GE Nuclear Energy

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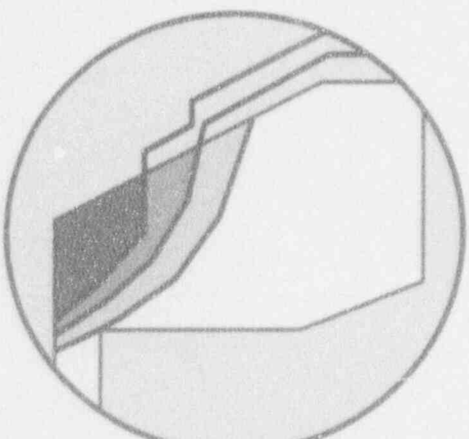
Licensing Topical Report

Reactor Stability Long-Term Solution: Enhanced Option I-A



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Reactor Stability Long-Term Solution: Enhanced Option I-A

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Boiling Water Reactor Owners' Group
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Contents of this Report

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This document was prepared by and for the BWR Owners' Group Enhanced Option I-A Committee. In accordance with established BWROG procedures, use of the information in this report is limited to the participating utilities that have sponsored this activity. The participating utilities of the Enhanced Option I-A Committee are identified below:

Boston Edison Company
Entergy
Northeast Utilities Services Company
PECO Energy

Employees of Entergy and PECO Energy have substantially contributed to this report, including the core boiling boundary stability control described in this document. GE has been informed that these Entergy and PECO Energy employees have applied for a patent on application of core boiling boundary as a stability control. The boiling boundary stability control is included in this document with permission of Entergy and PECO Energy.

A best-estimate frequency domain stability code, which is proprietary to GE, has been utilized in performing analysis in support of the Enhanced Option I-A stability solution. The results from this code are not proprietary to GE and are reported herein. However, the description of this code, which was requested by the NRC, is proprietary to GE and is provided separately as a Class III supplement to NEDO-32339.

TABLE OF CONTENTS

TAPLE OF CONTENTS
(Continued)

TABLE OF CONTENTS
(Continued)

TABLE OF CONTENTS
(Continued)

TABLE OF CONTENTS
(Continued)

TABLE OF CONTENTS
(Continued)

LIST OF TABLES

LIST OF TABLES
(Continued)

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LIST OF FIGURES

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ABSTRACT

BWR cores are susceptible to coupled neutronic/thermal-hydraulic reactor instabilities in certain portions of the core power/flow operating domain. General Design Criterion 12, which requires that established fuel thermal limits not be violated as a result of such instabilities, can be complied with by preventing them altogether. Stability Long-Term Solution Option I-A was developed to provide a methodology for prevention of reactor instabilities. However, several concerns were identified with Option I-A by the NRC.

Enhancements to Option I-A have been developed to address these concerns. A stability control that limits the destabilizing effects of highly skewed core power distributions is defined and demonstrated to assure the conservative nature of the boundary to the region susceptible to reactor instabilities. The use of appropriately qualified stability codes permits meaningful validation of the solution methodology for reasonably limiting events as reflected in the plant initial application and fuel cycle reload review procedures. The result is a process that (1) accounts for all reactor parameters important to stability, (2) can be applied to any fuel design, and (3) defines regions that are insensitive to normal fuel cycle variations in core design. Finally, the concept of defense-in-depth is introduced into the stability solution. An instability detection system, mandated operator actions, and specific changes to reactor trip setpoints provide diverse protection and remove reliance on a single system or methodology.

These enhancements are integrated into a robust and complete stability solution methodology, henceforth referred to as Enhanced Option I-A.

ACRONYMS AND ABBREVIATIONS

AOO	Anticipated Operational Occurrence
APRM	Average Power Range Monitor
ARO	All Rods Out
BWR	Boiling Water Reactor
BWROG	Boiling Water Reactor Owners' Group
CC	Current Cycle
COLR	Core Operating Limits Report
DR	Decay Ratio
DVM	Demonstration Validation Matrix
EFPD	Effective Full Power Days
EOC	End-of-Cycle
EOEC	End-of-Equilibrium Cycle
FCBB	Fraction of Core Boiling Boundary
FCTRC	Flow Control Trip Reference Card
FCV	Flow Control Valve
FRE	Flow Reduction Event
FWHOOS	Feedwater Heater Out of Service
GDC	General Design Criterion
GENE	General Electric Nuclear Energy
IFRE	Intermediate Flow Reduction Event
IVM	Initial Validation Matrix
LCO	Limiting Condition for Operation
LTS	Long-Term Solution
LOFH	Loss of Feedwater Heating
LPRM	Local Power Range Monitor
MCPR	Minimum Critical Power Ratio
N/C	Natural Circulation
NMS	Neutron Monitor System
PBA	Period-Based Algorithm
PBDS	Period-Based Detection System
RC	Reference Cycle
RPS	Reactor Protection System
RVM	Reload Validation Matrix
SC	Standard Cycle
SER	Safety Evaluation Report

DEFINITIONS

Analytical Region Boundary — A set of state conditions, which accounts for setpoint uncertainty, where validation analysis is performed to confirm the nominal region boundary.

Boundary Generation Stability Criteria — A set of core and hot channel decay ratio values that provide the basis for defining stability region boundaries considering both core-wide and regional mode instabilities.

Boundary Validation — The process of confirming the adequacy of a nominal region boundary by performing stability analysis at the corresponding analytical boundary using a best-estimate stability code applied to the Current Cycle design.

Boundary Validation Stability Criterion — A set of decay ratio values that provide the basis for validating stability region boundaries against both core-wide and regional mode instabilities using a best-estimate stability code. This criterion is code specific and incorporates the appropriate calculational uncertainties.

Core Average Boiling Boundary (Z_{bb}) — The elevation in the reactor core at which the core average bulk coolant reaches saturation.

Current Cycle (CC) — The actual reload fuel cycle design that is used in the region boundary validation process to ensure that the existing region boundaries are acceptable for use in the new fuel cycle.

Defense-in-Depth Methodology — A set of solution design features that provides significant diverse protection beyond the licensing methodology from unanticipated and hypothetical precursors to reactor instability. Defense-in-depth methodology features are not required to demonstrate protection of the MCPR Safety Limit.

Demonstration Validation Matrix (DVM) — A set of steady-state and transient conditions used to validate the region boundaries for a demonstration plant. The DVM analysis conditions result from an active search for limiting stability conditions.

Exclusion Region — The area of the licensed core power-flow operating domain where the reactor is susceptible to reactor instabilities.

Exclusion Region Flow Clamp — An upper core flow bound to the Exclusion Region which is generically set at 40% of rated core flow.

Flow-Biased Neutron Flux Scram and Control Rod Block Functions — The flow-biased neutron flux scram and control rod block functions generate reactor trip and control rod block signals that are a function of recirculation drive flow. When the APRM signal is equal to the corresponding core flow-biased trip reference setpoints, the protection function occurs. The APRM signal is representative of core average neutron flux. Some plant applications of the flow-biased neutron flux scram function include a filter of the APRM signal, which is considered to simulate the average core thermal power. However, for the purposes of the Enhanced Option I-A stability solution description, the APRM signal is referred to as neutron flux, regardless of any filtering that may occur.

The flow-biased neutron flux scram function provides automatic protection of the Exclusion Region boundary by immediate insertion of all control rods. The flow-biased neutron flux control rod block function provides automatic protection of the Restricted Region boundary by preventing withdrawal of any control rod.

Flow-Biased Scram Clamp	A clamp on the core flow-biased neutron flux scram function of the neutron monitoring system (NMS) above the highest licensed operating flow-control line in the Restricted Region.
Flow Control Trip Reference Card (FCTRC)	A card in the NMS that generates core flow-biased neutron flux and control rod block trip reference setpoints based on the reactor recirculation drive flow signal.
Hi Decay Ratio Alarm	An optional automatic alarm generated by the PBDS that can be used to indicate a reduction in core stability margin.
Hi-Hi Decay Ratio Alarm	An automatic alarm generated by the PBDS indicating that an unacceptable loss of stability margin has occurred. Immediate manual reactor scram is required upon receipt of the alarm.
Initial Validation Matrix (IVM)	A set of reasonably limiting steady-state and transient conditions used to validate new region boundaries for specific plant application. The IVM is a subset of the DVM where non-limiting DVM state points are excluded.
Licensing Methodology	A set of solution design features that provide automatic protection of the MCPR Safety Limit for anticipated reactor instability events.
Monitored Region	The area of the licensed core power/flow operating domain where the reactor is susceptible to reactor instabilities under conditions exceeding the licensing basis of the current reactor system.
Nominal Region Boundary	Region boundaries that are used to establish actual boundary setpoints and are determined based on the Reference Cycle (RC) design.

Period-Based Algorithm (PBA)
A neutron flux noise analysis technique based on the discrimination of confirmed power oscillation periods in LPRM signals.

Period-Based Detection System (PBDS)
A defense-in-depth feature that uses LPRM detectors and the Period-Based Algorithm to detect reductions in reactor stability margin. The PBDS automatically generates a Hi-Hi Decay Ratio alarm and an optional Hi Decay Ratio alarm.

Power Oscillations
Periodic time varying changes in global or regional core power which are excited by disturbances to the core state parameters that can be random in nature or caused by oscillating control systems. The power oscillations can decay in time, maintain limit cycle behavior, or grow, depending on the stability of the reactor system.

Power oscillations that challenge the MCPR safety limit occur under conditions of reactor coupled neutronic/thermal-hydraulic instability. Under these conditions, core power oscillations are either growing or have converged to limit cycle oscillations due to the non-linearities in the system dynamics.

Power oscillations that are present in a stable reactor decay in time and are not considered to challenge the MCPR safety limit.

Reactor Instability
The condition where power oscillations are either growing or have achieved limit cycle oscillations.

Reasonably Limiting Conditions A set of generically defined conditions that are limiting relative to expected steady-state and transient operating conditions, but are not necessarily bounding. These conditions were established based on validation feasibility analysis for the demonstration plant.

Reference Cycle (RC) A cycle design which is used to determine the nominal stability region boundaries, and which is designed to envelope the stability performance of anticipated future plant-specific fuel cycle designs.

Region Boundary Setpoint Setup A FCTRC feature that sets up the Restricted Region and lower portion of the Exclusion Region boundaries to higher power setpoints to permit required reactor maneuvering in the Restricted Region under controlled conditions.

Region Boundary Setpoint Uncertainty Power and flow uncertainties associated with instrument drift and calibration uncertainty at a stability region boundary.

Reload Validation Matrix (RVM) A set of reasonably limiting steady-state and transient conditions used to validate existing stability region boundaries for a plant-specific fuel cycle. The RVM is a subset of the IVM where non-limiting IVM state points are excluded.

Restricted Region The area of the licensed core power/flow operating domain where the reactor is susceptible to reactor instabilities in the absence of restrictions on core power distributions.

Restricted Region Entry Alarm An automatic alarm generated by the flow-biased control rod withdrawal block function upon inadvertent entry into the Restricted Region.

Stability Control (Fraction of Core Boiling Boundary - FCBB)	A licensing methodology feature required for controlled operation in the Restricted Region. The stability control ensures the validity of the Exclusion Region boundary with respect to core power distributions. Adherence to the FCBB limit maintains the core average boiling boundary greater than a predetermined value above active fuel bottom and provides significant reactor stability margin.
Standard Cycle (SC)	A cycle design that consists of a predetermined, generic fuel design common to all plant applications and a core configuration that is plant-specific. The SC design is used in the initial application process of the Enhanced Option I-A stability solution.

1. INTRODUCTION

1.1 Background

Under certain conditions, boiling water reactors (BWRs) are susceptible to coupled neutronic/thermal-hydraulic instabilities. Such instabilities, which are characterized by periodic core power and hydraulic oscillations, can compromise established fuel safety limits. Operational events and analytical studies have revealed that for most plants, existing neutron monitoring features of the reactor protection system do not assure automatic protection against this class of events. The stability Long-Term Solution (LTS) Option I-A, as described in the Licensing Topical Reports NEDO-31960 (Reference 1) and NEDO-31960 Supplement 1 (Reference 2), was intended to preclude reactor operation under conditions where coupled neutronic/thermal-hydraulic instabilities are possible. In this manner, compliance with GDC-10 and 12 of 10CFR50.55 Appendix A would be demonstrated.

The NRC has reviewed the description of LTS Option I-A documented in References 1 and 2, and provided feedback to the BWROG in the form of an SER (Reference 3) indicating that the general approach adopted in Option I-A was acceptable, with the following exceptions. First, core power distributions, during reactor operation near the Exclusion Region boundary, should be "...consistent with the assumptions of the exclusion boundary analysis...". Second, "...exclusion boundary setpoints should be sufficiently bounding to avoid [routine] changes on a cycle-by-cycle basis." Third, "specific reload confirmation procedures should be developed...[for use during each reload to]...confirm the applicability of old exclusion region settings or set a new exclusion region boundary."

LTS Option I-A originally provided the potential for an efficient and effective solution to the BWR stability issue. However, the NRC feedback described above indicated that the solution required further development. To preserve the instability prevention approach as a means for compliance with GDC-12, the NRC concerns delineated above require resolution.

1.2 Enhancements to Option I-A

The fundamental procedure, based on the FABLE/BYPSS stability code, for generating stability region boundaries was developed as described in Reference 1, and found to be generally acceptable by the NRC. It is therefore appropriate to maintain the process as the foundation for this stability solution. However, the impact of anticipated core power shapes on the FABLE/BYPSS procedure, which incorporates generically defined power distributions, must be examined in light of the issues raised in Reference 3. The effects of anticipated power distributions on an instability prevention solution can be resolved in one of two manners. The size of the reactor power/flow operating region susceptible to instabilities (Exclusion Region) can be increased to accommodate the destabilizing effects of any power shapes that may occur during reactor operation. This approach yields unacceptably large regions that inhibit necessary reactor maneuvering and can cause an increase in the number of unnecessary reactor scrams. Therefore, a better approach is to limit core power distributions near the stability Exclusion Region boundary to those that are consistent with the methodology used to generate that boundary. Development of this approach has resulted in an effective means to control the impact of core power distribution on reactor stability. Application of this control provides reasonable assurance that reactor operations outside the Exclusion Region remain stable.

A new region is defined outside the Exclusion Region (i.e., the Restricted Region) where stability controls are required. The optimized Exclusion Region can be based on the original FABLE/BYPSS methodology because of the stabilizing influence of controlled power distributions that are enforced at its boundary. The application of stability controls results in improved stability performance outside the Exclusion Region and allows optimization of the Exclusion Region size. Taken together, the Exclusion and Restricted Regions form a progressive, layered approach to instability prevention that is consistent with the graduated susceptibility to reactor instability within the reactor operating domain.

The development and confirmation of the new reactor stability control is made possible through the use of a more advanced frequency domain stability code than FABLE/BYPSS. This advanced type of code permits improved analysis of certain core configurations, in particular those which result in limiting stability performance. In addition, the reactor stability control concept can be used not only to generate stable reactor conditions, but also to identify which core power distributions exhibit limiting stability performance. Combined with the use of appropriately qualified stability codes, it permits plant-specific validation of all stability regions to ensure that the stability region setpoints are properly determined. These same tools are used in the initial application and reload review process to generate and validate plant-specific region boundaries.

1.3 Defense-in-Depth

Coupled neutronic/thermal-hydraulic core instability in BWR reactors is a complex phenomenon. Because of the intricate relationship among the many parameters that influence reactor core stability, analysis of realistic reactor state conditions does not yield precise results. Furthermore, the definition of credible events and conditions that yield limiting stability performance is exceedingly difficult and always subjective.

To address this issue, significant defense-in-depth features are incorporated into the solution. A Period-Based Detection System (PBDS) that is based on the Period-Based Algorithm (PBA) described in References 1 and 2 is introduced. This system provides an effective stability detection capability that has significantly faster response characteristics than conventional stability monitors. The stability detection system functions in an operating domain region (i.e., the Monitored Region) that not only encompasses the previously described regions, but also extends out to include all core power and flow states that could hypothetically result in instabilities. In addition, the flow-biased neutron flux scram setpoints are also adjusted downward in certain areas of the operating domain to provide backup protection against unanticipated combinations of limiting transients that may significantly affect stability margin. Finally, uniquely defined operator actions are provided for response to unanticipated situations. The defense-in-depth features are not part of the licensing methodology. They are incorporated into the solution to provide substantial protection from unanticipated reactor state conditions and transients. These features, which exist in a backup role to the licensing methodology, provide additional assurance for prevention of reactor instabilities. All defense-in-depth features are required in the implementation of this solution.

1.4 Implementation

Incorporation of a long-term stability solution into existing operating reactors must accommodate necessary interfaces with installed control systems. Some plants, especially the older generation product-lines, have instrumentation and control systems that although highly reliable, are not designed to current specifications (Class 1E). It is therefore not possible to generically construct an entire stability solution that contains only the latest design configurations for application to all plants. In view of this situation, special features are added to the stability hardware design to compensate for non-Class 1E interfaces with the existing control systems. These features, in concert with the defense-in-depth measures, provide reasonable assurance that anticipated hardware failures will not prevent the stability solution from performing its intended function.

1.5 Summary and Conclusions

Integrating all the licensing and defense-in-depth enhancements into a progressive, multi-region protection scheme provides significantly improved protection against reactor instabilities compared with the initial Option I-A stability solution (Reference 1). The robust nature of these enhancements also provides assurance of substantial protection against all contemplated core instability scenarios. As a result, the enhanced design resolves all concerns raised in the SER (Reference 3), and complies with the requirements of GDC-12.

This document describes in detail the nature of the enhancements to the LTS Option I-A solution described in References 1 and 2, and is henceforth referred to as Enhanced Option I-A. The general features and regions of Enhanced Option I-A are illustrated in Figure 1-1.

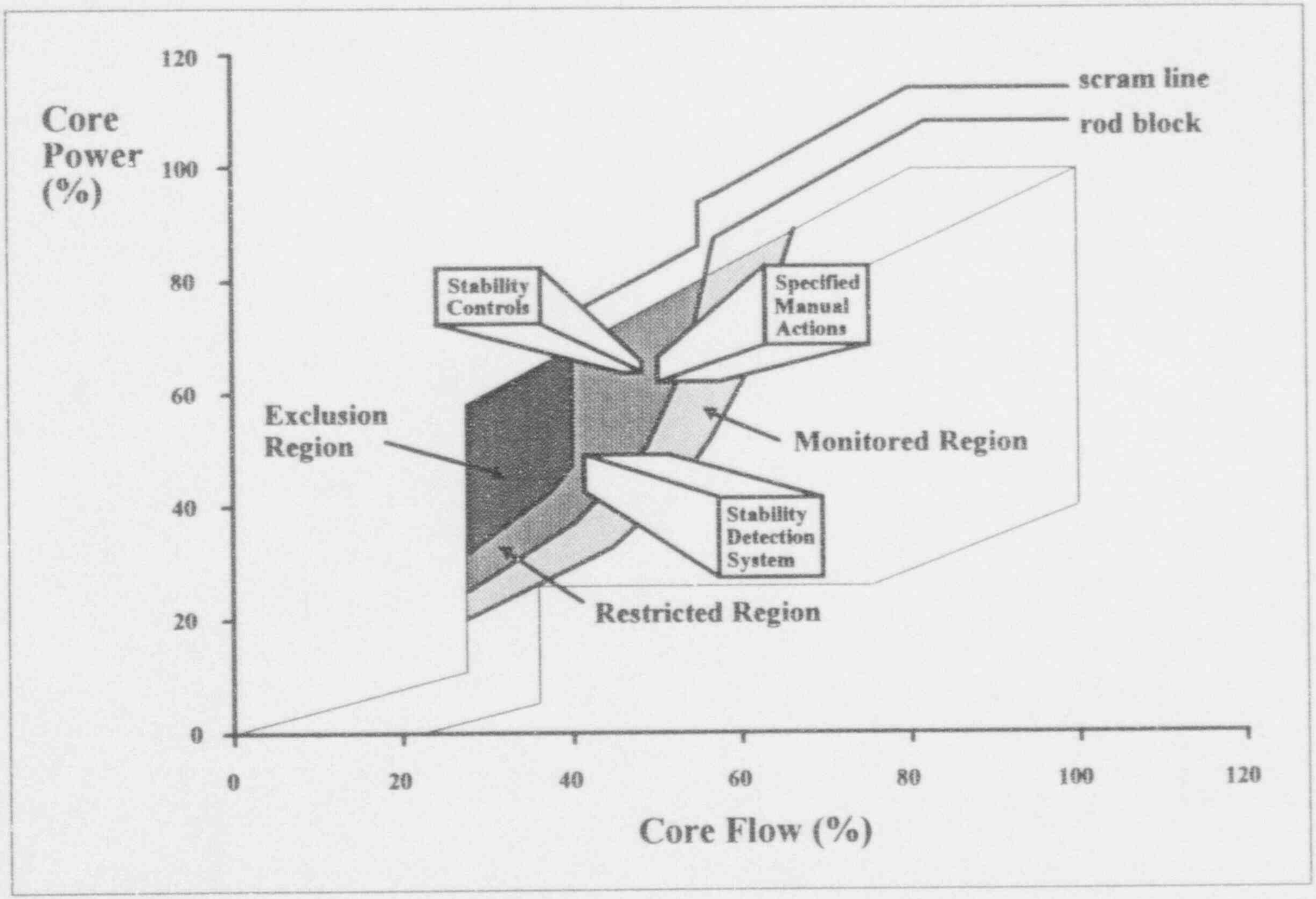
From a licensing methodology perspective, the Exclusion Region encompasses the core power and flow conditions susceptible to reactor instability. This region is therefore excluded from the licensed operating domain, and its boundary is automatically enforced by a flow-biased neutron flux scram. To ensure the validity of this region boundary with respect to the anticipated spectrum of core power distributions, stability controls are required in the Restricted Region. The Restricted Region boundary is automatically enforced by a flow-biased neutron flux control rod withdrawal block. The control rod withdrawal block can be temporarily set up to permit entry into the region if stability controls are enforced.

The defense-in-depth Period-Based Detection System (PBDS) operates inside the Monitored and Restricted Regions. In addition, the flow-biased neutron flux scram setpoint is adjusted downward above the Restricted Region to provide additional protection against unanticipated combinations of transients at high power. Finally, manual actions are specified for a number of conditions and transients that may occur within the Restricted and Monitored Regions, to enhance defense-in-depth. Many of these manual actions are prompted by automatic annunciation of alarms.

This report describes Enhanced Option I-A, which forms a complete solution to the reactor stability issue. The basic design philosophy that underlies all features of Enhanced Option I-A is described in Section 2. A detailed description of the solution is provided in Section 3, followed by discussions of the licensing methodology basis (Section 4) and the defense-in-depth methodology basis (Section 5). The corresponding design configuration for incorporation into the plants is described in Section 6. Initial plant-specific application of Enhanced Option I-A and the fuel cycle reload review processes are defined in Sections 7 and 8, respectively. Finally, a description of the reactor stability control is provided in Section 9. The appendices to this report contain specific procedures, specifications, and supporting analysis necessary to apply the Enhanced Option I-A solution to operating reactors.

Various figures in this report are provided for illustration purposes. In particular, stability region boundaries, validation analysis setpoints and flow-biased setpoints approximate the expected values. Plant-specific boundaries and setpoints will be generated during the stability solution initial application process for each plant. Appendix E contains feasibility analysis results for the demonstration plant. The figures and tables in this appendix reflect the actual analysis state conditions, stability region boundaries and boundary validation results based on the demonstration plant data.

Figure 1-1: Enhanced Option I-A General Features



2. SOLUTION DESIGN PHILOSOPHY

2.1 Systems Integration

Long-term stability solution Enhanced Option I-A takes a preventive approach to compliance with General Design Criterion 12. Instability prevention is desirable because reactor instabilities are avoided altogether and the need to predict transient thermal margin performance is obviated. However, the complexities of reactor coupled neutronic/thermal-hydraulic instabilities make precise analytical results difficult. This inherent difficulty is recognized and addressed in the design philosophy of this solution. In particular, the presence of backup and defense-in-depth features that utilize diverse means of preventing power oscillations eliminates complete reliance on a single system or methodology.

Integration of a stability solution into reactors should result in an improvement to overall reactor safety. A design feature that eliminates the potential for reactor instabilities at the expense of significantly increasing challenges to safety systems through unnecessary scrams is therefore inconsistent with the goals of such a solution. Enhanced Option I-A seeks, in the context of GDC-10 and GDC-12, to appropriately balance the needs for sufficiency of operating stability margin, performance of required reactor maneuvers, and reduction of solution reload dependency, while minimizing unnecessary challenges to safety systems. This is accomplished through comprehensive analysis of instability precursors to establish appropriate safeguards.

2.2 Stability Margin Protection

In addition to the uncertainty inherent in all stability calculations and measurements, susceptibility to reactor instability is a continuous function of core power and flow in the operating domain. In general, there is a gradual increase in the likelihood that instabilities will occur as core power is raised and core flow is lowered.

The Enhanced Option I-A solution design addresses the implications of these characteristics by affording progressively more restrictive operating requirements as susceptibility to reactor instability increases. Automatic prevention is warranted where susceptibility to instabilities is anticipated during normal operations and as a result of moderate frequency events. Protection from instabilities under conditions that are not anticipated or are beyond the existing design bases of the reactor systems should come from combined automatic and manual actions. This approach provides assurance that the degree of protection against instabilities is commensurate with the likelihood of occurrence. In addition, by providing protection against the entire spectrum of instability events, from the likely to the hypothetical, significant margin for future situations and developments is designed into the solution.

2.3 Design Features

The design philosophy of progressive protection is coupled with conservative stability region boundaries and mandated operator actions. This approach provides assurance that reactor instabilities are prevented, considering the complexity of the phenomenon and the attendant analytical uncertainties.

Furthermore, boundaries of the Enhanced Option I-A stability regions are maintained conservative when supplemented by the required actions during normal reactor operation. Not only is consideration of the specific required actions near the boundary of each stability region proper when determining the validity of the boundary location, but credit for such actions is an appropriate means to limit the size of the regions.

Operational experience and analytical results have conclusively demonstrated that for a given reactor design, the manner in which a reactor is operated is the overriding factor in determining stability performance. In recognition of this conclusion, limitations on steady-state reactor operating conditions are incorporated into the solution. Additional diverse operator actions are also mandated for both steady-state and transient operating conditions. The severity of these actions and the constraints on reactor operations in each stability region is consistent with the potential for instabilities to develop. This balanced approach to the issue of reactor stability is a central design philosophy of Enhanced Option I-A.

3. SOLUTION DESCRIPTION

3.1 Licensing Features

Enhanced Option I-A demonstrates compliance with GDC-12 solely through the use of licensing features that ensure reactor coupled neutronic/thermal-hydraulic instabilities will not occur considering reasonably limiting anticipated operating conditions. In this manner, protection of the fuel MCPR safety limit is assured.

3.1.1 Definition of Stability Licensing Regions

3.1.1.1 Exclusion Region (Region I)

The Exclusion Region of Enhanced Option I-A is defined to be that area of the licensed core power/flow operating domain where the reactor is susceptible to coupled neutronic/thermal-hydraulic instability, as illustrated in Figure 3-1. Reactor state conditions that are considered when deriving this region boundary result from steady-state scenarios and events of moderate frequency that are classified as Anticipated Operational Occurrences (AOOs). Since stability controls are applied outside the Exclusion Region boundary, applicable reactor state conditions are consistent with restrictions imposed on core power distributions.

The reactor is automatically protected from operating in this excluded region by the core flow-biased neutron flux reactor trip function of the Neutron Monitoring System (NMS). In effect, the reactor is precluded from operating in states where events of moderate frequency are anticipated to potentially result in unstable conditions.

3.1.1.2 Restricted Region (Region II)

The Restricted Region of Enhanced Option I-A (Figure 3-1) is defined to be that area of the licensed core power/flow operating domain where the reactor is susceptible to coupled neutronic/thermal-hydraulic instability without restrictions on core power distributions. Reactor state conditions that are considered when deriving this region include steady-state scenarios and events of moderate frequency (AOOs) that comply with specified restrictions on appropriate thermal limits, but with no restrictions on core power distribution.

The reactor is automatically protected from unintentional entry into the Restricted Region due to control rod withdrawal by the flow-biased neutron flux control rod withdrawal block function of the Neutron Monitoring System (NMS). This feature effectively increases the size of the Exclusion Region during normal reactor operation. Operation outside of the Restricted Region without specific stability related controls on power distribution is not anticipated to result in reactor instability. Anticipated transients that initiate outside the Restricted Region and terminate inside the Restricted Region are also not expected to result in reactor instability. However, continued operation in the Restricted Region following inadvertent entry is not permitted. The specific requirements to exit the Restricted Region following unintentional entry provide assurance that the reactor does not remain in this region with reduced stability margin. This mandated manual action is provided as a diverse defense-in-depth feature and is not necessary to demonstrate protection of the fuel MCPR safety limit.

Operation in the Restricted Region is permitted when specified administrative stability controls are in place. To facilitate intentional entry into the Restricted Region once stability controls are in place, the control rod withdrawal block and scram functions of the NMS may be temporarily setup as shown in Figures 3-2 and 3-3. The setup function sets up the control rod withdrawal block function to allow reactor operations throughout the licensed operating domain outside of the Exclusion Region. In addition, the bottom of the Exclusion Region boundary is setup by the power difference between the Restricted and Exclusion Regions at natural circulation. This Exclusion Region setup is included to accommodate power spikes associated with upshifting recirculation pumps to high speed. Reactor operation under these setpoint setup conditions does not reduce the necessary degree of protection from reactor instability because the reactor is maintained significantly stable by the stability controls. The setpoints are required to be manually restored to their normal values after the Restricted Region is exited. Conformance to stability controls is required as long as the setpoints are in the setup condition. In addition, the setpoints are automatically restored to normal after a specified core flow is exceeded. The setpoint setup feature permits required reactor maneuvering in the Restricted Region under controlled conditions.

