to obtain a core power distribution measurement.	The phrase "power distribution measurement" is appropriate for the use of either the PDMS or MIDS.
3/4 2-9	4.2.3.2b.	Change wording to "Increasing the measured $F^N_{\Delta H}$ by the applicable measurement uncertainty as specified in the COLR."	$F^N_{\Delta H}$ measurement uncertainty will now be applied to the measured $F^N_{\Delta H}$ for both PDMS and flux map measurements. The applicable measurement uncertainty will be specified in the COLR. This is consistent with Westinghouse Beacon PDMS and incore measurement methods.
3/4 2-10	4.2.4.2	Revise section to replace reference to movable detector with core power distribution measurement. Revise subsections a. and b. to allow either PDMS or incore measurements for quadrant power tilt ratio (QPTR) surveillance.	Above 25% RTP, the PDMS will be the preferred method for obtaining QPTR information when one power range channel is inoperable.
6-15	6.9.1.6a.8	Add heat flux hot channel factor measurement and manufacturing uncertainties to COLR requirements.	The PDMS uncertainties are dynamically calculated per the methods described in the Topical Report. Information about this calculation and MIDS uncertainties are relocated to the COLR.
6-15	6.9.1.6a.9	Add enthalpy rise hot channel factor measurement uncertainty to COLR requirements.	The PDMS dynamically calculates the $F^N_{\Delta H}$ measurement uncertainty per the topical report methods. Information about this calculation and MIDS uncertainty are relocated to the COLR.

Page #	Section	Proposed Change	Reason for Change
6-16	6.9.1.6b.11	Add WCAP-12472-P-A as reference that documents the analytical methods used to determine the core operating limits.	The addition documents the methodology of the NRC-approved PDMS.
3/4 4-6	3.4.1.4.2c	Change references to LCO 3.1.1 and LCO 3.1.1.2 to reference the COLR	**Editorial** - Shutdown margin limits and boron concentrations were moved to the COLR, but references to the old LCOs were inadvertently not changed.
3/4 4-38	4.4.9.3.4	Change footnote "2" in the second line to read footnote "5"	**Editorial** - The page was marked up correctly in the application, but the page was retyped incorrectly.
6-2	6.2.2b	Change reference to "6.2.2g" to read "6.2.2f"	**Editorial** - The reference was incorrect in the initial license amendment request.

The PDMS instrumentation itself does not meet the selection criteria set forth in 10 CFR 50.36(c)(2)(ii) for inclusion in the TS. Therefore, the PDMS instrumentation requirement will be contained in the STP Technical Requirements Manual (TRM). The TRM contains plant-specific administrative controls (similar to those provided by the TS) over plant equipment not required to be in the TS. Changes to the TRM are made in accordance with the provisions of 10 CFR 50.59.

The justification for not including PDMS instrumentation in the TS is outlined below. The purpose of this evaluation is to demonstrate that the structures, systems, and components associated with PDMS instrumentation are not required to be included in the TS. This evaluation is done in accordance with the requirements contained in 10 CFR 50.36(c)(2)(ii).

A TS Limiting Condition for Operation must be established for each item meeting one or more of the following criteria:

(A) Installed Instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.

PDMS instrumentation is not associated with monitoring of any aspect of the reactor coolant pressure boundary.

(B) A process variable, design feature, or operating restriction that is an initial condition of a design basis accident (DBA) or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

The limits for the power distribution parameters $F_Q(Z)$ and $F^N_{\Delta H}$ are operating restrictions, which ensure that all analyzed DBAs remain valid. These limits are included in the TS. The PDMS instrumentation, however, provides the capability to monitor these parameters more frequently than is currently required by the TS. Additionally, these limits can be determined independently of the operability of PDMS. Therefore, the PDMS instrumentation is not a process variable, design feature, or operating restriction that is an initial condition of a DBA or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

(C) A structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a DBA or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

PDMS instrumentation provides the capability to monitor core power distribution parameters more frequently than is currently required by TS. PDMS instrumentation does not change any of the key safety parameter limits or levels of margin as considered in the reference design basis evaluations. The PDMS instrumentation has no functions or actuations that mitigate any DBA or transient analysis that either assumes the failure of, or presents the challenge to the integrity of a fission product barrier.

(D) A structure, system, or component which operating experience or probabilistic risk assessment has shown to be significant to public health and safety.

PDMS instrumentation provides the capability to monitor power distribution parameters more frequently than is currently required by TS. PDMS instrumentation is used to monitor the core power distribution and has no impact on the results or consequences of any DBA or transient analysis. Therefore, it has no impact on public health and safety.

This evaluation shows that PDMS instrumentation does not meet any of the criteria for inclusion in the TS. The Technical Requirement for PDMS operability will reflect the minimum requirements presented in Reference 1, except for changes due to STPNOC's use of the BEACON TSM, according to vendor instructions.

In summary, the proposed amendment would allow the use of the Westinghouse proprietary three-dimensional nodal code BEACON for performing power distribution surveillances provided that the PDMS instrumentation is operable. This amendment would also allow the use of the MIDS for meeting power distribution surveillances and TS actions, and for calibration of BEACON.

The COLR and TS Bases will be revised to reflect the changes to the affected TS. The revised TS Bases are provided in Attachment 3 and a sample COLR is provided in Attachment 4, both for information only. The COLR changes will be implemented in accordance with TS 6.9.1.6 and the TS Bases changes will be implemented in accordance with TS 6.8.3m as part of the implementation of this amendment.

Editorial Changes

The editorial changes proposed in this license amendment request were part of past license amendment requests that were approved by the NRC, but contained editorial errors either in the marked-up TS pages submitted for review or in the retyped pages that were inserted in the TS.

TS Section 3.4.1.2.c

Reference 2 proposed and Reference 3 approved relocating the shutdown margin limits from TS 3.1.1.1 and 3.1.1.2 to the COLR and modifying certain boration requirements to be consistent with NUREG-1431. However, TS 3.4.1.4.2c referred to Limiting Conditions for Operation 3.1.1 and 3.1.1.2 and modifying TS 3.4.1.2c to refer instead to the COLR was overlooked in preparing Reference 2. Therefore, STPNOC proposes the change as part of this license amendment request.

TS 4.4.9.3.4

Reference 4 proposed and Reference 5 approved revising the cold overpressure mitigation curve to accommodate the replacement steam generators, and adding two surveillances for the centrifugal charging pumps and the emergency core cooling system accumulators to ensure operability of the cold overpressure protection mitigation system. The marked-up TS pages in Reference 4 were correct, but STPNOC made an error when retyping the TS pages for implementation. One footnote in TS 4.4.9.3.4 was incorrectly typed as "2" instead of "5." Therefore, STPNOC request correction of this editorial mistake.

TS 6.2.2.b

Reference 6 proposed and Reference 7 approved revising specific requirements of TS Section 6.0, "Administrative Controls," to be consistent with the Improved Westinghouse Standard Technical Specifications in NUREG-1431. Paragraph 6.2.2b was modified to reflect NUREG-1431 wording regarding shift crew composition. It incorrectly referred to paragraph 6.2.2g as providing further requirements. The reference should have been to paragraph 6.2.2.f. STPNOC requests correction of this editorial mistake.

3.0 BACKGROUND

As described in Reference 1, the Westinghouse BEACON PDMS was developed to improve the operational support for pressurized water reactors (PWRs). BEACON is an advanced core monitoring and support software package that utilizes existing plant instrumentation to provide incore thermocouple temperatures, reactor coolant system cold leg temperatures, control bank demand positions, power range detector output, and reactor power measurement data. These data are sent by the plant computer in the form of a file that BEACON can interpret to perform nodal power distribution prediction calculations.

The PDMS includes an on-line, three-dimensional nodal model that is continuously updated to reflect the current plant operating conditions. The nodal solution method used by the PDMS is consistent with the NRC-approved Westinghouse Advanced Nodal Code (ANC) core design code. The core-exit thermocouple and excore neutron flux detector readings are used with the reference three-dimensional power distribution to determine the measured power distribution. By coupling the measured three-dimensional power distribution with an on-line evaluation, actual core margins can be better understood. The PDMS provides an understanding of operating and design margins to address strategic fuel cycle changes. The BEACON methodology improves the quality of the surveillance process since it uses a depleted model to match the actual operational profile. The PDMS continuously monitors the limiting $F_Q(Z)$ and $F^N_{\Delta H}$.

The MIDS is available for core power distribution analysis if BEACON data becomes unavailable. The MIDS will also be used to calibrate BEACON.

STPNOC intends to utilize the BEACON PDMS to take advantage of the capability for continuous monitoring of the limiting core thermal peaking factors, $F_Q(Z)$ and $F^N_{\Delta H}$, without the need to obtain a full-core flux map. The BEACON PDMS will allow operational support for TS compliance, and the continuous monitoring feature will permit instantaneous identification of core anomalies and predictive capabilities for both operators and reactor engineers.

4.0 TECHNICAL ANALYSIS

The proposed changes allow STPNOC to use the BEACON PDMS as an alternate means to perform core power distribution surveillances when operating at greater than 25% RTP. The PDMS maintains an on-line three-dimensional nodal model that is continuously updated to reflect the current plant operating conditions. The following is a summary/excerpt of Brookhaven National Laboratory's Technical Evaluation Report (TER) for WCAP-12472-P-A.

4.1 BEACON Core Monitoring Methodology (TER 2.1)

The BEACON core monitoring system uses the NRC-approved Westinghouse SPNOVA nodal method for core power distribution measurements. The SPNOVA data libraries and core models are consistent with the NRC-approved Westinghouse PHOENIX/ANC design models and have been benchmarked against operating reactor measurements.

The BEACON core monitoring process is carried out in three steps. In the first step, the SPNOVA model, individual thermocouples, and the excore axial offset are calibrated to the full-core incore flux measurement. In the second step, the SPNOVA model is updated based on the most recent operating history, and adjusted using the thermocouple and excore measurements. The continuous monitoring is performed in the third step using the thermocouples and excores to update the BEACON model. (TER 2.1.3)

The BEACON power distribution calculation is updated using thermocouple and excore detector measurements. Thermocouple measurements are interpolated/extrapolated radially using the spline fit. The BEACON system provides both a full three-dimensional nodal power distribution calculation and a simplified, more approximate one-dimensional calculation. The BEACON on-line limits evaluation will be performed in three dimensions and the one-dimensional calculation will only be used as a scoping tool in predictive analysis. (TER 3.1)

The continuous core monitoring of the current reactor statepoint (fuel burnup, xenon distribution, soluble boron concentration, etc.) provided by BEACON allows a more precise determination of the parameters used in the transient analyses, and therefore relaxes the requirement to limit the transient initial conditions via power distribution control. As part of the continuous monitoring, the fuel limits are calculated using the standard Westinghouse methods. (TER 2.1.3)

In applying BEACON to STP, STPNOC will not take credit for the continuous monitoring of the power distribution. Instead, STPNOC will use BEACON as a TS monitor of present peaking factor limits and the transient initial condition limits will not be relaxed.

With BEACON operable, the criteria for the core-exit thermocouples require at least 25% of the thermocouples to be operable (at least two per quadrant) with the added requirement that the operable pattern normally covers all internal fuel assemblies within a chess "knight" move (an adjacent plus a diagonal square away) or more frequent calibration is required. Calibration with the MIDS is required every 180 effective full-power days. However, calibration is required every 30 days when the knight's move requirement is not satisfied. The accuracy of the power distribution information with decreased incore instrumentation operability has been analyzed by Westinghouse and penalties are applied to the calculated peaking factors (refer to TER section 2.3). The review concluded that the minimum available incore instrumentation, when coupled with the increased uncertainty penalties, provides reasonable and acceptable power distribution information.

