BWR OWNERS' GROUP

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Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Mail Station P1-137
Washington, D.C. 20555

Attention: Martin J. Virgilio, Director
Division of System Safety & Analysis

Subject: BWR Owners' Group Guidelines for Stability Interim Corrective Action

Reference: Letter, L.A. England to M.J. Virgilio, "BWR Owners' Group Improved Guidelines for Stability Interim Corrective Actions", April 4, 1994

Attached is a copy of the improved BWR Owners' Group (BWROG) Guidelines for Stability Interim Corrective Action. This material is being submitted to inform the NRC of the guidance being provided to the utilities by the BWROG, formal NRC approval of this material is not being requested. It should be noted that the feedback provided by the NRC Staff on the previously submitted draft (Reference) has been factored into this final version.

Previous BWROG guidelines for stability interim corrective actions (ICAs) were developed to assist utilities in the implementation of the requirements of NRC Bulletin 88-07, Supplement 1. The improved guidance contained in the attachment to this letter was developed primarily to better address operation at low power and flow such as exists during startup and pump shifts. Stability controls were also expanded for operation at very high rod lines and reduced flow to provide additional margin. This material has been carefully reviewed and approved in accordance with BWROG procedures. Based on this review, it has been concluded that the ICAs provide appropriate guidance to reduce the likelihood of an instability and to enhance early detection in the very unlikely event that some stability threshold is exceeded in spite of the ICA guidelines. It is important to note that this guidance is intended for application only until the stability long-term solutions are in place. If certain elements of the ICAs are

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appropriate for application in conjunction with some of the stability long-term solutions, this should be addressed in the licensing of the individual solutions. These improved guidelines are being distributed to BWROG member utilities for use in assessing their interim (until long-term solutions are in place) operating procedures and operator training programs relative to stability. The BWROG recommends these revised guidelines but does not consider the use of these particular guidelines obligatory. If you have any questions please contact either T.J. Rausch (Stability Committee Chairman) on (708) 663 6645, H.C. Pfefferlen (Stability Committee Program Manager) on (408) 925 3392, or the undersigned.

The material contained in the enclosure has been endorsed by a substantial number of the members of the BWR Owners' Group; however, it should not be interpreted as a commitment of any individual member to a specific course of action. Each member must formally endorse the BWR Owners' Group position in order for that position to become the member's position.

Sincerely,

LAEngland

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nrcica4/hcp

cc: RA Pinelli, BWROG Vice-Chairman
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BWR Owners' Group

Guidelines for Stability Interim Corrective Action

6/6/94

SECTION 1.0 INTRODUCTION

This document contains the BWR Owners Group (BWROG) recommended revised guidelines for reactor stability Interim Corrective Actions (hereafter referred to as Revised-ICA). The guideline revisions were made to better address the August 15, 1992 power oscillation event at WNP-2. The BWROG considers these recommendations to be an acceptable way to implement the USNRC requirement for stability ICAs in Bulletin 88-07, Supplement 1 given the event at WNP-2 (Note: these recommendations are not applicable to Big Rock Point). Alternative approaches based on plant specific evaluations or analysis may also be acceptable and are not precluded by these BWROG recommendations. Such alternative approaches might consider different regions as well as different actions and oscillation detection guidelines, based on appropriately supported knowledge of specific plant characteristics.

The interim corrective actions in this document incorporate all of the relevant recommendations and guidance provided by the BWROG since NRC Bulletin 88-07, Supplement 1. Accordingly, this document is intended to supersede all previous BWROG transmittals related to stability ICAs (Reference 1, 2 & 3), and to update the GE/BWROG recommended actions included as Attachment 1 of NRC Bulletin 88-07, Supplement 1. In issuing this document, the BWROG is providing interim guidance to reduce the potential for a significant instability occurrence prior to the implementation of the Long-Term Solutions (LTS). Consistent with previous discussions between the BWROG and the NRC, these ICA recommendations are applicable only until the LTS are in place.

These recommendations reflect lessons learned from recent industry experience (References 4 and 5), as well as new analytical progress in understanding power oscillations. The guidance and actions recommended by this document emphasize instability prevention. Prevention is emphasized because of the burden placed on an operator when monitoring for the onset of power oscillations.

The applicability of each recommendation to various plant organizations (i.e., operations and reactor engineering) is identified by the document section containing the specific guidance. Each section provides direction for specific training and procedural enhancements that are appropriate, considering the guidance provided therein.

SECTION 2.0 OPERATOR GUIDANCE

The region of the power-flow operating domain considered to be susceptible to thermal-hydraulic instability has been revised based on recent industry experience and analysis. Figure 2.1 illustrates the reactor stability Revised-ICA operating regions.

2.1 Revised-ICA Operating Regions and Actions

2.1.1 Scram Region (Region I)

Size:

1. Operating domain region below 40% rated core flow and above the 100% rated rod line (recirculation flow control line).

2. Natural core circulation while in the RUN mode (see Note 1). Natural circulation conditions are defined as no recirculation pumps operating (see Note 2).

   Notes:

   1) This guidance reflects the NRC bulletin 88-07, Supplement 1 requirement for manual scram at natural circulation while in the RUN mode and is, therefore, not technically consistent with the other BWROG ICA region definitions.

   2) When the reactor recirculation pumps in flow control valve plants are transferred from one speed to another, the pump motors are momentarily deenergized. This condition is not considered natural circulation, since pump rotation still exists.

Action:

Immediate manual scram upon determination that the region has been entered. If entry is an unavoidable, and well known, consequence of an event, early scram initiation is appropriate (see Reference 4).

2.1.2 Exit Region (Region II)

Size:

Operating domain region below 45% rated core flow and above the 80% rod line, and, for those plants licensed for operation at the maximum extended load line (i.e., MELLLA plants), the region below 50% rated core flow and above the 108% rod line, excluding Region I.

Action:

1. Inadvertent/Forced Entry: Immediate exit from region. The region may be exited by control rod insertion or core flow increase. Increasing core flow by either restarting or

upshifting (for FCV plants) a reactor recirculation pump is not an acceptable method of exiting the region.