3.1.2 Reactor Stability Control

Intentional operation in the Restricted Region on a continued basis is permitted only when administrative stability controls are in place. Use of stability controls provides significant protection against unacceptable loss of stability margin while operating in the Restricted Region, and assures that the plant continues to operate within analyzed reactor state conditions.

A stability control limit -- Fraction of Core Boiling Boundary (FCBB) -- is defined and applied during intentional operation within the Restricted Region. Adherence to this limit maintains the elevation of core average bulk coolant saturation greater than a predetermined value of 4 feet above active fuel bottom. The limit is normalized consistent with other core thermal limits, and is expressed as the following relationship:

$$\mathrm{FCBB} = \frac{\frac{1}{n}\sum_{i=1}^{\overline{Z}_{bb}=4.0'} AP_i}{0.264\,\frac{W \times DHS}{P}}, \qquad \text{where } \frac{1}{n}\sum_{i=1}^{n} AP_i = 1.0 \qquad (3\text{-}1)$$

and:

$\overline{Z}_{bb}$ = core average boiling boundary limit (4 ft),
AP_i = relative nodal axial power normalized to n,
n = total number of core axial nodes,
W = core flow rate (Mlb_m/hr),
DHS = core inlet subcooling (Btu/lb_m), and
P = core thermal power (MW_t).

In the Restricted Region, steady-state operation requires that the limit satisfy the condition:

$$FCBB \leq 1.0 \qquad (3\text{-}2)$$

Full details of the FCBB limit formulation and supporting analyses are provided in Section 9.

The FCBB control concept limits the core-average two-phase column length and therefore reduces the void sweeping time. This provides inherent core stability. The control incorporates all important core parameters affecting stability, including axial power shape. When implemented, the control demonstrates relative insensitivity to variations in all major parameters affecting reactor stability. This assures that stability can be directly influenced by the FCBB control alone, without concern for variations in these other parameters.

3.1.3 Licensing Features Summary

The Enhanced Option I-A licensing methodology features summarized in Table 3-1 are necessary and sufficient to prevent reactor operations under conditions anticipated to be susceptible to reactor instability. Therefore, the requirements of GDC-12 are satisfied.

3.2 Defense-in-Depth Features

Enhanced Option I-A utilizes the concept of defense-in-depth to improve overall reactor safety. In addition to providing diverse methods and systems to prevent the onset of reactor instability, defense-in-depth gives protection against unanticipated and hypothetical events that can result in an unstable reactor. The defense-in-depth features of this stability solution are not used to demonstrate compliance with the MCPR safety limit.

3.2.1 Definition of Stability Defense-in-Depth Region

3.2.1.1 Monitored Region (Region III)

The Monitored Region of Enhanced Option I-A (Figure 3-4) is defined to be that area of the licensed core power/flow operating domain where the reactor is hypothetically susceptible to coupled neutronic/thermal-hydraulic instability. The extent of the Monitored Region bounds the area of the licensed operating domain where potential for reactor instability exists under any conditions. Therefore, no stability related constraints are necessary in the licensed operating domain outside the Monitored Region.

Continued operation within the Monitored Region requires the presence of an automatic stability detection system. This defense-in-depth feature is provided to preclude reactor instability under unanticipated conditions. When the stability detection system is not operable, continued operation inside the Monitored Region Boundary is not permitted.

Operational requirements of the automatic stability detection system extend into the Restricted Region.

3.2.2 Period-Based Detection System

The instability detection system utilized in Enhanced Option I-A uses the Period-Based Algorithm (PBA) described in NEDO-31960 to detect the onset of power oscillations. This defense-in-depth feature, termed the Period-Based Detection System (PBDS), uses LPRM detectors and provides a completely independent method of ensuring reactor stability. The PBA noise analysis technique is inherently simple and fast. Discrete, well-defined stability margin detection levels are provided based on successive power oscillation period confirmation counts. The PBDS generates alarms when loss of stability margin has occurred, as discussed in Section 3.2.3.3. Following installation, the PBDS will be tuned to the unique conditions present at each reactor to ensure proper performance.

The PBA is based on a discrimination process that examines the neutron flux signal to identify time intervals between successive maxima and successive minima with a period which is characteristic of coupled neutronic/thermal-hydraulic reactor instability. The PBA effectiveness in recognizing the approach to reactor instability relies on the random nature of the LPRM signal signature at stable, low decay ratio conditions.

In certain situations, periodic perturbations can be introduced into the thermal-hydraulic behavior of the reactor system (e.g., from control system feedback). These perturbations can potentially drive the neutron flux to oscillate within a frequency range expected for reactor instability. The presence of these oscillations will be recognized by the PBA as reactor instability independent of the actual stability of the reactor. This situation would render the PBA useless for detecting reductions in stability margin. Therefore, reactors that exhibit power oscillations which lie within the characteristic frequency range, but are not associated with neutronic/thermal-hydraulic instability, cannot rely on the PBDS as an instability detection system. In such cases, the PBDS can be substituted by a different system for detecting the approach to core instability. An example of this type of system is the conventional stability monitor which evaluates core decay ratio based on regression analysis of LPRM signals. Qualification and demonstration of any alternatives or substitutes to the PBDS for plant-specific application of the Enhanced Option I-A solution must be addressed separately.

3.2.3 Defense-in-Depth Automatic Features

3.2.3.1 Restricted Region Entry Alarm

The boundary of the Restricted Region is monitored by the flow-biased control rod withdrawal block function, which provides an automatic alarm upon inadvertent entry into the region (Restricted Region Entry Alarm). This defense-in-depth feature triggers mandated operator actions designed to improve the overall stability protection afforded by this solution. In particular, core flow reduction events (FREs) that terminate in the Restricted Region are automatically indicated to the control room operator by this alarm. The mandated execution of associated operator actions improves the reactor stability margin following a FRE.

3.2.3.2 Flow-Biased Neutron Flux Trip Clamp

Unanticipated combinations of events near the high flow-control line in the Restricted Region (Figure 3-1) may result in large core power increases and losses of stability margin. To provide additional protection from these events, the flow-biased neutron flux reactor trip setpoint above the Restricted Region is adjusted downward (i.e., clamped). Placing the setpoint near the highest operating flow-control line results in termination of these events by automatic reactor scram.

Since this feature is not used in the licensing methodology, explicit setpoint analysis is not performed. Rather, the setpoint is above the highest normal APRM signal value observed during operation on the highest actual flow-control line in the Restricted Region. The highest flow-control line passes through the rated core power/minimum core flow state point in the licensed operating domain.

3.2.3.3 PBDS Alarms

The PBDS generates an alarm when an unacceptable loss of stability margin has occurred (Hi-Hi Decay Ratio). The Hi-Hi Decay Ratio alarm is a required solution feature. It triggers mandated operator action to prevent the onset of unanticipated reactor instability. The PBDS also generates an optional alarm that can be used to indicate reduced core stability margin (Hi Decay Ratio). Because this system is a defense-in-depth feature of the solution and is not used as part of the licensing methodology, explicit demonstration that the Hi-Hi Decay Ratio alarm provides protection for the MCPR safety limit is not necessary.

3.2.3.4 Stability Region Setpoint Setdown

An automatic setdown of the flow-biased neutron flux control rod block and trip setpoints is provided. The setdown function occurs when a specified core flow is exceeded during reactor power ascension. This is a backup feature that provides additional assurance of appropriate setpoint configuration.

3.2.4 Defense-in-Depth Manual Features

Consistent with the progressive, layered approach to instability prevention employed by Enhanced Option I-A, increasingly restrictive mandated operator actions are specified for the Monitored Region and Restricted Region. These manual actions are generally associated or triggered by the automatic defense-in-depth features of the solution.

3.2.4.1 Restricted Region Manual Features

3.2.4.1.1 Uncontrolled Entry into Restricted Region

Immediate action to exit the Restricted Region is required following any type of uncontrolled entry. This operator action is triggered upon receipt of the Restricted Region Entry alarm. Since adherence to FCBB is not assured, continued operation in this region under these conditions is not appropriate.

If the PBDS is not operable under conditions of inadvertent entry, manual scram without delay is required. This action is enforced because operation within the FCBB limit cannot be assured immediately after a transient, and no backup protection to prevent reactor instability is available with the PBDS inoperable. Therefore, the requirement is consistent with the layered approach to instability prevention for continued reactor operation.

Manual scram without delay is required upon receipt of the PBDS Hi-Hi Decay Ratio alarm. Annunciation of this alarm is indicative not only of a loss of acceptable stability margin, but also that an unanticipated condition exists. Manual scram without delay under these situations is appropriate.

3.2.4.1.2 Controlled Operation Inside Restricted Region

Immediate action to exit the Restricted Region is required following initiation of any unplanned transient that occurs while operating in the region. During an unplanned transient, adherence to the FCBB limit cannot be assured. Therefore, continued operation in the Restricted Region is not appropriate.

If the PBDS becomes inoperable during controlled operation in the Restricted Region, exit is required. Although compliance with the FCBB limit is still maintained, no backup protection to prevent reactor instability is available. This requirement ensures that the layered approach to instability prevention is restored prior to continued reactor operation in the Restricted Region.

Manual scram without delay is required upon receipt of the PBDS Hi-Hi Decay Ratio alarm. Annunciation of this alarm is indicative not only of a loss of acceptable stability margin, but also that an unanticipated condition exists. Manual scram without delay under these situations is appropriate.

3.2.4.2 Monitored Region Manual Features

If the PBDS becomes inoperable during operation in the Monitored Region, exit is required. Although reactor instability is not anticipated except under extreme conditions that significantly exceed the licensing basis of the current reactor system, no other designed backup protection is available.

Manual scram without delay is required upon receipt of the PBDS Hi-Hi Decay Ratio alarm. Annunciation of this alarm is indicative not only of a loss of acceptable stability margin, but also that an unanticipated condition exists. Manual scram without delay under these situations is appropriate.

Entry into the Monitored Region with the PBDS inoperable is allowed for limited duration for the purpose of controlled reactor shutdown. Prior to entry into the region for this purpose, and during power decension in the region, adherence to the FCBB limit is required.

3.2.5 Defense-in-Depth Features Summary

The Enhanced Option I-A defense-in-depth methodology features are summarized in Tables 3-2 and 3-3. They provide significant protection from unanticipated and hypothetical precursors to reactor instability and are designed to be progressively more restrictive as the susceptibility to reactor instability increases.

3.3 Solution Summary

Integrating the licensing methodology and defense-in-depth methodology of Enhanced Option I-A yields a progressive, multi-region protection scheme that provides significant protection against reactor instability. The robust nature of these features also provides assurance of substantial protection against all contemplated instability scenarios. As a result, this stability solution design provides robust resolution to all concerns raised in the SER (Reference 3), and complies with the requirements of GDC-12. The full stability solution is depicted in Figure 3-5, and the features are summarized in Tables 3-4 and 3-5.

Table 3-1: Licensing Methodology Functional Requirements

Region:	**Exclusion Region**	**Restricted Region**	
Types of Entry:	**Any Entry**	**Uncontrolled Entry**	**Controlled Entry**
Flow-Biased Scram	Initiates Automatic Scram	N/A	N/A
Flow-Biased Rod Block	N/A	Initiates Rod Block	Enforces Boundary
Stability Controls	N/A	N/A	Required for Rod Block Setup

Table 3-2: Defense-in-Depth Methodology Automatic Features Summary

<table>
<tr><th>Region:</th><th>Exclusion</th><th colspan="2">Restricted</th><th>Monitored</th></tr>
<tr><th>Types of Entry:</th><th>Any</th><th>Uncontrolled</th><th>Controlled</th><th>Any</th></tr>
<tr><td>Restricted Region Entry Alarm</td><td rowspan="4">N/A</td><td></td><td colspan="2">N/A</td></tr>
<tr><td>Period-Based Detection System and Alarm</td><td colspan="3">Alarm to Operator</td></tr>
<tr><td>Stability Regions Setpoint Setdown</td><td colspan="2">Setpoint Setdown After Exit</td><td rowspan="2">N/A</td></tr>
<tr><td>Flow-Biased Scram Clamp Above Restricted Region</td><td colspan="2">Automatic SCRAM</td></tr>
</table>

Table 3-3: Defense-in-Depth Manual Features Summary

Region:	Exclusion	Restricted		Monitored
Type of Entry:	Any	Uncontrolled	Controlled	Any
Rod Block Alarm	N/A	Region EXIT	N/A	N/A
Transient	N/A	Region EXIT	Region EXIT	N/A
PBDS Inoperable	N/A	Manual SCRAM	Region EXIT	Region EXIT
PBDS Hi-Hi Decay Ratio Alarm	N/A	Manual SCRAM	Manual SCRAM	Manual SCRAM

Table 3-4: Enhanced Option I-A Stability Features Summary

Licensing Features
1. Exclusion Region
2. Restricted Region
3. Flow-Biased Trip
4. Flow-Biased Control Rod Block
5. Reactor Stability Controls
6. Stability Regions Setpoint Setup

Defense-in-Depth Features
1. Period-Based Detection System
2. Restricted Region Entry Alarm
3. Flow-Biased Scram Clamp
4. Monitored Region
5. Mandated Operator Actions
6. Stability Regions Setpoint Setdown

Table 3-5: Enhanced Option I-A Stability Protection Summary

Region:	Exclusion	Restricted		Monitored
Type of Entry:	**Any**	**Uncontrolled**	**Controlled**	**Any**
Licensing Methodology Features	Auto SCRAM	N/A	Rod Block & Stability Controls	N/A
Defense-in-Depth Methodology Features	N/A	Auto Scram		N/A
		Conditions: Exceed Flow-Biased Scram Clamp Above Region	Conditions: Exceed Flow-Biased Scram Clamp Above Region	
		Manual Scram		
		Conditions: - Hi-Hi Decay Ratio Alarm - PBDS Inop	Conditions: - Hi-Hi Decay Ratio Alarm	Conditions: - Hi-Hi Decay Ratio Alarm
		Immediate Exit		
		Conditions: - Restricted Region Entry Alarm	Conditions: - PBDS Inop - Transient	Conditions: - PBDS Inop

Figure 3-1: Enhanced Option I-A Licensing Methodology Regions

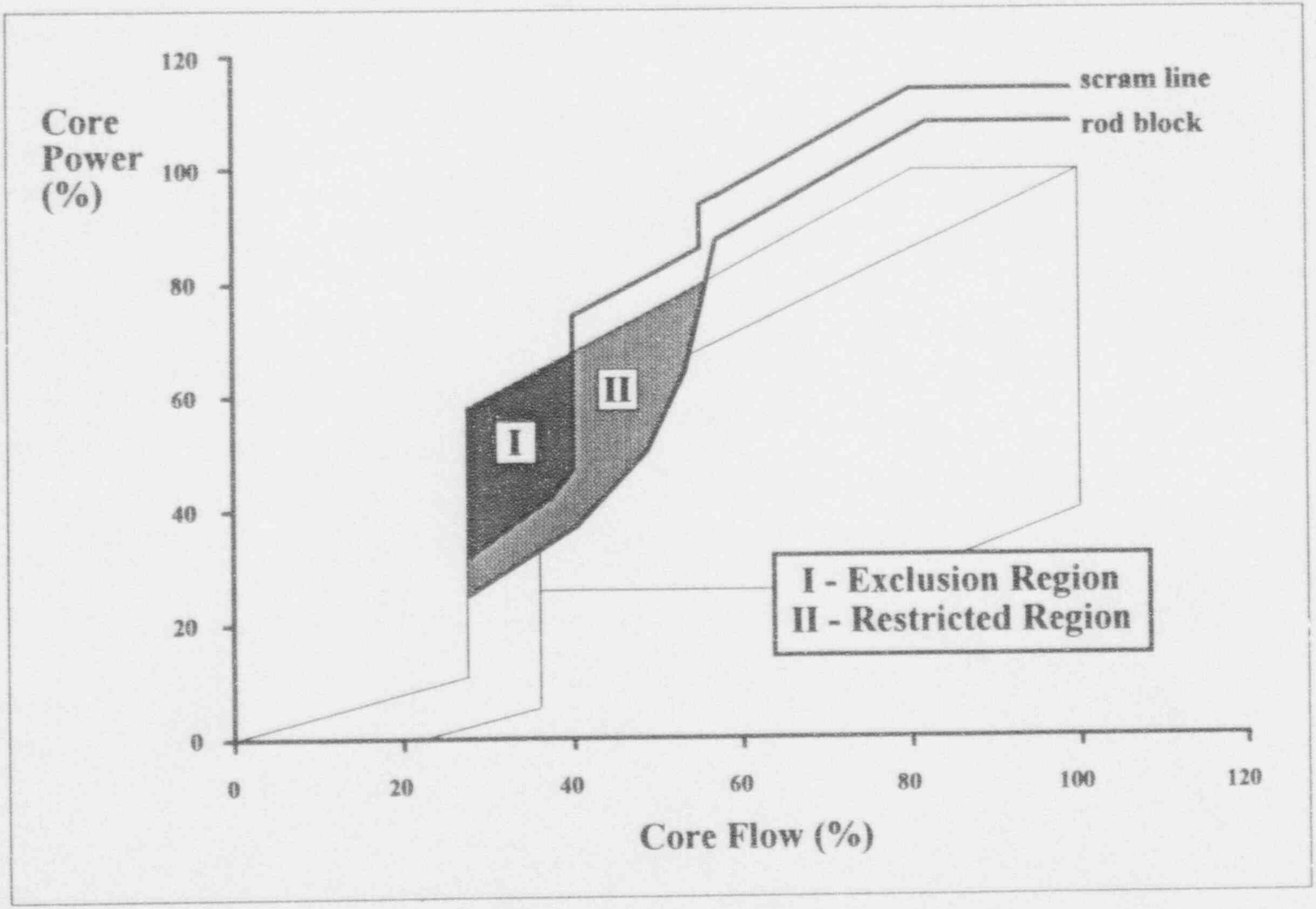


Figure 3-2: Setup Configuration for Flow-Biased Scram Setpoint

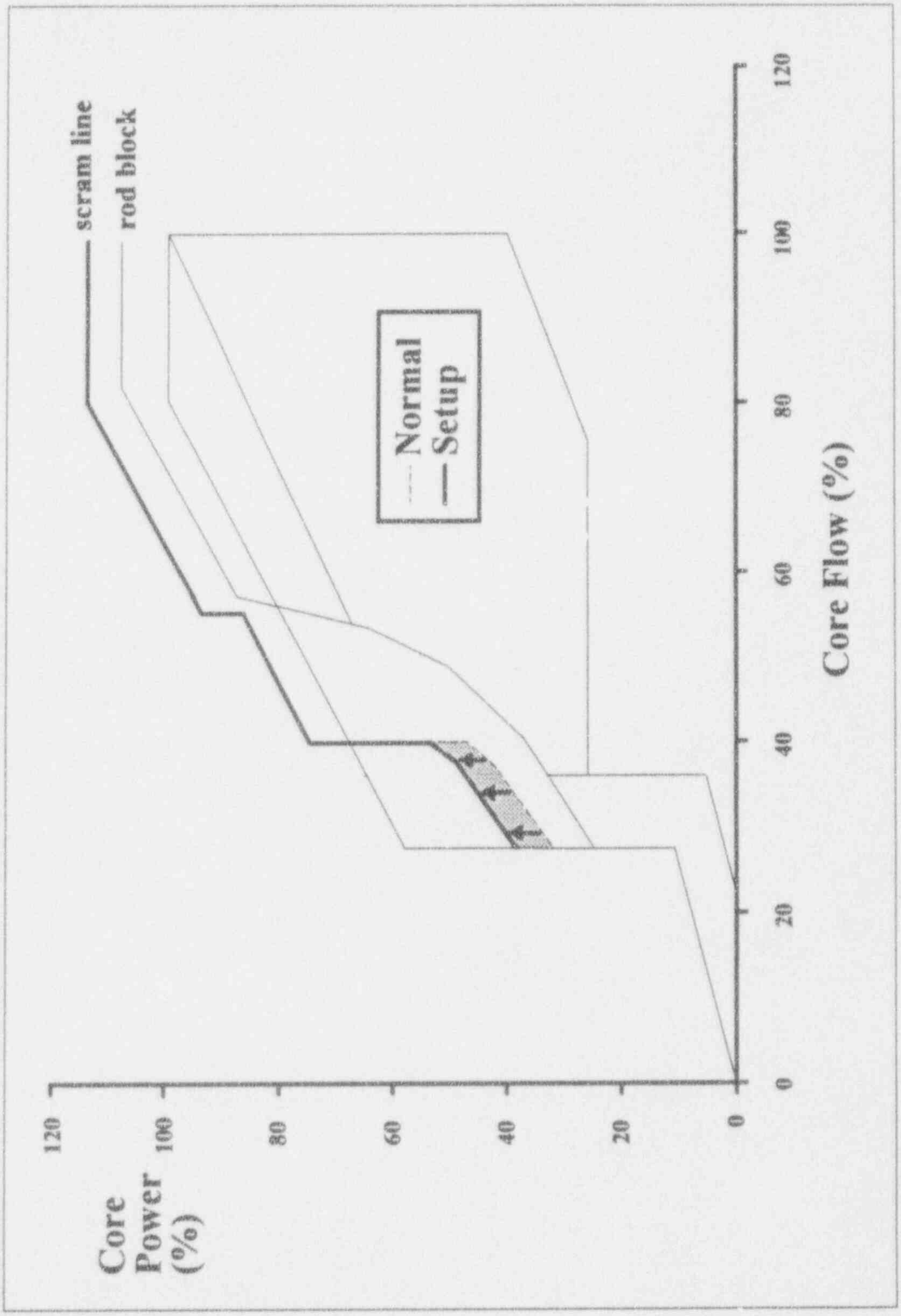


Figure 3-3: Setup Configuration for Flow-Biased Control Rod Block

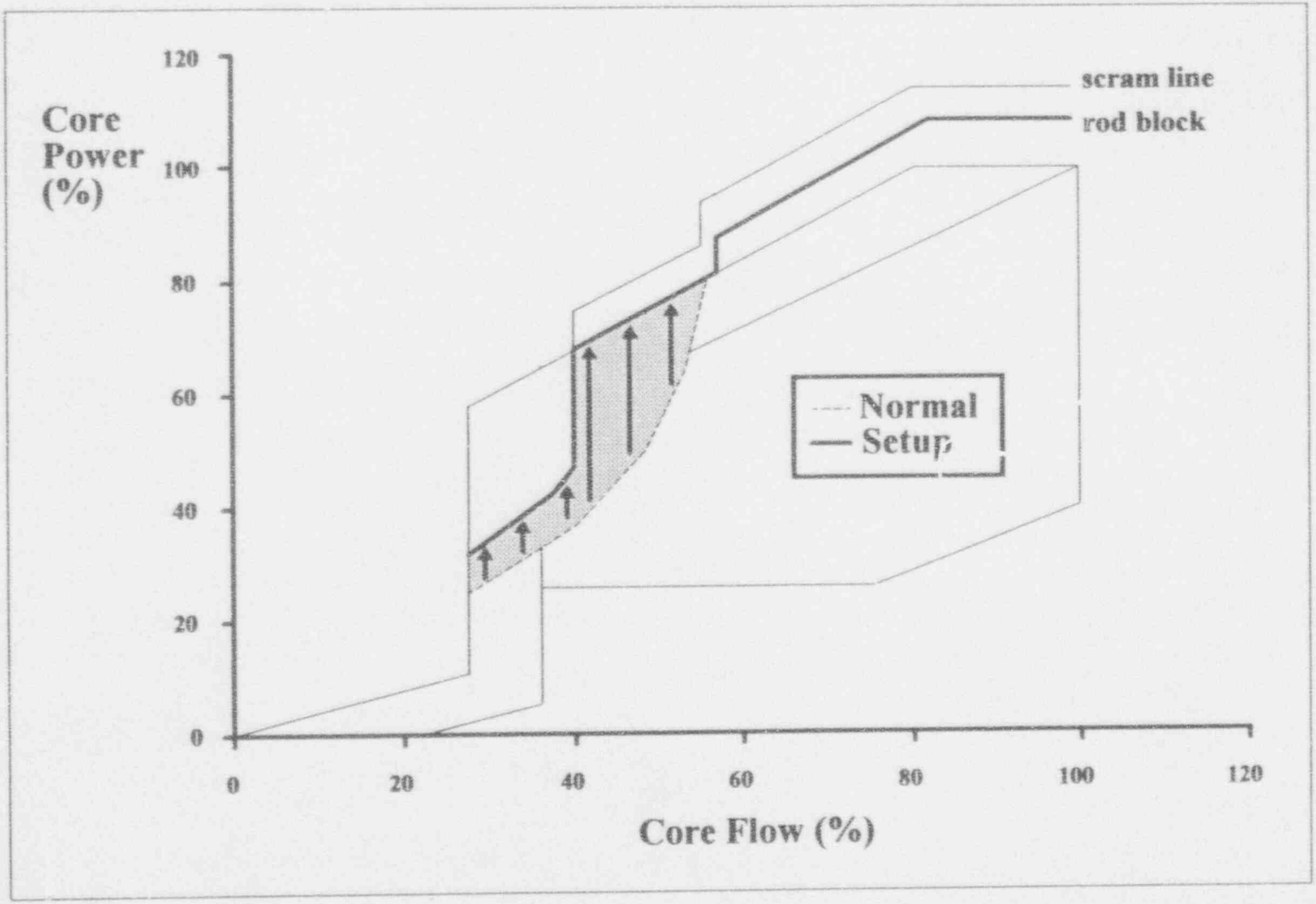


Figure 3-4: Defense-in-Depth Methodology Region

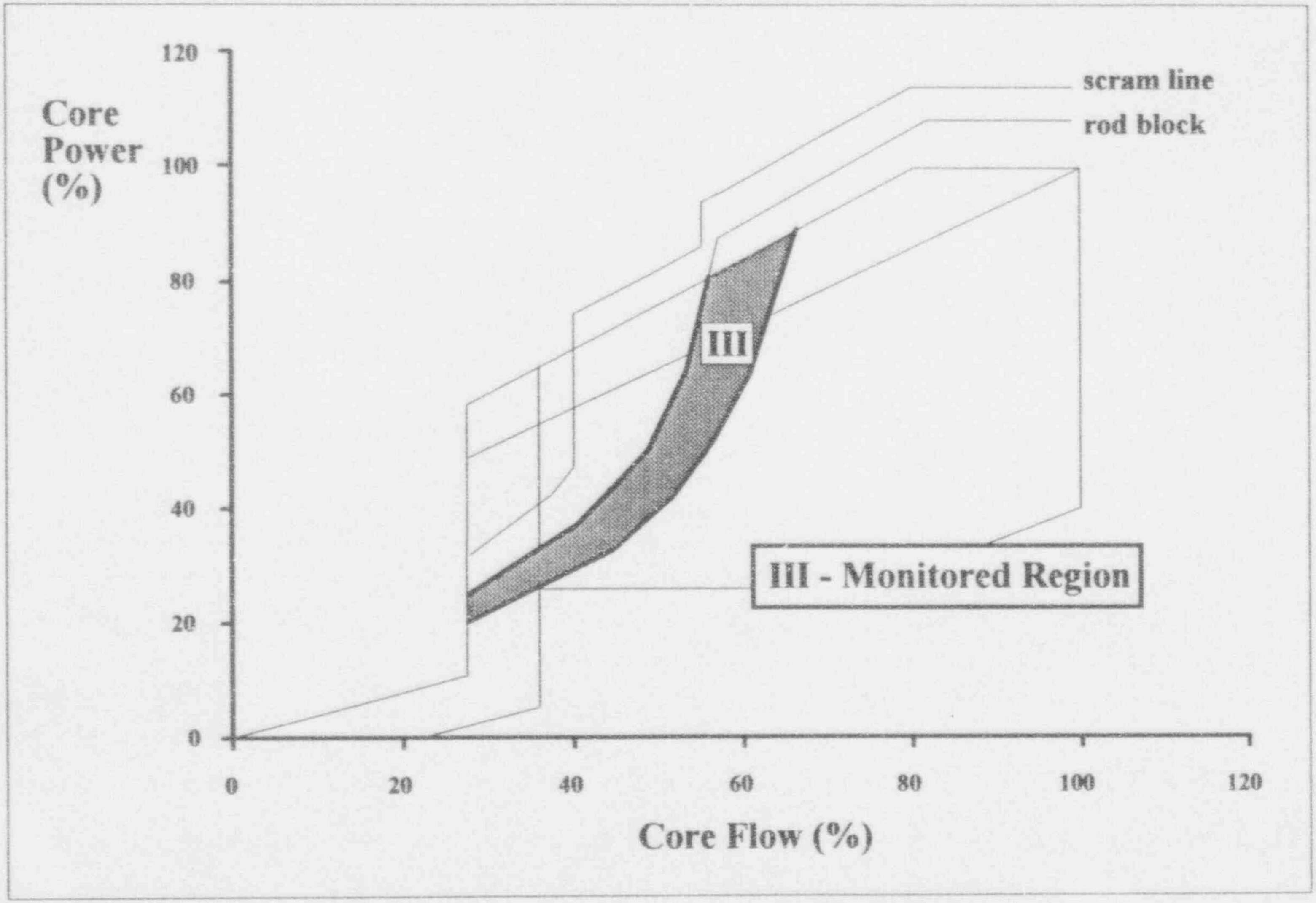
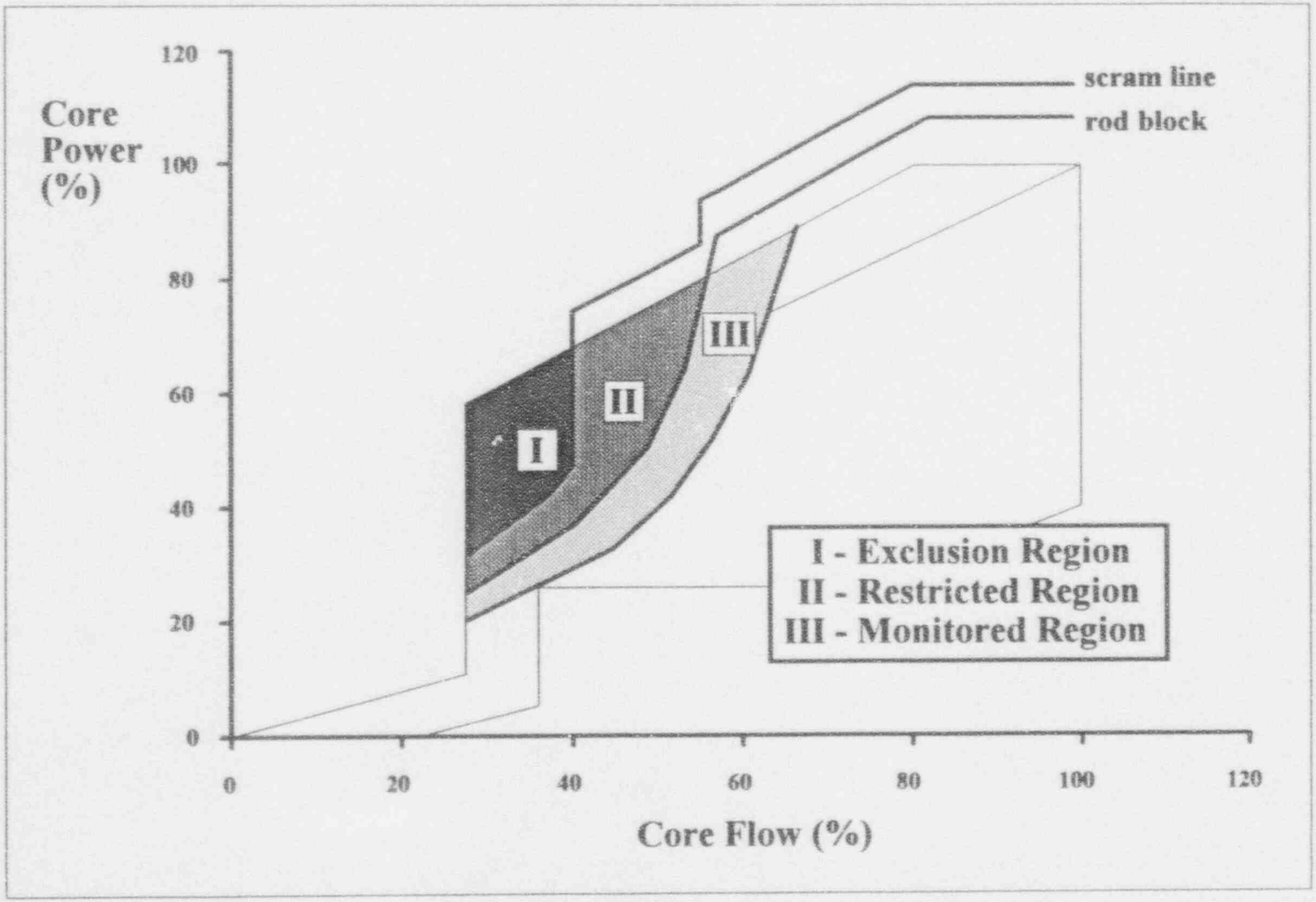