4.2 Model Calibration and Uncertainty (TER 2.1.2)

BEACON uses the incore flux detector measurements, core-exit thermocouples, and excore detectors for local calibration of the SPNOVA three-dimensional power distribution methodology. The SPNOVA-predicted detector reaction rates are normalized to the incore measurements at the incore radial locations and over an axial mesh. The thermocouple adjustment is two-dimensional and is made by normalizing the SPNOVA radial power distribution to the assembly power inferred from the core-exit thermocouples. The thermocouple assembly power measurement is periodically calibrated to the incore-measured assembly power.

The incore detectors and core-exit thermocouples do not provide complete coverage of the core; BEACON employs a two-dimensional spline fit to interpolate/extrapolate these measurements to the unmonitored assemblies. The spline fit includes a tolerance factor which controls the degree to which the fit is forced to match the individual measurements. If, for example, the measurements are believed to be extremely accurate (inaccurate) a low (high) tolerance factor is used and the SPNOVA solution is (not) forced to be in exact agreement with the measurements.

The BEACON axial power shape is adjusted to ensure agreement with the axial offset measured by the excore detectors. This adjustment is made by adding a sinusoidal component to the SPNOVA-calculated axial power shape. The SPNOVA excore axial offset is determined by an appropriate weighting of the peripheral assembly powers. The excore detector axial offset is periodically calibrated to the incore detector measurement.

As an initial assessment of the power distribution calculation, Westinghouse performed detailed comparisons of BEACON results to the predictions of the INCORE system presently used at Westinghouse plants. INCORE is a data analysis code written to process information obtained by the MIDS in Westinghouse PWRs. These comparisons were made for three plants over four cycles, and included a range of fuel burnup, core loadings, power level, and control rod insertion. From the results of this study, which includes a comparison of the standard deviations between predicted and measured assembly-wide reaction rates and a comparison of inferred assembly power between BEACON and INCORE, Westinghouse concluded that the BEACON processing of the incore flux map and the inferred assembly power distribution accuracy is statistically consistent with the INCORE computer code. (Refer to Section 4.1.1 and Table 4-6 of WCAP-12472-P-A.)

The uncertainties applied to the BEACON power distribution measurements are different from those applied to the traditional flux map systems because BEACON uses a more comprehensive set of instrumentation. An uncertainty analysis of the BEACON power distribution measurement is reported in WCAP-12472-P-A. Portions of the TER for WCAP-12472-P-A relevant to the uncertainty analysis are summarized/excerpted as follows:

Model Calibration (TER 2.3)
Due to the change in reactor statepoint, SPNOVA modeling approximations and instrumentation error, a model calibration uncertainty is introduced into the BEACON predictions. Westinghouse evaluated this uncertainty by comparing BEACON predicted and measured incore reaction rates over four cycles and a range of operating conditions, and found that the model calibration uncertainty was very small and varied only slightly for these comparisons.

Thermocouple Calibration (TER 2.3.2)
The thermocouple calibration uncertainty is due to the change in reactor statepoint and to instrument error. Westinghouse has evaluated this uncertainty by comparing the assembly powers inferred from the thermocouples to SPNOVA incore-corrected assembly powers. Comparisons for three plants and a range of operating conditions indicate a difference of less than a few percent at full power. The observed calibration uncertainty increased at lower powers due to the reduced enthalpy rise and changes in cross-flow.

Axial Power Distribution Uncertainty (TER 2.3.3)
In order to determine the axial power distribution uncertainty, Westinghouse has compared SPNOVA incore-updated and SPNOVA excore-updated predictions of the axial power shape. These comparisons included a range of fuel burnups and rod insertions, and

indicated a 95/95 upper tolerance limit of less than a few percent with a slight dependence on rod movement since calibration.

Calibration Interval (TER 2.3.4)
Based on an extensive set of calibration data, the model calibration uncertainty is observed to increase as the calibration interval (in units of fuel burnup) increases. Using the observed fuel burnup dependence, an additional assembly power uncertainty is determined to account for the effects of increased calibration interval.

Inoperable Detectors (TER 2.3.5)
The failure of incore detector thimbles and incore thermocouples used by the BEACON system results in a relaxation of the local calibration to measurement and an increase in the power distribution uncertainty. The effect of random failures of the incore detector thimbles and incore thermocouples on the assembly power was evaluated for failure rates of up to 75%. The assembly power uncertainty was found to increase linearly with incore detector failure and quadratically with the failure of thermocouples.

Local Power Distribution Uncertainty (TER 2.3.6)
The BEACON calculation requires local power distribution factors for: (1) the ratio of assembly power-to-detector response; (2) assembly local peaking factor; and (3) the grid power-depression factor (correction factor to the assembly axial power distribution to take the power depression, due to the grid of the assembly, into account). The BEACON uncertainty analysis employs previously approved upper tolerance values for the assembly power-to-detector response ratio and the local peaking factor. The grid (power depression) factor uncertainty was determined by comparison to measured flux traces and is found to be relatively small.

Determination of BEACON Power Peaking Uncertainty (TER 2.4)
The uncertainty in the BEACON power peak resulting from errors in the SPNOVA model calibration and thermocouple calibration is determined using an analog Monte Carlo error propagation technique. In this analysis, the BEACON three-step calibration model update and power distribution update procedure is simulated. The SPNOVA model and thermocouple calibration factors are subjected to random variations (based on their uncertainties) and the resulting variations in the BEACON power distribution are used to determine the 95% probability upper tolerance limit on the assembly power for the twenty highest powered assemblies.

The analysis is performed for a range of operating conditions including off-normal power distributions and extended calibration intervals. A typical set of thermocouple uncertainties is used together with a relatively large tolerance factor which results in substantial smoothing of the thermocouple measurements. The upper tolerance limit on the assembly power peaking factor is calculated and found to increase as the square-root of the thermocouple uncertainty.

4.3 Acceptance Criteria

In the NRC Safety Evaluation Report on WCAP-12472-P-A, the NRC staff evaluated the BEACON methodology, the uncertainty analysis, and the operation of the overall system, and concluded that the BEACON system is acceptable for performing core monitoring and operations support functions for Westinghouse PWRs subject to certain conditions. These conditions are listed below followed by the STPNOC evaluation of compliance.

1. *In the cycle-specific application of BEACON, the power peaking uncertainties $U_{\Delta H}$ and U_Q must provide 95% probability upper tolerance limits at the 95% confidence level.*

   Evaluation of Compliance:

   Although not specifically described in this submittal, cycle-specific BEACON calibrations performed before startup and at beginning-of-cycle (BOC) will ensure that power peaking uncertainties provide 95% probability upper tolerance limits at the 95% confidence level. These calibrations are performed using the Westinghouse approved methodology. Until these calibrations are complete, more conservative default uncertainties will be applied. The calibrations will be documented and retained as records.

2. *In order to ensure that the assumptions made in the BEACON uncertainty analysis remain valid, the generic uncertainty components may require reevaluation when BEACON is applied to plant or core designs that differ sufficiently to have a significant impact on the WCAP-12472-P database.*

   Evaluation of Compliance:

   South Texas Project Units 1 and 2 have Westinghouse 4-loop Nuclear Steam Supply Systems (NSSS) with Westinghouse movable incore instrumentation. All fuel is presently of Westinghouse manufacture. Therefore, STP does not currently differ significantly from the plants that form the WCAP database and no additional review of WCAP applicability to STP is necessary.

   During review of the Westinghouse topical report WCAP-12472-P, the NRC requested additional information on how BEACON treats core loadings with fuel designs from multiple fuel vendors and the impact to the BEACON uncertainty analysis. Westinghouse responded that for all BEACON applications, the previous operating cycle is reviewed against established reference uncertainties. This examination accounts for loading of fuel supplied by multiple vendors by comparing a BEACON model to actual operating data over the cycle. At BOC, thermocouple data is verified and calibration/uncertainty components are updated as necessary. In addition, the initial flux mapping at BOC ensures model calibration factors reflect the actual fuel in the reactor before the BEACON system is declared operable.

3. *The BEACON Technical Specifications should be revised to include the changes described in Section 3 concerning Specifications 3.1.3.1 and 3.1.3.2 and the Core Operating Limits Report.*

   Evaluation of Compliance:

   This condition does not directly apply to this submittal because the WCAP describes an application of BEACON where the core operating limits are changed and STPNOC proposes to use BEACON as a core TS monitor of the current limits.

5.0 REGULATORY SAFETY ANALYSIS

5.1 No Significant Hazards Consideration

STPNOC has evaluated whether a significant hazards consideration is involved with the proposed amendment by focusing on the three standards set forth in 10 CFR 50.92:

1. Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

   Response: No

   The PDMS performs continuous core power distribution monitoring. This system utilizes the NRC-approved Westinghouse proprietary computer code BEACON to provide data reduction for incore flux maps, core parameter analysis, load follow operation simulation, and core prediction. It in no way provides any protection or control system function. Fission product barriers are not impacted by these proposed changes. The proposed changes occurring with PDMS will not result in any additional challenges to plant equipment that could increase the probability of any previously evaluated accident. The changes associated with the PDMS do not affect plant systems such that their function in the control of radiological consequences is adversely affected. These proposed changes will therefore not affect the mitigation of the radiological consequences of any accident described in the Updated Final Safety Analysis Report Update (UFSAR).

   Continuous on-line monitoring through the use of PDMS provides significantly more information about the power distributions present in the core than is currently available. This results in more time (i.e., earlier determination of an adverse condition developing) for operator action prior to having an adverse condition develop that could lead to an accident condition or to unfavorable initial conditions for an accident.

   Each accident analysis addressed in the UFSAR is examined with respect to changes in cycle-dependent parameters, which are obtained from application of the NRC-approved reload design methodologies, to ensure that the transient evaluations of reload cores are

bounded by previously accepted analyses. This examination, which is performed in accordance with the requirements set forth in 10 CFR 50.59, ensures that future reloads will not involve a significant increase in the probability or consequences of any accident previously evaluated.

The three editorial changes only correct typographical errors made in previously approved TS changes. They do not affect plant operation or structures, systems, and components important to safety.

Therefore, the proposed changes do not involve a significant increase in the probability or consequence of an accident previously evaluated.

2. Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

   Response: No

   The implementation of the PDMS has no influence or impact on plant operations or safety, nor does it contribute in any way to the probability or consequences of an accident. No safety-related equipment, safety function, or plant operation will be altered as a result of this proposed change. The possibility for a new or different type of accident from any accident previously evaluated is not created since the changes associated with implementation of the PDMS do not result in a change to the design basis of any plant component or system. The evaluation of the effects of using the PDMS to monitor core power distribution parameters shows that all design standards and applicable safety criteria limits are met.

   The proposed changes do not result in any event previously deemed incredible being made credible. Implementation of the PDMS will not result in more adverse conditions and will not result in any increase in the challenges to safety systems. The cycle-specific variables required by the PDMS are calculated using NRC-approved methods. The TS will continue to require operation within the required core operating limits and appropriate actions will be taken if limits are exceeded.

   The three editorial changes only correct typographical errors made in previously approved TS changes. They do not affect plant operation or structures, systems, and components important to safety.

   Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any previously evaluated.

3. Does the proposed change involve a significant reduction in a margin of safety?

   Response: No

The margin of safety is not affected by implementation of the PDMS. The margin of safety provided by current TS is unchanged. The proposed changes continue to require operation within the core limits that are based on NRC-approved reload design methodologies. Appropriate measures exist to control the values of these cycle-specific limits. The proposed changes continue to ensure that appropriate actions will be taken if limits are violated. These actions remain unchanged.

The three editorial changes only correct typographical errors made in previously approved TS changes. They do not affect plant operation or structures, systems, and components important to safety.

Therefore, the proposed changes do not involve a significant reduction in a margin of safety.