2. Deliberate Entry: No deliberate entry into the Exit Region is permitted.

2.1.3 Controlled Entry Region (Region III)

Size:
Operating domain region below 40% rated core flow and above the 70% rod line, excluding Regions I and II.

Action:
1. Inadvertent/Forced Entry: Immediate exit from region. The region may be exited by control rod insertion or core flow increase. Increasing core flow by either restarting or upshifting (for FCV plants) a reactor recirculation pump is not an acceptable method of exiting the region.

2. Deliberate Entry: Deliberate entry into the Controlled Entry Region requires compliance with one of the stability controls outlined in Appendix A prior to entry. If adherence to stability controls cannot be maintained or an unintended reactor transient occurs while operating in the Controlled Entry Region, then immediate exit, in the same manner as for Inadvertent/Forced entry, is required.

2.1.4 Forced Entry into Revised-ICA Operating Regions

Other plant procedures may require immediate operator actions that result in entry into these regions to protect fuel integrity or plant equipment. Transient entry into any Revised-ICA operating regions is therefore permitted for the following conditions:

1. Conditions exist which challenge safety or fuel operating limits (e.g., fuel preconditioning violation).

2. Protection of plant equipment whose failure could adversely impact plant safety (e.g., reactor recirculation pump high temperature alarm).

3. Entry into a region (such as the Controlled Entry Region) is required in order to comply with a requirement to exit another region.

Following a forced entry into a Revised-ICA region, the appropriate actions for that region, as described in Section 2.1, should be performed.

2.2 Stability Region Boundaries

The boundaries of the Revised-ICA stability regions of Figure 2.1 do not provide a constant margin to power oscillations. Relative reactor stability margin at the region boundaries is dependent on specific operating conditions (e.g., reactor power shape, core inlet subcooling/feedwater heating, xenon concentration). Caution should be used whenever operating near (i.e., within approximately 5% in rod line or core flow) the region boundary, and it is best to minimize the amount of time spent operating near the stability region. The stability regions are not exact, but rather an approximate region where oscillations have been known (or predicted) to occur. Furthermore, at lower powers and flows or when operating in single loop, the uncertainties in measuring power and flow increase.

2.3 Instability Detection and Operator Response

Conditions conducive to reactor instability are possible even when operating within the guidance of these Revised-ICA's. Operator awareness of this potential when the reactor is in or near the stability regions, and the means to recognize and mitigate any resulting power oscillations are important components of the Revised-ICA's.

2.3.1 Power Oscillation Detection

Operator training/retraining programs should emphasize oscillation recognition. GE SIL 380, Revision 1 provided guidance on oscillation recognition. The most relevant points of this SIL, considering operation under the ICAs and operating experience subsequent to the SIL, are included in this section.

One or more of the following conditions is an indication of reactor instability induced power oscillations when operating in or near the identified regions:

1. A sustained increase in APRM and/or LPRM peak-to-peak signal noise level, reaching two or more times its initial level at reduced core flow conditions (for plants with very low inherent noise, a threshold noise level of approximately 5% is appropriate). Any noticeable increase in noise level warrants closer monitoring of the LPRM signals. Whenever closer monitoring is needed, LPRMs should be selected and monitored from several different areas of the core. This can be done by selecting control rods in each quadrant or octant of the core in a sequential manner. When the control rod is selected, the surrounding LPRMs will be displayed and can be monitored.

   The increased noise occurs with a characteristic period of less than 3 seconds. Periodicity is best observed on the LPRM or reactor period meter. (Note that the period meter does not indicate "oscillation period", but rather the swing of the period meter indicator will occur at the characteristic period.)

2. LPRM and/or APRM upscale and/or downscale annunciators that alarm with a characteristic period of less than 3 seconds.

2.3.2 Power Oscillation Mitigation

Indication of reactor instability induced power oscillation, as described in section 2.3.1, requires an immediate manual scram.

2.4 Procedures and Training

2.4.1 Operator Training

The following topics and issues should be included in licensed operator training:

1. Operators should be trained and retrained to scram the reactor when thermal-hydraulic oscillations are observed. The training should emphasize that a scram is required, even if the magnitude is below 10% on the APRMs and LPRM upscale or downscale alarms have not occurred. Such training minimizes the potential for a safety limit violation should a regional oscillation occur, and is consistent with a proactive reactivity management philosophy.

2. Operators should be trained and retrained on how to recognize thermal hydraulic oscillations.

3. Operators should be trained and retrained on the Revised-ICA stability regions and required operator actions within those regions.

4. Operators should be trained and retrained to recognize that the region boundaries are not an absolute indicator of the potential for instability under all conditions (see Section 2.2).

2.4.2 Operating Procedures

The following actions and instructions should be incorporated into appropriate operating procedures:

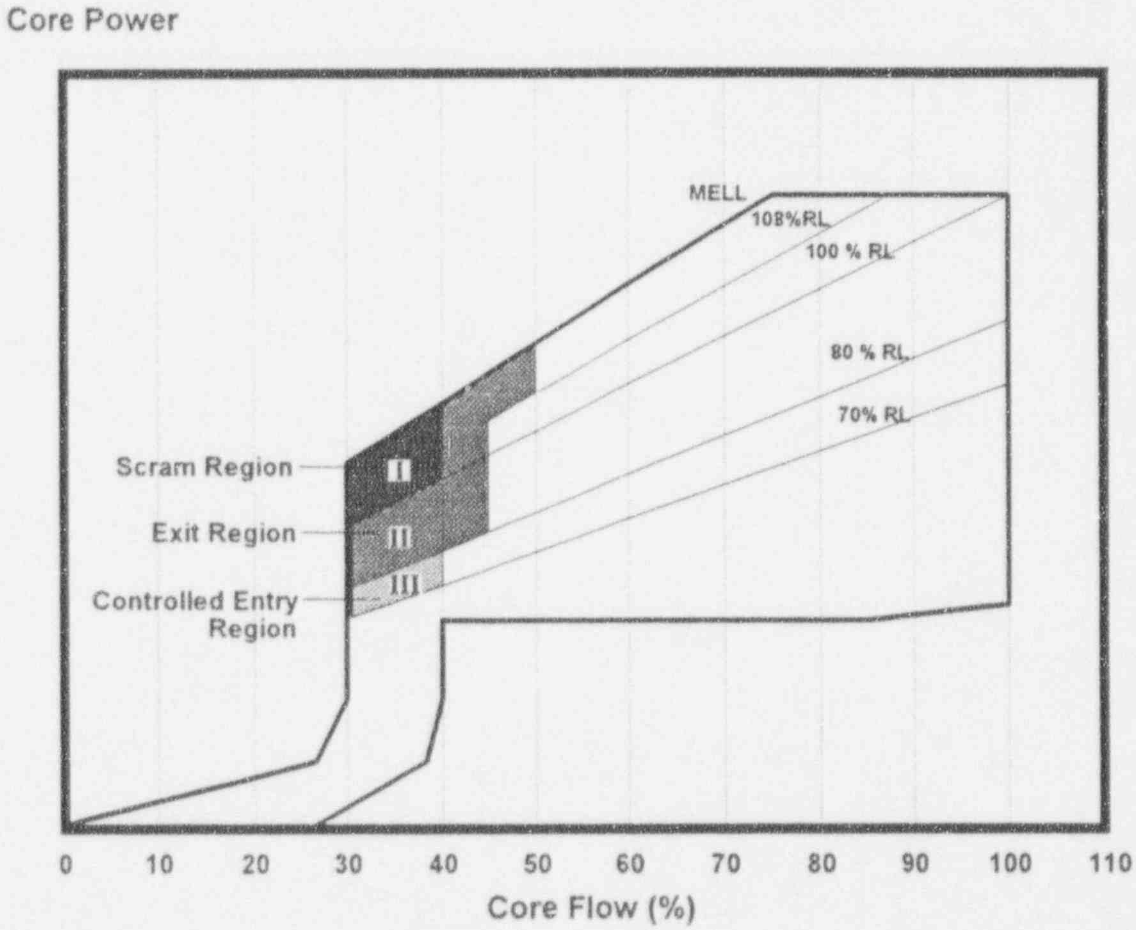
1. ICA Operating Regions described in Section 2.1

2. ICA Operating Region's Actions described in Section 2.1

3. ICA indications of power oscillations and the appropriate response as described in Section 2.3.

4. Guidance for operation near the stability regions as described in Section 2.2.

Figure 2.1



Stability Interim Corrective Actions Operating Domain Regions

Section 3.0 REACTOR ENGINEERING GUIDANCE

Analysis and industry experience have demonstrated that for a given core power and flow condition, the core power shape is the primary factor affecting reactor stability. Flat radial power shapes are more stable than highly peaked radial flux shapes. Axial power shapes which enhance reactor stability can be established by controlling core average boiling boundary. For those plants planning to operate in the controlled entry region, a discussion of recommended controls (including boiling boundary) is provided in Appendices A and B.

Core average boiling boundary encompasses all of the operational factors important to reactor stability and can be used as a parameter for improving reactor stability. However, boiling boundary control is not compatible with existing Technical Specifications or flux shaping practices at some plants, and is therefore not a practical solution in all cases. If this is the case, consideration of parameters such as radial power shape, axial power shape, feedwater temperature, and xenon concentration is suggested when developing plant specific procedural guidance for reactor maneuvering in and near the stability regions (See Section 3.1).

It should be noted that if entry into the Controlled Entry Region is not planned, the guidelines of Appendix A do not apply.

3.1 Guidance for Reactor Maneuvering near the Stability Regions

3.1.1 Radial Power Shape

Radial power shapes with peaks substantially higher than design radial peaking at rated conditions have been demonstrated to reduce reactor stability margin. Control rod patterns which insert peripheral control rods to allow withdrawal of interior rods can lead to radial power shapes that reduce stability. However, the insertion of peripheral control rods for the purpose of Technical Specification operability, testing, fuel leak suppression, etc. is acceptable. Reactor Engineers should consider the stability impact of insertion of peripheral control rods when operating near the stability regions.

3.1.2 Axial Power Shape

Axial power shape has a major influence on reactor stability. Extreme axial power shapes (i.e., very bottom peaked or significantly "double humped") can be destabilizing; however, a simple relationship with axial power peaking, or location of the axial power peak, has not been identified. High core average boiling boundaries, as described in Appendix B, improve reactor stability margin. Reactor engineers should consider the stability impact of the core axial power shape when operating near the stability regions.

3.1.3 Feedwater Temperature

Significant reductions in feedwater heating (i.e., lower than normal feedwater temperature) can reduce the reactor stability margin. Increased awareness should be exercised when operating near the stability regions with reduced feedwater heating which results in lower than normal feedwater temperature for the operating condition.

3.1.4 Xenon Concentration

Operating under low xenon or xenon-free conditions can reduce the stability margin of the reactor by creating unusual axial power shapes; therefore, increased awareness should be exercised when operating near the stability regions with off-normal xenon concentration.

3.2 Guidance for Deliberate Operation in the Controlled Entry Region

Guidance for deliberate operation in the controlled entry region is provided in Appendix A. Some plants can avoid deliberate operation in any of the regions. For these plants, the Controlled Entry Region may be treated as an exit region and the guidelines of Appendix A are not applicable.

3.3 Procedures and Training

3.3.1 Reactor Engineering Training

Reactor Engineers should become familiar with specific plant instability regions and actions, guidance for maneuvering near/into the regions, as well as parameters that can affect stability margin. The following topics and issues should be included in reactor engineering training:

1. For a given core and fuel design and power and flow condition, reactor stability is primarily influenced by the core power shape.