Figure 3-5: Enhanced Option I-A Stability Regions



4. LICENSING METHODOLOGY BASIS

4.1 Approach to Licensing Methodology

4.1.1 Fuel Thermal Safety Limit Protection

General Design Criterion 12 mandates protection of fuel thermal safety limits from conditions caused by coupled neutronic/thermal-hydraulic instability. The existence of significant power oscillations in an unstable reactor generates transient conditions where boiling transition may occur. In these situations, compliance with the MCPR safety limit cannot be assured.

Enhanced Option I-A protects the MCPR safety limit by preventing the occurrence of conditions anticipated to be susceptible to reactor instability in the licensed operating domain. Prevention is accomplished by a combination of two features. First, reactor operation is automatically excluded in a specific region of the licensed operating domain susceptible to reactor instability. Second, stable reactor state conditions are maintained using stability controls where unrestricted steady-state reactor operation may otherwise result in conditions susceptible to instability.

The exclusion function is accomplished by modifying the existing core flow-biased neutron flux scram function of the reactor protection system (RPS) to conform with the boundary of the Exclusion Region. Any event that causes the reactor state trajectory to cross the Exclusion Region boundary results in an automatic scram, thereby preventing conditions susceptible to reactor instabilities.

Adherence to stable reactor state conditions is accomplished by use of the stability control. Analysis demonstrates that application of the stability control in the Restricted Region, which is located just outside the Exclusion Region, assures reactor stability under all anticipated operating conditions. Inadvertent entry into this region during off-rated reactor maneuvering is prevented by modification of the existing flow-biased neutron flux control rod withdrawal block to conform with the boundary of the Restricted Region. Controlled operation inside the

Restricted Region is permitted only with the stability control in place, and is accomplished by setting up the flow-biased neutron flux control rod withdrawal block to conform with the boundary of the Exclusion Region.

4.1.2 Determination of Stability Region Boundaries

The licensing stability regions of Enhanced Option I-A are based on the FABLE/BYPSS procedure methodology described in NEDO-31960 (Reference 1). The FABLE/BYPASS procedure utilizes a combination of conservative and nominal inputs to establish a decay ratio base line. Next, analysis with a properly qualified best-estimate stability code is performed to establish a decay ratio bias correction for application to the FABLE baseline results. The adjusted FABLE baseline data is then evaluated against an established region boundary generation stability criterion which considers regional mode oscillations as well as core-wide mode oscillations to establish the stability region boundaries. The stability region boundaries are used to define nominal region boundary setpoints that properly account for uncertainties considered in the standard setpoint methodology process (i.e., ISA Standard 67.04 and Regulatory Guide 1.105).

4.1.3 Region Boundary Setpoint Validation

Enhanced Option I-A makes use of a properly qualified best-estimate stability code to validate the appropriateness of the nominal region boundaries, considering reasonably limiting events for each plant-specific application. Validation of all licensing methodology setpoints, including the 40% flow clamp of the Exclusion Region, on a plant-specific basis provides significant assurance that the region boundaries are properly established.

Validation of the nominal region boundaries is performed at defined analytical setpoints. The power and flow state conditions of the analytical setpoints are chosen such that when standard setpoint methodology is applied to the analytical setpoints, the result validates the region boundaries used to define the nominal setpoints. This is illustrated in Figure 4-1.

4.2 Stability Regions Licensing Basis

Prevention of conditions susceptible to reactor instability is demonstrated through analysis of conditions that are reasonably limiting precursors to the onset of unstable reactor states. Reasonably limiting state conditions are divided into those that occur during anticipated steady-state reactor operations, and those that occur as a result of anticipated transients.

4.2.1 Protection for Steady-State Operations

Anticipated steady-state and startup core power distributions at the boundary of the Exclusion Region, if not restricted by stability controls, may be more severe than those assumed in the FABLE procedure as the basis for the Exclusion Region boundary definition. This situation, if not addressed, could result in reactor instability outside the Exclusion Region.

To address this issue, a Restricted Region is created immediately outside of the Exclusion Region. The boundary of the Restricted Region is defined such that the onset of reactor instability is not anticipated outside the region. In particular, the boundary is chosen such that severe power shapes, that do not violate existing operating limits and limiting operating practices, are not anticipated to result in reactor instabilities outside the Restricted Region. The Restricted Region is automatically protected against inadvertent manual entry by a control rod withdrawal block.

The existence of the Restricted Region effectively enlarges the Exclusion Region size during steady-state and startup operations as illustrated in Figure 4-2. Control rod withdrawal cannot continue once the region boundary is encountered and operation outside the region is anticipated to remain stable based on analysis. Therefore, operation outside the Restricted Region, with reactor conditions unrestricted by stability controls, will not challenge the MCPR safety limit, since reactor instability is not anticipated.

Intentional reactor operation in the Restricted Region remains consistent with the methodology used to define the Exclusion Region boundary when core power distribution is restricted. Therefore, stability controls are required for operation inside the Restricted Region in order to force core power distributions into more stable configurations than those used as inputs to the FABLE procedure methodology. In this manner, intentional steady-state and startup operations near the Exclusion Region boundary remain bounded by the methodology assumptions.

The stability control, FCBB, incorporated into Enhanced Option I-A is designed to ensure effective management of core power distributions in the Restricted Region. Since FCBB is the only stability control required, it is designed to be effective without additional constraints on reactor operation. This is accomplished by incorporating the following global core parameters important to stability into the FCBB:

- Core Power,
- Core Flow,
- Axial Power Shape, and
- Core Inlet Subcooling.

By appropriate choice of the boiling boundary elevation limit, $\overline{Z}_{bb}$, reactor stability is demonstrated to be insensitive to variations in radial power shape and peaking. Therefore, these parameters may be ignored for stability considerations within the Restricted Region when utilizing the FCBB control.

In addition, when operating at or above $\overline{Z}_{bb}$, reactor stability is insensitive to variations in all the parameters described above. This feature provides significant operational flexibility to maneuver the reactor within the Restricted Region, while also assuring that stable core power distributions are maintained. Analysis demonstrates that as long as fuel thermal limits and reactor system parameters are maintained within licensed limits, use of FCBB is sufficient to assure stable steady-state operation within the Restricted Region under all anticipated operating conditions. The adequacy and effectiveness of FCBB as a stability control is described in Section 9.

To facilitate entry and deliberate operation in the Restricted Region, the automatic control rod withdrawal block function that protects the region boundary may be setup after stability controls are applied. The setup function also adjusts the lower boundary of the Exclusion Region as illustrated in Figure 4-3.

When stability controls are applied, the effective region size susceptible to reactor instability is reduced, and steady-state operation outside the setup Exclusion Region boundary remains stable. Mandated compliance with the stability controls in this situation results in reactor state conditions that are significantly more stable at the Exclusion Region boundary than those required by the region boundary generation stability criterion. When the Restricted Region is exited, the stability region boundaries are setdown to the normal setpoints before removing the stability control requirements, thereby reinstating protection for situations where power distributions are not restricted by stability considerations.

4.2.2 Protection for Limiting Transients

The establishment of the Exclusion and Restricted Regions assures the stability of anticipated terminal reactor state conditions following plant transients. The transients that result in limiting reactor stability conditions are Loss Of Feedwater Heating (LOFH) and core Flow Reduction Events (FREs). Events whose reactor state trajectories would otherwise enter the Exclusion Region terminate with automatic scram at the region boundary. The treatment of events that terminate within the Restricted Region depends upon whether they initiate inside or outside of the Restricted Region.

Because adherence to stability controls results in extremely stable reactor conditions, LOFH or FRE transients that initiate within the Restricted Region (Figure 4-4) and do not enter the Exclusion Region, remain stable. The presence of the stability controls in the Restricted Region makes these transients non-limiting. The LOFH and FRE initial and final conditions are evaluated at the analytical region boundaries.

Limiting transients may also initiate outside the Restricted Region and result in unintentional entry into the region. Limiting transients that initiate outside the region without stability controls in place and do not enter the Exclusion Region are demonstrated to be stable at the events' terminal state condition.

Reasonably limiting LOFH events that initiate at the Restricted Region boundary without stability controls in effect and terminate prior to reaching the Exclusion Region boundary are shown in Figure 4-5. The available stability margin at the initial condition is demonstrated to be sufficient to ensure adequate stability margin at the terminal condition.

The limiting FREs initiate from rated power conditions. FREs that would otherwise terminate at or near natural circulation conditions are not limiting because the events end with an automatic reactor scram upon reaching the Exclusion Region boundary. The location of the Exclusion Region boundary at high flow-control lines is therefore established such that Intermediate Flow Reduction Events (IFREs) which terminate within the Restricted Region (Figure 4-6) are stable.

Anticipated IFREs result in low decay ratios immediately following the core flow reduction due to the presence of rated feedwater temperature and the corresponding relatively low core inlet subcooling. Following the flow reduction (which results in an automatic rod block alarm upon entry to the Restricted Region), the reactor system approaches a new equilibrium feedwater temperature slowly. Analysis and operational experience indicate that this feedwater heating time constant is approximately 5-7 minutes. The IFREs that terminate in the Restricted Region are anticipated to be stable, as supported by operational experience and demonstration plant analysis. In addition, the time constant associated with the reactor inlet subcooling transient allows for the defense-in-depth mandated operator action to exit the Restricted Region following inadvertent entry.

4.3 Stability Regions Boundary Generation

4.3.1 Boundary Generation Process Overview

Establishment of the boundaries that define the stability regions of Enhanced Option I-A is a multi-step generic process that accommodates all fuel and reactor designs, yet remains grounded in the conservative nature of the FABLE procedure methodology. The steps beyond the FABLE procedure of Reference 1 are employed to accommodate, in a generic manner, specific core and fuel design features, as well as permit interface with any qualified best-estimate stability code.

Unique stability region boundaries are determined on a plant-specific basis using this process.

The Exclusion Region boundary is constructed by adjusting the FABLE decay ratio baseline based on a plant-specific cycle design and evaluating the result against the region boundary generation stability criterion of Reference 1. The final region boundary incorporates a clamp at 40% core flow. The core flow clamp optimizes the solution methodology to improve overall reactor safety by avoiding unnecessary challenges to the reactor safety systems from reactor scrams initiated when the reactor is stable.

The Restricted Region boundary is constructed using the same process. To this end, the stability criterion of Reference 1 is expanded for use in the Restricted Region generation. The Restricted Region stability criterion shown in Figure 4-7, which is a function of core decay ratio (DR_{core}) and hot channel decay ratio (DR_{ch}), is significantly more conservative than the Exclusion Region criterion. The stability criterion for the Restricted Region serves only as a tool for generating the location of the Restricted Region boundary, and is not intended to serve as a specification for the behavior of reasonably limiting events at the boundary of the region.

The Enhanced Option I-A methodology does not require demonstration that the specific parameters that serve as inputs to the FABLE procedure are limiting for each application. The FABLE methodology is an overall conservative procedure that serves as the basis for the generation of the stability regions. The adequacy of the region boundaries is validated for reasonably limiting events through the validation process described in Section 4.4.

The boundary generation process results in nominal region boundaries. No increase in the size of the regions generated by this process occurs due to uncertainties in core power and flow. Instead, the validation analysis used to demonstrate that the nominal region boundaries are adequate is performed at the analytical setpoints. Setpoint methodology is applied to the nominal region boundaries to determine the analytical setpoints.

4.3.2 Standard Cycle Design Basis

The first step in generating the nominal stability region boundaries is to develop the Standard Cycle (SC) design. The SC design uses a predetermined, generic fuel design that is common to all plant applications. The SC core configuration contains plant-specific features to accommodate variations in core size and performance. The SC design captures all the unique stability-related features of a particular reactor system design that are independent of the fuel design. The SC design is used to develop a baseline decay ratio that can later be adjusted depending on specific fuel characteristics. The SC design yields baseline core and hot channel decay ratios along a standard high flow-control line and at natural circulation conditions. This baseline forms the basis for further fuel-specific calculations that determine the plant-specific stability region boundaries.

The FABLE methodology is applied to the SC in a consistent manner for all plant applications and yields overall conservative results. By decoupling the fuel design from the reactor design, the reactor-specific fuel designer can use qualified best-estimate stability codes to complete the process independently of the initial SC design performed using FABLE. In addition, future changes in fuel designs can be accommodated without FABLE reanalysis.

The specifications that describe the Standard Cycle fuel and core characteristics are delineated in Appendix C. The neutronic and thermal-hydraulic performance of all current fuel designs are sufficiently similar to the SC fuel design so that significant departures in the relative stability characteristics will not occur. Therefore, the results of the plant and fuel-specific analysis are expected to yield decay ratio values that are relatively close to the SC calculated decay ratios. This provides assurance that deviations from the overall conservative results of the FABLE procedure are minimized.

4.3.3 Reference Cycle Design Basis

The second step in the region boundary generation process is to develop the Reference Cycle (RC) design. The RC design analysis process uses a qualified best-estimate stability code to transition from the SC design, which uses a generic fuel design, to the actual fuel design existing in the plant-specific reactor core. In addition to the specific fuel, the RC design provides a mechanism to account for specific features of reactor and fuel cycle operating practices that affect stability performance. The RC also permits the introduction of design allowances to compensate for potential variations in future designs.

To establish the stability performance of the Reference Cycle, the SC is reanalyzed at preselected core power and flow state conditions using a qualified best-estimate stability code. The same state conditions are then analyzed using the RC. A comparison of the RC and SC analysis results yields a bias correction factor that accounts for the reactor-specific fuel design stability performance. This bias is applied to the SC baseline decay ratio previously established using the FABLE methodology to generate FABLE-based RC decay ratios. Using the RC adjusted decay ratios and the appropriate region boundary generation stability criterion, the Reference Cycle nominal region boundary intercepts are determined.

To generate the stability region boundaries, a generic boundary shape function is applied at the boundary intercepts. The shape function is derived from an analysis of Exclusion Region boundaries that have been explicitly calculated (Appendix F). The shape function conservatively generates region boundaries based on this database.

4.3.4 Exclusion Region Flow Clamp

4.3.4.1 Flow Clamp Basis Overview

The final step in the region boundary generation process is to clamp the Exclusion Region at 40% core flow. Repositioning the upper portion of the Exclusion Region boundary to the 40% flow clamp reduces unnecessary challenges to reactor safety systems and thereby improves overall reactor safety. The flow clamp is generically established based on analysis of the demonstration plant (Appendix E). This analysis has been performed with a best-estimate frequency domain stability code (ODYSY) and confirms that application of the Enhanced Option I-A licensing features to the operating domain region affected by the Exclusion Region flow clamp maintains large stability margins. The resultant region is validated during the boundary validation process of each plant-specific application. Plant operating experience and tests support the assessment that the reactor remains stable under anticipated operating conditions near the Exclusion Region flow clamp.

4.3.4.2 Protection From Steady-State Conditions and LOFH Events

When the Exclusion Region is clamped, the size of the Restricted Region at high flow-control lines is commensurately extended downward to lower flows. In the Enhanced Option I-A stability solution, protection from anticipated events that initiate from within the Restricted Region is provided by the presence of the stability controls. Figures 4-8 and 4-9 display analysis results from the best-estimate frequency domain stability code (ODYSY) for the demonstration plant steady-state and LOFH events at the analytical reactor state point, A_{fr}, corresponding to the most limiting conditions along the core flow clamp. The clamped Exclusion Region boundary setpoint is extremely conservative with respect to events that initiate within the extended Restricted Region because of the mandated reactor stability control which forces the initial reactor power shape into a very stable configuration. The reduction in stability associated with the LOFH event is not large compared with the total margin to instability created by the stability control. The forced core recirculation also contributes to reactor stability at core flows above the Exclusion Region clamp by reducing the void sweep time through the two-phase region of the core. Analysis with stability controls in place demonstrates $DR_{core} \cong 0.1 - 0.2$ and $DR_{ch} \cong 0.0$ near point A_{fr}. The core and hot channel decay ratios for steady-state and LOFH events are compared to the ODYSY boundary validation stability criterion shown in Figure 4-9.

In general, the boundary validation stability criterion is a set of decay ratio values that provide the basis for validating stability region boundaries against both core-wide and regional mode instabilities using a best-estimate stability code. This criterion is code specific and incorporates the appropriate calculational uncertainties.

The Exclusion Region clamp has no effect on the validity of the Restricted Region boundary. As a result, prevention of reactor instability for steady-state conditions and LOFH events that occur at the Restricted Region boundary is not impacted.

The location of the Exclusion Region clamp setpoint, A_{fr}, is therefore not limited by LOFH events and steady-state operation, but is solely defined by IFREs.

4.3.4.3 Protection From Intermediate Flow Reduction Events

Analysis has been performed on the demonstration plant to confirm that establishing the nominal Exclusion Region flow clamp at 40% core flow provides reactor instability prevention protection against reasonably limiting IFREs. The validity of the 40% flow clamp is confirmed for each plant-specific application. The IFRE analysis considers flow reduction events that initiate from rated power and various flow-control lines between 100% and 120%. The initial power distributions for these events are End-of-Cycle (EOC) Haling. The Haling conditions are reasonably conservative since they are associated with low boiling boundary and negative void coefficient. Additionally, some analysis includes Feedwater Heater Out-of-Service (FWHOOS) conditions. The terminal state for analysis of anticipated IFREs is defined to be the immediate post-flow reduction reactor conditions, including the initial rated feedwater temperature and rated equilibrium Xenon concentration.

Figure 4-10 provides a sample of the analysis results performed to investigate anticipated IFREs. The analysis demonstrates that not only are the immediate post-flow reduction state conditions significantly stable, but also that the equilibrium feedwater temperature state conditions are expected to remain stable. The core and hot channel decay ratios calculated by ODYSY are compared to the ODYSY boundary validation stability criterion as shown in Figure 4-10. Additional details regarding IFRE analysis with an Exclusion Region flow clamp are provided in Appendix E.

4.3.4.3.1 Definition of Anticipated Intermediate Flow Reduction Events

Establishment of the anticipated IFRE conditions is important since this event is used to validate the location of the Exclusion Region flow clamp for plant-specific application. However, the ability of the Enhanced Option I-A stability solution to prevent reactor instability is made relatively insensitive to the definition of what constitutes an anticipated IFRE because the strict licensing methodology is only a part of the total stability solution. A broad spectrum of instability prevention features incorporated into Enhanced Option I-A are available for practical mitigation of stability margin reductions associated with IFREs. While many of these features constitute defense-in-depth, a discussion of these features is appropriate here to illustrate how anticipated events and the associated stability protection relate to the entire protective scheme of the Enhanced Option I-A stability solution. By examining this relationship, the appropriateness of the anticipated IFRE can be determined. Definition of what constitutes an anticipated IFRE is supported by an assessment of four factors.

First, the normal core inlet subcooling transient that is associated with any core flow reduction is slow. Actual plant data as well as analytical models confirm that approximately 5-7 minutes elapse before the off-rated equilibrium feedwater temperature is achieved. As described above, the immediate post-flow reduction state conditions are very stable. Since the Enhanced Option I-A defense-in-depth methodology mandates immediate initiation of actions to exit the Restricted Region and the reduction in stability margin occurs slowly during this period, the stable reactor conditions immediately following the flow reduction are expected to be maintained. The process of exiting the Restricted Region is itself stabilizing since the reduction in total core power which occurs as the region is exited improves reactor stability. It is important to reiterate that analysis of anticipated IFREs at equilibrium reactor state conditions indicates that the reactor is expected to remain stable regardless of whether the mandated operator actions are completed.

Second, a completely diverse method of instability prevention is provided that does not rely on the location of the Exclusion Region flow clamp boundary. This defense-in-depth feature, the Period-Based Detection System, is continuously operating to provide automatic indication if an unacceptable loss of stability margin has occurred as a result of an unanticipated event. Upon receipt of an alarm indicating this condition, manual reactor scram without delay is required. The presence of this system reduces the consequences associated with the occurrence of an unanticipated event, and therefore, also reduces the stability solution's sensitivity to the exact conditions associated with an anticipated IFRE.

Third, IFREs that are combined with unanticipated power increases at the post-flow reduction reactor state are terminated by automatic scram from the flow-biased scram clamp above the Restricted Region. A significant part of the reduction in reactor stability during an IFRE is associated with the core inlet subcooling transient. In anticipated IFREs this transient does not completely eliminate stability margin. However, to provide protection from larger unanticipated subcooling transients associated with IFREs initiating from the highest flow-control lines and terminating very near the Exclusion Region flow clamp, the flow-biased scram setpoint above the Restricted Region is clamped. Because LOFH transients raise total core power, this feature provides effective protection from IFREs that are followed by severe subcooling transients.

Fourth, the Exclusion Region flow clamp at 40% core flow is supported by operational experience. For GE BWRs, no IFREs from rated power have resulted in reactor instability. This is significant since hundreds of IFREs, stemming from various types of flow runbacks, flow control valve (FCV) closures, and single recirculation pump trips, have occurred. The only flow reduction event from rated conditions that has resulted in reactor instability was the LaSalle event, which reduced flow to natural circulation conditions (i.e. not an IFRE). All other GE BWR instability events have occurred under startup conditions or due to a LOFH transient. A summary of this data is shown in Figure 4-11. Therefore, operational experience is consistent with the IFRE analysis results and supports a clamp at 40% core flow.

In summary, two conclusions are generated from the previous discussion. First, the licensing features of Enhanced Option I-A that provide protection from anticipated IFREs are validated on a plant-specific basis to demonstrate that reactor instability is prevented. Second, the stability solution is not sensitive to the exact conditions associated with IFREs, in particular the terminal feedwater temperature, because the licensing protection is only one part of the robust design of the solution. These conclusions support a definition of the anticipated IFRE based on a rated initial feedwater temperature for immediate post-flow reduction conditions. Since IFREs constitute the largest fraction of all events that approach the Exclusion Region, optimization of the Exclusion Region size with a flow clamp at 40% core flow significantly reduces the number of unnecessary challenges to safety systems caused by scrams when the reactor is still significantly stable.

4.3.4.3.2 Potential Intermediate Flow Reduction Scenarios

The previous section discusses how Enhanced Option I-A provides protection from reactor instability during IFREs. The manner in which the features of the stability solution function to prevent reactor instability resulting from IFREs is illustrated in Figures 4-12 through 4-15. The scenarios described in the figures illustrate both licensing and defense-in-depth features of Enhanced Option I-A.

Figure 4-12 shows the Enhanced Option I-A solution response to anticipated IFREs that terminate with less than 40% core flow. All IFREs with core flow reductions to less than 40% terminate in the Exclusion Region, where an immediate automatic scram is generated and the onset of any reactor instability is precluded.

Figure 4-13 shows the Enhanced Option I-A solution response to anticipated IFREs that terminate within the Restricted Region. The immediate post-flow reduction reactor conditions are very stable. Upon receipt of the Restricted Region Entry alarm, the operator immediately initiates action to exit the Restricted Region. Because the feedwater temperature transient is slow, and the action of exiting the Restricted Region is in itself stabilizing, the reactor retains significant stability margin. Once the Restricted Region is exited, the reactor is no longer susceptible to reactor instability, regardless of the core power distribution or feedwater temperature.

Figure 4-14 shows the Enhanced Option I-A solution response to an anticipated IFRE into the Restricted Region followed by an unanticipated event that reduces stability margin. This scenario initially proceeds as described above for Figure 4-13, and large stability margin is initially present. At this point, an unanticipated event is postulated to occur. In response to this unexpected erosion in stability margin, the PBDS generates an automatic Hi-Hi Decay Ratio alarm. Manual action to scram the reactor without delay then occurs to terminate the loss in stability margin.

Figure 4-15 shows the Enhanced Option I-A solution response to an anticipated IFRE into the Restricted Region followed by an unanticipated power increase event. The most limiting event of this type is one which initiates from the minimum core flow permitted at rated power. The immediate post-flow reduction state conditions are therefore near the highest flow-control line at the Exclusion Region boundary (40% core flow). Again, this event initially proceeds in a manner similar to that described above for Figure 4-13, and large stability margin is initially present. At this point, an unanticipated core power increase is postulated to occur with an associated loss of stability margin. Prior to the natural termination of the event, the flow-biased scram clamp above the Restricted Region automatically generates a reactor scram. In this manner, the onset of reactor instability is precluded because the core power increase itself places the reactor at the scram setpoint.

4.4 Stability Regions Boundary Validation

4.4.1 Region Validation Process Overview

The stability regions generated using the process described in Section 4.3 are the plant-specific nominal region boundaries. They are used to define the flow-biased neutron flux scram and control rod block nominal setpoints. In order to confirm the adequacy of these nominal setpoints, a validation process is defined that considers anticipated reactor operating scenarios which result in reasonably limiting stability conditions. The analysis is performed at the core power and flow conditions corresponding to the analytical setpoints associated with each stability region. Results of the analysis are considered to validate the corresponding nominal region boundary if the calculated decay ratios, which quantify the susceptability to core wide and regional modes of reactor instability, conform to the boundary validation stability criterion established for the best-estimate stability code used for the analysis.

The limiting events that validate the nominal region boundaries are determined based on examination of all analyses performed for the demonstration plant during development of the validation methodology (see Appendix E for details). The number of cases in the validation set is dependent on whether the initial region boundaries established by the Reference Cycle design are being validated, or the previously validated Reference Cycle design is being re-validated during the course of a reload review process. Analysis is performed for reactor state conditions that are potentially limiting for LOFH events, IFREs, and steady-state operations.