Based on the above, STPNOC concludes that the proposed amendment regarding the use of the PDMS and BEACON and three proposed editorial corrections involves no significant hazards consideration under the standards set forth in 10CFR50.92, and a finding of "no significant hazards consideration" is justified.

5.2 Applicable Regulatory Requirements/Criteria

10 CFR 50, Appendix A, General Design Criterion 13 states:

Criterion 13 - Instrumentation and control. Instrumentation shall be provided to monitor variables and systems over their anticipated ranges for normal operation, for anticipated operational occurrences, and for accident conditions as appropriate to assure adequate safety, including those variables and systems that can affect the fission process, the integrity of the reactor core, the reactor coolant pressure boundary, and the containment and its associated systems. Appropriate controls shall be provided to maintain these variables and systems within prescribed operating ranges.

Implementation of the PDMS at STP Units 1 and 2 does not replace, eliminate, or modify existing plant instrumentation. The PDMS software runs on a workstation connected to the plant process computer. The PDMS combines inputs from currently installed plant instrumentation and design data generated for each fuel cycle. Together, this provides a means to continuously monitor the power distribution limits including limiting peaking factors and quadrant power tilt ratio.

In conclusion, based on the considerations discussed above, (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

6.0 ENVIRONMENTAL CONSIDERATION

STPNOC has evaluated the proposed changes and determined the changes do not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluents that may be released offsite, or (iii) a significant increase in the individual or cumulative occupational exposure. Accordingly, the proposed changes meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9), and an environmental assessment of the proposed changes is not required.

7.0 REFERENCES AND PRECEDENTS

7.1 References

1. WCAP-12472-P-A, "BEACON - Core Monitoring and Operations Support System," August 1994 (NRC-approved version with Safety Evaluation Report included)

2. Letter, J. J. Sheppard to Document Control Desk, "License Amendment Request - Proposed Revision to Relocate Shutdown Margin Limits from the Technical Specifications to the Core Operating Limits Report and to Modify Action Statements Consistent with NUREG-1431," dated May 23, 2002 (ML021550040)

3. Letter, M. Thadani to W. T. Cottle, "South Texas Project, Units 1 and 2 - Issuance of Amendments on Relocation of Shutdown Margin Limits to Core Operating Limits Report (TAC Nos. MB5155 and MB5156)," dated February 4, 2003 (ML030360230)

4. Letter, J. J. Sheppard to Document Control Desk, "Revised Proposed Amendment to Technical Specification 3.4.9.3 to Reflect Cold Overpressure Mitigation Curve Associated with Replacement Steam Generators," dated August 18, 1999 (NOC-AE-000521)

5. Letter, T. W. Alexion to W. T. Cottle, "South Texas Project, Units 1 and 2 - Issuance of Amendments re: Cold Overpressure Mitigation Curve Associated with Replacement Steam Generators (TAC Nos. MA3519 and MA3520)," dated November 9, 1999 (ML993240276)

6. Letter, J. J. Sheppard to Document Control Desk, "License Amendment Request - Proposed Amendment to South Texas Project Technical Specifications to Revise Administrative Control Requirements," dated November 5, 2001 (ML013510314)

7. Letter, J. L. Minns to J. J. Sheppard, "South Texas Project, Units 1 and 2 - Issuance of Amendments to Revise Specific Requirements of Technical Specification 6.0, 'Administrative Controls,' (TAC Nos. MB3589 and MB3593)," dated April 24, 2003 (ML031140670)

7.2 Precedents

The BEACON Technical Specification Monitor was approved by the NRC for use at the following plants:

Summer, Amendment 142, April 9, 1999 (ML012260068)

Salem, Amendments 237/218, November 6, 2000 (ML003761792)

Diablo Canyon, Amendments 164/166, March 31, 2004 (ML040920245)

This STP license amendment request is consistent with these amendments.

These changes were also approved for:

Byron, Amendments 116 and Braidwood, Amendments 110, February 13, 2001 (ML010510325).

The application at Byron/Braidwood uses BEACON to take credit for the direct and continuous monitoring of departure from nuclear boiling ratio, whereas the application of BEACON at STP is only for power distribution surveillances.

Attachment 2

Proposed Technical Specification Changes (Mark-up)

No Changes This Page

3/4.1.3 MOVABLE CONTROL ASSEMBLIES

GROUP HEIGHT

LIMITING CONDITION FOR OPERATION

3.1.3.1 All full-length shutdown and control rods shall be OPERABLE and positioned within ± 12 steps (indicated position) of their group step counter demand position.

APPLICABILITY: MODES 1* and 2*.

ACTION:

a. With one or more full-length rods inoperable due to being immovable as a result of excessive friction or mechanical interference or known to be untrippable, determine that the SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is satisfied within 1 hour and be in HOT STANDBY within 6 hours.

b. With one full-length rod trippable but inoperable due to causes other than addressed by ACTION a., above, or misaligned from its group step counter demand height by more than ± 12 steps (indicated position), POWER OPERATION may continue provided that within 1 hour:

   1. The rod is restored to OPERABLE status within the above alignment requirements, or

   2. The rod is declared inoperable and the remainder of the rods in the group with the inoperable rod are aligned to within ± 12 steps of the inoperable rod while maintaining the rod sequence and insertion limits as specified in the Core Operating Limits Report (COLR). The THERMAL POWER level shall be restricted pursuant to Specification 3.1.3.6 during subsequent operation, or

   3. The rod is declared inoperable and the SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is satisfied. POWER OPERATION may then continue provided that:

      a) A reevaluation of each accident analysis of Table 3.1-1 is performed within 5 days; this reevaluation shall confirm that the previously analyzed results of these accidents remain valid for the duration of operation under these conditions;

      b) The SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is determined at least once per 12 hours;

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*See Special Test Exceptions Specifications 3.10.2 and 3.10.3.

ACTION (Continued)

c) A core power distribution measurement ~~map~~ is obtained ~~from the movable incore detectors~~ and $F_Q(Z)$ and $F_{\Delta H}^N$ are verified to be within their limits within 72 hours; and

d) The THERMAL POWER level is reduced to less than or equal to 75% of RATED THERMAL POWER within the next hour and within the following 4 hours the High Neutron Flux Trip Setpoint is reduced to less than or equal to 85% of RATED THERMAL POWER.

c. With more than one rod trippable but inoperable due to causes other than addressed by ACTION a. above, POWER OPERATION may continue provided that:

1. Within 1 hour, the remainder of the rods in the bank(s) with the inoperable rods are aligned to within ± 12 steps of the inoperable rods while maintaining the rod sequence and insertion limits as specified in the COLR. The THERMAL POWER level shall be restricted pursuant to Specification 3.1.3.6 during subsequent operation, and

2. The inoperable rods are restored to OPERABLE status within 72 hours.

d. With more than one rod misaligned from its group step counter demand height by more than ± 12 steps (indicated position), be in HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

4.1.3.1.1 The position of each full-length rod shall be determined to be within the group demand limit by verifying the individual rod positions at least once per 12 hours except during time intervals when the rod position deviation monitor is inoperable, then verify the group positions at least once per 4 hours.

4.1.3.1.2 Each full-length rod not fully inserted in the core shall be determined to be OPERABLE by movement of at least 10 steps in any one direction at least once per 31 days.

POSITION INDICATION SYSTEMS - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.3.2 The Digital Rod Position Indication System and the Demand Position Indication System shall be OPERABLE and capable of determining the control rod positions within ± 12 steps.

APPLICABILITY: MODES 1 and 2.

ACTION:

a. With a maximum of one digital rod position indicator per bank inoperable either:

   1. Determine the position of the nonindicating rod(s) indirectly by the movable incore detectors **or a core power distribution measurement** at least once per 8 hours and immediately after any motion of the nonindicating rod which exceeds 24 steps in one direction since the last determination of the rod's position, or

   2. Reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within 8 hours.

b. With a maximum of one demand position indicator per bank inoperable either:

   1. Verify that all digital rod position indicators for the affected bank are OPERABLE and that the most withdrawn rod and the least withdrawn rod of the bank are within a maximum of 12 steps of each other at least once per 8 hours, or

   2. Reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within 8 hours.

SURVEILLANCE REQUIREMENTS

4.1.3.2 Each digital rod position indicator shall be determined to be OPERABLE by verifying that the Demand Position Indication System and the Digital Rod Position Indication System agree within 12 steps at least once per 12 hours except during time intervals when the rod position deviation monitor is inoperable, then compare the Demand Position Indication System and the Digital Rod Position Indication System at least once per 4 hours.

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:

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

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

Where: F_Q^{RTP} = the F_Q limit at RATED THERMAL POWER (RTP) *specified in the Core Operating Limits Report (COLR).*

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

K(Z) = the normalized $F_Q(Z)$ as a function of core height specified in the COLR.

APPLICABILITY: MODE 1.

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 Setpoint 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 Setpoint has been reduced at least 1% for each 1% $F_Q(Z)$ exceeds the limit.

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~~ **core power distribution measurement** to be within its limit.

POWER DISTRIBUTION LIMITS

SURVEILLANCE REQUIREMENTS

4.2.2.1 The provisions of Specification 4.0.4 are not applicable.

4.2.2.2 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~~ **Obtaining a core power distribution measurement**,

b. Increasing the measured F_{xy} **by the applicable manufacturing and measurement uncertainties as specified in the COLR** ~~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 4.2.2.2b., above to:

1) The F_{xy} limits for RATED THERMAL POWER (F_{xy}^{RTP}) for the appropriate measured core planes given in Specification 4.2.2.2e. and f., below, and

2) The relationship:

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

Where F_{xy}^L is the limit for fractional THERMAL POWER operation expressed as a function of F_{xy}^{RTP}, PF_{xy} is the power factor multiplier F_{xy} specified in the COLR, and P is the fraction of RATED THERMAL POWER at which F_{xy} is 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 **core** power distribution **measurements** ~~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% RATED THERMAL POWER or greater, the THERMAL POWER at which F_{xy}^C was last determined, or

b) At least once per 31 Effective Full Power days (EFPD), whichever occurs first.

2) When F_{xy}^{C} is less than or equal to the F_{xy}^{RTP} limit for the appropriate measured core plane, additional **core** power distribution **measurements** ~~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} limits used in the Constant Axial Offset Control analysis for RATED THERMAL POWER (F_{xy}^{RTP}) shall be provided for all core planes containing bank "D" control rods and all unrodded core planes as specified in the COLR per Specification 6.9.1.6;

f. The F_{xy} limits of Specification 4.2.2.2e, above, are not applicable in the following core planes 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 22.4 $\pm$ 2%, 34.2 $\pm$ 2%, 46.0 $\pm$ 2%, 57.8 $\pm$ 2%, 69.5 $\pm$ 2% and 81.3 $\pm$ 2%, inclusive, and

4) Core plane regions within $\pm$ 2% of core height ($\pm$ 3.36 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.

4.2.2.3 When $F_Q(Z)$ is measured for other than F_{xy} determinations, an overall measured $F_Q(Z)$ shall be obtained from a **core** power distribution **measurement** ~~map~~ and increased by **the applicable manufacturing and measurement uncertainites as specified in the COLR** ~~3% to account for manufacturing tolerances and further increased by 5% to account for measurement uncertainty~~.

3/4.2.3 NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR

LIMITING CONDITION FOR OPERATION

3.2.3 $F^N_{\Delta H}$ shall be less than $F^{RTP}_{\Delta H}$ [1.0 +$PF_{\Delta H}$ (1-P)]

Where: $F^{RTP}_{\Delta H}$ = the $F^N_{\Delta H}$ Limit at RATED THERMAL POWER (RTP) specified in the Core Operating Limits Report (COLR).

$PF_{\Delta H}$ = the Power Factor Multiplier for $F^N_{\Delta H}$ specified in the Core Operating Limits Report (COLR).

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

APPLICABILITY: MODE 1.