2. The influence of radial power shape on stability.

3. The influence of axial power shape on stability.

4. The influence of core average boiling boundary on stability.

5. How feedwater heating affects reactor stability.

6. How xenon affects reactor stability.

7. How relative reactor stability changes throughout the operating cycle.

8. Differences between regional and core wide instabilities.

9. How to recognize thermal-hydraulic instabilities.

10. Recognition that the region boundaries are not absolute.

3.3.2 Operating Procedures

The following actions and instructions should be incorporated into appropriate reactor maneuvering procedures:

1. ICA Operating Regions described in Section 2.1.

2. ICA Operating Region's Actions described in Section 2.1.

3. Guidance for deliberate operation in the Controlled Entry Region as described in Section 3.2.

4. Guidance for maneuvering near the stability regions as described in Section 3.1.

SECTION 4.0 REFERENCES

1. Letter, DN Grace to A Thadani, "NRC Bulletin No. 88-07, Supplement 1, "Power Oscillations in Boiling Water Reactors'", January 26,1989 (BWROG-8905).

2. Letter, RD Binz to BWR Owners' Group Executive Committee and Primary Representatives, "Implementation Guidance for Stability Interim Corrective Actions", March 18, 1992 (BWROG-92030).

3. Letter, CL Tully to BWR Owners' Group Executive Committee, "Qualitative Guidance on Stability", January 27, 1993 (BWROG-93011).

4. INPO SER 23-93, "Delayed Manual Scram Following a Core Flow Reduction Event and Entry into a Region of Core Instability", October 25,1993.

5 NRC Information Notice 92-74, "Power Oscillations at WNP-2", November 10 1992.

SECTION 5.0 APPENDICES

Appendix A	Controlled Entry Region Operation
Appendix B	Boiling Boundary Implementation and Basis

APPENDIX A

CONTROLLED ENTRY REGION OPERATION

A1.0 Controlled Entry Region Operation

Guidance for deliberate operation in the Controlled Entry Region is provided for plants expecting to operate in this region. Some plants can avoid operation in any of the regions. For these plants, the Controlled Entry Region may be treated as an exit region and the guidelines of this Appendix are not applicable.

A1.1 Guidance for Deliberate Operation in the Controlled Entry Region

Options for stability controls for operation in the Controlled Entry Region are provided in this section.

A1.1.1 Stability Control Option Descriptions

Four primary alternative stability control options have been developed and reviewed. They are defined below, and any of the four may be selected for implementation by an individual utility, based on plant-specific requirements. A fifth option is also discussed which is intended to be used only if a primary option is not available and entry into the Controlled Entry Region is required to perform mandated testing, or to conduct plant startups.

Option 1 specifies absolute limits on core average boiling boundary elevation. Adherence to these limits provides assurance that sufficient stability margin exists to operate within the Controlled Entry Region.

Option 2 specifies operating limits relative to a reference reactor state point that has been demonstrated to be stable (a threshold core average boiling boundary is also specified). Adherence to these restrictions, with appropriate compensation for deviations from the reference state point during the current reactor maneuver, provides assurance that sufficient stability margin exists within the Controlled Entry region.

Option 3 is provided for consideration by those plants which currently have, or are procuring, on-line core stability monitors. Appropriate use of an on-line stability monitor provides assurance that sufficient stability margin exists to operate within the Controlled Entry region without relying on boiling boundary or radial peaking restrictions. Note that inclusion of this option does not imply that plants with stability monitors must adopt Option 3, nor that plants without on-line stability monitors must procure them for the purposes of adopting this option.

Option 4 provides for the capability to pre-analyze or predict stability performance during reactor maneuvers which traverse the Controlled Entry Region. Adherence to pre-analyzed reactor state point conditions calculated to be stable provides sufficient assurance that power oscillations will not occur.

Option 5 provides adequate protection against the effects of power oscillations by supplanting stringent power distribution controls with significant monitoring requirements. This option, which is provided as a backup, has a greater reliance on detection and suppression.

A1.1.2 Plant Specific Implementation of Stability Controls

It is expected that Option 1 can be utilized at many plants. The size of the operating domain region where stability controls will be required is sufficiently small that routine entry during reactor startups and shutdowns can be minimized or completely avoided. If the high boiling boundary can be achieved, Option 1 provides an efficient solution since it is independent of other plant information. It also provides flexibility for reactor maneuvering since the controls are not path or state dependent.

Option 2 is provided for those reactors which must routinely traverse the controlled region during reactor maneuvers, and for whom implementation of the controls in Option 1 is impractical. Adoption of this option can potentially result in less restrictive power distribution limits because they are rendered plant-specific, at the expense of greater restrictions on reactor states and maneuvering trajectories in the controlled region. Significant preplanning and analysis may be required to support this option.

Options 3 and 4 provide stability assurance with tools that are not generally available to utilities. Thus, although they can provide significant protection, general adoption of these options is not expected in the near term.

The power distribution controls in Option 1, 2, 3 and 4 have an equivalent basis in analysis. The controls outlined in each of these options, when adhered to, provide similar assurance that power oscillations will not occur in the controlled region. The controls in Option 5 provide stability protection by relying on continuous monitoring to assure early detection of oscillations.

A1.1.3 Stability Control Option Requirements

Option 1 : **Limits**

Maintain core average Boiling Boundary (BB) $\geq$ 4.0 ft.

Implementation

The core average boiling boundary is calculated using parameters usually available from the core monitoring computer edit. First determine the fraction of core power needed to achieve coolant saturation. Second, add relative powers for each axial node until the fraction determined above is reached. This is the boiling boundary elevation. (The specific formula is in Appendix B.)

Note: Controls for Option 1 can be programmed into core monitoring software for display in a fractional format similar to the current Thermal Limits. In this case, no manual calculations would be necessary to implement this option.

Option 2: **Limits**

1. Maintain core average Boiling Boundary (BB) ≥Reference AND ≥ 3.0 ft.

2. Maintain radial peaking factor (RPF) ≤ Reference.

The reference values are demonstrated to maintain the plant and cycle specific core stable at a particular rod line and core flow. Operation in the controlled region is permitted at or below this rod line and at or above this core flow. Demonstration of stable operation is based on appropriate prior operating experience in the region, or appropriately analyzed values.