4.4.2 Region Boundary Initial Validation

The initial validation of stability region boundaries established for a particular plant encompasses events that are potentially limiting, based on an assessment of the demonstration plant validation results. This prescribed set of validation analysis conditions, which is common to all initial plant applications, constitutes the Initial Validation Matrix (IVM). Calculations to determine decay ratios are performed with a qualified best-estimate stability code using inputs generated by three-dimensional core simulator calculations at the selected state conditions. The IVM results are compared to the boundary validation stability criterion of the applicable best-estimate stability code. The Enhanced Option I-A region boundaries are validated if the criterion is met. The RC design may include allowance for future variations in fuel cycle stability performance to reduce the potential for changes in the nominal region boundary setpoints.

4.4.2.1 Validation for Steady-State Operations

In order to validate the licensing stability regions for steady-state conditions, reasonably limiting conditions are established at specific locations on each regions' analytical boundary. The reactor operating states which are analyzed conform with fuel operating limits, required stability control, realistic control rod patterns, and power peaking limits contained in plant-specific Technical Specifications. For each analytical region boundary, reasonably limiting analyses are performed at selected state points at natural circulation and on the maximum flow-control line. The analysis at natural circulation is performed with Xenon-free conditions to emulate reasonably limiting startup conditions. The analyzed conditions must result in decay ratios that meet the boundary validation stability criterion of the applicable best-estimate stability code in order to validate the nominal region boundaries.

4.4.2.2 Validation for Intermediate Flow Reduction Events

For immediate flow reduction events, the location of the Exclusion Region nominal boundary is defined such that reasonably limiting initial conditions at rated power result in decay ratios that meet the boundary validation stability criterion at the terminal state conditions on the Exclusion Region analytical boundary. The initial reactor operating states are specified as EOC Haling in order to achieve reasonably limiting axial flux shapes and kinetic response.

Inadvertent entry into Single-Loop Operation (SLO) due to a flow reduction event caused by tripping one reactor recirculation pump does not challenge the integrity of the Exclusion Region boundary. The APRM flow-biased neutron flux trip reference signal, which provides automatic enforcement of the Exclusion Region, uses recirculation drive flow signals as a measure of core flow. Although the uncertainties in measured core flow (from jet pump flows) during SLO increase, the drive flow uncertainty is unchanged. In addition, the signal remains representative of core flow, since the losses associated with parallel recirculation pump operation, which are removed, have a positive effect on total core flow. This compensates for the losses associated with reverse flow through the idle recirculation loop jet pumps. Additional discussion of the stability region setpoint requirements for SLO is provided in Appendix G.

4.4.2.3 Validation for Loss of Feedwater Heating Transients

Loss of Feedwater Heating (LOFH) transients that result in conditions susceptible to reactor instability can initiate either outside or inside the Restricted Region.

4.4.2.3.1 LOFH Transients Initiating Outside the Restricted Region

Operation outside the Restricted Region is not constrained by reactor stability controls. Therefore, relatively severe core power distributions can occur during anticipated reactor operation near the Restricted Region boundary. LOFH transients that initiate under these conditions can challenge reactor stability margin. Therefore, these events are considered for the purposes of validating the location of the Restricted Region boundary.

For LOFH events initiating outside of the Restricted Region, the location of the nominal Restricted Region boundary is defined such that reasonably limiting initial conditions immediately outside the Restricted Region analytical boundary result in decay ratios that meet the boundary validation stability criterion at the terminal state conditions of the event. Because such events increase core power, they are expected to be the limiting event validating the Restricted Region boundary. The initial reactor operating state for these LOFH events is defined as EOC Haling with core flow reduced to the Restricted Region boundary, in order to achieve reasonably limiting axial flux shapes. The magnitude of these LOFH transients is consistent with operational experience for events of this type.

4.4.2.3.2 LOFH Transients Initiating Inside the Restricted Region

Operation within the Restricted Region is constrained by the reactor stability control. Therefore, relatively severe core power distributions cannot occur during anticipated reactor operation near the Exclusion Region boundary. LOFH transients that initiate under these conditions generally do not challenge reactor stability margin.

For LOFH events that initiate inside the Restricted Region, the location of the nominal Exclusion Region boundary is defined such that reasonably limiting events that initiate immediately outside the Exclusion Region analytical boundary result in decay ratios that meet the boundary validation stability criterion. For the purpose of this validation analysis, the flow-biased scram setpoint along the Exclusion Region is assumed to be setup. Because such events initiate with the stability control in place, they are very stable and therefore not part of the IVM analysis. Appendix E provides the demonstration plant validation analysis which supports this conclusion.

4.4.3 Region Boundary Reload Validation

The extent of the validation of existing stability region boundaries during the reload review process is dependent on the significance of any design changes that affect the stability performance for the new fuel cycle. Determination of how to validate the previously established setpoints is based on a defined reload review procedure.

The reload validation of existing stability region boundaries established for a particular plant encompasses all events that are potentially limiting, based on an assessment of the plant-specific initial validation results. This prescribed set of validation analysis conditions, which is common to all plant reload review applications, constitutes the Reload Validation Matrix (RVM). The RVM analysis results are compared to the boundary validation stability criterion of the applicable best-estimate stability code. The Enhanced Option I-A region boundaries are validated if the criterion is met.

In the unlikely event that the RVM analyses do not meet the boundary validation stability criterion, the Reference Cycle design must be re-established. Any margin originally added to the Reference Cycle to accommodate future changes in fuel design and operating practices is reflected in the nominal region boundary setpoints and tends to reduce the likelihood of this situation.

4.4.3.1 Reload Validation Review Criteria Basis

Small deviations in fuel cycle performance that meet the reload review design change criteria do not require RVM analysis. For more significant changes that do not meet the criteria, performance of the RVM analysis is required. Under circumstances where changes in reactor or fuel design invalidate the RC design, a more significant analysis, which is outside the scope of the Reload Validation Matrix, is required. The more extensive analyses may encompass a new Reference Cycle design and the generation of new region boundaries. The Standard Cycle design is considered to remain applicable because the effect of the design change on decay ratio is small and is adequately addressed by the best-estimate stability code. When reactor design modifications significantly alter the stability characteristics of the reactor system, a complete reconstitution of the initial application process, including the Standard Cycle design, may be appropriate. The reanalysis of the Standard Cycle is required only if the effect of the design change on decay ratio is large and a large decay ratio adjustment to the FABLE SC baseline is necessary.

4.4.3.2 Reload Validation Limiting Events

The scope of the RVM analysis is dependent on identification of the limiting events analyzed in the plant-specific IVM. Only one FRE and one LOFH event are analyzed based on the lowest margin to the boundary validation code stability criterion identified by the IVM analysis. Steady-state validation analysis is performed only outside the Restricted Region, since any state points inside the Restricted Region are non-limiting as a result of the required stability control. Analysis at these conditions is sufficient to confirm that the design changes introduced by the new fuel cycle remain bounded by the existing stability region setpoints.

4.4.4 Region Boundary Validation for SLO

The initial application and reload review validation processes reasonably bound limiting conditions during operation with one reactor recirculation loop in service. No additional analysis is required to validate the stability region boundaries for this operating mode. Supporting information is contained in Appendix G.

The SLO operating mode does not alter reactor stability during steady-state operation compared with two-loop operation at the same power/flow state point. Stability controls provide the same level of protection within the Restricted Region. The stability performance for natural circulation conditions is not affected by the choice of recirculation operating modes. In addition, operation in the SLO mode generally requires a reduction in the highest licensed flow-control line. This situation makes events and conditions at the SLO equivalents of points A_{fr} and A' (A^{fr}_{SLO} and A'_{SLO} shown in Figure 4-16) more stable than those evaluated in the validation process conducted for two-loop operation.

The stability margin for reasonably limiting LOFH events that occur when operating in the SLO mode is also not altered compared with similar events during two recirculation loop operation. These LOFH events initiate from steady-state conditions that are unchanged or more stable than the normal validation conditions. Since the change in feedwater temperature for these events is not dependent on recirculation system operating modes, the stability of reasonably limiting LOFH events is bounded by the initial application and reload review validation analyses.

The magnitude of a core flow reduction during a FRE is affected by the recirculation system operating mode. In SLO, the maximum core flow attainable with one recirculation pump in service is approximately one-half of rated core flow. In addition, MEOD operation is not generally permitted in the SLO mode, which reduces the maximum terminal power for any flow reduction event. Therefore, the stability of flow reduction events in SLO is bounded by the initial application and reload review validation analyses for two-loop operation.

4.4.5 Setup Stability Region Boundary Validation

To intentionally enter and operate in the Restricted Region, the flow-biased control rod block and flow-biased trip function setpoints that define the Restricted and Exclusion Region boundaries are placed in the setup conditions. In this situation, the entire Restricted Region is made available for reactor maneuvering with stability controls in place. The setup boundary setpoints are validated as part of the boundary validation process since setpoint setup is assumed for the validation analysis inside the Restricted Region.

At natural circulation, the Restricted Region boundary is setup from point B' to B and the Exclusion Region boundary is setup from point B to B_s, as illustrated in Figure 4-17. Since the control rod block prevents deliberate power increases above B, the steady-state conditions and LOFH validation analysis performed at point B confirms protection during normal setup operations. The presence of stability controls under these conditions assures that the reactor remains stable as confirmed by the validation process. Adherence to the stability controls and the presence of forced circulation also assures significant reactor stability at point A_{fr}, which was demonstrated as part of the initial application (refer to Figure 4-8).

FREs are non-limiting under setup conditions. Prior to setting up the stability region boundaries for entry into the Restricted Region, power distributions are manipulated to meet the stability control requirement. This process makes the reactor very stable when the region boundaries are initially setup. The limiting FRE for stability region boundary setup conditions corresponds to the largest possible flow reduction that terminates at the highest flow-control line without resulting in automatic scram, and is defined to be the event initiating at point A_s and terminating at point B_s (Figure 4-17).

The core average boiling boundary is relatively insensitive to reactor state condition variations along a constant flow-control line that result from changes in core flow. Therefore, the FRE terminal conditions at point B_s retain a high core average boiling boundary. In addition, analysis of steady-state conditions with stability controls in place along the natural circulation line near point B shows considerable stability margin. Since the terminal state conditions for these FREs is near point B (B_s) and have a high boiling boundary, they also retain considerable stability margin.

Flow reduction events from outside the Restricted Region (i.e., initial core flow above state point A_s) are addressed the analysis for normal setpoint conditions, because they initiate prior to setting up the stability region boundaries.

Based on this assessment of the relationship between the stability performance of reasonably limiting events for operation under stability region boundary setup conditions, and operation under normal setpoint conditions, validation of reactor stability when operating in the setup condition is addressed as part of the initial application and reload review validation analyses.

Figure 4-1: Region Boundaries and Setpoint Methodology Uncertainty

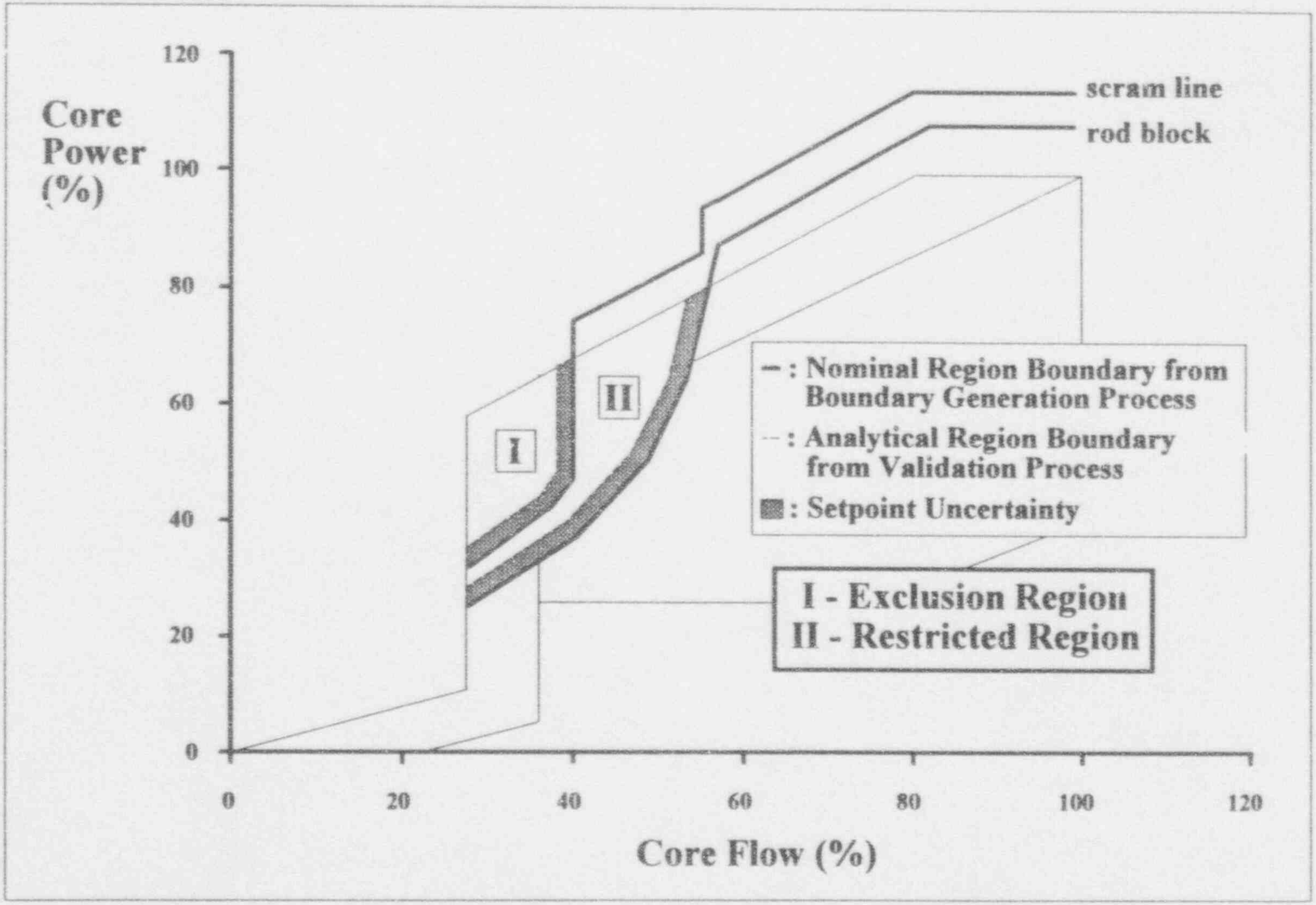


Figure 4-2: Effective Exclusion Region for Unrestricted Steady-State Operation

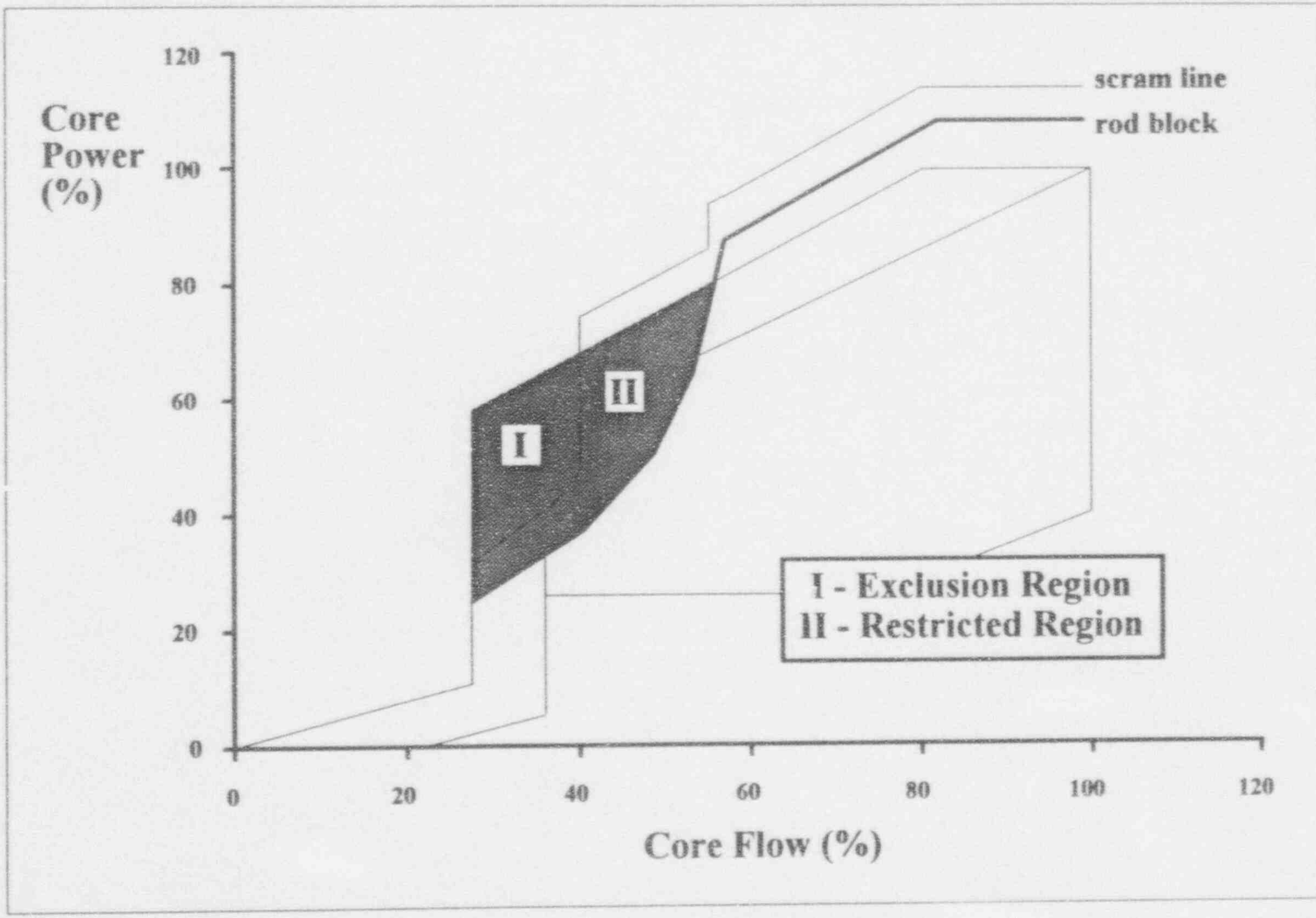


Figure 4-3: Setup Region Boundaries for Steady-State Operation with Stability Controls

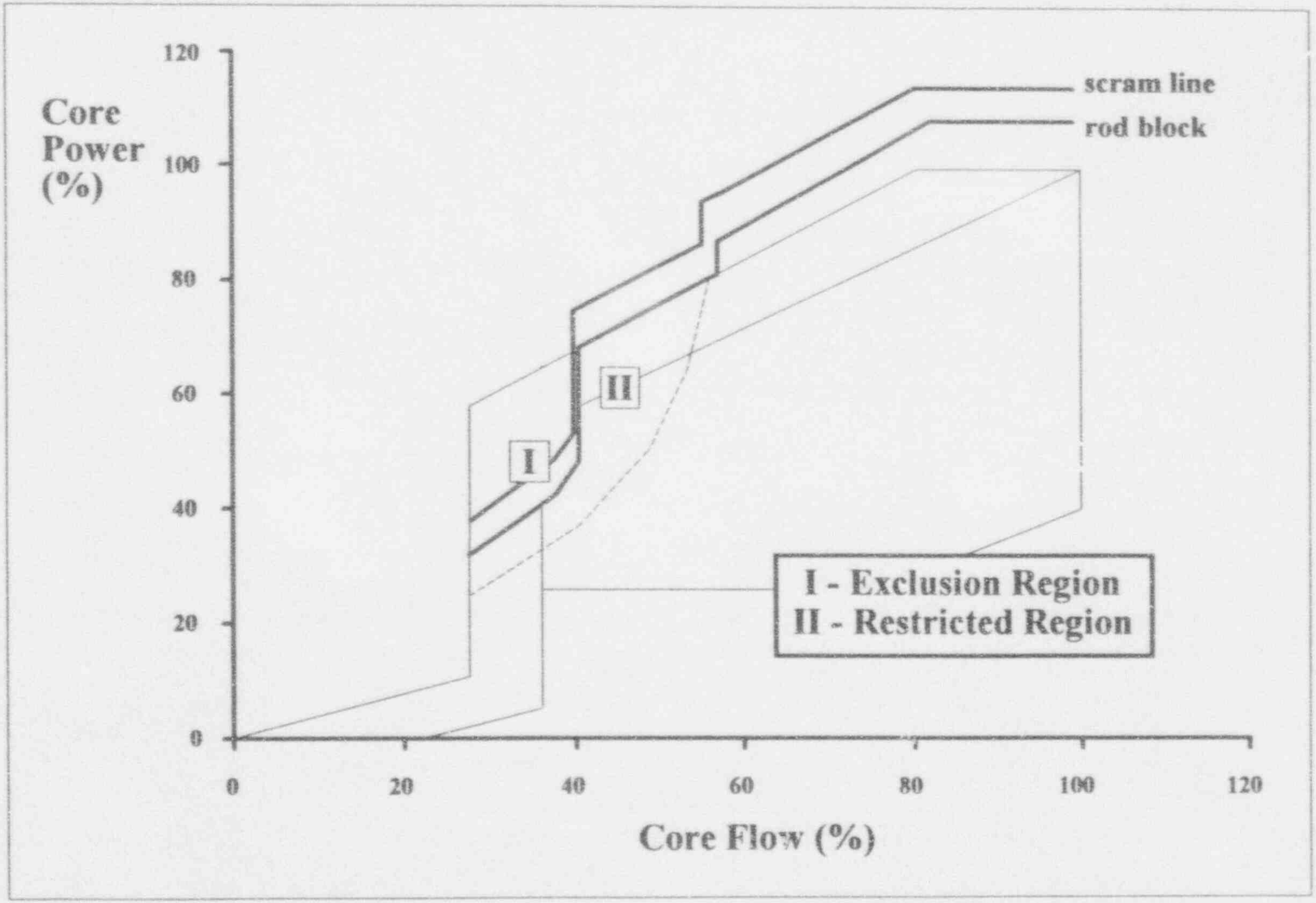


Figure 4-4: LOFH and FRE Transients From Inside Restricted Region

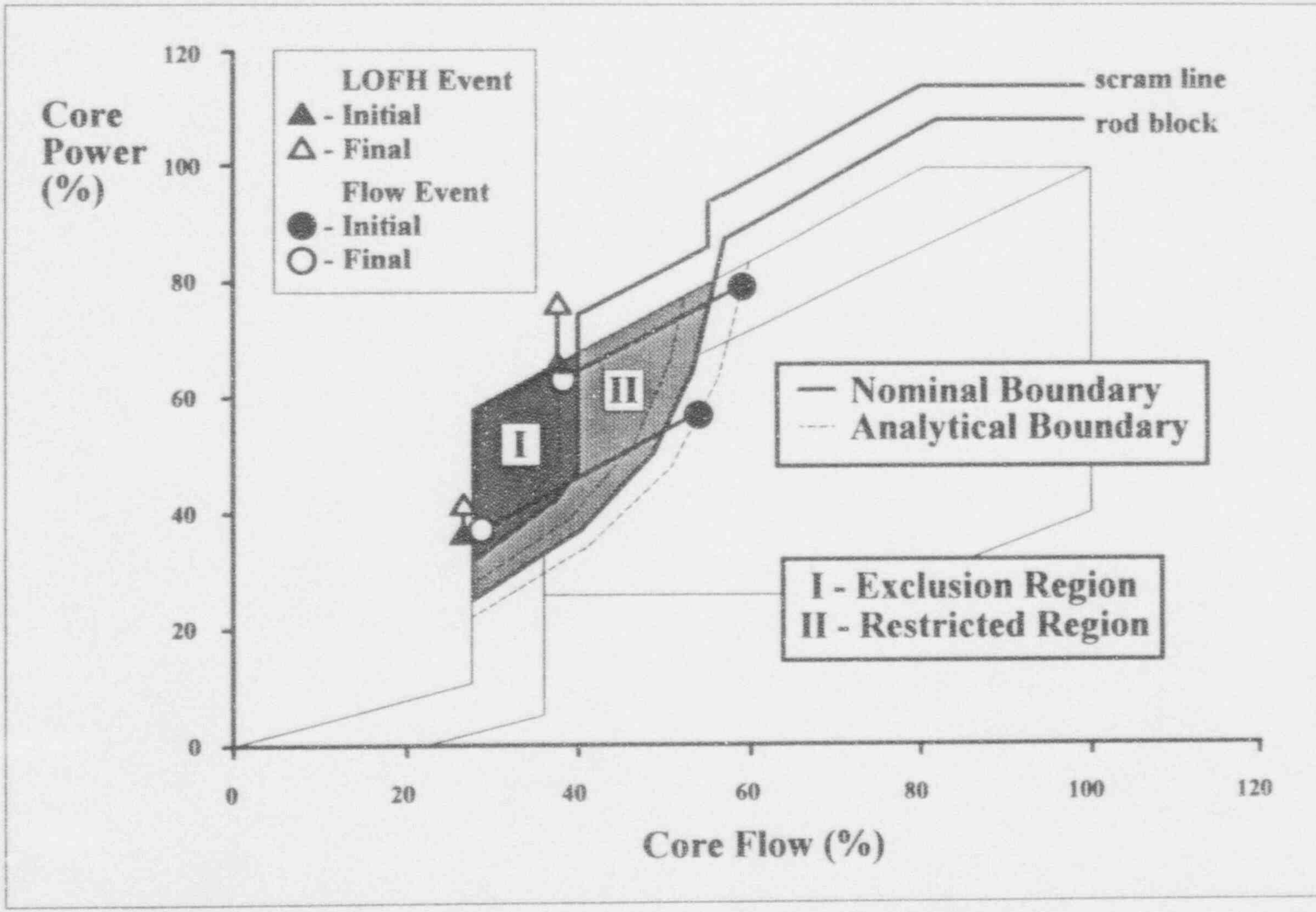


Figure 4-5: LOFH Transients From Outside Restricted Region

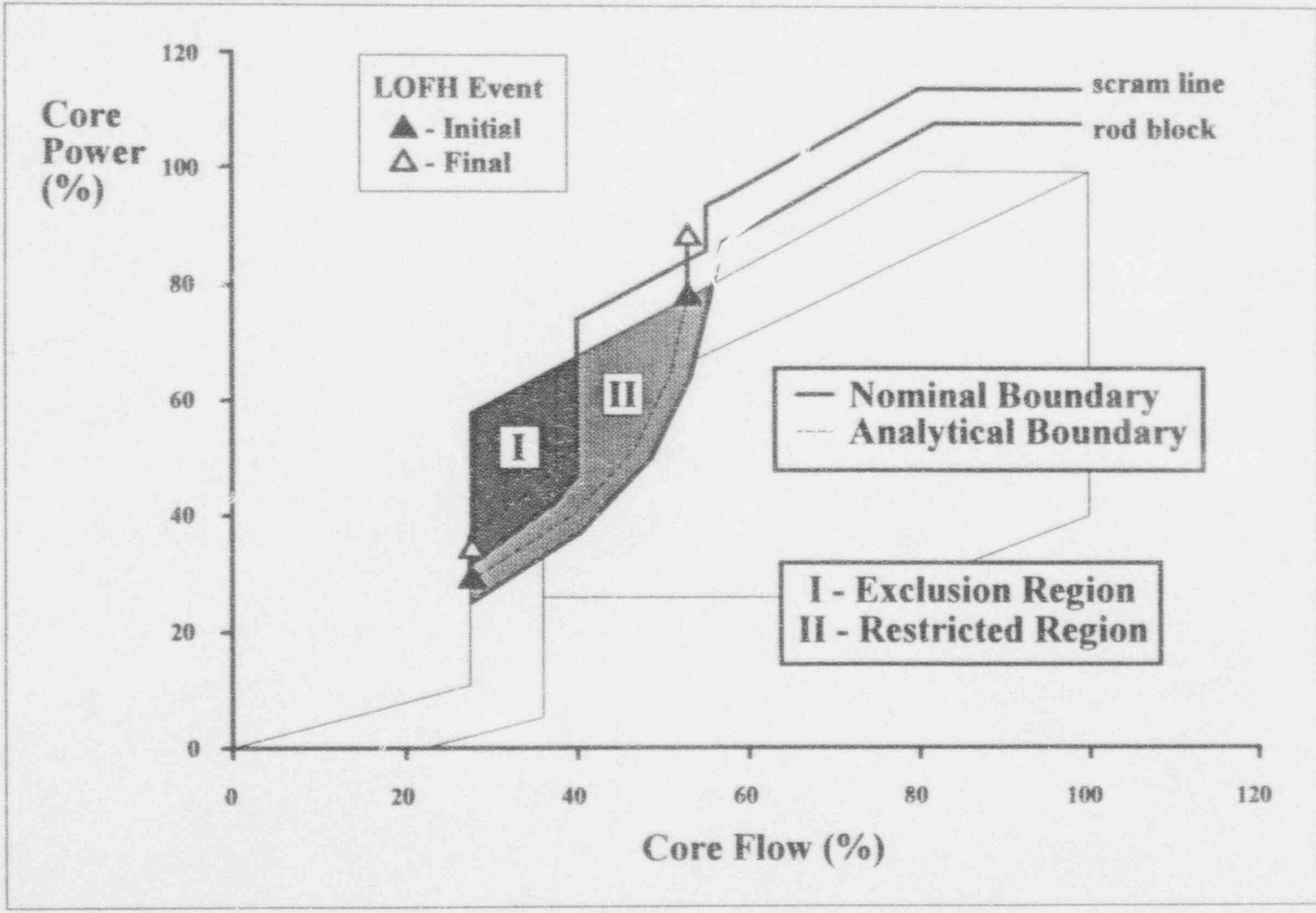


Figure 4-6: Intermediate Flow Reduction Events (IFREs)