ACTION:

With $F^N_{\Delta H}$ exceeding its limit:

a. Within 2 hours reduce the THERMAL POWER to the level where the LIMITING CONDITION FOR OPERATION is satisfied.

b. Identify and correct the cause of the out-of-limit condition prior to increasing THERMAL POWER above the limit required by ACTION a., above; THERMAL POWER may then be increased, provided $F^N_{\Delta H}$ is demonstrated through ~~incore mapping~~ **core power distribution measurement** to be within its limits.

SURVEILLANCE REQUIREMENTS

4.2.3.1 The provisions of Specification 4.0.4 are not applicable.

4.2.3.2 $F^N_{\Delta H}$ shall be demonstrated to be within its limit prior to operation above 75% RATED THERMAL POWER after each fueling loading and at least once per 31 EFPD thereafter by:

a. **Obtaining a core power distribution measurement** ~~Using the movable incore detectors to obtain a power distribution map at any THERMAL POWER greater than 5% RATED THERMAL POWER~~.

b. **Increasing** ~~Using~~ the measured value of $F^N_{\Delta H}$ **by the applicable** ~~which does not include an allowance for~~ measurement uncertainty **as specified in the COLR.**

3/4.2.4 QUADRANT POWER TILT RATIO

LIMITING CONDITION FOR OPERATION

3.2.4 The QUADRANT POWER TILT RATIO shall not exceed 1.02.

APPLICABILITY: MODE 1, above 50% of RATED THERMAL POWER*.

ACTION:

With the QUADRANT POWER TILT RATIO determined to exceed 1.02:

a. Within 2 hours reduce THERMAL POWER at least 3% from RATED THERMAL POWER for each 1% of indicated QUADRANT POWER TILT RATIO in excess of 1 and similarly reduce the Power Range Neutron Flux-High Trip Setpoint within the next 4 hours.

b. Within 24 hours and every 7 days thereafter, verify that $F_Q(Z)$ (by F_{xy} evaluation) and $F^N_{\Delta H}$ are within their limits by performing Surveillance Requirements 4.2.2.2 and 4.2.3.2. THERMAL POWER and setpoint reductions shall then be in accordance with the ACTION statements of Specifications 3.2.2 and 3.2.3.

SURVEILLANCE REQUIREMENTS

4.2.4.1 The QUADRANT POWER TILT RATIO shall be determined to be within the limit above 50% of RATED THERMAL POWER by:

a. Calculating the ratio at least once per 7 days when the alarm is OPERABLE, and

b. Calculating the ratio at least once per 12 hours during steady-state operation when the alarm is inoperable.

4.2.4.2 The QUADRANT POWER TILT RATIO shall be determined to be within the limit when above 75% of RATED THERMAL POWER with one Power Range channel inoperable by **measuring core power distribution** ~~using the movable incore detectors~~ to confirm indicated QUADRANT POWER TILT RATIO at least once per 12 hours by ~~either~~ **using**:

a. **The Power Distribution Monitoring System (PDMS), or**

b. **The movable incore detectors by either:**

1.~~a.~~ Using the four pairs of symmetric thimble locations, or

2.~~b.~~ Using the movable incore detection system to monitor the QUADRANT POWER TILT RATIO with a full incore map.

* See Special Test Exceptions Specification 3.10.2.

REACTOR COOLANT SYSTEM

COLD SHUTDOWN - LOOPS NOT FILLED

LIMITING CONDITION FOR OPERATION

3.4.1.4.2

a. At least two residual heat removal (RHR) loops shall be OPERABLE* and at least one RHR loop shall be in operation**, and

b. Each valve or mechanical joint used to isolate unborated water sources shall be secured in the closed position.

APPLICABILITY: MODE 5 with reactor coolant loops not filled.

ACTION:

a. With less than the above required RHR loops OPERABLE, immediately initiate corrective action to return the required RHR loops to OPERABLE status as soon as possible.

b. With no RHR loop in operation, suspend all operations that would cause introduction into the RCS of coolant with boron concentration less than required to meet SHUTDOWN MARGIN of LCO 3.1.1 and immediately initiate corrective action to return the required RHR loop to operation.

c. With a valve or mechanical joint used to isolate unborated water sources not secured in the closed position, immediately suspend all operations that would cause introduction into the RCS of coolant with boron concentration less than required to meet SHUTDOWN MARGIN ~~of LCO 3.1.1~~ **specified in the Core Operating Limits Report (COLR)** and initiate action to secure the valve(s) or joint(s) in the closed position and within 4 hours verify boron concentration is within limits specified in ~~LCO 3.1.1.2.~~ **the COLR**. The required action to verify the boron concentration within limits must be completed whenever ACTION c is entered. A separate ACTION entry is allowed for each unsecured valve or mechanical joint.

SURVEILLANCE REQUIREMENTS

4.4.1.4.2.1 At least one RHR loop shall be determined to be in operation and circulating reactor coolant at least once per 12 hours.

4.4.1.4.2.2 Each valve or mechanical joint used to isolate unborated water sources shall be verified closed and secured in position at least once per 31 days.

*Two RHR loops may be inoperable for up to 2 hours for surveillance testing provided the other RHR loop is OPERABLE and in operation.

**The RHR pump may be deenergized for up to 1 hour provided: (1) no operations are permitted that would cause introduction into the RCS of coolant with boron concentration less than that required to meet SHUTDOWN MARGIN of LCO 3.1.1, and (2) core outlet temperature is maintained at least 10°F below saturation temperature.

SURVEILLANCE REQUIREMENTS

4.4.9.3.1 Each PORV shall be demonstrated OPERABLE by:

a. Performance of an ANALOG CHANNEL OPERATIONAL TEST on the PORV actuation channel, but excluding valve operation, within 31 days prior to entering a condition in which the PORV is required OPERABLE and at least once per 31 days thereafter when the PORV is required OPERABLE;

b. Performance of a CHANNEL CALIBRATION on the PORV actuation channel at least once per 18 months; and

c. Verifying the PORV block valve is open at least once per 72 hours when the PORV is being used for overpressure protection.

4.4.9.3.2 The RCS vent(s) shall be verified to be open at least once per 12 hours[4] when the vent(s) is being used for overpressure protection.

4.4.9.3.3 The positive displacement pump shall be demonstrated inoperable[5] at least once per 31 days, except when the reactor vessel head is removed or when both centrifugal charging pumps are inoperable and secured, by verifying that the motor circuit breakers are secured in the open position.[2]

4.4.9.3.4 Verify at least once every 31 days that only one centrifugal charging pump is capable of injecting into the RCS[2,5], except when the reactor vessel head is removed, by verifying that the motor circuit breakers are secured in the open position.[2]

4.4.9.3.5 Verify at least once every 12 hours that each ECCS accumulator is isolated.

SPECIFICATION NOTATIONS

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[1] ECCS accumulator isolation is required only when ECCS accumulator pressure is greater than or equal to the maximum RCS pressure for the existing RCS cold leg temperature allowed by Figures 3.4-2 and 3.4-3.

[2] An inoperable centrifugal charging pump(s) and/or positive displacement charging pump may be energized for testing or pump switching provided the discharge of the pump(s) has been isolated from the RCS by a closed isolation valve with power removed from the valve operator, or by a manual isolation valve secured in the closed position. Reactor coolant pump seal injection flow may be maintained during the RCS isolation process.

[3] This ACTION may be suspended for up to 7 days to allow functional testing to verify PORV operability. During this test period, operation of systems or components which could result in an RCS mass or temperature increase will be administratively controlled. During the ASME stroke testing of two inoperable PORVS, cold overpressurization mitigation will be provided by two RHR discharge relief valves associated with two OPERABLE and operating RHR loops which have the auto closure interlock bypassed [or deleted]. If one PORV is inoperable, cold overpressure mitigation will be provided by the OPERABLE PORV and one RHR discharge relief valve associated with an OPERABLE and operating RHR loop which has the auto closure interlock bypassed [or deleted].

[4] Except when the vent pathway is provided with a valve that is locked, sealed, or otherwise secured in the open position, then verify these valves open at least once per 31 days.

No Changes This Page

[5] Entry into MODE 4 from MODE 3 while making all but one centrifugal charging pump incapable of injecting into the RCS pursuant to Specification 4.4.9.3.4, and for the positive displacement pump declared inoperable pursuant to Specification 4.4.9.3.3 provided that all but one centrifugal charging pump is made incapable of injecting into the RCS, and the positive displacement pump is declared inoperable within 4 hours after entry into MODE 4 from MODE 3 or prior to the temperature of one or more of the RCS cold legs decreasing below 325°F, whichever comes first.

6.2.1 Offsite and Onsite Organizations

Onsite and offsite organizations shall be established for unit operation and corporate management, respectively. The onsite and offsite organizations shall include the positions for activities affecting the safety of the nuclear power plant.

a. Lines of authority, responsibility, and communication shall be defined and established throughout highest management levels, intermediate levels, and all operating organization positions. These relationships shall be documented and updated, as appropriate, in organization charts, functional descriptions of departmental responsibilities and relationships, job descriptions for key personnel positions, or in equivalent forms of documentation. These requirements, including the plant-specific titles of those personnel fulfilling the responsibilities of the positions delineated in these Technical Specifications, shall be documented in the UFSAR and/or the Operations Quality Assurance Plan.

b. The plant manager shall be responsible for overall safe operation of the plant and shall have control over those onsite activities necessary for safe operation and maintenance of the plant.

c. A specified corporate officer shall have corporate responsibility for overall plant nuclear safety and shall take any measures needed to ensure acceptable performance of the staff in operating, maintaining, and providing technical support to the plant to ensure nuclear safety.

d. The individuals who train the operating staff, carry out radiation protection functions, or perform quality assurance functions may report to the appropriate onsite manager; however, these individuals shall have sufficient organizational freedom to ensure their independence from operating pressures.

6.2.2 Unit Staff

The unit staff organization shall include the following:

a. A total of three non-licensed operators for the two units is required in all conditions. At least one of the required non-licensed operators shall be assigned to each unit. When a unit is operating in MODES 1, 2, 3, or 4, two non-licensed operators are required to be assigned to that unit.

b. The shift crew composition may be one less than the minimum requirements of 10 CFR 50.54(m)(2)(i) and Specifications 6.2.2.a and 6.2.2.~~g~~ f for a period of time not to exceed 2 hours in order to accommodate unexpected absence of on-duty shift crew members, provided immediate action is taken to restore the shift crew composition to within the minimum requirements.

(continued)

6.9.1.6a (Continued)

8. Heat Flux Hot Channel Factor, K(Z), Power Factor Multiplier, ~~and~~ F_{xy}^{RTP}, **and $F_Q(Z)$ manufacturing and measurement uncertainties** for Specification 3/4.2.2,

9. Nuclear Enthalpy Rise Hot Channel Factor, ~~and~~ Power Factor Multiplier, **and $F_{\Delta H}^{N}$ measurement uncertainties** for Specification 3/4.2.3, and

10. DNB related parameters for Reactor Coolant System T_{avg} Pressurizer Pressure, and the Minimum Measured Reactor Coolant System Flow for Specification 3/4.2.5.

The COLR shall be maintained available in the Control Room.

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 9272-P-A, "Westinghouse reload safety evaluation methodology," July 1985 (W Proprietary).

(Methodology for Specification 3.1.1.1 – Shutdown Margin, Methodology for Specification 3.1.1.3 - Moderator Temperature Coefficient, 3.1.3.5 – Shutdown Rod Insertion Limit, 3.1.3.6 – Control Bank Insertion Limits, 3.2.1 – Axial Flux Difference, 3.2.2 – Heat Flux Hot Channel Factor, 3.2.3 – Nuclear Enthalpy Rise Hot Channel Factor, and 3.2.5 – DNB Parameters.)