Implementation

The limits specified for Option 2 may be monitored as outlined above for Option 1. The boiling boundary is required to be no less than 3 ft. since lower values do not correlate well with stability margin, and therefore will not provide meaningful indication or control. In addition, demonstration of significant stability margin is required since no direct monitoring of stability performance is required or performed in the controlled entry region. This demonstration is the responsibility of the individual owner, and is not prescribed. It is important to consider variations, relative to the reference value, in parameters such as xenon, feedwater temperature, and power distribution above the boiling boundary.

It is recognized that some plants may have difficulty in obtaining the necessary reactor parameters to explicitly calculate the above limits while operating in the controlled entry region; therefore, other methods of implicitly maintaining the required limits can be employed. One such method is to maintain a control rod pattern which has at least as many shallow rods inserted at each reactor state point as the reference trajectory (consistent with the above limits). Feedwater heating must be maintained at a similar value to that of the reference trajectory, and appropriate consideration given to the other parameters indicated above, for this method to be valid. In this manner, an adequate boiling boundary can be established without actually calculating its value.

Option 3: **Limits**

Maintain reactor core decay ratio (DR_{core})<0.6,
as calculated by an on-line core stability monitor.

Implementation

The core stability monitor must be operating continuously during controlled operation in the Controlled Entry Region, with the core decay ratio less than the limit. If the DR_{core} is outside the allowable limits then initiate action to restore stability margins or exit the region. If the stability monitor is not operating then an immediate exit from the region is required unless other stability control options are satisfied.

Option 4 **Limits**

The individual owner will determine appropriate limits for DR_{core} as calculated by a core stability predictor, or by preanalysis of a reactor state trajectory through the Controlled Entry Region.

Implementation

If either the DR_{core} is outside the allowable limits, or the stability predictor is not operating, then immediate exit from the region is required. If preanalysis is used, then any deviation from the reactor state trajectory beyond a predetermined limit requires immediate exit from the Controlled Entry Region.

Option 5: **Limits**

None.

Implementation

Continuous dedicated monitoring of real time control room neutron monitoring instrumentation is required. Monitoring will be performed in accordance with the guidance in Section 2.3 with the requirement for an immediate manual scram upon indication of a reactor instability induced power oscillation. Note that this option is intended to be used only if Options 1-4 are not available (see Section 1.1.1 of this appendix).

APPENDIX B

BOILING BOUNDARY IMPLEMENTATION AND BASIS

B1.0 Reactor Boiling Boundary Stability Control

Controlling power distribution by limiting the core average bulk coolant saturation elevation above a predetermined axial plane provides a simple and effective means to reliably influence the stability of a BWR. The relative insensitivity of the stability control to variations in all major parameters affecting reactor stability assures that core stability can be directly influenced by the control, without concern for variations in these other parameters. This stability control, its phenomenological basis, sensitivity to relevant reactor parameters, and method of implementation are described in this appendix.

B1.1 Stability Control Formulation

B1.1.1 Parameters Affecting Stability

Predicting and controlling reactor stability in an operational setting, where the fuel and core designs are fixed, is challenging because commonly used operational parameters for measuring core thermal-hydraulic and neutronic behavior do not provide sufficient insight into the basic mechanics of reactor stability.

The following is a broad discussion of the stability issue which provides background information for the boiling boundary stability control.

It is observed that:

1: BWR stability performance is dominated by the core void distribution for a given core design.

$$DR_{core} = f\{void\ distribution\}, \qquad \text{(B-1)}$$

where DR_{core} is the core decay ratio.

When a BWR is maneuvered throughout its power-flow operating domain, five global variables can have a significant influence on void distribution: core flow, core power, axial flux shape, radial flux shape, and core coolant inlet subcooling.

DR_{core}, which is influenced by the core void distribution, is therefore related to the following variables:

$$DR_{core} = f\{AP_i, RP_j, P, W, DHS\} \tag{B-2}$$

where: AP_i : axial power shape
RP_j : radial power shape
P : core thermal power
W : core flow
DHS : core inlet subcooling

The usefulness of this equation is severely limited because of the difficulty in defining the relationship between core decay ratio and the other variables. This is especially true of axial power shape since no unique relationship between it and core decay ratio has been demonstrated.

The development of a simple, reliable stability control based on a direct independent assessment of each parameter in equation (B-2) is therefore not feasible. The variables are either interdependent, or their influence on DR_{core} cannot be resolved.

The complexity of the physical process resulted in a search for a parameter which could be easily correlated to core stability characteristics. This led to the finding that it was largely the voided region of the core that determines reactor stability characteristics.

B1.1.2 Axial Power Shape Effects

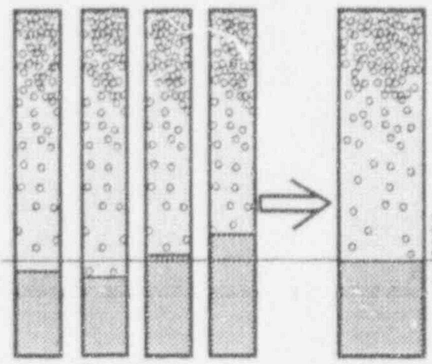
To illustrate the effects of axial power shape, a radially collapsed core, as depicted in Figure B-1, is initially assumed.

For an average fuel channel, equation (B-2) is simplified to:

$$DR_{core} = f\{AP_i, P, W, DHS\} \tag{B-3}$$

The presence of voids in the coolant flowing through the average channel divides the core into two distinct regions: the single-phase region (1ϕ), and the two-phase region (2ϕ). As a first order approximation, subcooled boiling is ignored.

Figure B-1: Core Radial Collapse



The thermal hydraulic feedback is dependent on void sweeping time and core void fraction. Both of these parameters are dependent on the location of the bulk coolant saturation elevation. This elevation determines the two-phase column length which, for a given coolant flow rate (W), defines the void sweeping time and therefore pressure drop feedback phase lag. The location of the bulk coolant boiling boundary, in conjunction with the axial power shape in the two-phase region, also determines the core void fraction for a given reactor state condition (P, W, and DHS). The magnitude of the core void fraction helps determine the feedback gain. Thus, by resolving the location of the core average boiling boundary, the specific effects that the axial power shape has on reactor stability can be elicited.