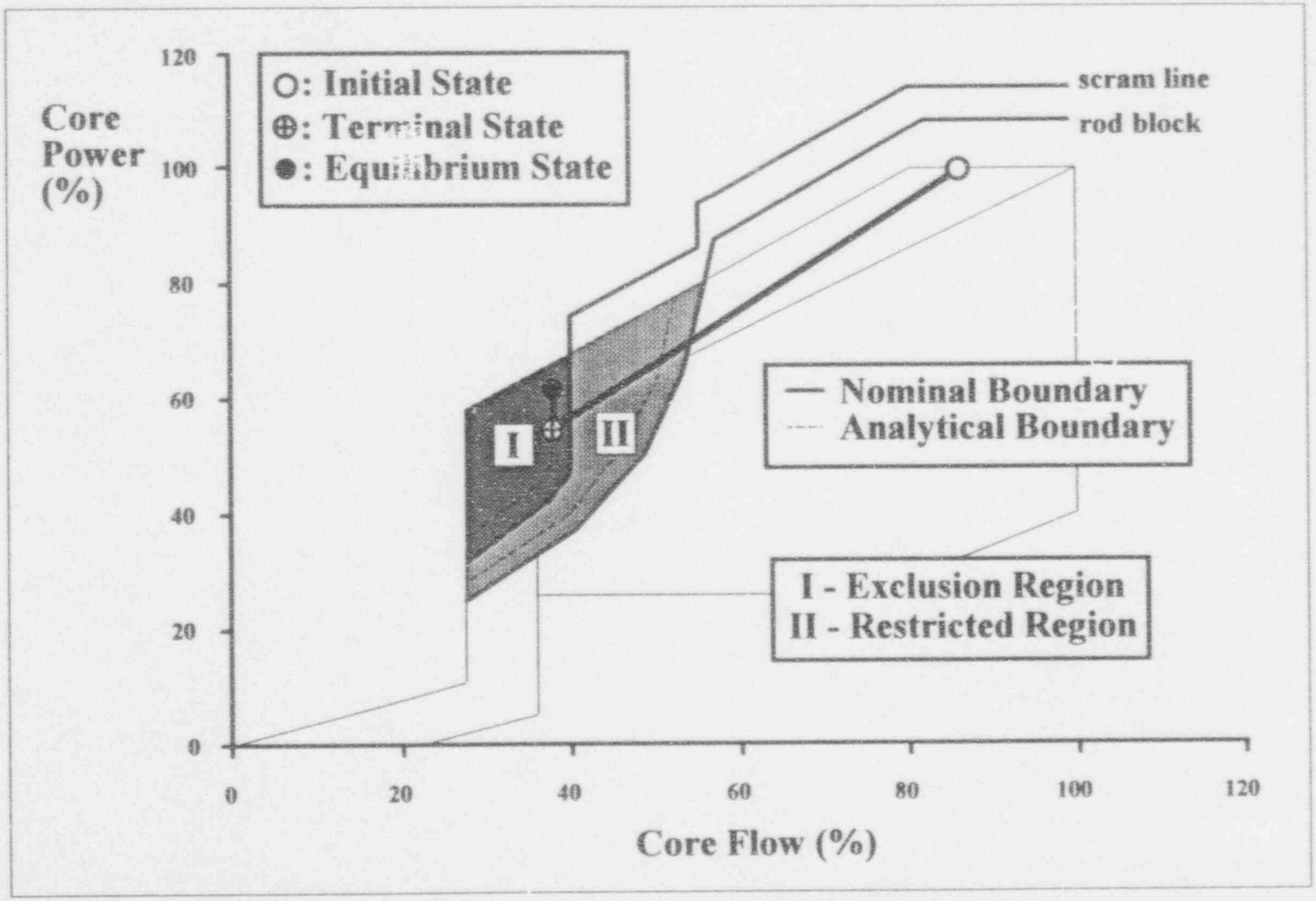


Figure 4-7: Enhanced Option I-A Licensing Boundary Generation Stability Criteria

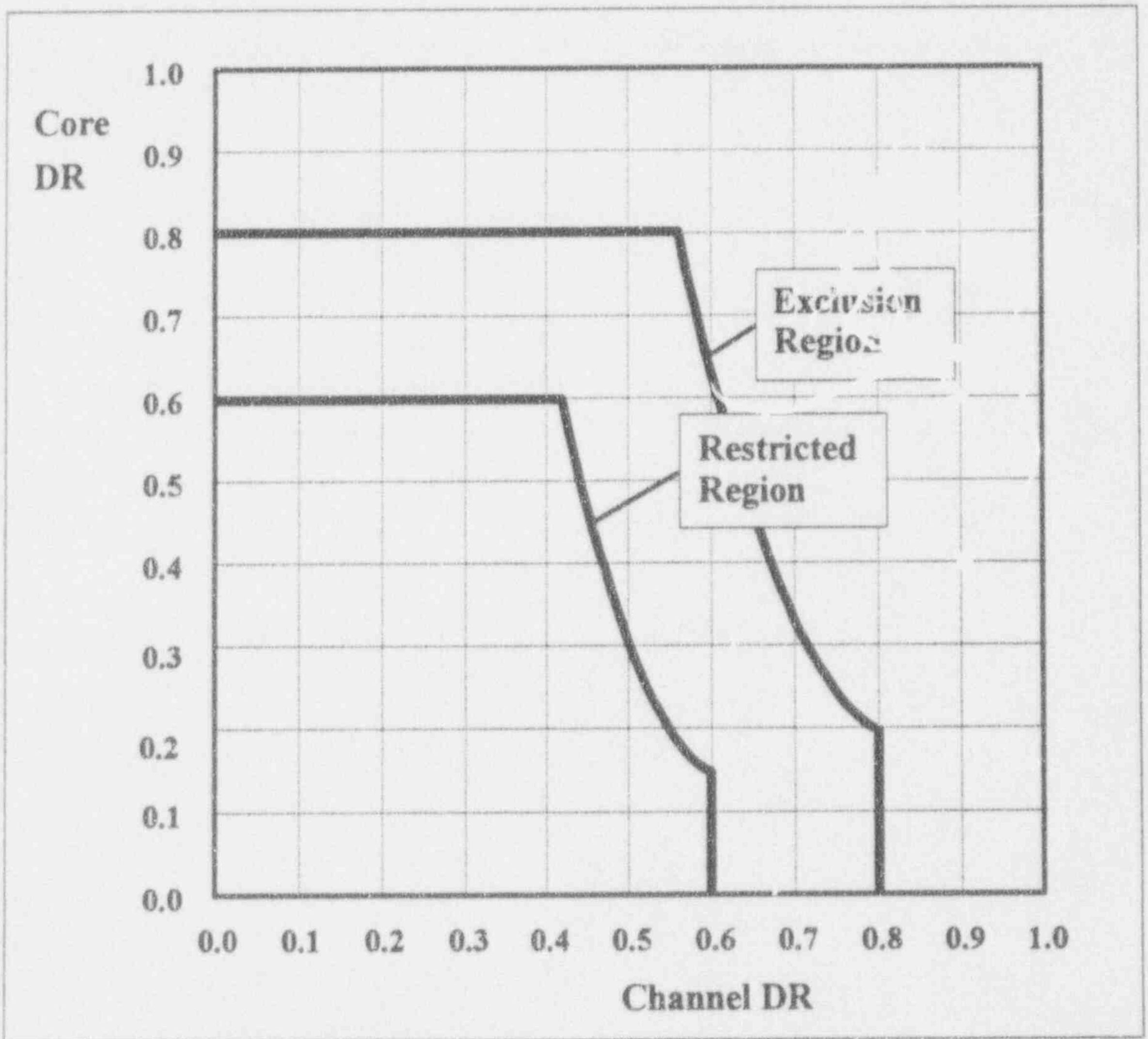


Figure 4-8: State Point Analysis with Stability Controls Supporting Exclusion Region Flow Clamp

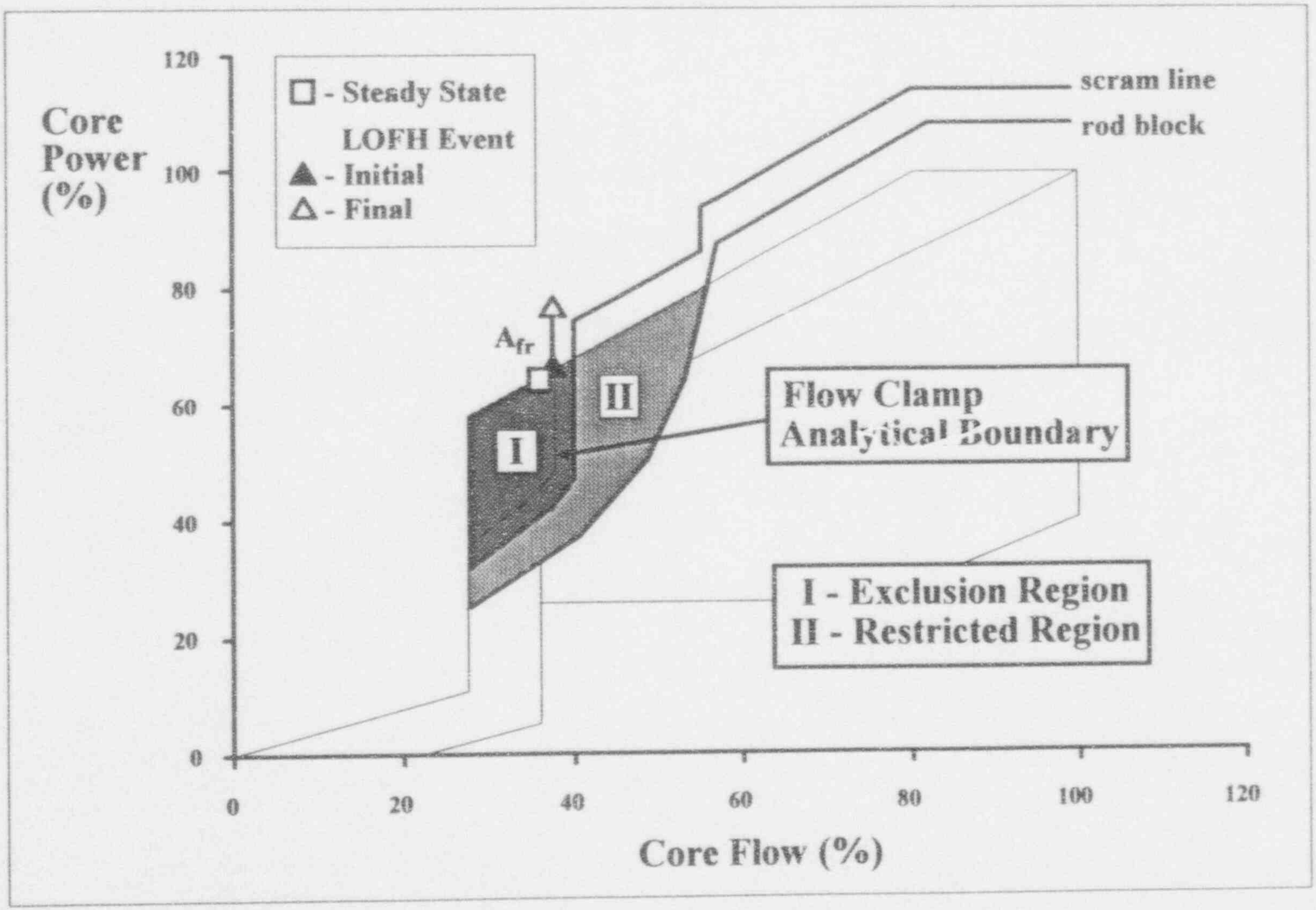


Figure 4-9: Restricted Region Events with Stability Controls

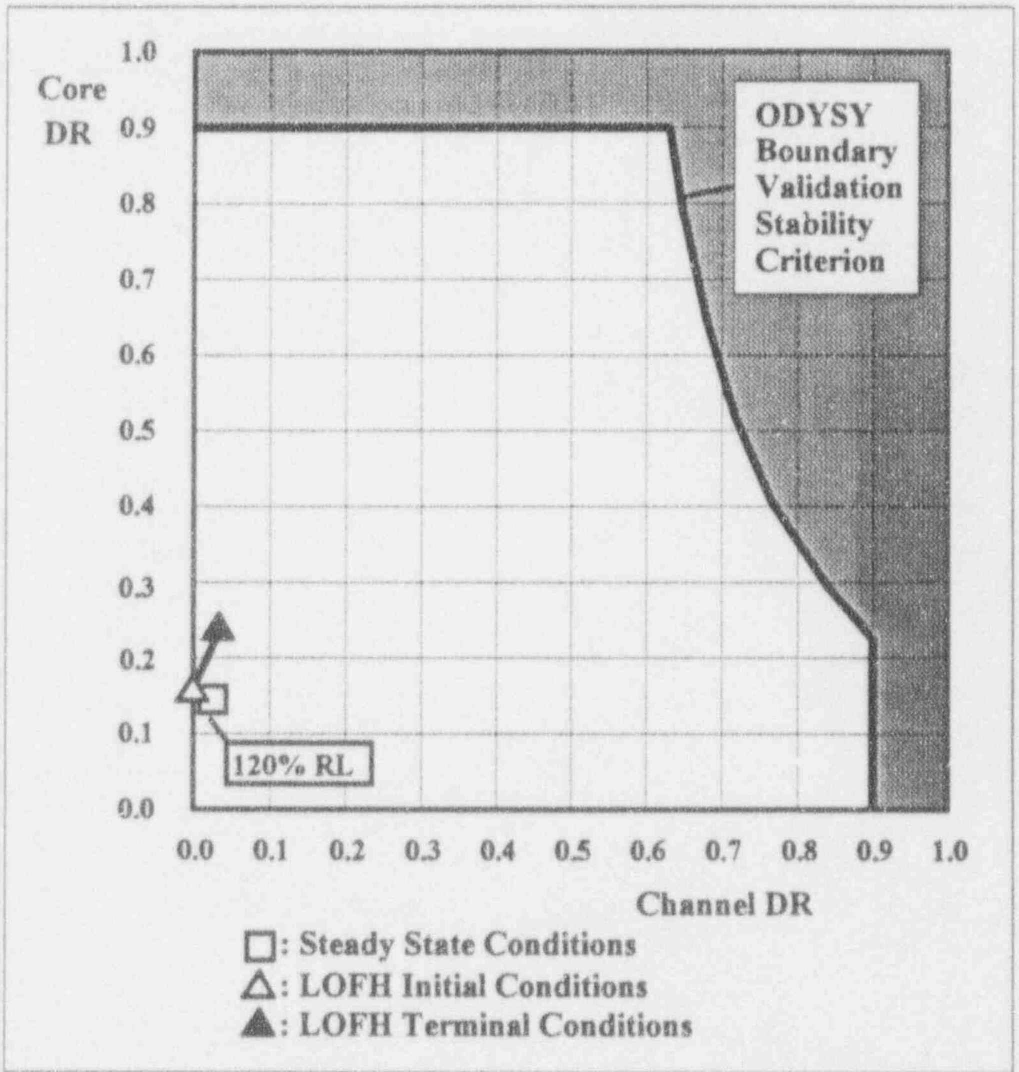


Figure 4-10: IFRE versus Stability Performance

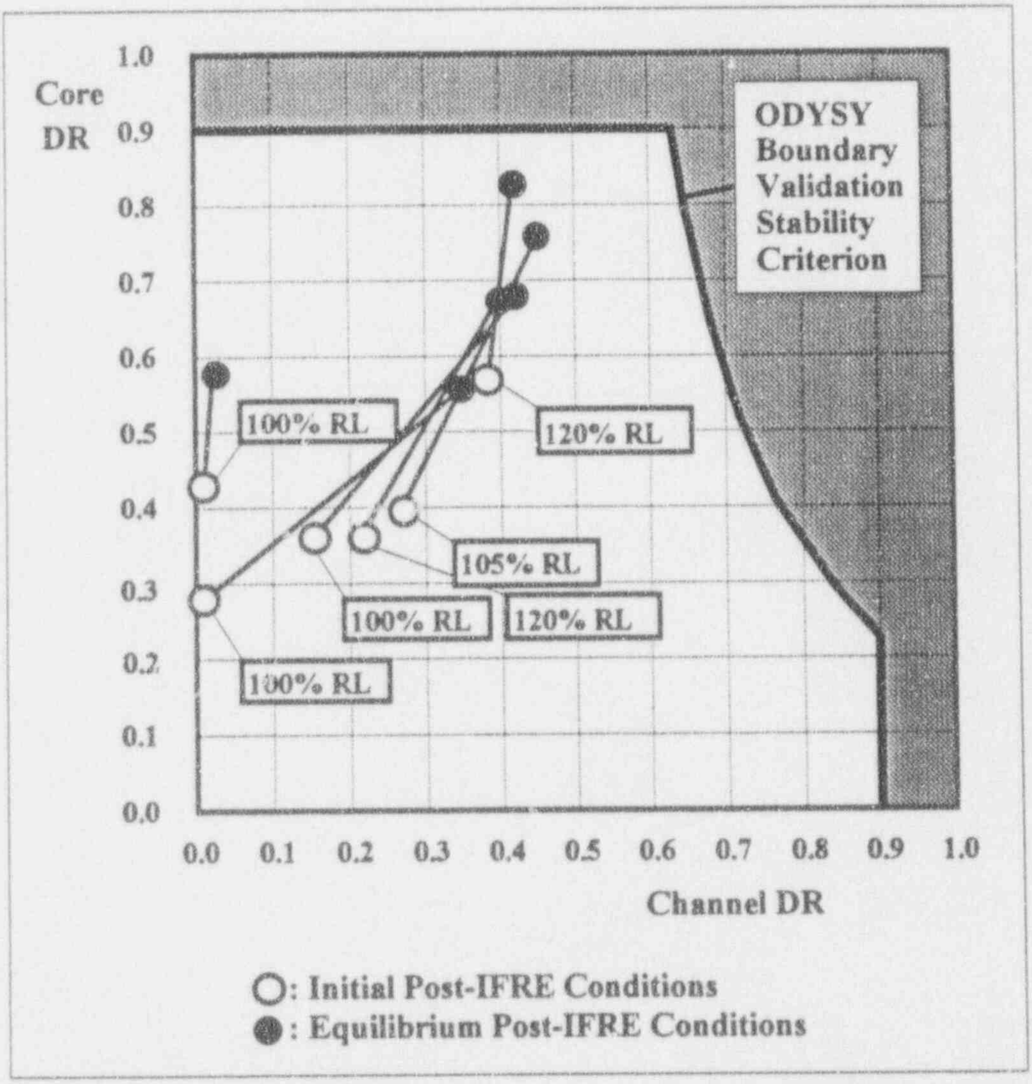


Figure 4-11: GE BWR Instability Events (Excluding Testing)

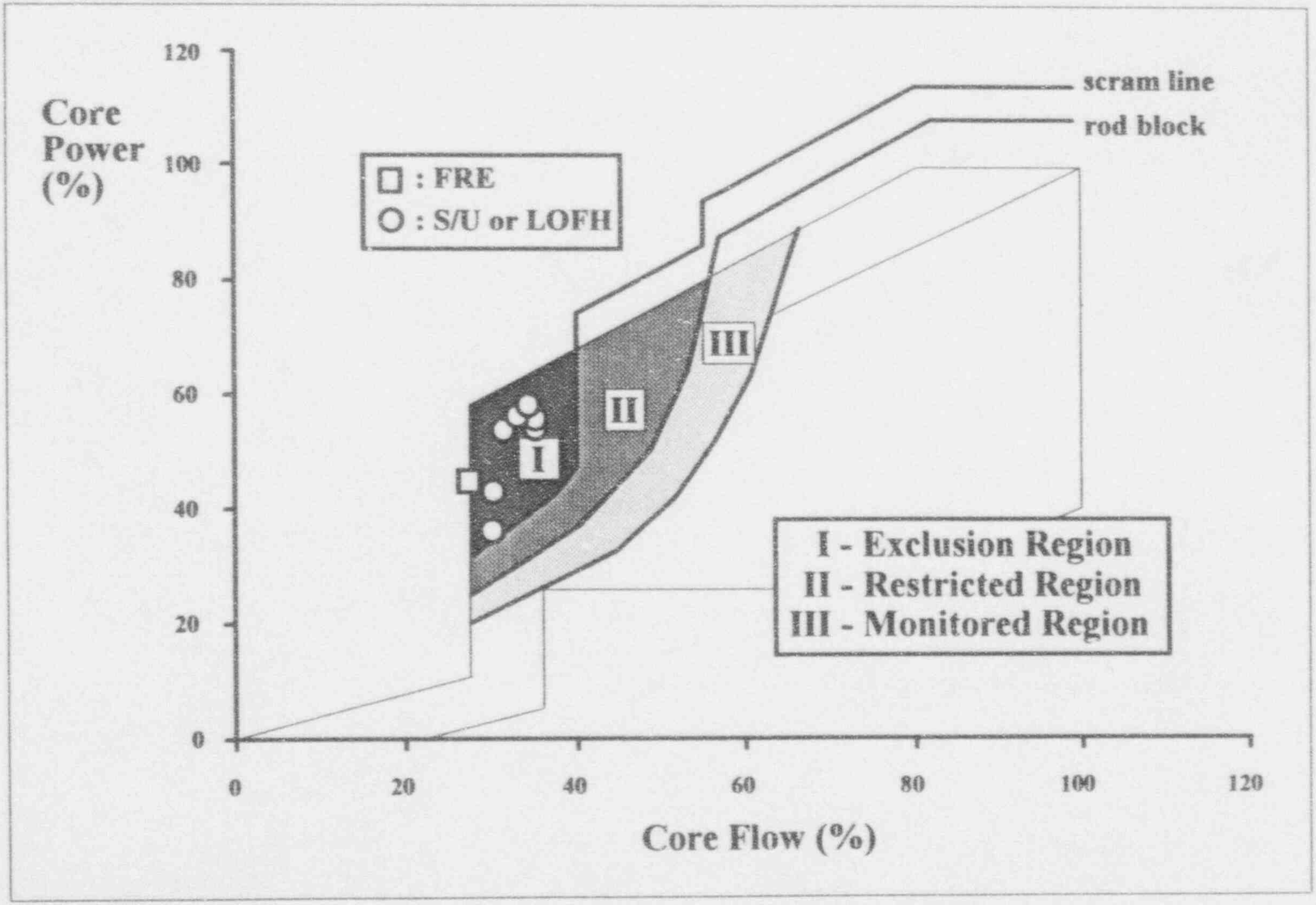


Figure 4-12: IFRE Scenario 1: Anticipated IFRE into Exclusion Region

Protective Features	Event Timeline		
	State 1	State 2	State 3
Licensing Features:	⇒	⇒	AUTO SCRAM
Defense-in-Depth Actions:	⇒	⇒	N/A
Defense-in-Depth Features:	PBDS monitoring stability	Restricted Region Entry Alarm	N/A

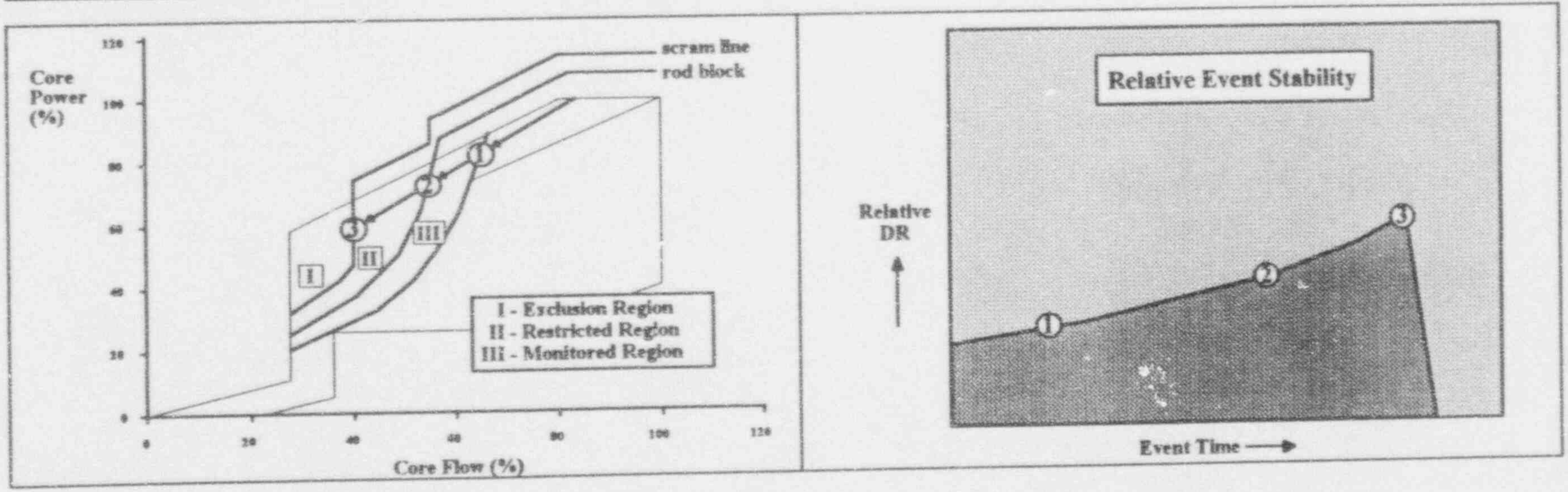


Figure 4-13: <u>IFRE Scenario 2</u>: Anticipated IFRE into Restricted Region

Protective Features	Event Timeline				
	State 1	State 2	State 3	State 4	State 5
Licensing Features:	—	—	—	—	—
Defense-in-Depth Actions:	—	—	Exit Restricted Region		—
Defense-in-Depth Features:	PBDS monitoring stability	Restricted Region Entry Alarm	⇒	⇒	⇒

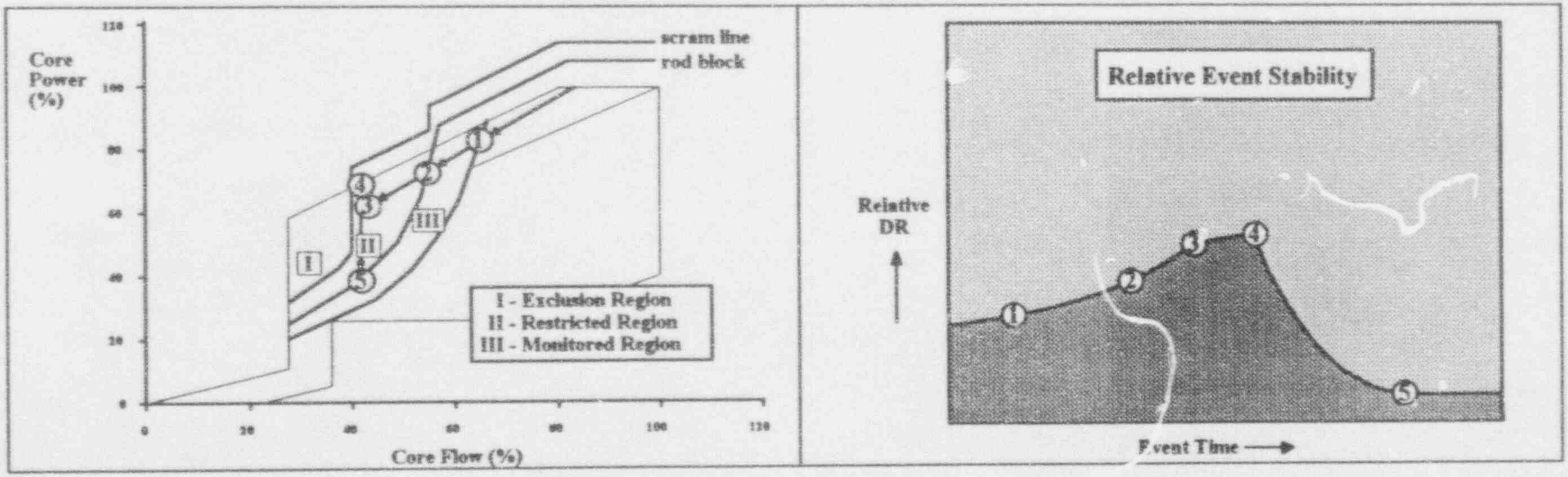


Figure 4-14: IFRE Scenario 3: Anticipated IFRE into Restricted Region followed by Unanticipated Event

Protective Features	Event Timeline				
	State 1	State 2	State 3	State 4	State 5
Licensing Features:	---	---	---	---	---
Defense-in-Depth Actions:	---	---	Exit Restricted Region		Manual SCRAM
Defense-in-Depth Features:	PBDS monitoring stability	Restricted Region Entry Alarm	⇒	PBDS Hi-Hi Decay Ratio Alarm	⇒