2. WCAP 12942-P-A "safety evaluation supporting a more negative eol Moderator temperature coefficient technical specification for the south texas project electric generating station units 1 and 2."

(Methodology for Specification 3.1.1.3 – Moderator Temperature Coefficient)

3. WCAP 8745-P-A, "Design Basis for the Thermal Overpower ΔT and Thermal Overtemperature ΔT Trip Functions," September 1986 (Westinghouse Proprietary Class 2).

(Methodology for Specification 2.1 – Safety Limits and 2.2 – Limiting Safety System Settings)

4. WCAP 8385, "power distribution and load following procedures topical report," September, 1974 (W Proprietary)

(Methodology for Specification 3.2.1 – Axial Flux Difference (Constant Axial Offset Control)

(continued)

6.9.1.6b (continued)

5. Westinghouse Letter NS-TMA-2198, T. M. Anderson (Westinghouse) to K. Kniel (Chief of Core Performance Branch, NRC) January 31, 1980 – Attachment: Operation and Safety Analysis Aspects of an Improved Load Follow Package.

(Methodology for Specification 3.2.1 – Axial Flux Difference (Constant Axial Offset Control). Approved by NRC Supplement No. 4 to NUREG-0422, January 1981, Docket Nos. 50-369 and 50-370.)

6. NUREG-0800, Standard Review Plan, U. S. Nuclear Regulatory Commission, Section 4.3, Nuclear Design, July, 1981. Branch Technical Position CPB 4.3-1, Westinghouse Constant Axial Offset Control (CAOC), Rev. 2, July 1981.

(Methodology for Specification 3.2.1 - Axial Flux Difference (Constant Axial Offset Control).)

7. WCAP 10266-P-A, Rev. 2, WCAP 11524-NP-A, Rev. 2, "The 1981 Version of the Westinghouse ECCS Evaluation Model Using the BASH Code," Kabadi, J. N., et al., March 1987; including Addendum 1-A, "Power Shape Sensitivity Studies," December, 1987 and Addendum 2-A, "BASH methodology Improvements and Reliability Enhancements," May 1988.

(Methodology for Specification 3.2.2 – Heat Flux Hot Channel Factor)

8. WCAP 12610-P-A, "Vantage+ Fuel Assembly Reference Core Report," April 1995 (W Proprietary)

(Methodology for Specification 3.2.2 – Heat Flux Hot Channel Factor)

9. CENPD-397-P-A, Revision 01, "Improved Flow Measurement Accuracy Using Crossflow Ultrasonic Flow Measurement Technology," May 2000.

(Methodology for operating at a RATED THERMAL POWER of 3,853 Mwt)

10. WCAP 13749-P-A, "Safety Evaluation Supporting the Conditional Exemption of the Most Negative EOL Moderator Temperature Coefficient Measurement," March 1997 (W Proprietary).

(Methodology for Specification 3.1.1.3 – Moderator Temperature Coefficient)

11. WCAP 12472-P-A, "BEACON Core Monitoring and Operations Support System," August 1994 (W Proprietary)

(Methodology for Specification 3.2.1 – Axial Flux Difference, 3.2.2 – Heat Flux Hot Channel Factor, 3.2.3 – Nuclear Enthalpy Rise Hot Channel Factor)

(continued)

Attachment 3

Proposed Changes to the Technical Specification Bases

(For Information Only)

Proposed Changes to the Technical Specification Bases
(For Information Only)

The Bases for TS

3/4.1.3.1	Movable Control Assemblies - Group Height
3/4.1.3.2	Position Indicating Systems - Operating
3/4.2.4	Quadrant Power Tilt Ratio
3/4.3.1	Reactor Trip Instrumentation

do not need to be revised to reflect the proposed changes.

The revised Bases for TS 3/4.2.2 and 3/4.2.3 are presented below:

3/4.2.2 and 3/4.2.3 Heat Flux Hot Channel Factor and Nuclear Enthalpy Rise Hot Channel Factor

The limits on heat flux hot channel factor and nuclear enthalpy rise hot channel factor ensure that: (1) the design limits on peak local power density and minimum DNBR are not exceeded and (2) in the event of a LOCA the peak fuel clad temperature will not exceed the 2200°F ECCS acceptance criteria limit.

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:

a. 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;

b. Control rod groups are sequenced with overlapping groups as described in Specification 3.1.3.6;

c. The control rod insertion limits of Specifications 3.1.3.5 and 3.1.3.6 are maintained; and

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

$F^N_{\Delta H}$ will be maintained within its limits provided Conditions a. through d. above are maintained. The combination of the RCS flow requirement (TS 3.2.5) and the requirement on $F^N_{\Delta H}$ guarantees that the DNBR used in the safety analysis will be met. The relaxation of $F^N_{\Delta H}$ a function of THERMAL POWER allows changes in the radial power shape for all permissible rod insertion limits.

An allowance for measurement uncertainty must be made for $F_{\Delta H}^{N}$ measurements. If the movable incore detector system is used to obtain the measurement, the correct measurement uncertainty is presented in the COLR. The appropriate measurement uncertainty for $F_{\Delta H}^{N}$ using the Power Distribution Monitoring System (PDMS) is developed using methodology described in WCAP-12472-P-A. Information on the PDMS uncertainty calculation is contained in the COLR. The PDMS will automatically calculate and apply the correct measurement uncertainty to the measured $F_{\Delta H}^{N}$

Fuel rod bowing reduces the value of DNB ratio. Margin has been maintained between the DNBR value used in the safety analyses and the design limit to offset the rod bow penalty and other penalties which may apply.

An allowance for both experimental error and manufacturing tolerance must be made when an F_Q measurement is taken. If the movable incore detector system is used to obtain the measurement, the correct allowances to be applied to the measurement are specified in the COLR. If the PDMS is used, the tolerances are calculated using methods described in WCAP-12472-P-A. Information on the PDMS measurement uncertainty and tolerance allowance is presented in the COLR. The PDMS automatically applies the appropriate measurement uncertainty and manufacturing allowance to the measured F_Q.

The Radial Peaking Factor, $F_{XY}(Z)$, is measured periodically to provide assurance that the Hot Channel Factor, $F_Q(Z)$, remains within its limit. The F_{xy} limit for RATED THERMAL POWER (F_{xy}^{RTP}) as provided in the Core Operating Limits Reports (COLR) was determined from expected power control manuevers over the full range of burnup conditions in the core.

Attachment 4

Sample Core Operating Limits Report

(For Information Only)

SOUTH TEXAS PROJECT

Unit 1 Cycle 13

CORE OPERATING LIMITS REPORT

Revision 0

1.0 CORE OPERATING LIMITS REPORT

This Core Operating Limits Report for STPEGS Unit 1 Cycle 13 has been prepared in accordance with the requirements of Technical Specification 6.9.1.6. The core operating limits have been developed using the NRC-approved methodologies specified in Technical Specification 6.9.1.6.

The Technical Specifications affected by this report are:

1)	2.1	SAFETY LIMITS
2)	2.2	LIMITING SAFETY SYSTEM SETTINGS
3)	3/4.1.1.1	SHUTDOWN MARGIN
4)	3/4.1.1.3	MODERATOR TEMPERATURE COEFFICIENT LIMITS
5)	3/4.1.3.5	SHUTDOWN ROD INSERTION LIMITS
6)	3/4.1.3.6	CONTROL ROD INSERTION LIMITS
7)	3/4.2.1	AFD LIMITS
8)	3/4.2.2	HEAT FLUX HOT CHANNEL FACTOR
9)	3/4.2.3	NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR
10)	3/4.2.5	DNB PARAMETERS

2.0 OPERATING LIMITS

The cycle-specific parameter limits for the specifications listed in Section 1.0 are presented below.

2.1 SAFETY LIMITS (Specification 2.1):

2.1.1 The combination of THERMAL POWER, pressurizer pressure, and the highest operating loop coolant temperature (T_{avg}) shall not exceed the limits shown in Figure 1.

2.2 LIMITING SAFETY SYSTEM SETTINGS (Specification 2.2):

2.2.1 The Loop design flow for Reactor Coolant Flow-Low is 98,000 gpm.

2.2.2 The Over-temperature ΔT and Over-power ΔT setpoint parameter values are listed below:

Over-temperature ΔT Setpoint Parameter Values

τ_1 measured reactor vessel ΔT lead/lag time constant, $\tau_1 = 8$ sec
τ_2 measured reactor vessel ΔT lead/lag time constant, $\tau_2 = 3$ sec
τ_3 measured reactor vessel ΔT lag time constant, $\tau_3 = 2$ sec
τ_4 measured reactor vessel average temperature lead/lag time constant, $\tau_4 = 28$ sec
τ_5 measured reactor vessel average temperature lead/lag time constant, $\tau_5 = 4$ sec
τ_6 measured reactor vessel average temperature lag time constant, $\tau_6 = 2$ sec
K_1 Overtemperature ΔT reactor trip setpoint, $K_1 = 1.14$
K_2 Overtemperature ΔT reactor trip setpoint T_{avg} coefficient, $K_2 = 0.028/°F$
K_3 Overtemperature ΔT reactor trip setpoint pressure coefficient, $K_3 = 0.00143$/psig
T' Nominal full power T_{avg}, $T' \leq 592.0$ °F
P' Nominal RCS pressure, P' = 2235 psig
$f_1(\Delta I)$ is a function of the indicated difference between top and bottom detectors of the power-range neutron ion chambers; with gains to be selected based on measured instrument response during plant startup tests such that:

(1) For $q_t - q_b$ between -70% and +8%, $f_1(\Delta I) = 0$, where q_t and q_b are percent RATED THERMAL POWER in the top and bottom halves of the core respectively, and $q_t + q_b$ is total THERMAL POWER in percent of RATED THERMAL POWER;

(2) For each percent that the magnitude of $q_t - q_b$ exceeds -70%, the ΔT Trip Setpoint shall be automatically reduced by 0.0% of its value at RATED THERMAL POWER; and

(3) For each percent that the magnitude of $q_t - q_b$ exceeds +8%, the ΔT Trip Setpoint shall be automatically reduced by 2.65% of its value at RATED THERMAL POWER.

Over-power ΔT Setpoint Parameter Values

τ_1 measured reactor vessel ΔT lead/lag time constant, $\tau_1 = 8$ sec
τ_2 measured reactor vessel ΔT lead/lag time constant, $\tau_2 = 3$ sec
τ_3 measured reactor vessel ΔT lag time constant, $\tau_3 = 2$ sec
τ_6 measured reactor vessel average temperature lag time constant, $\tau_6 = 2$ sec
τ_7 Time constant utilized in the rate-lag compensator for T_{avg}, $\tau_7 = 10$ sec
K_4 Overpower ΔT reactor trip setpoint, $K_4 = 1.08$
K_5 Overpower ΔT reactor trip setpoint T_{avg} rate/lag coefficient, $K_5 = 0.02/°F$ for increasing average temperature, and $K_5 = 0$ for decreasing average temperature
K_6 Overpower ΔT reactor trip setpoint T_{avg} heatup coefficient $K_6 = 0.002/°F$ for $T > T''$, and $K_6 = 0$ for $T \leq T''$
T'' Indicated full power T_{avg}, $T'' \leq 592.0$ °F
$f_2(\Delta I) = 0$ for all (ΔI)

2.3 SHUTDOWN MARGIN (Specification 3.1.1.1):

The SHUTDOWN MARGIN shall be:

2.3.1 Greater than 1.3% $\Delta\rho$ for MODES 1 and 2*
*See Special Test Exception 3.10.1

2.3.2 Greater than the limits in Figure 2 for MODES 3 and 4.

2.3.3 Greater than the limits in Figure 3 for MODE 5.

2.4 MODERATOR TEMPERATURE COEFFICIENT (Specification 3.1.1.3):

2.4.1 The BOL, ARO, MTC shall be less positive than the limits shown in Figure 4.

2.4.2 The EOL, ARO, HFP, MTC shall be less negative than –62.6 pcm/°F.

2.4.3 The 300 ppm, ARO, HFP, MTC shall be less negative than –53.6 pcm/°F (300 ppm Surveillance Limit).

Where: BOL stands for Beginning-of-Cycle Life,
EOL stands for End-of-Cycle Life,
ARO stands for All Rods Out,
HFP stands for Hot Full Power (100% RATED THERMAL POWER),
HFP vessel average temperature is 592 °F.