The neutronic feedback is related to the core void fraction and the axial flux shape in the two-phase region ($AP_i^{2\phi}$). No significant neutronic feedback can occur in the single-phase region because moderator density variations are small. Again, knowledge of the bulk coolant saturation elevation is critical to evaluating this feedback mechanism. Since void reactivity is dependent on local flux squared, $AP_i^{2\phi}$ can have a significant impact on stability margin if axial flux peaks high in the voided region of the core. These concepts lead to the following observation:

***2*: The two phase column length and neutron flux shape in the two-phase region of a reactor core are the major factors influencing reactor stability.**

$$DR_{core} = f\left\{\frac{L_{2\phi}}{L_{1\phi}+L_{2\phi}}, AP_i^{2\phi}\right\}, \qquad \text{(B-4)}$$

where $L_{x\phi}$ is phase column length.

The separation of the 1ϕ and 2ϕ regions of the core is dependent on identifying the average axial bulk coolant saturation elevation, Z_{bb}. On a core-average basis, this boiling boundary is a function of:

$$Z_{bb} = f\{AP_i, P, W, DHS\} \tag{B-5}$$

This leads to the conclusion that AP_i has two distinct impacts on the stability feedback mechanisms. First, the integrated AP_i at the core bottom determines the location of Z_{bb} and thus the 2ϕ column length. Second, the AP_i above Z_{bb} influences the void reactivity feedback:

$$\rho_v = f\{AP_i, i > Z_{bb}\} \tag{B-6}$$

Without knowledge of the location of Z_{bb} (which is not available independent of P,W, and DHS), the impact of axial power shape on each stability feedback mechanism is indeterminate.

The expression that relates core average boiling boundary, Z_{bb}, to the core average parameters important to stability is:

$$\sum_{i=1}^{Z_{bb}} AP_i = C\frac{W \times DHS}{P} \tag{B-7}$$

where C is a constant.

Variations in each parameter of Equation (B-7) result in an appropriate change in the core average boiling boundary as tabulated in Table B-1.

Table B-1: Limiting Changes in Z_{bb}

Parameter Value	Boiling Height
$W << W_{nom}$	$Z_{bb} \rightarrow 0$
$W >> W_{nom}$	$Z_{bb} \rightarrow H_{core}$
$P << P_{nom}$	$Z_{bb} \rightarrow H_{core}$
$P >> P_{nom}$	$Z_{bb} \rightarrow 0$
$DHS << DHS_{nom}$	$Z_{bb} \rightarrow 0$
$DHS >> DHS_{nom}$	$Z_{bb} \rightarrow H_{core}$
AP_i= top peak	$Z_{bb} \rightarrow H_{core}$
AP_i= bottom peak	$Z_{bb} \rightarrow 0$

where: X_{nom} =nominal value
H_{core}= core height

B1.1.3 Radial Power Shape Effects

One variable that can significantly influence stability, but is not captured within the Z_{bb} expression, is the radial power shape, RP_j. This parameter was initially collapsed by performing a radial averaging of the fuel channels. In fact, the boiling boundary of each fuel assembly lies above or below the core average, depending on the assembly's relative thermal hydraulic condition (see Figure B-1). The hot channel boiling boundary, Z_{bb}^{ch}, is usually located below the core average because of its high power output. Therefore, the hot channel is expected to be thermal-hydraulically less stable than an average channel. To identify the parameters important in controlling hot channel stability, the fraction of core average power required for coolant saturation is expressed as follows:

$$f = \frac{DHS \times \frac{W}{N}}{\frac{P}{N}}, \tag{B-8}$$

where N is the number of fuel assemblies in the core.

Define $\overline{w}$ = average channel active flow, and $\overline{p}$ = average channel power, such that:

$$f = \frac{DHS \times \overline{w}}{\overline{p}} \tag{B-9}$$

The fraction of power required for coolant saturation for the hot channel (f_{ch}) can be written as follows:

$$f_{ch} = \frac{DHS_{ch} \times w_{ch}}{p_{ch}} \tag{B-10}$$

Comparing the hot channel power fraction, f_{ch}, to the core average bundle power fraction, f, the following observations can be made:

$$\begin{aligned} &DHS_{ch} = DHS, \\ &p_{ch} = RP_j^{ch} \times \overline{p}, \\ &and \\ &w_{ch} \cong \overline{w}; \end{aligned} \tag{B-11}$$

where RP_j^{ch} is hot channel radial peaking.

The single, most important factor relating the core average to the hot channel power fraction required for saturation is RP_j^{ch}, or:

$$f_{ch} \cong \frac{f}{RP_j^{ch}} \tag{B-12}$$

The axial power shape affects boiling boundary elevation. Hot channels are generally completely uncontrolled, and therefore the hot channel axial power shape, AP_i^{ch}, can be significantly more bottom peaked than the average channel. However, reducing the average power generated at the core bottom will limit the length of the hot channel two phase column length, thus positively affecting the stability performance of the hot channels.

The influence of the high power fuel bundles on the stability of the entire reactor core can be large; therefore, an effective stability control should also limit the hot channel decay ratio, DR_{ch}.

The capability to resolve the influence of core axial power shape on coupled neutronic-thermal hydraulic feedback mechanisms is achieved by dividing the axial flux into two components. These components are defined by the bulk coolant saturation elevation which provides the basis for a reliable, effective stability control. This leads to the following observations:

3: ***If the core average boiling boundary, Z_{bb}, is maintained sufficiently high, then the core will remain stable ($DR_{core} << 1$) during normal reactor operations in regions susceptible to power oscillations.***

and:

4: ***When Z_{bb} is sufficiently high, then variations in other parameters that affect stability will produce only second order effects on DR_{core} and may be ignored.***

Stable reactor performance is assured with a high boiling boundary primarily because of the impact of a short two-phase column on the thermal hydraulic and neutronic feedback mechanisms important to stability. The effect of variations in the two-phase axial power shape cannot render the core unstable at sufficiently high boiling boundaries.