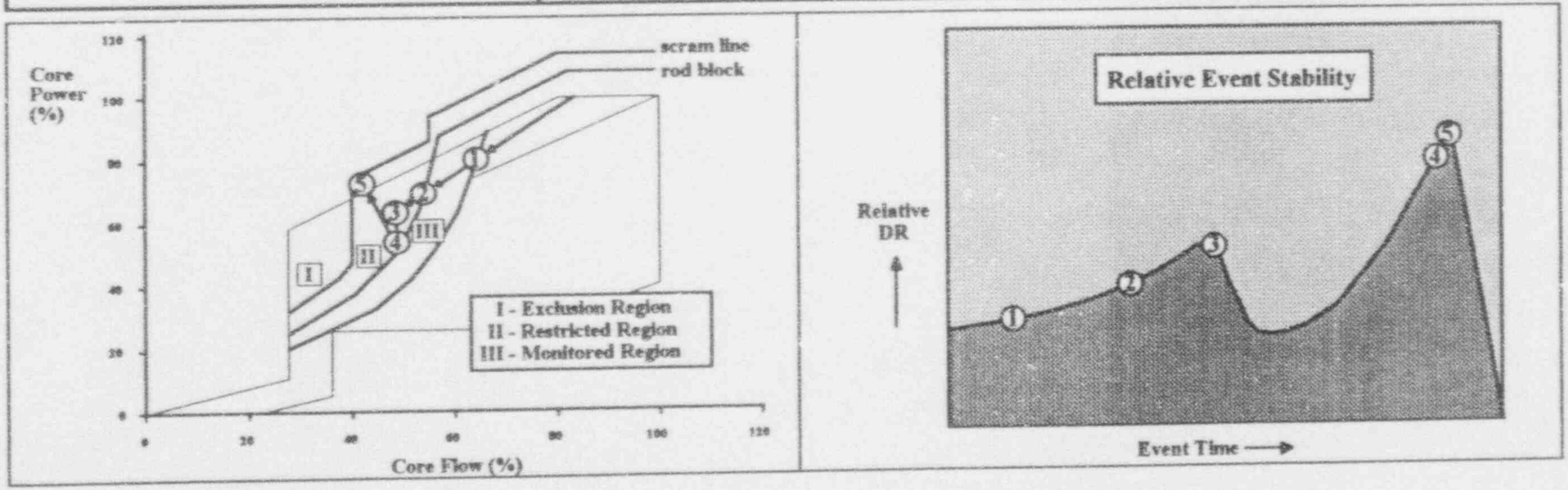


Figure 4-15: IFRE Scenario 4: Anticipated IFRE into Restricted Region followed by Unanticipated Power Increase

Protective Features	Event Timeline			
	State 1	State 2	State 3	State 4
Licensing Features:	—	—	—	—
Defense-in-Depth Actions:	—	—	Exit Restricted Region	⇒
Defense-in-Depth Features:	PBDS monitoring stability	Restricted Region Entry Alarm	⇒	Auto-SCRAM by flow-biased SCRAM clamp

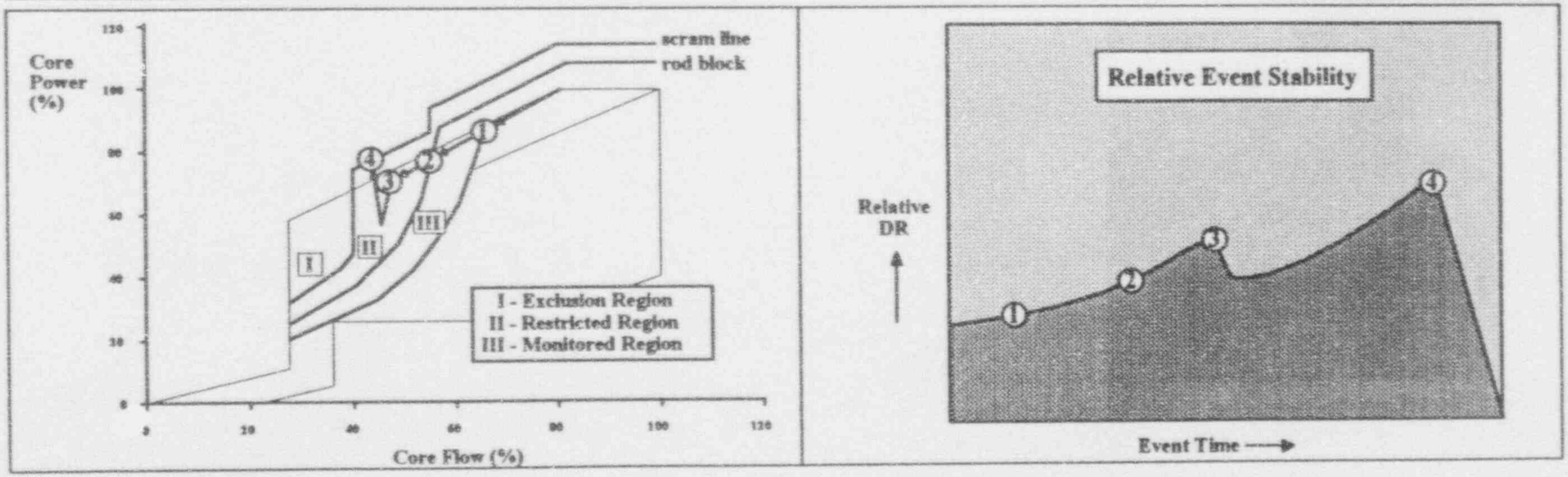


Figure 4-16: Region Boundary Validation for Single-Loop Operation

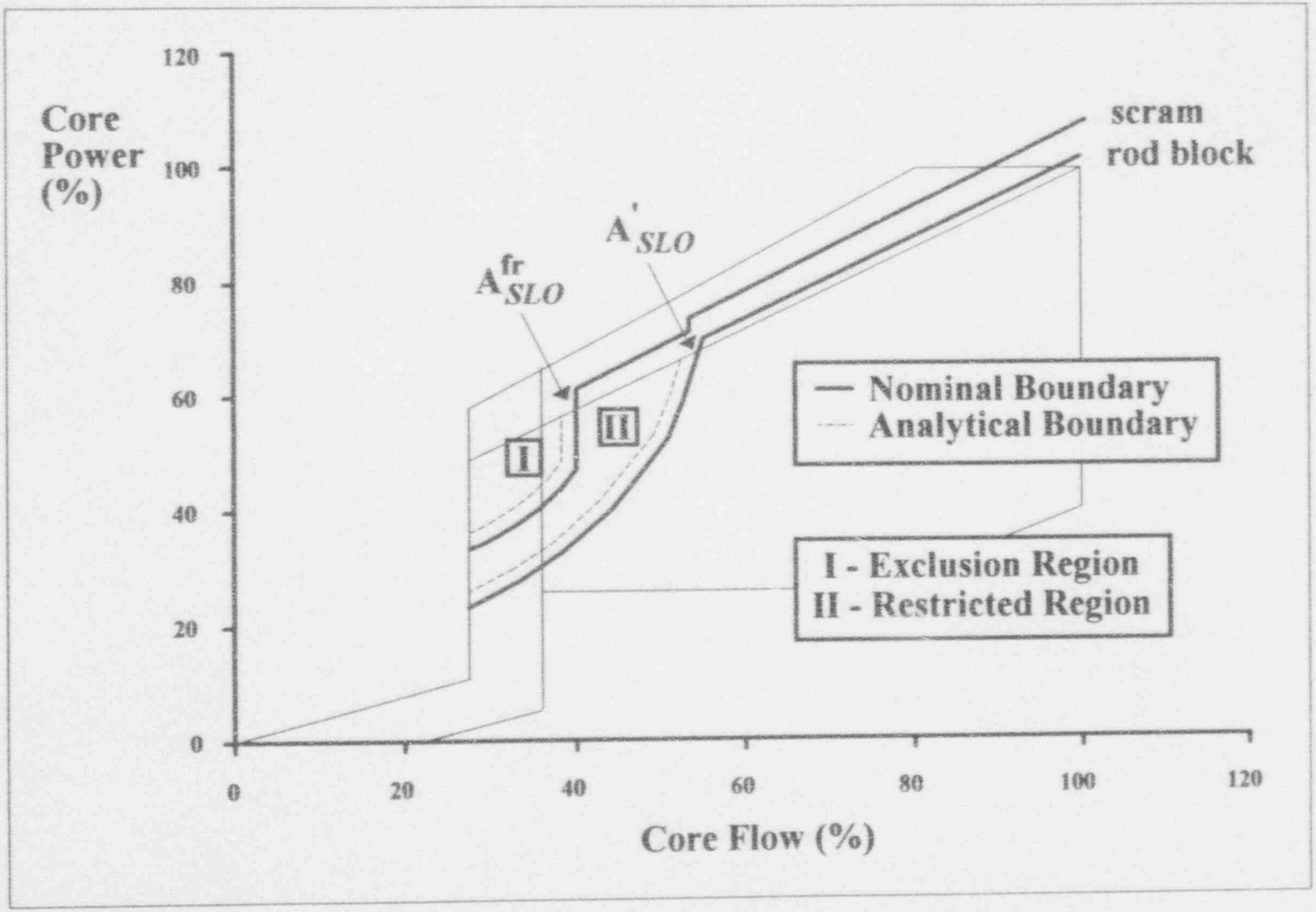
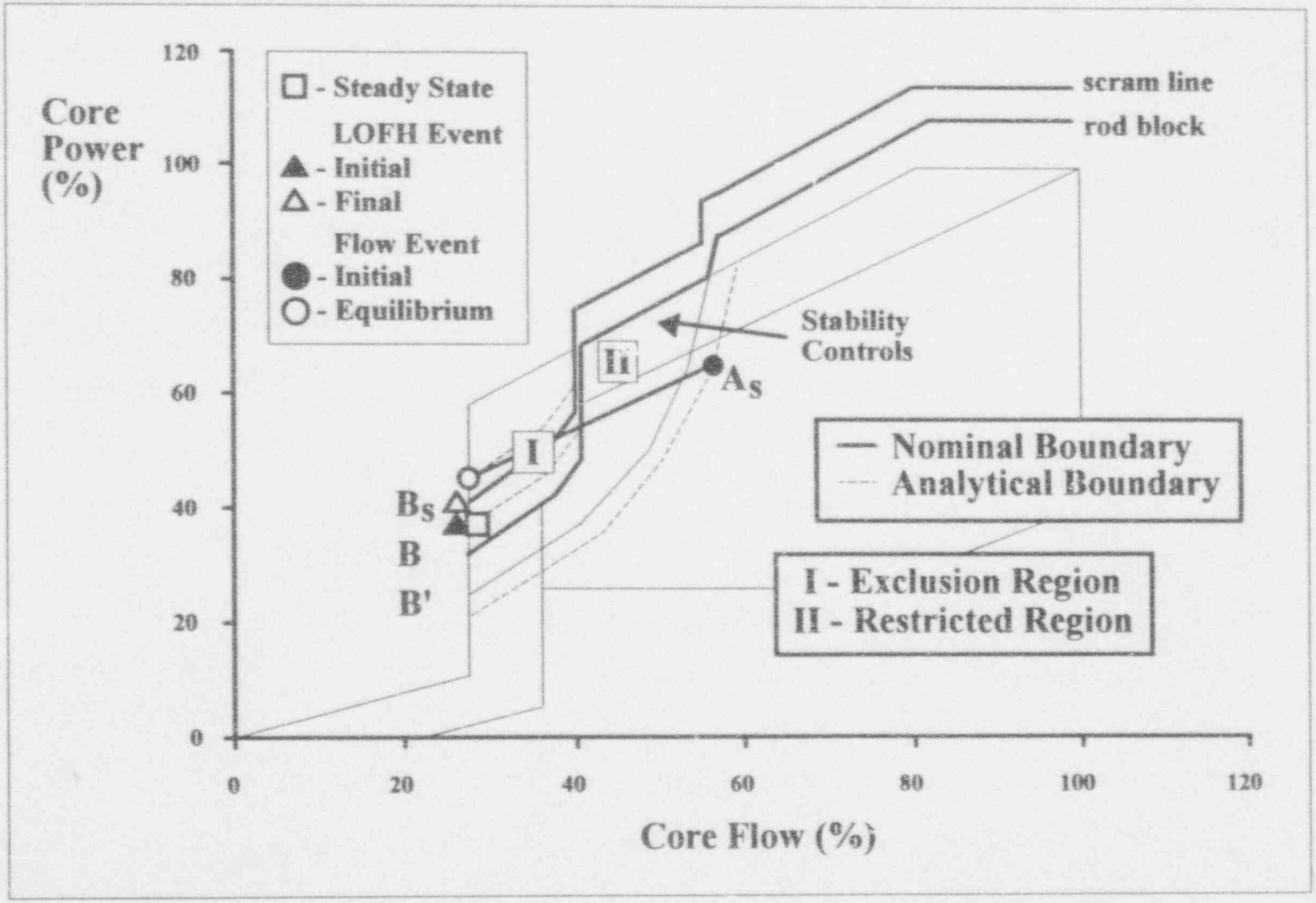


Figure 4-17: Setup Stability Region Boundary Validation Conditions



5. DEFENSE-IN-DEPTH METHODOLOGY BASIS

5.1 Purpose and Relationship to Licensing Methodology

The Enhanced Option I-A stability solution demonstrates compliance with GDC-12 based solely on the licensing methodology features described in Section 4. Reactor instabilities under anticipated operating conditions are prevented by these features. The preventive features of the licensing methodology automatically protect the MCPR safety limit from reasonably limiting precursors to reactor instability.

In strict terms, therefore, the licensing methodology provides the minimum features necessary to comply with the design criteria. The design philosophy of Enhanced Option I-A, however, requires that the solution provide robust protection from reactor instability. This requirement is driven by recognition that the inherent complexities of reactor instabilities make precise analytical results difficult. As a result, the concept of defense-in-depth is introduced into the stability solution.

The defense-in-depth concept employs a series of automatic and mandated manual operator actions. These defense-in-depth features are consistent with and are an extension of the solution design philosophy. Their presence provides a diverse means of preventing reactor instability and eliminates complete reliance on a single system or methodology. Furthermore, protection can be extended to potential and hypothetical events that are beyond anticipated operational occurrences. The existence of protection for a broad spectrum of events also reduces reliance on precise identification of what constitutes a reasonably limiting event. The defense-in-depth features are applied to the solution such that protection becomes more restrictive as the probability of reactor instability increases.

In general, defense-in-depth methodologies differ from licensing methodologies because explicit demonstration that safety limits are protected from anticipated events is not required. The approach utilized in this solution is to introduce extremely conservative actions where appropriate. The defense-in-depth features provide diverse protection against reactor instabilities irrespective of the licensing methodology solution features.

5.2 Monitored Region Basis

The Enhanced Option I-A stability solution includes the Period-Based Detection System (PBDS) as a defense-in-depth feature. The PBDS is required to continuously operate in the Monitored Region, where reactor instability is considered hypothetical. The PBDS is also required to be operable in the Restricted Region. The combined region where the PBDS is required to be operable is shown in Figure 5-1. To maintain methodology consistency with the licensing features, the Monitored Region boundary is generated using the same FABLE/BYPSS-based process utilized for the Exclusion and Restricted Region boundaries. The Monitored Region boundary stability criterion, shown in Figure 5-2, is a function of core and channel decay ratios, and is significantly more conservative than the Restricted Region criterion.

Because of the similarities between the defense-in-depth methodology for generating the Monitored Region, and the licensing methodology for generating the Exclusion and Restricted Regions, the plant-specific application of this solution creates all region boundaries under one procedure. This process, which is outlined in Section 4.3, utilizes the same Standard Cycle and Reference Cycle designs.

However, because the Monitored Region is part of the defense-in-depth features, specific validation of the region boundary is not necessary. Additional rationale for not validating the Monitored Region boundary comes from the definition of the region, which provides protection against hypothetical events. These events are classified as hypothetical because the conditions necessary for reactor instability outside the Restricted Region require the presence of core power distributions and other stability related parameters that are outside the current licensing basis of any BWR. Therefore, the concept of a reasonably limiting event or state condition is not defined for the Monitored Region. The nominal setpoints of the Monitored Region remain valid until the Reference Cycle design is reperformed due to the requirements of the licensing features. No independent reload confirmation of the Monitored Region Boundary is required.

5.3 Period-Based Detection System

5.3.1 PBDS Integration Basis

The primary purpose of the stability detection system is to provide a redundant, diverse means of preventing reactor instability that could challenge the MCPR safety limit. The stability detection system utilizes the core power oscillation period confirmation process of the Period-Based Algorithm (PBA) for detecting the onset of reactor instability, and is therefore termed the Period-Based Detection System (PBDS). The PBA is fully described in NEDO-31960 and Supplement 1 (References 1 and 2). Because the PBA inputs are restricted to LPRM detectors, the PBDS is completely independent of the licensing methodology that relies on core flow indication to generate a reactor trip reference.

Incorporation of the PBDS into Enhanced Option I-A is consistent with the instability prevention solution design philosophy. The PBDS provides automatic indication of reductions in stability margin to alert the operator when mandated corrective actions are necessary. For this solution, any transient that can cause reactor instability will either result in an immediate reactor scram or cause a gradual erosion of reactor stability margin. Therefore, with an appropriately selected alarm setpoint, sufficient time for manual operator action exists. Because the PBDS is a defense-in-depth feature, no explicit demonstration of MCPR safety limit protection or setpoint uncertainty analysis is required, although the PBA methodology is able to provide formal MCPR safety limit protection when coupled to an automatic reactor trip function. This automatic trip function is necessary when the PBA is utilized as a stand-alone licensing methodology feature that must protect against instability events that can occur near natural circulation conditions. Because Enhanced Option I-A automatically excludes operation under these conditions, the PBDS need only operate under conditions where reactor instability is not anticipated. Therefore, manual response to the PBDS alarms provides conservative defense-in-depth protection.

A PBA based stability detection system was chosen for inclusion in Enhanced Option I-A because it provides rapid response to changes in reactor stability margin; conventional noise analysis type stability monitors have significant lags in their response. Furthermore, the PBDS is amenable to applications where specific alarm setpoints are necessary.

5.3.2 Period-Based Detection System

The PBDS utilizes specific LPRM signal inputs to monitor the core for the presence of precursors to reactor instability. LPRMs are chosen from the existing group associated with any one APRM channel. In some BWR types, the two LPRM groups that are not associated with any particular APRM still qualify for use as an input set for a PBDS channel. The PBDS utilizes all LPRMs within a group, except for the upper, D level, detectors.

With the core stable (DR < 1.0), global neutron flux noise perturbations decay rapidly. As reactor stability is reduced (DR → 1.0), these global neutron flux noise perturbations begin to decay more slowly. The PBA algorithm has the ability to detect reduced reactor stability margin based on the number of power oscillation period confirmations. The manner in which the PBA detects neutron flux oscillations is documented in Reference 1.

Enhanced Option I-A incorporates two PBDS channels into the Neutron Monitoring System. This arrangement provides redundant defense-in-depth capability since only one channel is required to be operable. Since the PBDS is redundant, one system channel can be inoperable to perform maintenance or testing while operating within the Restricted or Monitored Regions. When operating outside the Monitored Region, no specific requirements for PBDS operability exist, and the system may remain inoperable indefinitely.

5.3.3 PBDS Setpoint Methodology

The PBDS is designed to generate two stability related alarm signals. The Hi-Hi Decay Ratio alarm indicates an unacceptable loss of stability margin and requires a manual scram without delay. The Hi Decay Ratio alarm, which is an optional solution feature, can provide an early indication of reduced stability margin. In order to establish the corresponding period count setpoints, N_{HH} and N_H, a simple period confirmation count model is introduced, which relates core decay ratio to the period confirmation count. This model and desired PBDS performance characteristics are then used in determining the PBDS setpoints.

5.3.3.1 Period Confirmation Count Model

The ability of the PBA to identify the onset of reactor instability as reflected in the characteristics of core power oscillations in recorded plant data is demonstrated in Reference 3. Analysis of recorded steady-state plant data with the PBA demonstrated that very low period confirmation counts are associated with low core decay ratio operations.

The relationship between core decay ratio and successive period confirmation count is readily apparent for asymptotic situations. For a high decay ratio (DR $\simeq$ 1.0), the period count approaches infinity; for a very low decay ratio (DR $\simeq$ 0), the period count approaches zero. The number of observable oscillation periods resulting from a disturbance to a stable system is related to the core decay ratio and is reduced from infinity to zero as the decay ratio changes from 1.0 to 0.0. The two asymptotic situations are illustrated in Figure 5-3.

The period confirmation count associated with intermediate core decay ratios increases with core decay ratio. The most important mechanism that can cause interruptions in the PBA successive period confirmation count is the inherent stochastic neutron noise in the LPRM signal. To establish the behavior of the period confirmation count as core decay ratio transitions between 0.0 and 1.0, a model that accounts for the signal noise characteristics is required. The primary objective of the period confirmation count model is to establish a general functional relationship between core decay ratio and period confirmation count for a given plant-specific neutron noise signature. To this end, a simple reactor core neutron noise model is presented.

5.3.3.1.1 Neutron Noise Components

The neutron noise is assumed to consist of two independent and temporally random components: (1) a global noise component that is core-wide and caused by perturbations to the dynamic behavior of the reactor coolant, and (2) a local noise component caused by local stochastic phenomena in the reactor coolant.

The source of the global noise component is external to the reactor core. Typical global neutron noise sources are reactor pressure and flow perturbations. These perturbations are temporally random, and can have a coherent effect on the entire core. The global noise component characteristics are plant-specific and have a maximum perturbation amplitude of approximately 2 to 5% of rated power under stable reactor operating conditions.

The source of the local noise component is internal to the reactor core. Local neutron noise sources consist of local stochastic perturbations in channel flow and coolant boiling. These random perturbations only affect nearby LPRMs. The local noise component is characterized by a small average amplitude that is typically less than 1% of rated core power. The local noise spans a wider range of frequencies than the global noise, including frequencies that are much higher than those expected from power oscillations induced by coupled neutronic/thermal-hydraulic instability.

In the PBDS, LPRM signals are conditioned to filter out all frequencies above a certain corner frequency, set above the known frequency range for reactor instability. Signal conditioning reduces the disruptive effect of the local noise component of the LPRM signal on the period confirmation count process for successive oscillations.

5.3.3.1.2 Neutron Noise Decay Phenomena

Core decay ratios greater than or equal to unity result in power oscillations that have achieved or are converging to limit cycle oscillations with a constant period. These power oscillations are associated with reactor instability and are not compatible with the preventive design philosophy of Enhanced Option I-A. Therefore, the following discussion only considers decay ratios that range between zero and one, so that any perturbation of core neutron flux will eventually decay in time.

In a stable reactor, occasional global noise perturbations occur that are sufficiently large to create a coherent core response that is regionally coupled and decays with an exponential attenuation. The oscillatory signature of this core response decays until either global perturbations out of phase with the initial perturbation occur or until the decaying oscillation is sufficiently weak that local neutron noise disrupts the coherent response of the system. For this model, a threshold noise level is defined to be the power oscillation amplitude at which small global perturbations and local noise effects disrupt an oscillation created by a large global perturbation to the extent that the PBA cannot discriminate the oscillation period.

The threshold noise level is a function of the LPRM signal conditioning, since removal of the high frequency component of the noise by the conditioning filter reduces the disruptive effects on the successive period count process and therefore effectively lowers the threshold noise level.

Typically, multiple global noise perturbations can occur within any one period of a given power oscillation, creating random changes in frequency and amplitude, and preventing successive oscillation period confirmations. However, since the noise is random, occasionally a large global perturbation decays in the absence of other large global perturbations.

The probability that the response to a global noise perturbation will decay to the threshold noise level undisturbed is higher for low decay ratio conditions, since the decay is rapid. However, the likelihood of successive period counts is very low because of the rapid decay. As decay ratio increases, a higher successive period count is possible because of the slower decay. However, the probability for additional large global noise perturbations during the decay that would terminate the confirmation process is higher. Nevertheless, occasionally the global perturbation decays to the threshold noise level, resulting in a higher period count. At intermediate decay ratios, a subsequent global perturbation during the decay may have a reduced effect on the oscillation frequency because the reactor response is more coherent. This is expected to further increase the probability for a full decay to the threshold noise level, which will result in a higher period count.

For high decay ratio conditions, the oscillatory perturbation signature approaches limit cycle or decays very slowly. Since the core response is increasingly coherent and strongly coupled, subsequent global perturbations do not alter the oscillation frequency, but only enhance or diminish the oscillation amplitude. This behavior can be clearly observed for the filtered LPRM signal shown in Figure 5-4. The figure also demonstrates that some level of signal conditioning, which filters out high frequency responses outside the range expected for reactor instability, is essential for application of the PBA and for construction of a neutron noise model.

5.3.3.1.3 Analytical Model

The power oscillation period confirmation count can be related to the core decay ratio using the neutron noise model of the LPRM signal signature described above. Specifically, a single global noise perturbation representing a typically large perturbation amplitude, δA_1, is introduced and allowed to decay to the threshold noise level, δA_0. Over the interval in which the perturbation amplitude decreases from δA_1 to δA_0, successive period confirmations occur. When the perturbation amplitude reaches δA_0, the successive period confirmation count is terminated.

The PBA establishes the base period after one oscillation period and obtains the first period confirmation in the next half-period. Subsequent period confirmations are obtained every half-period at the signal maxima and minima. The period confirmation count model is illustrated in Figure 5-5. For a stable reactor, with decay ratio (DR) less than unity, the amplitude reduction per half-period is $\sqrt{DR}$. Since the first period is associated with two half-periods, the perturbation amplitude after N successive period confirmations are identified (N+2 half-periods) is given by $\delta A_1 \times (\sqrt{DR})^{N+2}$. Since the successive period count is assumed to terminate when the perturbation amplitude reaches δA_0, it follows that

$$\delta A_1 \times (\sqrt{DR})^{N+2} = \delta A_0 \qquad (5\text{-}1)$$

where N is the period confirmation count when the amplitude reaches δA_0.

The decay ratio can be related to the period count by

$$DR = \left(\frac{\delta A_0}{\delta A_1}\right)^{\frac{2}{N+2}} \qquad (5\text{-}2)$$

or

$$\mathrm{DR} = \Re^{\frac{2}{N+2}}, \qquad (5\text{-}3)$$

where $\Re = \dfrac{\delta A_0}{\delta A_1}$.

The noise factor, $\Re$, is plant-specific and represents an effective ratio of the threshold noise level to a typically large global noise perturbation as observed in the conditioned LPRM signal. The period confirmation count, N, represents the maximum number of successive confirmations that are expected to be observed for a given core decay ratio condition. Equation 5-3 represents the relationship between the core decay ratio and the successive period confirmation count parametric in the noise factor $\Re$. This model relationship is illustrated for a wide range of noise factors in Figure 5-6.

For convenience, Figure 5-6 makes use of the inverse of the noise ratio factor, $\Re^{-1}$. In the limit where the global and threshold noise amplitudes are the same ($\Re^{-1} = 1$), successive period confirmations are not possible for decay ratios less than unity. For a very small noise factor (e.g., $\Re^{-1} = 10000$), a high period count occurs at low decay ratios. This is an expected outcome of the model, since the threshold noise is effectively removed (e.g., use of large period tolerance). The model shows that intermediate noise factors provide varying levels of confirmation count responsiveness for a given core decay ratio.

The plant-specific application of the PBDS will adjust the PBDS setpoints to provide a PBA sensitivity consistent with an intermediate $\Re$ value. This is done to avoid excessively responsive configurations (e.g., $\Re^{-1} = 10000$) or unresponsive configurations (e.g., $\Re^{-1} = 1$). The functional shape of the intermediate ranges of $\Re$ is consistent with the characteristics of other noise-based detection systems. Specifically, for a given range of noise factors, the decay ratio range (or uncertainty) associated with a period confirmation count is large for low decay ratios and decreases significantly for high decay ratios.

5.3.3.2 Setpoint Selection

The PBDS setpoints consist of specific period confirmation count values that represent varying levels of stability margin. Equation 5-3 relates the core decay ratio to the specific period confirmation count associated with a given signal characteristic ($\Re^{-1}$). To determine the PBDS setpoints, it is necessary to identify the expected decay ratio range during reactor operations and the desired conditioned LPRM signal characteristics. The determination of the PBA parameters' values needed to achieve the PBA detection objectives and to maintain the target conditioned LPRM signal characteristics for a plant-specific application is addressed below.

5.3.3.2.1 Decay Ratio Range

The core decay ratio between zero and one can be divided into three ranges:

a. Low decay ratio,
b. Intermediate decay ratio, and
c. High decay ratio.

Reasonable values for low, intermediate and high decay ratio ranges are 0 to 0.5, 0.5 to 0.8, and 0.8 and higher, respectively. A summary of the decay ratio ranges is provided in Table 5-1. The low decay ratio range is expected to envelop normal reactor operations. For Enhanced Option I-A, reactor operation in the Restricted Region with the required stability controls or outside the Restricted Region ensures a low decay ratio during startup and shutdown evolutions.

Deviations from normal operating conditions near or inside the stability regions may result in a moderate reduction in stability margin. As a result, the core decay ratio increases into the intermediate range. An optional PBDS alarm is available to alert the operator to the decay ratio increase. This alarm, defined as the Hi Decay Ratio alarm, can allow the operator to take timely preventive actions to mitigate further reductions in core stability margin, but is not required as part of the Enhanced Option I-A stability solution since no reactor safety issue is associated with the alarm.

Extreme operating conditions and unanticipated transients can result in a significant loss of stability margin. As a result, the core decay ratio may increase into the high decay ratio range. In this range the reactor is either unstable or is on the verge of becoming unstable. The PBDS provides an alarm that requires the operator to manually scram the reactor without delay if the high decay ratio range has been reached. This alarm is defined as the Hi-Hi Decay Ratio alarm.

Normal off-rated operations are expected to result in a decay ratio well within the low decay ratio range. Actual decay ratios during normal operations cannot be established precisely because of the large uncertainty in decay ratio measurements in the low decay ratio range. If used, the Hi Decay Ratio alarm should be set in the intermediate decay ratio range above a decay ratio of 0.5 and can be reasonably associated with a decay ratio range of 0.6 to 0.7.