2.4.4 The Revised Predicted near-EOL 300 ppm MTC shall be calculated using the algorithm from T.S. 6.9.1.6.b.10:

Revised Predicted MTC = Predicted MTC + AFD Correction - 3 pcm/°F

If the Revised Predicted MTC is less negative than the S.R. 4.1.1.3b limit and all of the benchmark data contained in the surveillance procedure are met, then an MTC measurement in accordance with S.R. 4.1.1.3b is not required.

2.5 ROD INSERTION LIMITS (Specification 3.1.3.5 and 3.1.3.6):

2.5.1 All banks shall have the same Full Out Position (FOP) of at least 250 steps withdrawn but not exceeding 259 steps withdrawn.

2.5.2 The Control Banks shall be limited in physical insertion as specified in Figure 5.

2.5.3 Individual Shutdown bank rods are fully withdrawn when the Bank Demand Indication is at the FOP and the Rod Group Height Limiting Condition for Operation is satisfied (T.S. 3.1.3.1).

2.6 AXIAL FLUX DIFFERENCE (Specification 3.2.1):

2.6.1 AFD limits as required by Technical Specification 3.2.1 are determined by CAOC Operations with an AFD target band of +5, -10%.

2.6.2 The AFD shall be maintained within the ACCEPTABLE OPERATION portion of Figure 6, as required by Technical Specifications.

2.7 HEAT FLUX HOT CHANNEL FACTOR (Specification 3.2.2):

2.7.1 $F_Q^{RTP} = 2.55$.

2.7.2 K(Z) is provided in Figure 7.

2.7.3 The F_{xy} limits for RATED THERMAL POWER (F_{xy}^{RTP}) within specific core planes shall be:

2.7.3.1 Less than or equal to 2.102 for all cycle burnups for all core planes containing Bank "D" control rods, and

2.7.3.2 Less than or equal to the appropriate core height-dependent value from Table 1 for all unrodded core planes.

2.7.3.3 $PF_{xy} = 0.2$.

These F_{xy} limits were used to confirm that the heat flux hot channel factor $F_Q(Z)$ will be limited by Technical Specification 3.2.2 assuming the most-limiting axial power distributions expected to result for the insertion and removal of Control Banks C and D during operation, including the accompanying variations in the axial xenon and power distributions, as described in WCAP-8385. Therefore, these F_{xy} limits provide assurance that the initial conditions assumed in the LOCA analysis are met, along with the ECCS acceptance criteria of 10 CFR 50.46.

2.7.4 Core Power Distribution Uncertainties for the heat flux hot channel factor:

2.7.4.1 If the Power Distribution Monitoring System is OPERABLE, as defined in the Technical Requirements Manual, the core power distribution measurement uncertainty (U_{FQ}) to be applied to the $F_Q(Z)$ and $F_{xy}(Z)$ using the PDMS shall be calculated by:

$$\mathbf{U_{FQ} = (1.0 + (U_Q/100))*U_E}$$

Where:

U_Q = Uncertainty for power peaking factor as defined in Equation 5-19 of T.S. 6.9.1.6.b.11.
U_E = Manufacturing uncertainty (or tolerance) factor of 1.03.

This uncertainty is calculated and applied automatically by the Beacon computer code in the Core Monitor Surveillance Report.

2.7.4.2 If the moveable incore detectors are used, the core power distribution measurement uncertainty (U_{FQ}) to be applied to the $F_Q(Z)$ and $F_{xy}(Z)$ shall be calculated by:

$$\mathbf{U_{FQ} = U_{QU} * U_E}$$

Where:

U_{QU} = FQ measurement uncertainty of 1.05.
U_E = Manufacturing uncertainty (or tolerance) factor of 1.03.

2.8 ENTHALPY RISE HOT CHANNEL FACTOR (Specification 3.2.3):

2.8.1 $F_{\Delta H}^{RTP} = 1.62$ [1]

2.8.2 $PF_{\Delta H} = 0.3$

[1] Applies to all fuel in the Unit 1 Cycle 13 Core.

2.8.3 Core Power Distribution Measurement Uncertainty for the enthalpy rise hot channel factor

2.8.3.1 If the Power Distribution Monitoring System is OPERABLE, as defined in the Technical Requirements Manual, the core power distribution measurement uncertainty ($U_{F\Delta H}$) to be applied to the measured $F^N_{\Delta H}$ using the PDMS shall be calculated by:

$U_{F\Delta H} = 1.0 + (U_{\Delta H}/100)$

Where:

$U_{\Delta H}$ = Uncertainty for power peaking factor as defined in Equation 5-19 of T.S. 6.9.1.6.b.11.

This uncertainty is calculated and applied automatically by the Beacon computer code in the Core Monitor Surveillance Report.

2.8.3.2 If the moveable incore detectors are used, the core power distribution measurement uncertainty ($U_{F\Delta H}$) to be applied to the measured $F^N_{\Delta H}$ shall be calculated by:

$U_{F\Delta H} = 1.04$

2.9 DNB PARAMETERS (Specification 3.2.5):

2.9.1 The following DNB-related parameters shall be maintained within the following limits: [1]

2.9.1.1 Reactor Coolant System T_{avg}, $\leq$ 595 °F [2],

2.9.1.2 Pressurizer Pressure, > 2200 psig [3],

2.9.1.3 Minimum Measured Reactor Coolant System Flow [4] > 403,000 gpm.

[1] A discussion of the processes to be used to take these readings is provided in the basis for Technical Specification 3.2.5.

[2] Includes a 1.9 °F measurement uncertainty per Reference 3.3.

[3] Limit not applicable during either a Thermal Power ramp in excess of 5% of RTP per minute or a Thermal Power step in excess of 10% RTP. Includes a 9.6 PSI measurement uncertainty as read on QDPS display per Reference 3.4.

[4] Includes a 2.8% flow measurement uncertainty.

3.0 REFERENCES

3.1 Letter from D. E. Robinson (Westinghouse) to D. F. Hoppes (STPNOC), "Unit 1 Cycle 13 Final Reload Evaluation (RE) Revision 1," ST-UB-NOC-05002532, Rev. 1, March 29 2005.

3.2 NUREG-1346, Technical Specifications, South Texas Project Unit Nos. 1 and 2.

3.3 STPNOC Calculation ZC-7035, Rev. 2, "Loop Uncertainty Calculation for RCS Tavg Instrumentation," Section 10.1, effective July 22, 2003.

3.4 STPNOC Calculation ZC-7032, Rev. 4, "Loop Uncertainty Calculation for Narrow Range Pressurizer Pressure Monitoring Instrumentation," Section 2.3, Page 9, effective July 22, 2003.

3.5 Condition Report Engineering Evaluation 03-6461-9, Revision 0, "Reload Safety Evaluation and Core Operating Limits Report for South Texas Unit 1 Cycle 13 Modes 1, 2, 3, 4, and 5."

Figure 1

Reactor Core Safety Limits - Four Loops in Operation

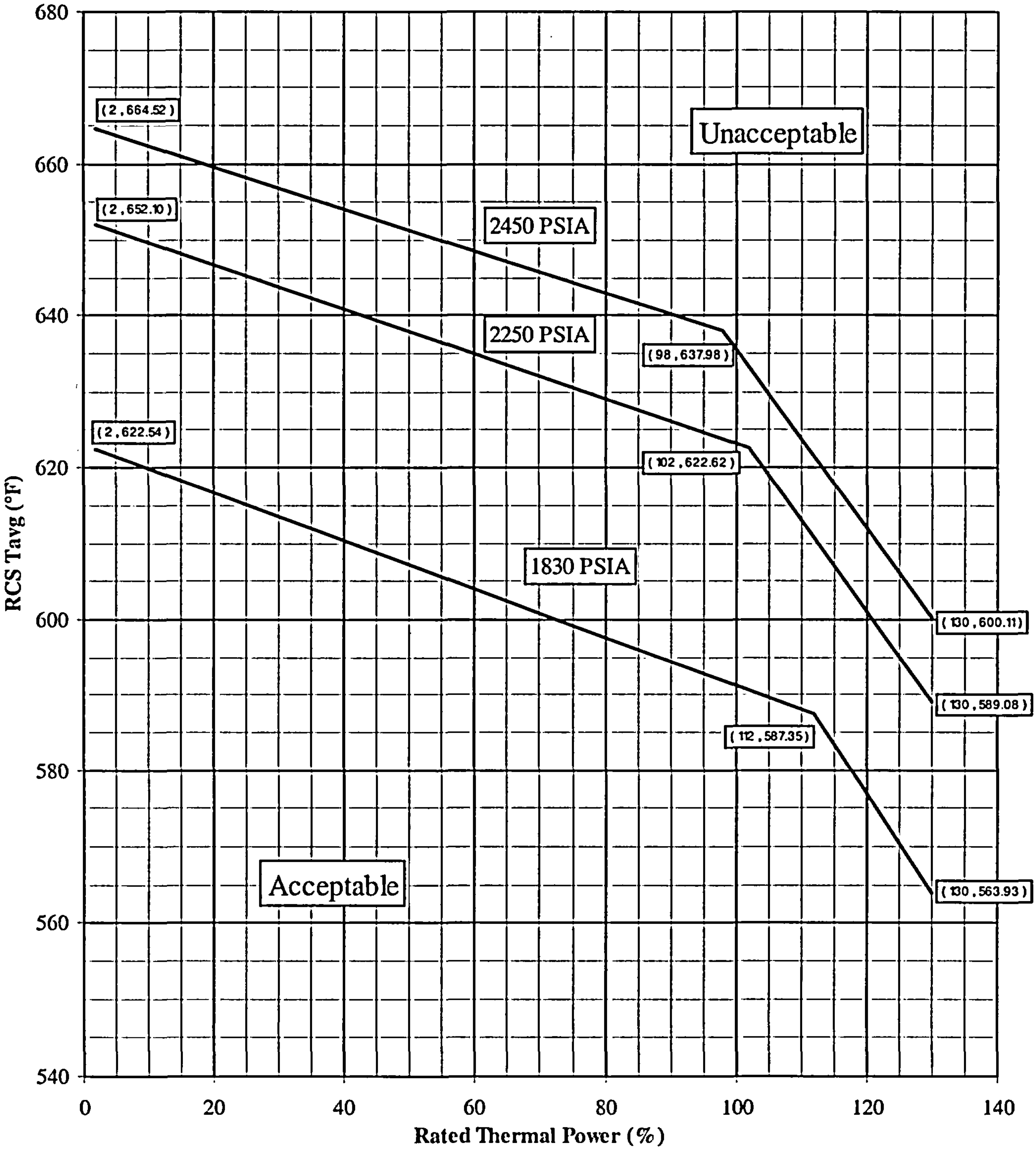
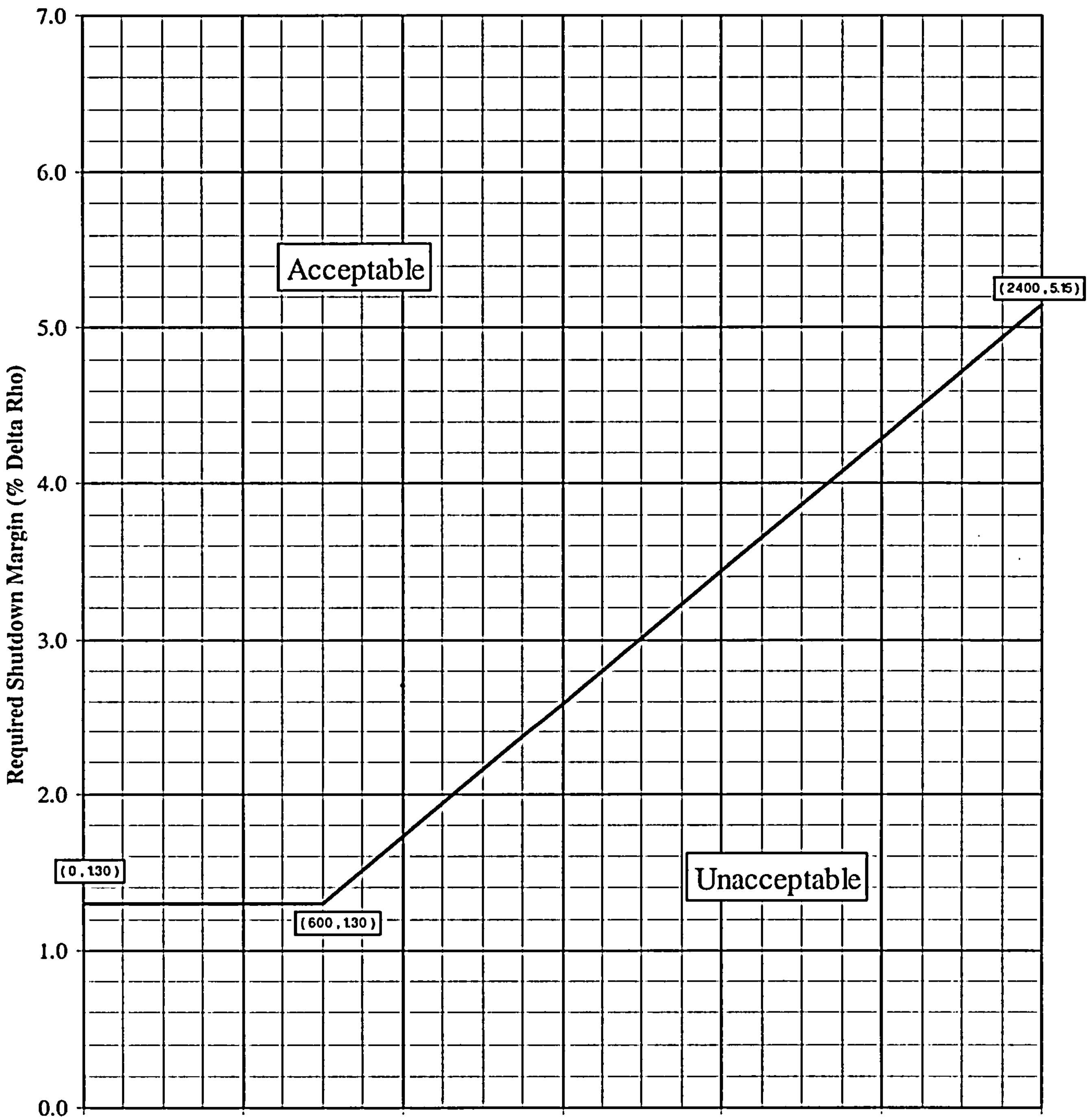