Finally, a high boiling boundary limits the influence of radial power shape, RP_j, on stability. A significantly low integrated axial power in the core bottom is required to generate a high boiling boundary. This low average power in the core bottom limits the hot channel two-phase column length and therefore maintains its relative stability. Highly skewed power shapes could result in low boiling boundaries in parts of the core, but such shapes are not compatible with maintenance of existing fuel thermal limits.

B1.2. Derivation of Stability Control Limit

The Z_{bb} stability control, which utilizes well known core average parameters, can be employed to define the stability limit.

A total bypass flow of 10% of core flow, which corresponds to an active core flow fraction (F_{af}) of 0.9, is generically selected for application to the stability control limit. Assuming that 100% of core power is deposited in the active fuel channel flow (conservative, since actual value is approximately 98%), the fraction of core power (f) required for coolant saturation is:

$$f = F_{af} \frac{W \times DHS}{P} \tag{B-13}$$

where:

F_{af} = active core flow fraction at off-rated conditions (0.9)
W = core flow rate
DHS = core inlet subcooling
P = total core thermal power

or following unit conversion:

$$f = \frac{0.293 \times F_{af} \times W \times DHS}{P}, \tag{B-14}$$

where:

W	in	($10^6 lb_m/hr$)
DHS	in	(BTU/lb)
P	in	(MW_{th}).

The core axial plane where this fraction of core power occurs is dependent upon the average axial power shape. For a core divided into n axial nodes, generating a relative nodal axial power AP_i, the axial power distribution is assumed to be normalized as follows:

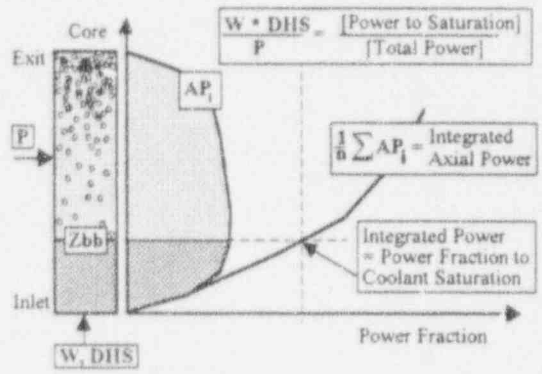
$$\frac{1}{n}\sum_{i=1}^{n} AP_i = 1.0. \tag{B-15}$$

The axial elevation where the integral of the average axial power (from the bottom of the fuel) equals f defines the core average bulk coolant boiling boundary (Z_{bb}):

$$\frac{1}{n}\sum_{i=1}^{Z_{bb}} AP_i \equiv 0.293 \frac{F_{af} \times W \times DHS}{P} \tag{B-16}$$

The relationship of the core average boiling boundary to all core average parameters that are important to stability, is illustrated in Figure B-2.

Figure B-2: Core Average Boiling Boundary



To control the core average boiling boundary during reactor operations, the boiling boundary (Z_{bb}) can be compared to a predetermined minimum elevation limit, $\overline{Z}_{bb}$. This boiling boundary stability control is enforced by requiring the actual boiling boundary (Z_{bb}) to exceed the limit, $\overline{Z}_{bb}$:

$$Z_{bb} \geq \overline{Z}_{bb}. \tag{B-17}$$

This expression is now converted from an elevation limit into a core power fraction limit. Specifically, the core power fraction up to the boiling boundary limit, $\overline{Z}_{bb}$, must be less than the power fraction required for bulk coolant saturation:

$$\frac{1}{n}\sum_{i=1}^{\overline{Z}_{bb}} AP_i \leq 0.293\frac{F_{af} \times W \times DHS}{P}. \tag{B-18}$$

Thus, the power required for coolant saturation must be larger than the actual power generated up to elevation $\overline{Z}_{bb}$ and therefore the boiling boundary will occur, on a core average basis, above $\overline{Z}_{bb}$.

The stability control is now normalized, by defining a limit of Fraction of Core Boiling Boundary (FCBB) as follows:

$$FCBB = \frac{\frac{1}{n}\sum_{i=1}^{\overline{Z}_{bb}} AP_i}{0.293\frac{F_{af} \times W \times DHS}{P}} \tag{B-19}$$

This normalized limit should satisfy the condition:

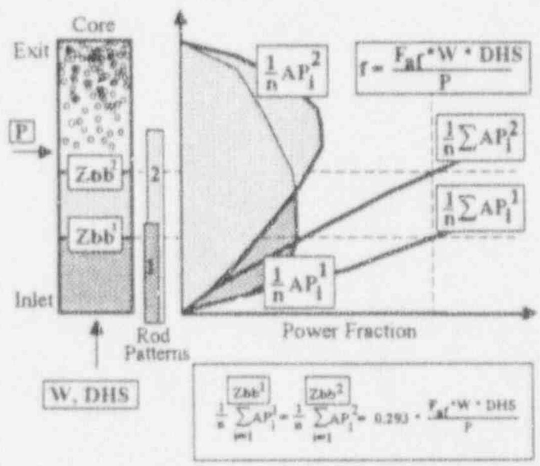
$$\text{FCBB} \le 1.0 \qquad \text{(B-20)}$$

Adherence to the FCBB limit ensures that the actual core average boiling boundary, Z_{bb}, is equal to or higher than $\overline{Z}_{bb}$.

Use of the boiling boundary concept provides a mechanism for operational control of reactor stability and the stability control is readily derived from core average parameters normally available to a reactor operator. The normalized stability control limit, FCBB, can be incorporated into core monitoring software for automatic display to reactor operators.