The Hi-Hi Decay Ratio alarm is set at the low-end of the high decay ratio range and is associated with a decay ratio near 0.8. The 0.8 decay ratio is selected because, during the transition from 0.8 core decay ratio to 1.0, the core's oscillatory response to perturbations becomes increasingly coherent and accelerated such that each LPRM response becomes coupled to the behavior of the entire core and cannot be considered random. Since the period confirmation count model is based on an isolated LPRM decay signature and does not account for the increasingly self-sustained oscillatory behavior of the core at high decay ratios, the modeling assumptions of Equation 5-3 break down and the model is expected to under-predict the oscillation period count in the high decay ratio range. The alarms, associated operating conditions and decay ratio ranges are summarized in Table 5-2.

5.3.3.2.2 LPRM Signal Characteristic

In addition to establishing the decay ratio range in order to determine the PBDS setpoints, it is necessary to identify the desired signal characteristic ($\Re^{-1}$). This is accomplished by comparing the signal characteristic, based on Equation 5-3, to the decay ratio range identified in Table 5-2. The decay ratio comparison is illustrated in Figure 5-7. From Figure 5-7, different $\Re^{-1}$ and successive period confirmation count values intersecting the three decay ratio ranges can be discerned.

Target conditioned LPRM signal characteristics are determined based on the PBDS objectives. A choice of a very low or very high $\Re^{-1}$ value is not appropriate for the PBDS. A low value (e.g., 2) will not provide any period confirmation count during normal operations and will result in a Hi-Hi Decay Ratio alarm at a very low confirmation count, making the PBDS insensitive to changes in decay ratio. A high $\Re^{-1}$ value (e.g., 15) will result in a high period confirmation count during normal operations (up to 7) and over 20-period count for the Hi-Hi Decay Ratio alarm (Figure 5-6), making the PBDS overly sensitive to changes in decay ratio. Therefore, an intermediate value of $\Re^{-1}$ is selected.

The total number of successive period confirmations required for the Hi-Hi Decay Ratio alarm must be limited to a value between approximately 10 and 15. This is done to ensure quick system and operator response to avoid prolonged loss of stability margin for conditions where the core decay ratio approaches unity. As can be seen from Figure 5-7, the value of $\Re^{-1}$ should not exceed approximately 6 to meet this requirement.

The low period count range (N equals 2 to 3) is reserved for PBDS calibration during normal off-rated operations. The calibration process is necessary to determine the plant-specific values of the PBA parameters. An occasional period confirmation count in the 2 to 3 range during normal operations is necessary to ensure that the PBDS is sufficiently sensitive to respond to an increase in the core decay ratio. As can be seen from Figure 5-7, the value of $\Re^{-1}$ should not fall below approximately 4 to meet this requirement. Selecting the range of $\Re^{-1}$ between approximately 4 and 6 corresponds to 1 to 3 successive confirmation count for core decay ratio between 0.3 and 0.5.

5.3.3.2.3 Setpoint Determination

Consideration of the PBDS detection objectives and the target decay ratio ranges can be used to establish the PBDS setpoints. As illustrated in Figure 5-8, the $\Re^{-1}$ range of 4 to 6 results in a maximum period confirmation count range of 2 to 3 during normal operations, 4 to 8 for the Hi Decay Ratio alarm, and 11 to 15 for the Hi-Hi Decay Ratio alarm.

To determine the Hi and Hi-Hi Decay Ratio alarm setpoints, the calibration of the PBDS at normal off-rated operating conditions is first considered. For example, the PBDS parameters may be set to achieve a maximum count of 2 to 3, so that the system's confirmation count is sensitive to variations in decay ratio and to maintain a low maximum count during expected normal operations. The Hi Decay Ratio alarm setpoint may be selected within the period confirmation count range of Figure 5-8 and above the normal period count. To maintain sufficient separation between the confirmation count at the low decay ratio range (i.e., 2 to 3), the Hi Decay Ratio alarm range may be selected corresponding to a decay ratio of 0.7, which results in a confirmation count of 6 to 8. Implementation of the Hi Decay Ratio alarm is optional. Selection of a specific Hi Decay Ratio alarm setpoint value and definition of associated operator actions is not addressed in this report.

The Hi-Hi Decay Ratio alarm may be selected within the period confirmation count range of Figure 5-8 at the 0.8 decay ratio threshold. As illustrated, the appropriate count value for the Hi-Hi Decay Ratio alarm is 11. The alarm setpoints and the target PBDS sensitivity during normal off-rated operations are summarized in Table 5-3 and illustrated in Figure 5-9.

Since the PBDS calibration process results in a maximum period confirmation count of 2 to 3, core decay ratios below 0.5 during normal operations will result in a $\Re^{-1}$ value above 6 and, therefore, greater system sensitivity. This calibration process is therefore conservative. Increased system sensitivity is expected to affect the approach to the Hi-Hi Decay Ratio alarm. However, the actuation of the Hi-Hi Decay Ratio alarm is not expected to be affected because of the alarm logic employed (see below).

The process of determining the PBDS period count setpoints is plant-independent and results in a generic PBDS Hi-Hi Decay Ratio alarm setpoint. However, plant application of the PBDS requires selection of plant-specific values for certain PBA parameters to establish the necessary noise characteristics of the conditioned LPRM signal.

5.3.3.2.4 PBDS Alarm Logic

The Hi-Hi Decay Ratio alarm setpoint (11 successive period confirmation counts) is expected to be reached or exceeded as the core approaches the high decay ratio range. During the transition into high decay ratio conditions, the probability for a single occurrence of a 11-confirmation count increases. The probability of a sustained period confirmation count is high, however, only when the core is on the verge of instability. Moreover, most of the LPRM signals are expected to exhibit a higher confirmation count, and overlapping 11-confirmation counts from different LPRM signals are expected.

The Hi-Hi Decay Ratio alarm logic is based on a single PBDS card output. However, since multiple 11-confirmation counts from different LPRM signals are expected to overlap, the alarm is based on a two-out-of-all-LPRMs per channel logic. This logic increases the reliability of the Hi-Hi Decay Ratio alarm. If an analog output is available, observable from the reactor controls in the control room, and is operable, verification against the PBDS card analog output may be performed without delay prior to the manual scram. Upon validation of the Hi-Hi Decay Ratio alarm based on the analog output, the reactor is manually scrammed without delay. A summary of the PBDS alarm logic is provided in Table 5-4.

If the Hi Decay Ratio alarm is implemented, the setpoint (6 to 8 successive period confirmation counts) is expected to be reached following a substantial reduction in stability margin. The Hi Decay Ratio alarm is optional and the setpoint is plant-specific, selected based on operational and plant reliability considerations. Since only one PBDS card is required to be operable, the alarm logic is based on the output from a single card. Furthermore, since multiple period counts from different LPRM signals are not expected to overlap in the intermediate decay ratio range, the alarm logic is based on a single period count occurrence that reaches the selected Hi Decay Ratio alarm setpoint.

5.3.3.2.5 Spurious Alarm Considerations

The Hi-Hi Decay Ratio alarm setpoint is conservatively selected at the low-end of the high decay ratio band in Figure 5-7. The following discussion demonstrates that the selected setpoint will not lead to unnecessary alarms (i.e., manual scrams) during normal off-rated operations.

During operations near or inside the stability regions, core decay ratios are expected to remain in the low range. The period confirmation count remains, based on the period confirmation count model, in the 2 to 3 range. This count represents the occasional situation where a global perturbation is permitted to decay to the threshold noise level. Typically, the confirmation count remains at zero (i.e., successive time periods not satisfying the base period criteria) because the large global noise perturbations overlap.

Overlapping global noise perturbations (perturbations occurring within a single oscillation period) will usually disrupt and terminate the successive period count. It is conceivable, however, that subsequent global perturbations will be introduced in phase relative to the initial perturbation and will reinforce and sustain the successive count even at low core decay ratios. The Hi-Hi Decay Ratio alarm setpoint has to be sufficiently high to avoid an alarm as a result of an in-phase series of random global noise perturbations.

The period confirmation count model is used to estimate the probability of reaching or exceeding the Hi-Hi Decay Ratio alarm setpoint during low decay ratio operations. To simplify the discussion, the first 3-confirmation count is assumed to be uninterrupted. At this time, the initial perturbation is attenuated to the threshold noise level. To sustain a successive period count to the Hi-Hi Decay Ratio alarm setpoint, 8 additional periods, equal to the base period, are needed. This is accomplished by introducing 8 global noise perturbations at the appropriate time intervals.

If the 8 additional periods are generated by less than 8 independent global noise perturbations, then more than one confirmation per perturbation is required. Since two or three successive confirmations occur only every few minutes, the probability for 8 successive additional confirmations is lower. For the purpose of this demonstration, only the 8 independent global noise perturbation model is considered.

Specifically, the subsequent global noise perturbations are introduced once per half-oscillation period (which corresponds to each successive period count), inside the period tolerance band and in phase with the previous oscillation period. The period tolerance band (i.e., 2ε) is set at 10% of the oscillation period. The perturbation can be either positive or negative. It must be negative at a peak and positive at a minimum to sustain the oscillations and, therefore, has a 50% probability of occurrence. Assuming only a single global perturbation per oscillation period and the appropriate value of $\Re$, the probability for sustaining 8 successive confirmations is given by:

$$p_s = \left(\frac{1}{10}\right)^8 \times \left(\frac{1}{2}\right)^8 \cong 4 \times 10^{-11} \qquad (5\text{-}4)$$

where p_8 represents the probability for 8 successive confirmations, provided that a 3-confirmation count had occurred. If a five-second time interval is conservatively assumed between 3-confirmation count occurrences, the probability per second for 8 successive confirmations, $\overline{p}_8$, is given by:

$$\overline{p}_8 = p_8 / (5\,\text{second}) \cong 8 \times 10^{-12} / \text{second} \tag{5-5}$$

It is possible to sustain a successive count by introducing a global perturbation every other period count, which will appear to increase the probability calculated in Equation 5-4. However, the possibility for overlapping global perturbations within one oscillation period, and the variations in global perturbation amplitudes that are conservatively ignored in the derivation of Equation 5-4 more than compensate for the possibility of not having any perturbation during the oscillation period. Equations 5-4 and 5-5, therefore, represent a conservative probability estimate.

The probability of 11 successive confirmations occurring for a single LPRM during reactor operations near or inside the stability regions in a plant life-time is estimated next. With a plant life-time of 40 years with 5% of the operation near or inside the stability regions, the total probability, $\mathbf{P}_8$, is obtained by multiplying $\overline{p}_8$ of Equation 5-5 by the operation time:

$$\mathbf{P}_8 = \overline{p}_8 \times 40 \times 0.05 \times \frac{\text{year}}{\text{second}} = 5 \times 10^{-4}. \tag{5-6}$$

A more realistic assumption for the time interval between 3-confirmation count occurrences at low decay ratios is on the order of minutes. Therefore, the probability of Equation 5-6 can reasonably be applied to all PBDS LPRMs (approximately 30). Equation 5-6 demonstrates that with an appropriate PBDS calibration to achieve a 2 or a 3-confirmation count during low decay ratio operations, the probability for spurious 11-confirmation count due to random noise perturbations is negligible. Moreover, a spurious Hi-Hi Decay Ratio alarm, which is based on a setpoint logic requiring two overlapping confirmation counts to reach the alarm setpoint, is not credible.

The probability assessment for the Hi-Hi Decay Ratio alarm can be applied in a similar fashion to the Hi Decay Ratio alarm. The Hi Decay Ratio alarm can be selected between 6 and 8 successive period confirmation counts as compared to a 11 successive period confirmation count for the Hi-Hi Decay Ratio alarm. Therefore, Equations 5-4 and 5-5 when applied to the Hi Decay Ratio alarm range for a setpoint of 6 counts results in:

$$\overline{p}_3 \quad (\frac{1}{10})^3 \quad (\frac{1}{2})^3 / (5 \text{ seconds}) \quad 3 \quad 10^{-5} / \text{second} \quad 0.1 / \text{hour} \qquad (5\text{-}7)$$

and for a setpoint of 8 counts results in:

$$\overline{p}_5 = (\frac{1}{10})^5 \times (\frac{1}{2})^5 / (5 \text{ seconds}) \cong 6 \times 10^{-8} / \text{second} \cong 2 \times 10^{-4} / \text{hour} \qquad (5\text{-}8)$$

where $\overline{p}_3$ represents the probability during low decay ratio operations for 3 successive confirmations subsequent to the initial 3-confirmation count, and $\overline{p}_5$ the probability for 5 successive confirmations.

According to Equations 5-7 and 5-8, for a source of random noise, a 6 to 8 period confirmation count occurs during low decay ratio operations approximately once every ten hours to five thousand hours for a single LPRM. This is a conservative estimate, since it is based on a five-second separation between subsequent occurrences of a 3-confirmation count. If a few minute interval is used Equations 5-7 and 5-8 the results can be applied for all PBDS LPRMs. This assessment is conservative, since overlapping global perturbation within one oscillation period and variations in global perturbation amplitudes are ignored. If a Hi Decay Ratio alarm is implemented, considerations of spurious alarm frequency based on this assessment can be used for setpoint selection.

5.3.3.3 Plant-Specific Application

The implementation of the PBDS for a specific plant requires an appropriate determination of the PBA parameters' values to ensure the applicability of the PBDS alarm setpoints. The PBA parameters are divided into a set of generic parameters and a set of plant-specific parameters. The generic parameters are common to all plants and are determined based on the BWR thermal-hydraulic instability phenomenon and PBA performance requirements. The plant-specific parameters are determined based on specific plant neutron noise characteristics by appropriately calibrating the PBDS at normal off-rated operating conditions.

5.3.3.3.1 PBA Generic Parameters Specifications

The basis for the selection of the PBA generic parameters and their values follows:

1. Sample Interval (t_S)

The sample interval should be selected to allow sufficient resolution of an oscillation cycle to determine the magnitude and temporal location of the peaks and minima of the oscillation input signal. It should also be selected to ensure sufficient flexibility in the selection of the period tolerance. The sample interval is therefore set to 50 milliseconds, which represents a short sample interval for the PBDS application.

2. Minimum Oscillation Period (Tmin)

Tmin should provide a lower bound to all expected oscillation periods. On the basis of experience and analysis, the oscillation frequency for an external recirculation pump GE BWR design is less than 0.8 Hz. Tmin is therefore set to 1.2 seconds in order to bound this frequency range.

3. Maximum Oscillation Period (Tmax)

Tmax should provide an upper bound to all expected oscillation periods. On the basis of experience and analysis, the oscillation frequency is greater than 0.25 Hz. Tmax is therefore set to 4.0 seconds in order to bound this frequency range.

4. Conditioning Filter Order (P_C)

PBA testing has demonstrated that a two-pole filter is adequate for the PBDS application. The performance of the conditioning filter depends on the corner frequency (a more direct and convenient parameter to achieve a desired frequency range) in addition to the filter order. The filter order is set equal to 2.

The basis and values of the generic PBDS parameters that are common to all plants are summarized in Table 5-5.

5.3.3.3.2 Determination of PBA Plant-Specific Parameters

The neutron noise characteristic is plant-specific. Since generic PBDS setpoints are used for all plants, the noise characteristic, $\Re^{-1}$, is targeted in a common, narrow range ($4 < \Re^{-1} < 6$). In the period confirmation count model, $\Re^{-1}$ is defined as the effective ratio between a typical global perturbation amplitude and the threshold noise level. The oscillation period confirmation count is determined by the PBA based on raw plant-specific noise signatures and the PBA parameters. Therefore, $\Re^{-1}$ in Equation 5-3 depends on the specific values of the PBA parameters and is the ratio between the initial amplitude of typically large global neutron noise perturbations and the threshold noise amplitude where the oscillation period confirmation process is disrupted, as seen by the PBA. $\Re^{-1}$ is controlled by varying the LPRM signal conditioning filter and the period confirmation tolerance. Since the order of the conditioning filter is fixed, only the filter corner frequency is considered in this process. The period tolerance and the filter corner frequency are set during normal off-rated operating conditions to ensure that the appropriate successive period count is achieved.

LPRM signal conditioning is necessary to allow PBDS oscillation confirmation count capability at the low to intermediate decay ratio range. Without signal conditioning, the high frequency component of the noise will prevent the PBA from discerning oscillation periods until the reactor becomes unstable and the oscillations grow large. However, a residual high frequency component just above the expected frequency range for thermal-hydraulic instabilities is useful in controlling the effective threshold noise level. A lower filter corner frequency reduces the high frequency component of the noise, which results in a low effective threshold noise level and therefore high $\Re^{-1}$ value.

The filter corner frequency is set just above the expected oscillation frequency range (0.3 to 0.7 Hz). A target value of 2.0 Hz is appropriate. Although the PBDS performance is primarily affected by the selection of the period tolerance, the filter corner frequency is also allowed to be adjusted to ensure sufficient design flexibility to achieve the target $\Re^{-1}$ value. For example, with a sufficiently high corner frequency, the period count can be completely suppressed for all decay ratios less than one. The range needed to support the PBDS calibration during normal off-rated conditions is 1.0 to 3.0 Hz.

The period tolerance is used to fine-tune the PBDS successive period count performance. The minimum value that can be selected for the period tolerance is the sample interval, t_S. Period confirmation for this setting is difficult at low and intermediate decay ratio ranges due to variations in measured time periods. As the period tolerance increases, larger variations in the measured time periods can be accommodated and the successive period count increases. For a sufficiently large period tolerance, high period counts can be achieved at any decay ratio.

The period tolerance is selected above the sample interval value (50 milliseconds). A target value of 150 milliseconds is appropriate based on PBA testing. The range needed to support PBDS calibration during normal off-rated conditions is 50 to 300 milliseconds.

The target values and calibration ranges for the period tolerance and the filter corner frequency are summarized in Table 5-6. The PBDS card hardware design allows on-line adjustment of these parameters. For each plant, appropriate PBDS performance is established during expected normal off-rated conditions, with the core decay ratio in the low range, by adjusting these parameters to achieve maximum successive period confirmation counts between 2 and 3.

5.3.3.3.3 Validation of PBDS Calibration Process

A comparison of available test data of PBA performance against actual plant data is used to validate the following PBDS features and setpoint modeling assumptions:

1. The relationship between core decay ratios and successive period confirmation counts, parametric in $\Re^{-1}$.
2. The effect of the adjustable PBA parameters on $\Re^{-1}$.
3. The need for plant-specific PBDS calibration.

The PBA test data was generated by applying the PBA to LPRM signals recorded for different plant operating conditions, including stable and unstable conditions. The same data was also analyzed using conventional noise-analysis methods to establish the decay ratio at each reactor operating condition. In general, the results at high decay ratios are more reliable; noise analysis methods are less accurate when applied to data at low decay ratios.

The application of the PBA to recorded LPRM data generates successive period confirmation counts as a function of recording time. The same data is used to estimate the core decay ratio as a function of recording time. The maximum period confirmation count generated for a time period where the decay ratio is approximately constant is identified for different decay ratio values. This information is compared to the period confirmation count model predictions (Equation 5-3). The correlation between the calculated decay ratio and the confirmation count is expected to be approximate since large variations exist in decay ratios calculated by noise analysis, particularly for the low decay ratio range.

Figures 5-10, 5-11 and 5-12 provide comparisons of estimated decay ratios to successive confirmation counts for a single plant at different fuel cycle conditions and different PBA settings. For all cases, the sample interval is 50 milliseconds. The corner frequency and period tolerance used in each case are indicated in the figures. The test data in the figures correspond to several LPRM signals. This is done to simulate the actual PBDS output which is based on up to 18 LPRM signals. The scatter in the confirmation count is due to uncertainties in calculated decay ratios and variations in LPRM response.

Figure 5-10 illustrates a set of PBA parameter values that result in an appropriate PBDS calibration for the analyzed reactor. A corner frequency of 1.5 Hz and period tolerance of 100 milliseconds result in a decay ratio to confirmation count relation nominally associated with $\Re^{-1} \cong 5$. Calibration of the PBDS with these settings for this plant and fuel cycle will result in a maximum of 2 to 3 successive confirmations at the low decay ratio range during normal operations and a Hi-Hi Decay Ratio alarm when a 0.8 to 0.9 decay ratio is approached. An increasing decay ratio corresponding to greater confirmation counts, consistent with Equation 5-3 is evident.

Figure 5-11 illustrates two different sets of PBA parameters with different filter corner frequency values that are applied to the same test data. The period tolerance is 150 millisecond and the filter corner frequencies is either 1.5 or 2.5 Hz. The trends of the decay ratio as a function of successive period confirmation counts are consistent with the confirmation count model (Equation 5-3) for both settings. In addition, lowering the filter corner frequency results in a general increase in period confirmation count for a given decay ratio. This change increases the PBDS sensitivity. The observed trend is consistent with the expected effect of lowering the filter corner frequency to reduce the disruptive effects of the high frequency noise component on the confirmation process.

The PBA sensitivity, in particular with the high corner frequency, is low as indicated by the response trends which lie above the target $\Re^{-1}$. The PBDS sensitivity should be increased in the low decay ratio range to achieve a maximum of 2 to 3 successive confirmation counts. This can be achieved by either lowering the filter corner frequency further (e.g., to 1.0 Hz) or by increasing the period tolerance (e.g., to 200 milliseconds).

Figure 5-12 illustrates the PBDS response for two sets of PBA parameters with different period tolerance values. These are applied to the test data used to generate the results of Figure 5-10. Only high decay ratio data are shown for this comparison. The trends of decay ratio as a function of successive confirmation count are shown for different values of the period tolerance. A higher confirmation count is observed for the higher period tolerance value. This trend is consistent with the expected effect that higher period tolerance has on the reduction in discrimination of identified periods relative to the base period.

The low period tolerance setting (i.e., 100 milliseconds) provides an appropriate PBDS calibration for the analyzed reactor conditions that are consistent with the conditions in Figure 5-10. The higher period tolerance value (i.e., 150 milliseconds) results in a PBA performance below the target. This setting is too sensitive and will result in excessive confirmation count in the low decay ratio range.

A comparison of the parameter settings used to process the LPRM data shown in Figure 5-10, to the low filter corner frequency parameter setting used to process LPRM data from an earlier cycle for the same plant as shown in Figure

5-11, demonstrates that variations in plant noise characteristics between cycles may require cycle-specific tuning of certain PBA parameters. This tuning is necessary to ensure performance of the PBDS consistent with the confirmation count model. The parameter settings of Figure 5-11, which are more relaxed relative to Figure 5-10 (i.e., relative higher period tolerance for the same filter corner frequency), should result in a higher $\Re^{-1}$ if applied to the same plant conditions represented by Figure 5-10. However, the resulting $\Re^{-1}$ in Figure 5-11 is lower for the specific plant cycle characteristics. Therefore, the reload cycle represented in Figure 5-11 has a more noisy LPRM signal than the initial cycle in Figure 5-10. This data confirms the need for cycle-specific verification of the appropriate sensitivity of the PBDS in the low decay ratio range during reactor startup. The appropriate calibration of the PBDS can be verified or adjusted as necessary at the beginning of each cycle by ensuring that the period confirmation count reaches a maximum of 2 to 3.

5.4 Automatic Features

5.4.1 Flow-Biased Scram Clamp

The Exclusion Region is demonstrated to provide protection for the MCPR safety limit for all anticipated events, including core flow reduction events. For flow reduction events, the location of the Exclusion Region clamp is specifically validated against reasonably limiting IFREs that terminate within the Restricted Region. Since these events initiate from areas of the operating domain where no specific stability related restrictions on core power distribution exist, they potentially pose a challenge to the stability solution. For these events, analysis demonstrates that reasonably limiting IFREs that terminate outside of the Exclusion Region remain stable. However, unanticipated combinations of plant transients involving an IFRE terminating within the Restricted Region may potentially result in unacceptable loss of stability margin.

Specifically, LOFH transients that are not normally associated with IFREs can cause high core decay ratios if initiated immediately following the IFRE. These unanticipated scenarios are characterized by large power increases following the core flow reduction. The power increase is caused by the core inlet subcooling transient associated with the LOFH. The combination of low initial core average boiling boundary and increased power can diminish reactor stability margin under these unanticipated conditions.

To provide protection from this and other unexpected combinations of transients at high flow-control lines within the Restricted Region, the core flow-biased neutron flux scram function of the Neutron Monitoring System has its power axis intercept clamped above the highest licensed operating flow-control line in the Restricted Region. This defense-in-depth feature provides substantial protection from the most severe unanticipated stability related transients, which occur at high flow-control lines. Because these events usually involve large core power increases, the transient is automatically terminated when the clamped flow-biased scram setpoint is reached. The protection provided by the clamped flow-biased scram is illustrated in Figure 5-13.

Because this feature is not part of the licensing methodology for Enhanced Option I-A, setpoint methodology and explicit analysis are not necessary. Furthermore, the clamped flow-biased scram setpoint cannot be adjusted too low, or the frequency of unnecessary reactor scrams and associated challenges to reactor safety systems will be adversely affected. Based on these considerations, the actual clamped flow-biased scram setpoint is placed such that it is approximately 5% above the highest normal APRM signal value, including noise, that is present when operating on the highest flow-control line in the Restricted Region. This setpoint is plant-specific.

5.4.2 Restricted Region Entry Alarm

Inadvertent entry into the Restricted Region may result in reactor operating states that are not in compliance with the stability control limit. Since immediate confirmation of compliance with the stability control limit is not possible, and terminal reactor stability conditions are unknown during a transient, continued operation within the Restricted Region is not consistent with the overall solution philosophy, even though the analysis of these anticipated events demonstrates that the reactor remains stable. To address this situation, an automatic Restricted Region Entry alarm is added to the stability solution as a defense-in-depth feature (Figure 5-14).

This automatic alarm prompts mandated manual actions to exit the Restricted Region. The alarm is generated by the flow-biased control rod withdrawal block function of the Neutron Monitoring System. This function defines the Restricted Region boundary, and prohibits inadvertent entry by control rod withdrawal as a licensing feature of the stability solution. The same Neutron Monitoring System hardware that generates a flow-biased reference for the control rod block function also generates the Restricted Region Entry alarm whenever the reactor state crosses the Restricted Region boundary and the setpoints are not setup. Because the same hardware is used, this defense-in-depth feature provides a highly reliable automatic indication to the operator that immediate corrective action is required.

5.4.3 PBDS Alarms

The PBDS generates two alarms. The Hi-Hi Decay Ratio alarm annunciates following an unacceptable loss of stability margin. Since the stability solution is designed such that all events anticipated to result in conditions susceptible to power oscillations lie within the Exclusion Region, this automatic defense-in-depth feature is not expected to activate under anticipated operating conditions. Therefore, annunciation of this alarm implies that the reactor is in an unanticipated condition, and continued reactor operation is not appropriate. Operator action to manually scram the reactor without delay upon receipt of this alarm is required. The Hi Decay Ratio alarm is optional and can be used to annunciate reductions in stability margin.

5.4.4 Automatic Stability Region Setpoint Setdown

Setdown of the Exclusion Region and Restricted Region setpoints is performed upon exiting the Restricted Region. This requirement is part of the licensing methodology of the Enhanced Option I-A stability solution, and provides the necessary protection from limiting FREs, prior to removing the stability control.

In order to provide defense-in-depth for this action, an automatic setdown of the Exclusion and Restricted Region boundaries is provided. Since the majority of reactor operating time is spent at rated power, this feature is designed to provide assurance that the stability region boundaries are at their normal setpoints when operating in this condition. This function is accomplished with a core flow referenced trigger that switches the FCTRC setpoint algorithms from the setup condition to the appropriate normal values. The flow referenced setpoint, $W_{S/D}$, is specified to be:

$$W_{S/D} = W_{A'_{nom}} + 5\% \tag{5-9}$$

This value provides appropriate margin to the Restricted Region boundary to prevent inadvertent setdown near the high flow corner, A'_{nom}. However, the value is sufficiently low to ensure that the stability region boundary setpoints are in the normal setting during plant power operation.

5.5 Manual Features

Enhanced Option I-A makes extensive use of manual operator action to provide defense-in-depth protection. Although not acceptable as part of the solution licensing basis, operational experience has clearly demonstrated the efficacy of operator action to mitigate reactor transients. To bolster the reliability of manual actions, most actions are contingent upon receipt of automatic alarms generated by defense-in-depth features that are continuously and autonomously monitoring the reactor state. Furthermore, the manual actions are pre-specified and mandated by plant Technical Specifications. The combination of strict administrative controls and responses that are keyed to automatic defense-in-depth features results in a highly reliable backup means of preventing reactor instability.

5.5.1 Restricted Region Actions

5.5.1.1 Uncontrolled Entry into Restricted Region

The Restricted Region is defined as the area of the licensed operating domain susceptible to reactor instability in the absence of adherence to stability controls. Transients that initiate outside of the Restricted Region and terminate within it do not necessarily conform to these stability controls. In addition, the stability controls utilized in Enhanced Option I-A are not designed to maintain control of power distributions during transients. In consideration of these limitations, explicit demonstration that these type of reasonably limiting transients do not result in reactor instability is incorporated into the validation process of the licensing stability regions.

Consistent with the solution design philosophy of defense-in-depth and layered protection, additional protection from this class of events is appropriate. Enhanced Option I-A provides two additional protective features for transients that terminate in the Restricted Region.

Any uncontrolled entry into the Restricted Region will automatically generate a Restricted Region Entry alarm. Based upon receipt of this alarm, an instruction to immediately initiate action to exit the Restricted Region is administratively specified. Generally, this action will be accomplished through control rod insertion. The ability of control rod insertion to mitigate increases in core decay ratio is well documented. The limiting stability transients that terminate in the Restricted Region are LOFH events and IFREs. Analysis and operational experience demonstrate that the core inlet subcooling transient associated with these events, which reduces stability margin, occurs slowly. Because the stability transient is slow, and operator action is keyed by an automatic alarm, this defense-in-depth feature provides a highly reliable means of maintaining reactor stability.