Figure 2

Required Shutdown Margin for Modes 3 & 4



RCS Critical Boron Concentration (ppm)
(for ARI minus most reactive stuck rod)

Figure 3

Required Shutdown Margin for Mode 5

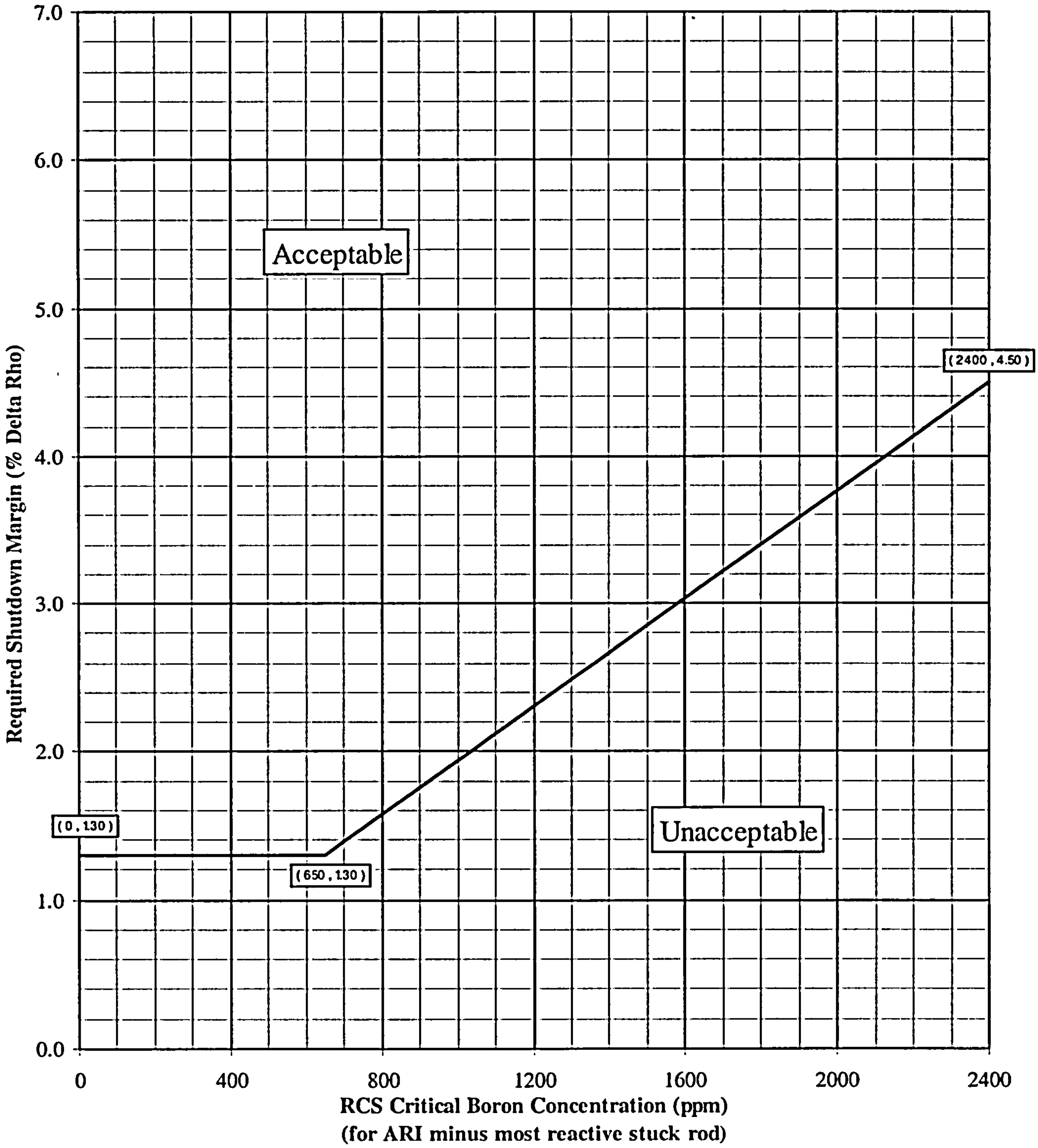


Figure 4

MTC versus Power Level

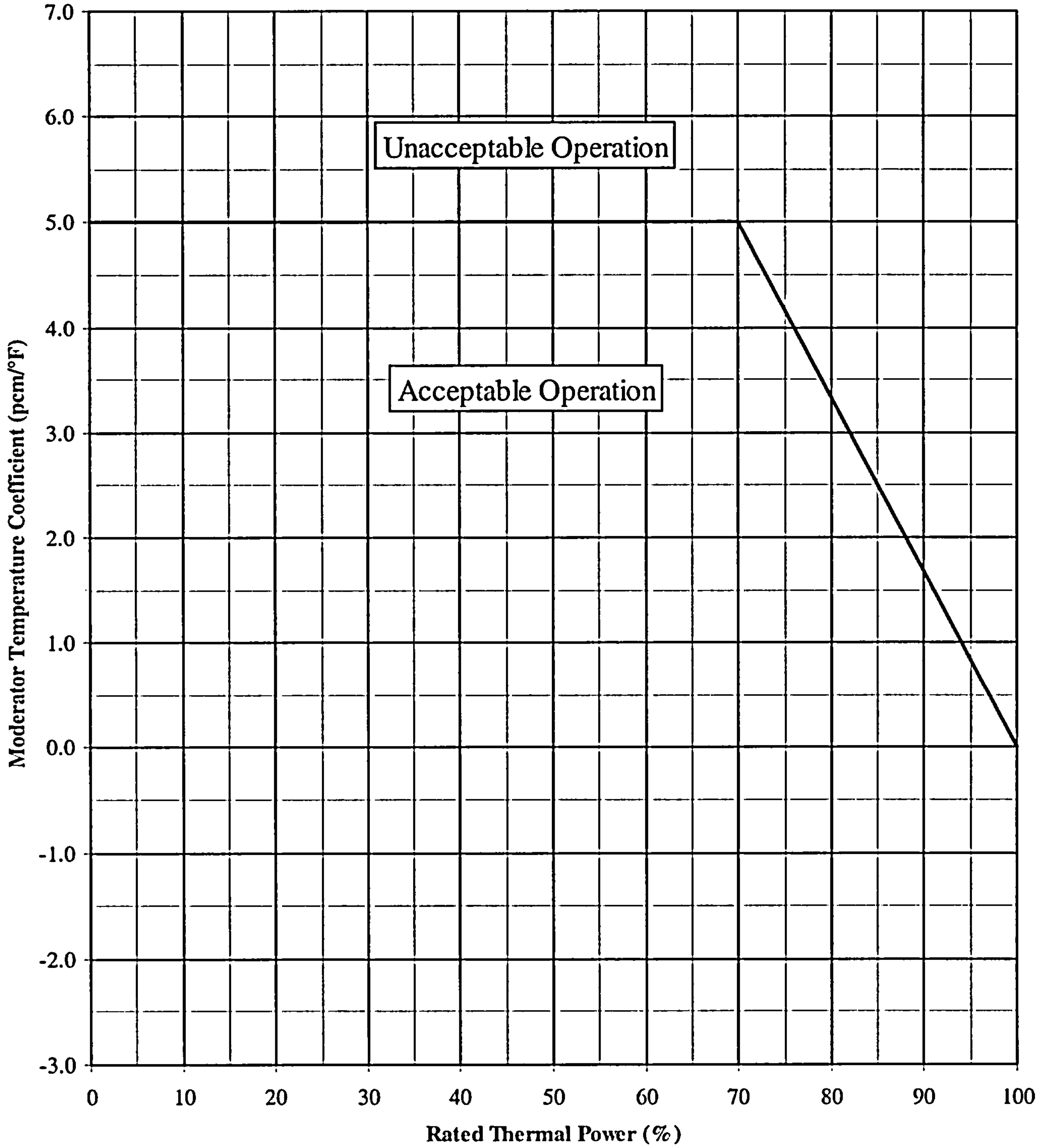
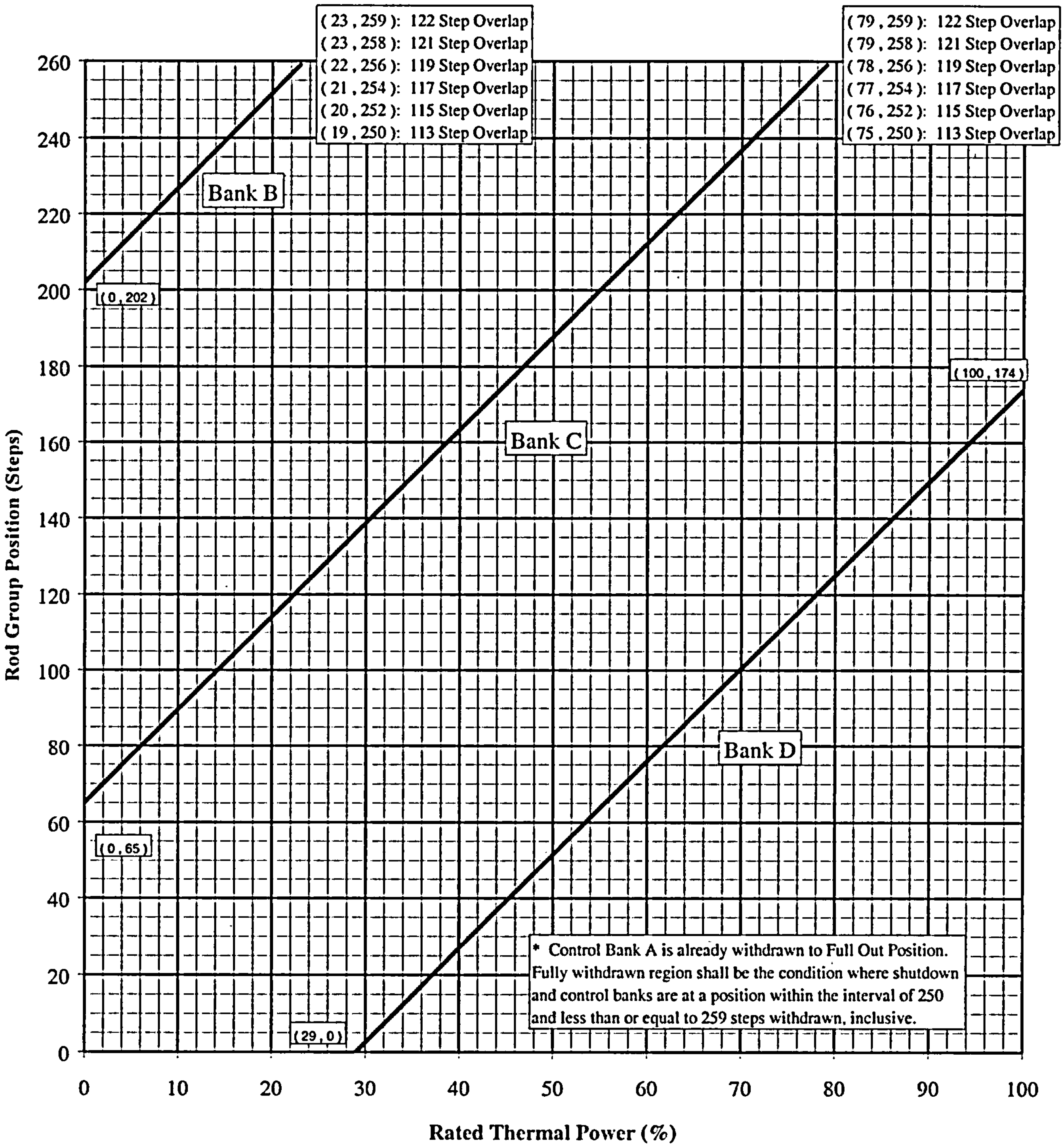