For example, the change in Z_{bb}, caused by the repositioning of control rods, is reflected in Figure B-3. Control rod pattern 1 represents a bottom peaked power shape with an associated boiling boundary Z^1_{bb} that is assumed to cause FCBB>1.0. To rectify this situation, control rod pattern 2 is adopted. This change raises the boiling boundary to Z^2_{bb}, where $Z^2_{bb} > Z^1_{bb}$, in order that FCBB<1.0. The effect of raising the boiling boundary is a shortened two-phase column length, which improves the reactor stability margin as outlined in section B1.1.2.

Figure B-3: Stability Evaluation of Changes to Core Power Shapes



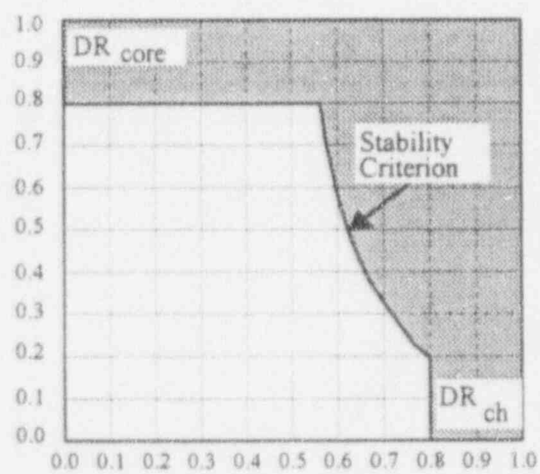
B1.3 Stability Control Analysis

B1.3.1 Background

BWR stability analysis was performed to determine the feasibility and effectiveness of using the core average boiling boundary, Z_{bb}, as a stability control. The primary objective of the analysis was to calculate a core average boiling boundary elevation limit, $\overline{Z}_{bb}$, that meets the following requirements. First, $\overline{Z}_{bb}$ is sufficiently high in the core to provide significant stability margin. Second, the core has a very low sensitivity to all parameters important to stability. Third, adherence to the limit, when appropriate, is operationally feasible for many units. An analysis has been performed for a specific BWR design. Due to the generic nature of the physics and gross reactor systems design of modern BWRs, similar results are expected for all designs.

The generic analysis used both the the core decay ratio (DR_{core}) and hot channel decay ratio (DR_{ch}), and applied the same stability criterion as described in NEDO 31960 "BWROG Long-Term Stability Solutions Licensing Methodology" (see Figure B-4). The stability criterion accounts for susceptibility to the fundamental and higher order harmonic modes of power oscillations.

Figure B-4: Generic Stability Criterion



B1.3.2 Methods and scope of analysis

The stability analysis was performed using a best-estimate frequency domain code. The core and fuel configurations used in the analysis are simulated with a three-dimensional neutronic steady-state code that generates the necessary input to the frequency domain analysis. The frequency domain code utilizes a one-dimensional reactor kinetics model with void and Doppler reactivity feedback based on flux-squared weighted kinetics parameters. A one-dimensional thermal hydraulic model for multiple channel types is also included.

Stable core conditions may be associated with significant axial heterogeneities in void, neutron flux and control rod density distributions in the core. In particular, a high Z_{bb} core configuration is associated with high control rod density in the single-phase region of the core, and a power distribution significantly skewed toward the two-phase region. Voiding is also shifted significantly toward the top of the core. Any stability code used to analyze these configurations must include modeling capability to adequately account for these heterogeneities. Specifically, the void reactivity treatment must include adequate power shape weighting and specific accounting for control density distribution.

A demonstration of the stability control concept was accomplished by varying Z_{bb} at a given core power and flow state point on the boundary of the operating domain region susceptible to instabilities. The core average boiling boundary was moved upward in the core using an incremental insertion of shallow (shaping) control rods. A target limit, $\bar{Z}_{bb}$, was then established that corresponds to a core average boiling boundary associated with significant stability margin.

Next, a sensitivity analysis of parameters and conditions that can affect core stability margin was performed at the target $\bar{Z}_{bb}$. The objective of the sensitivity analysis was to demonstrate that, for the target $\bar{Z}_{bb}$, core, stability performance is insensitive to variations in these parameters and conditions.

The sensitivity analysis was performed at the boundary of the power-flow operating domain region susceptible to instabilities. The analysis considers previously identified parameters that affect core void distribution. Core power and flow define the instability region boundary, and therefore remained constant in this analysis. The parameters for which sensitivity analysis was performed include:

1. Axial flux shape above Z_{bb},
2. Radial flux peaking, and
3. Core inlet subcooling.

In addition, special conditions were investigated, including:

4. 'Hot' radial region, and
5. Cycle depletion effects.

These conditions are associated with operating strategies that can affect stability performance. The hot radial region sensitivity represents situations where non-uniform control rod distributions can create limited, uncontrolled regions in the core that are potentially destabilizing. The cycle depletion sensitivity represents the changing conditions and combinations of parameters that occur throughout the operating cycle.

The sensitivity analysis was performed, in general, at reasonably limiting conditions. This was done to generate higher decay ratios that are more meaningful in assessing the sensitivity analysis results. For example, xenon free conditions that consistently produce higher decay ratios were assumed.

The xenon free state requires a significant increase in control rod density for reactivity control. This can limit the ability to achieve the target $\overline{Z}_{bb}$. Therefore, the use of a xenon free condition also provides a conservative assessment of the boiling boundary control implementation feasibility.

B1.3.3 Analysis Summary

The analysis demonstrates that an effective stability control is achieved by a single control, $\overline{Z}_{bb}$, that consists of a predetermined elevation of the core average boiling boundary. Parametric analysis relates stability performance to the core average boiling boundary. It demonstrates that a $\overline{Z}_{bb}$ limit of 4.0 feet in conjunction with existing fuel thermal limits, provides significant stability margin when applied to reactor state conditions just outside the operating domain region susceptible to instability.

Implementation of this limit ensures that the effects of variations in parameters important to stability become secondary. These include core axial and radial flux shapes and feedwater temperature (i.e., inlet subcooling). In addition, the effectiveness of the boiling boundary limit in ensuring adequate stability control in the presence of extreme radial power-peaked core regions and for varying cycle conditions, has been demonstrated.