Figure 5

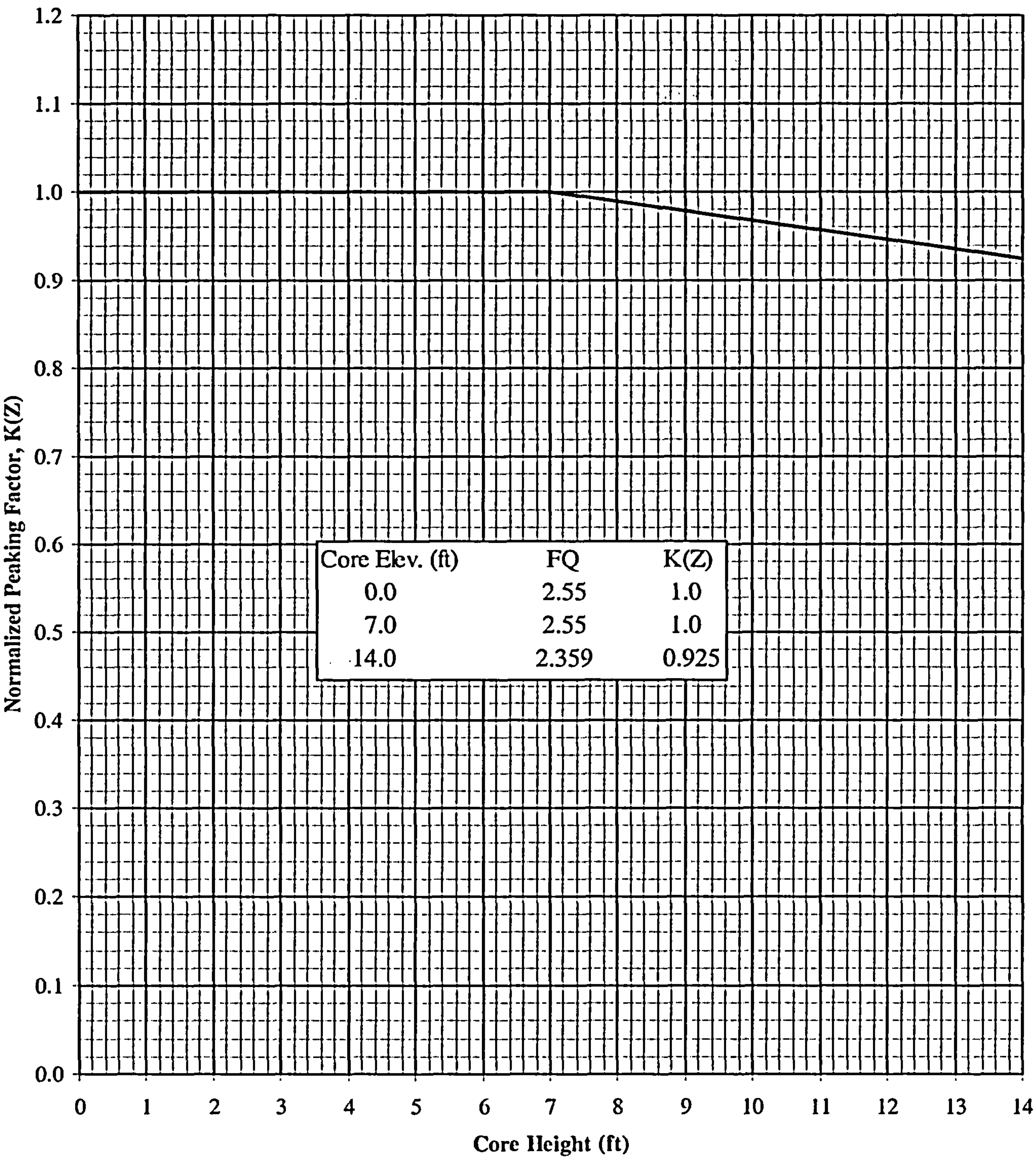
Control Rod Insertion Limits* versus Power Level



* Control Bank A is already withdrawn to Full Out Position. Fully withdrawn region shall be the condition where shutdown and control banks are at a position within the interval of 250 and less than or equal to 259 steps withdrawn, inclusive.

Figure 7

K(Z) - Normalized FQ(Z) versus Core Height



Core Elev. (ft)	FQ	K(Z)
0.0	2.55	1.0
7.0	2.55	1.0
14.0	2.359	0.925

Table 1 (Part 1 of 2)

Unrodded F_{xy} for Each Core Height

for Cycle Burnups Less Than 10500 MWD/MTU

Core Height (Ft.)	Axial Point	Unrodded Fxy	Core Height (Ft.)	Axial Point	Unrodded Fxy
14.00	1	4.633	6.80	37	1.872
13.80	2	3.954	6.60	38	1.861
13.60	3	3.275	6.40	39	1.853
13.40	4	2.596	6.20	40	1.845
13.20	5	2.276	6.00	41	1.841
13.00	6	2.068	5.80	42	1.840
12.80	7	2.070	5.60	43	1.839
12.60	8	2.046	5.40	44	1.840
12.40	9	2.023	5.20	45	1.844
12.20	10	1.995	5.00	46	1.850
12.00	11	1.979	4.80	47	1.860
11.80	12	1.971	4.60	48	1.870
11.60	13	1.971	4.40	49	1.878
11.40	14	1.975	4.20	50	1.885
11.20	15	1.979	4.00	51	1.890
11.00	16	1.984	3.80	52	1.893
10.80	17	1.984	3.60	53	1.893
10.60	18	1.984	3.40	54	1.895
10.40	19	1.982	3.20	55	1.899
10.20	20	1.985	3.00	56	1.904
10.00	21	1.987	2.80	57	1.903
9.80	22	1.989	2.60	58	1.911
9.60	23	1.997	2.40	59	1.921
9.40	24	2.003	2.20	60	1.932
9.20	25	2.011	2.00	61	1.929
9.00	26	2.017	1.80	62	1.920
8.80	27	2.023	1.60	63	1.905
8.60	28	2.027	1.40	64	1.907
8.40	29	2.032	1.20	65	1.909
8.20	30	2.037	1.00	66	1.920
8.00	31	2.033	0.80	67	1.994
7.80	32	2.016	0.60	68	2.124
7.60	33	1.983	0.40	69	2.281
7.40	34	1.945	0.20	70	2.439
7.20	35	1.911	0.00	71	2.596
7.00	36	1.889			

Table 1 (Part 2 of 2)
Unrodded Fxy for Each Core Height
for Cycle Burnups Greater Than or Equal to 10500 MWD/MTU

Core Height (Ft.)	Axial Point	Unrodded Fxy	Core Height (Ft.)	Axial Point	Unrodded Fxy
14.00	1	4.778	6.80	37	2.144
13.80	2	4.129	6.60	38	2.139
13.60	3	3.480	6.40	39	2.127
13.40	4	2.831	6.20	40	2.115
13.20	5	2.450	6.00	41	2.101
13.00	6	2.153	5.80	42	2.089
12.80	7	2.149	5.60	43	2.077
12.60	8	2.121	5.40	44	2.067
12.40	9	2.096	5.20	45	2.057
12.20	10	2.055	5.00	46	2.047
12.00	11	2.032	4.80	47	2.038
11.80	12	2.026	4.60	48	2.028
11.60	13	2.019	4.40	49	2.017
11.40	14	2.022	4.20	50	2.006
11.20	15	2.027	4.00	51	1.994
11.00	16	2.031	3.80	52	1.982
10.80	17	2.033	3.60	53	1.969
10.60	18	2.035	3.40	54	1.957
10.40	19	2.036	3.20	55	1.945
10.20	20	2.041	3.00	56	1.931
10.00	21	2.048	2.80	57	1.916
9.80	22	2.056	2.60	58	1.891
9.60	23	2.065	2.40	59	1.858
9.40	24	2.072	2.20	60	1.841
9.20	25	2.078	2.00	61	1.839
9.00	26	2.084	1.80	62	1.835
8.80	27	2.089	1.60	63	1.836
8.60	28	2.095	1.40	64	1.853
8.40	29	2.100	1.20	65	1.853
8.20	30	2.106	1.00	66	1.877
8.00	31	2.112	0.80	67	2.127
7.80	32	2.119	0.60	68	2.605
7.60	33	2.127	0.40	69	3.195
7.40	34	2.135	0.20	70	3.786
7.20	35	2.142	0.00	71	4.377
7.00	36	2.145			