In addition, the PBDS functions as a tertiary protective layer against the onset of reactor instability. The PBDS automatically monitors the reactor for loss of stability margin. Since no other means of monitoring the stability effects of the core power distribution exists during a transient, this defense-in-depth feature must be functioning for continued reactor operation to be appropriate. Therefore, if the PBDS is not operating, immediate manual scram is necessary following uncontrolled entry into the Restricted Region.

Since no anticipated reactor transients that terminate in the Restricted Region are expected to result in reactor instability, receipt of the PBDS Hi-Hi Decay Ratio alarm should not occur. A immediate manual scram is required if the alarm does occur, since possible operation under unanalyzed conditions may be occurring. This defense-in-depth feature is provided to protect against hypothetical events that lie outside the licensed design basis of the reactor.

5.5.1.2 Controlled Entry Into Restricted Region

The licensing methodology of Enhanced Option I-A requires adherence to the reactor stability control limit, FCBB, during controlled operations within the Restricted Region. Under these conditions, the reactor is extremely stable due to the presence of a high core average boiling boundary. The validation analyses performed as part of the licensing methodology confirm that all reasonably limiting events initiating within the Restricted Region remain stable due to the very low decay ratios at the beginning of the transients.

During any such transient, however, adherence to the FCBB limit cannot be confirmed, and the terminal conditions of the reactor are not known. Consistent with the concept of defense-in-depth, manual operator action to immediately initiate action to exit the Restricted Region following the initiation of any transient is required. This action prevents operation in the Restricted Region when reactor stability margin is indeterminate. Since any flow reduction event that occurs while operating in the Restricted Region at a high flow-control line is likely to terminate in automatic scram due to the proximity of the Exclusion Region, the highest frequency events expected to result in operator actions in this situation are the LOFH event and the flow reduction event at low flow-control rod lines. LOFH events evolve slowly, and sufficient time to conduct corrective manual actions is expected. Flow reduction events within the Restricted Region are associated with a small core flow change and very stable initial conditions.

Again, the PBDS functions as a tertiary protective layer against the onset of reactor instability. The PBDS automatically monitors the reactor for loss of stability margin. Since no other means of monitoring the stability effects of the core power distribution exists during a transient, this defense-in-depth feature must be functioning for continued reactor operations to be appropriate. Therefore, if the PBDS is not operating during a transient within the Restricted Region, immediate initiation of action to exit the region is necessary.

Since no anticipated reactor transients that initiate in the Restricted Region are expected to result in reactor instability, receipt of the PBDS Hi-Hi Decay Ratio alarm should not occur while exiting the region. An immediate manual scram is required if the alarm does occur, since possible operation under unanalyzed conditions may be occurring. This defense-in-depth feature is provided to protect against hypothetical events that lie outside the licensed design basis of the reactor.

5.5.2 Monitored Region Actions

The Monitored Region extends over the area of the power/flow operating domain that bounds reactor states where reactor instability is hypothetically possible under extreme conditions that exceed the licensed design basis of the reactor. Because the design philosophy of the solution demands that the degree of protection be commensurate with the probability for reactor instability, the Monitored Region requires the least amount of protection. Reactor instability under any anticipated operating conditions is not expected within this region. Therefore, the only defense-in-depth feature in the Monitored region is the PBDS. Receipt of any PBDS alarm is sufficient to warrant corrective action because high core decay ratios are not expected during anticipated reactor operations or transients within the region. An immediate manual scram is required following annunciation of the PBDS Hi-Hi Decay Ratio alarm, since possible operation under unanalyzed conditions may be occurring.

Because the PBDS is the only defense-in-depth feature in the Monitored Region, it must be functioning in order to operate within the region. In addition, any transient that terminates within the Monitored Region with the PBDS inoperable requires action to exit the region because no backup protection is available.

Entry into the Monitored Region is allowed for a limited duration with the PBDS inoperable for the purpose of controlled shutdown. Since defense-in-depth features are not available in the Monitored Region under these conditions, compliance with the stability control limit, FCBB, is required. Compliance with the FCBB limit ensures significant stability margin during power decension in the Monitored Region and obviates unnecessary reactor scrams.

Table 5-1: Decay Ratio Range Summary

Decay Ratio Range	Decay Ratio Values
Low	$0.0 \leq DR < 0.5$
Intermediate	$0.5 \leq DR < 0.8$
High	$DR \geq 0.8$

Table 5-2: Alarms Conditions Summary

Operating Conditions	Alarm	Required Response	Decay Ratio Range
Normal	None	None	0.0-0.5
Anticipated events	Hi Decay Ratio (early warning)	Optional	0.6-0.7
Extreme operating conditions and unanticipated events	Hi-Hi Decay Ratio (manual scram)	Mandatory	≥ 0.8

Table 5-3: PBDS Alarm Setpoints

Alarm/Operating Conditions	Successive Period Confirmation Count
Expected off-rated operations	2-3
Hi Decay Ratio alarm (Optional)	6-8
Hi-Hi Decay Ratio alarm	11

Table 5-4: PBDS Alarm Logic

Alarm	Alarm Logic (per channel)
Hi-Hi Decay Ratio (Mandatory)	Two-out-of-all-LPRMs, Once
Hi Decay Ratio (Optional)	One-out-of-all-LPRMs, Once

Table 5-5: PBDS Algorithm Generic Parameters

PBDS Parameter	Value	Basis
t_S - Sample Interval (milliseconds)	50	Oscillation period resolution and flexibility in selecting the period tolerance ε as a multiple of t_S
Tmin - Minimum Oscillation Period (seconds)	1.2	Lower bound of observed, expected oscillation period
Tmax - Maximum Oscillation Period (seconds)	4.0	Upper bound of observed, expected oscillation period
P_C - Filter Order for Conditioning Filter	2	Algorithm testing

Table 5-6: PBDS Algorithm Plant-Specific Parameters

PBDS Parameter	Target Value	Expected Calibration Range
ε - Period Tolerance (milliseconds)	150	50 to 300
f_C - Corner Frequency for Conditioning Filter (Hz)	2.0	1.0 - 3.0

Figure 5-1: PBDS Application Region

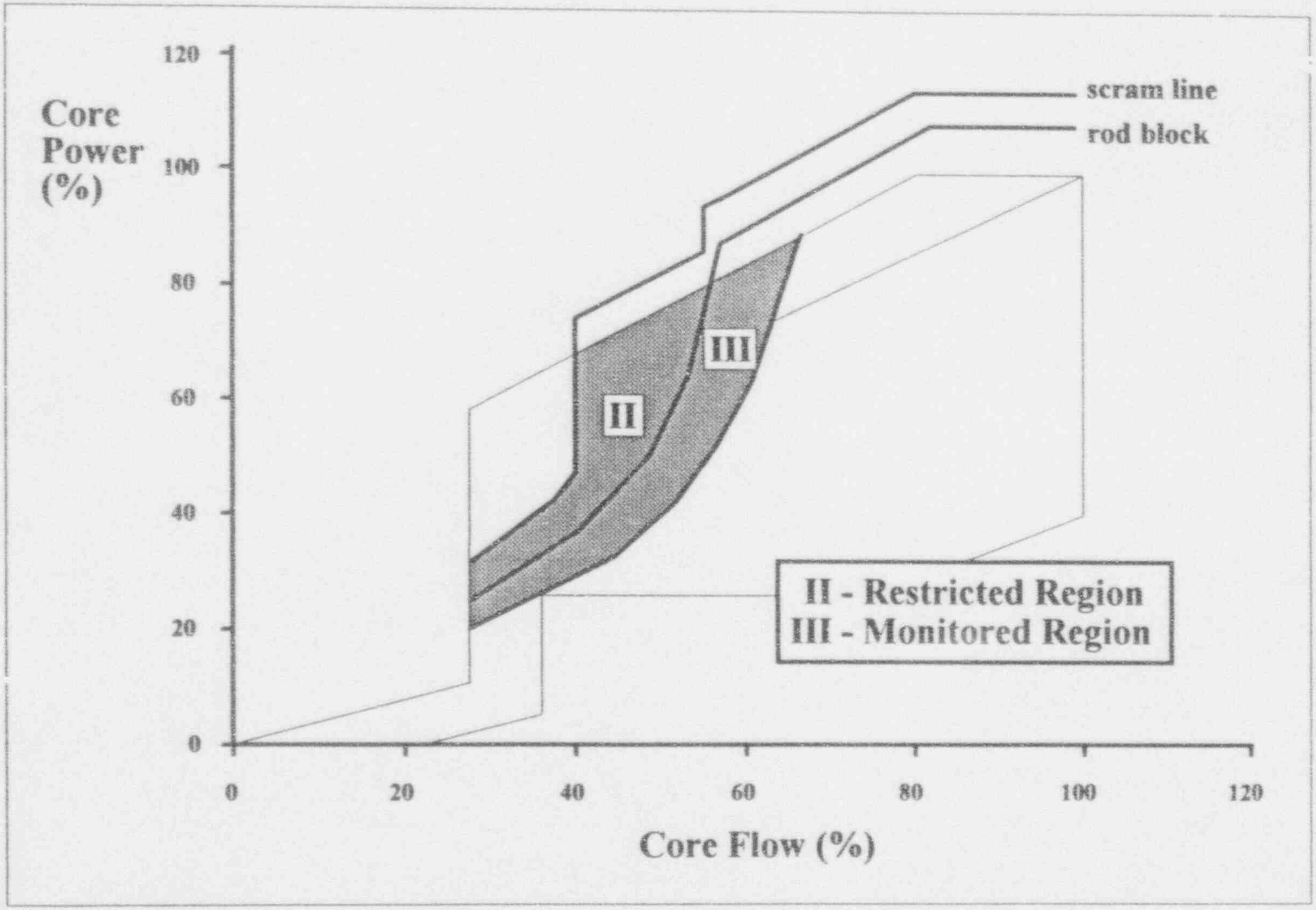


Figure 5-2: Monitored Region Boundary Generation Stability Criterion

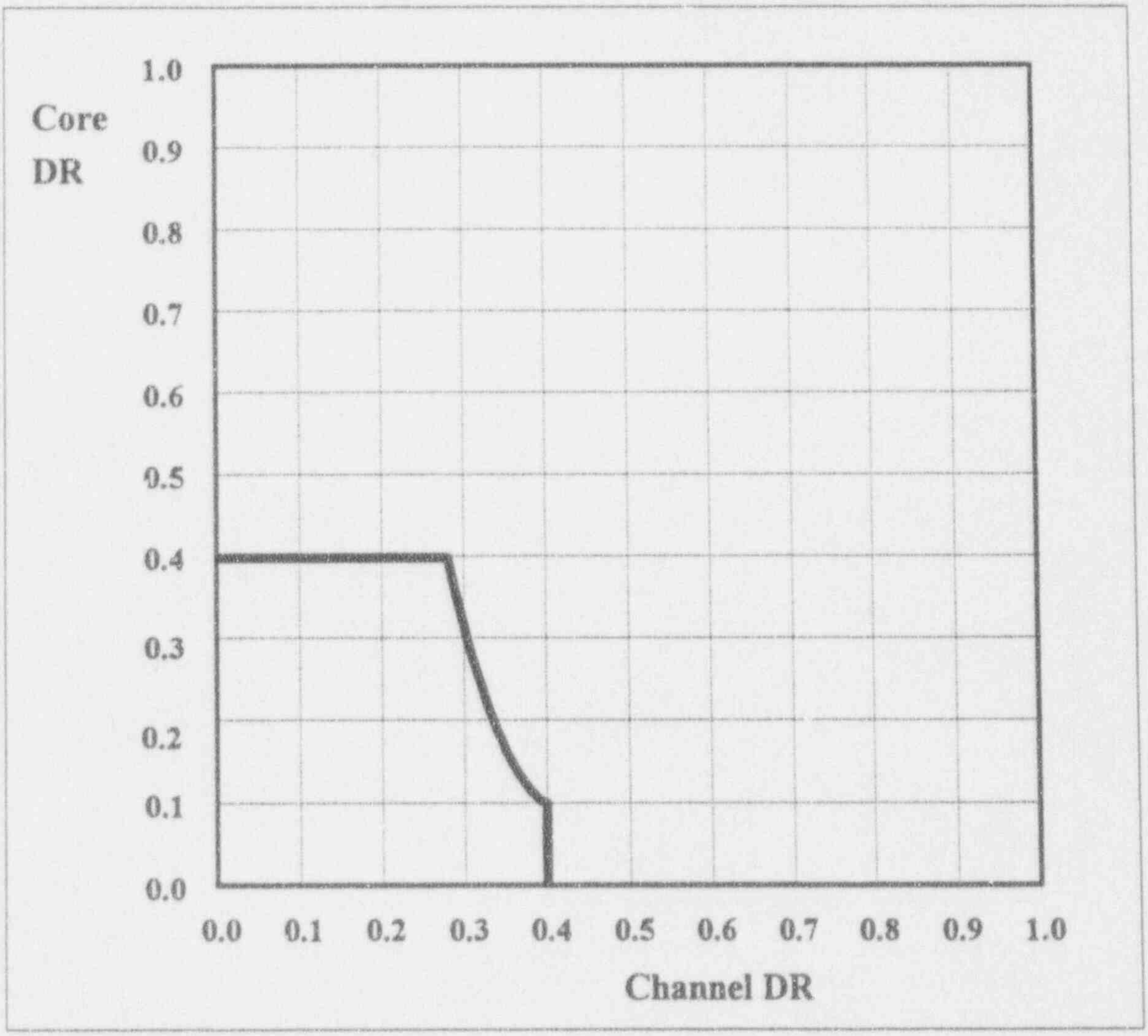


Figure 5-3: Asymptotic Amplitude Behavior

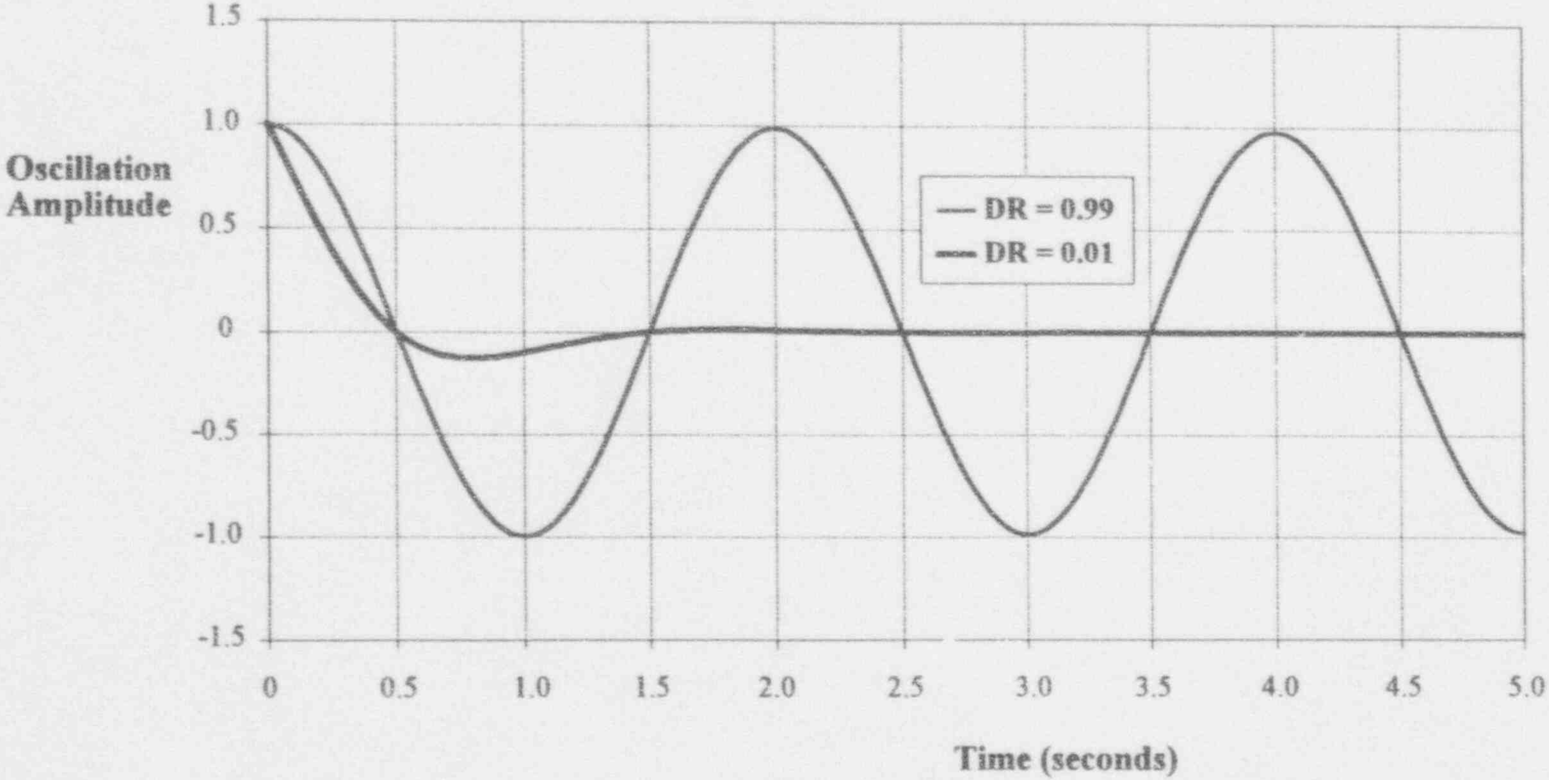
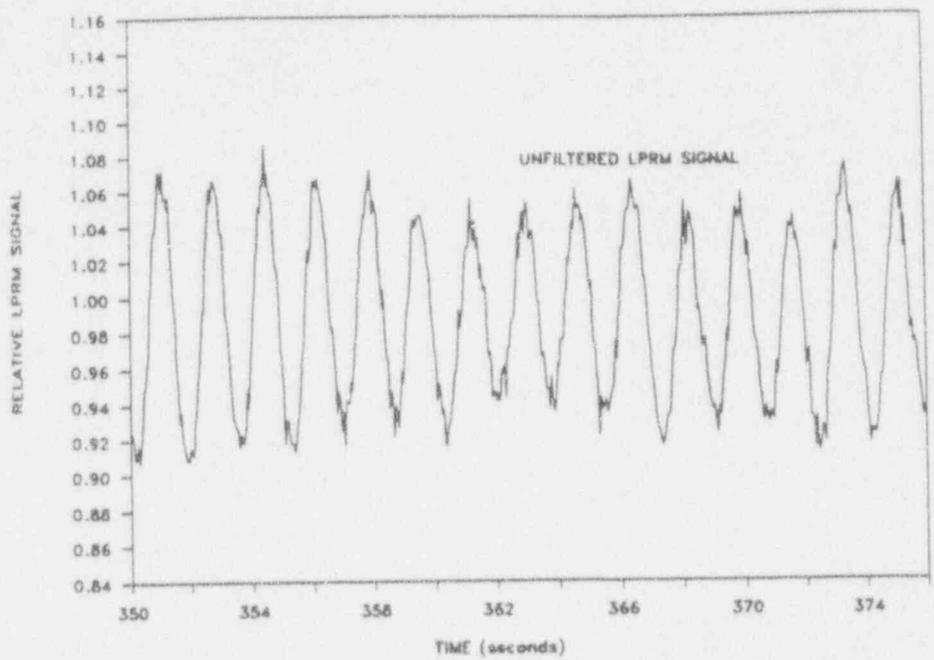


Figure 5-4: Conditioning Filter - Limit Cycle Data



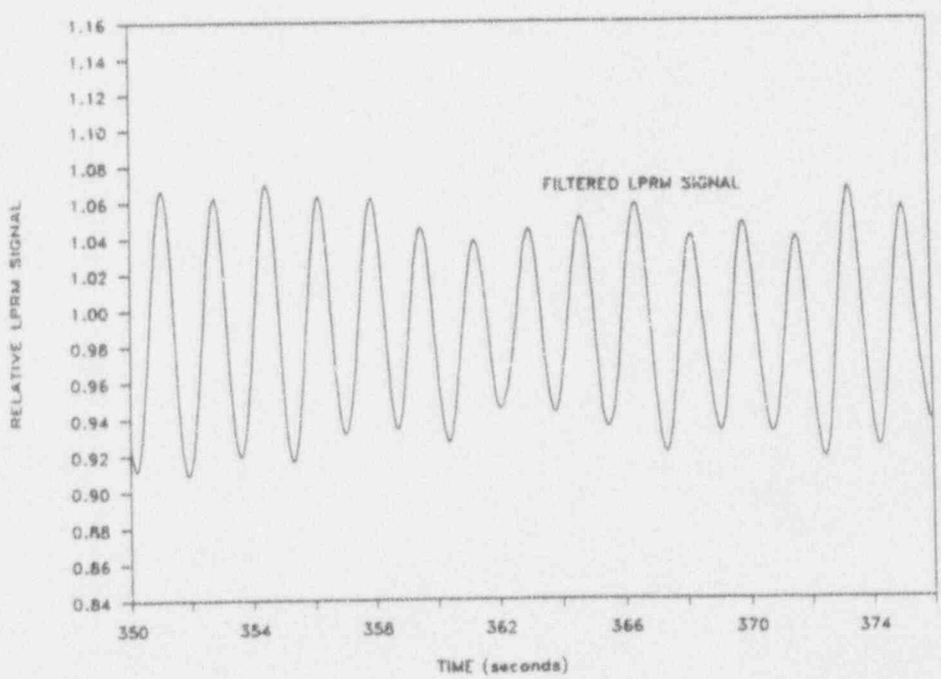


Figure 5-5: Period Confirmation Count Model

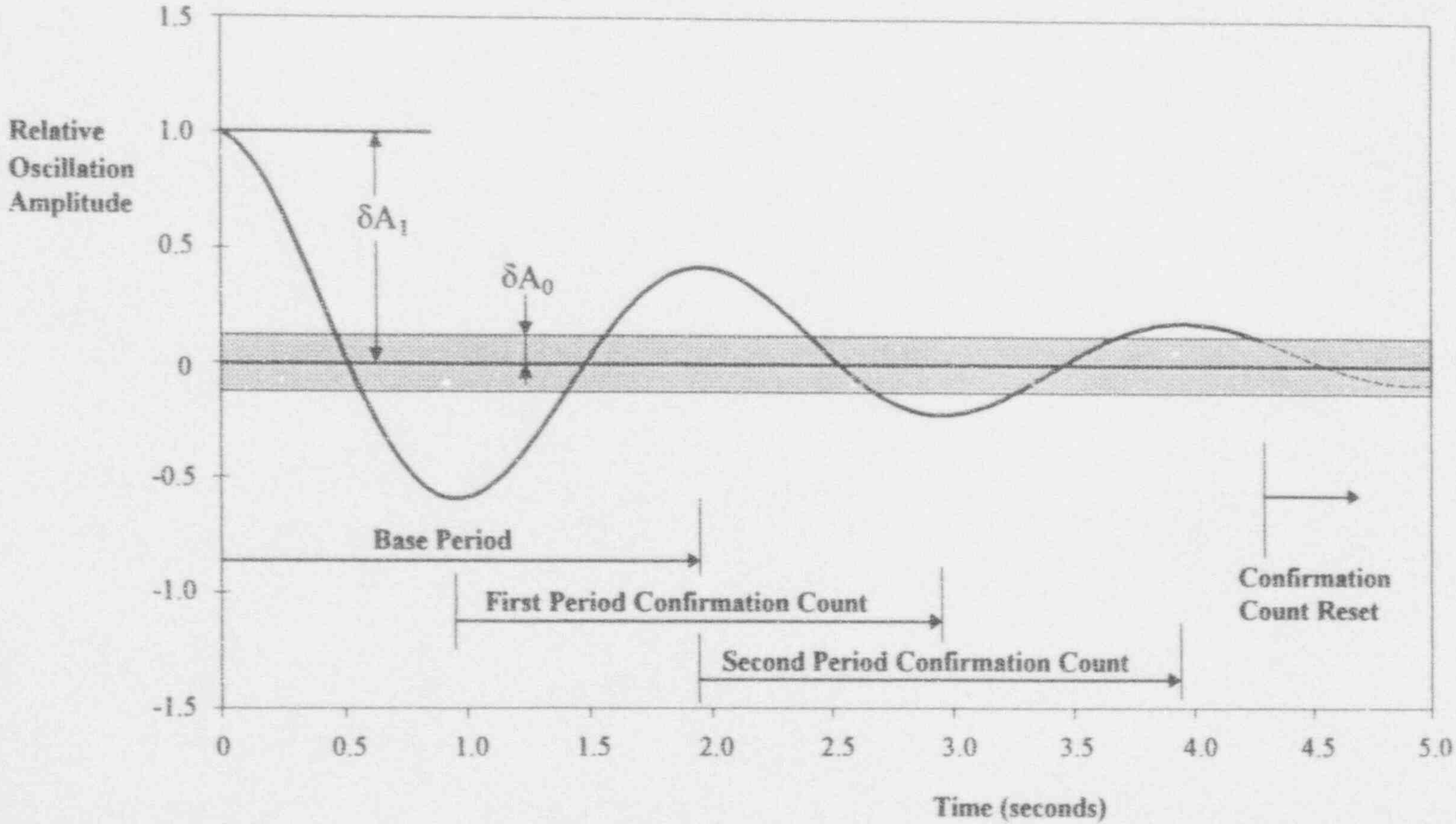


Figure 5-6: Noise Factor Effect on Stability Performance

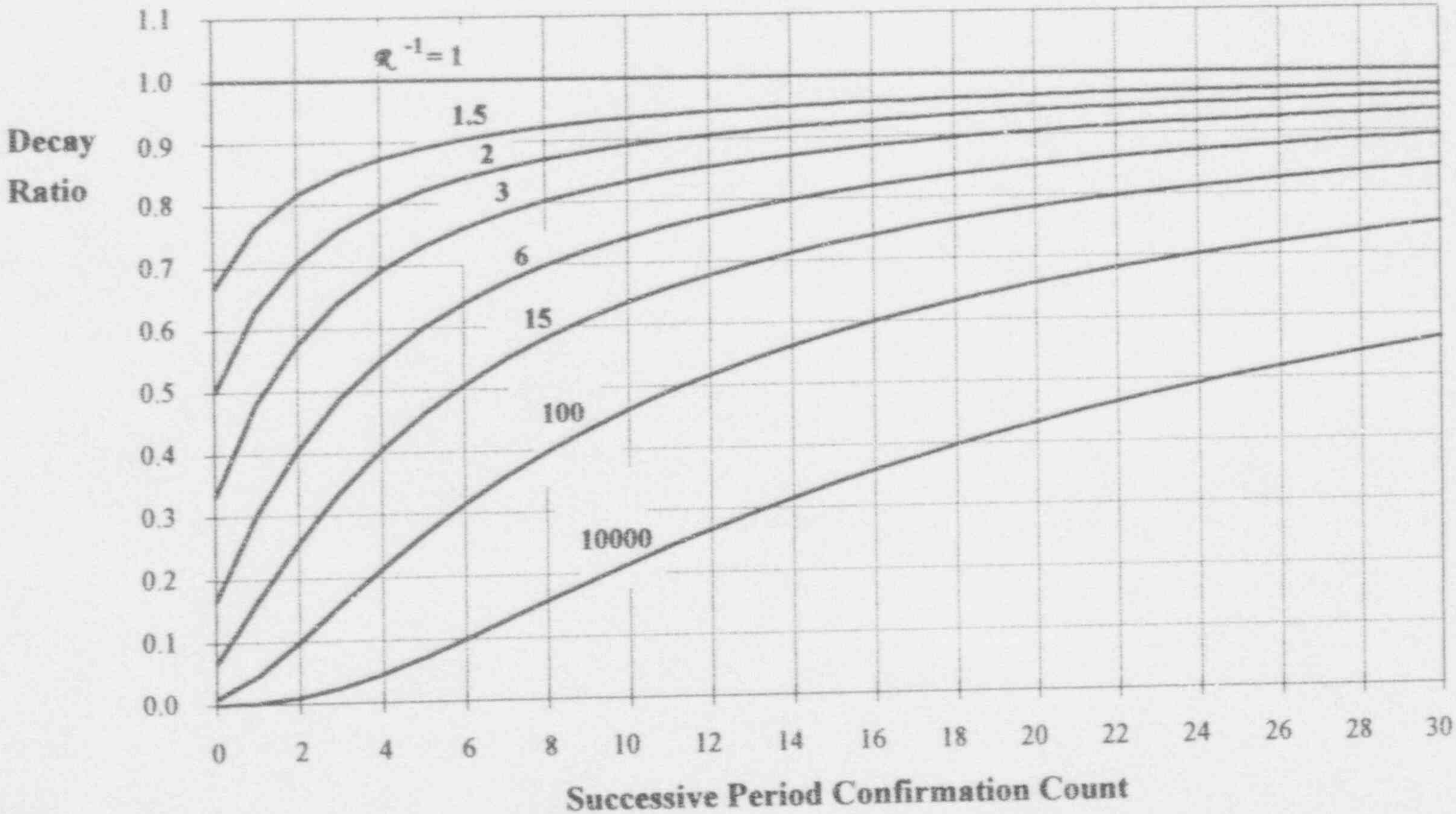


Figure 5-7: Stability Margin Consideration for Alarm Setpoint Determination

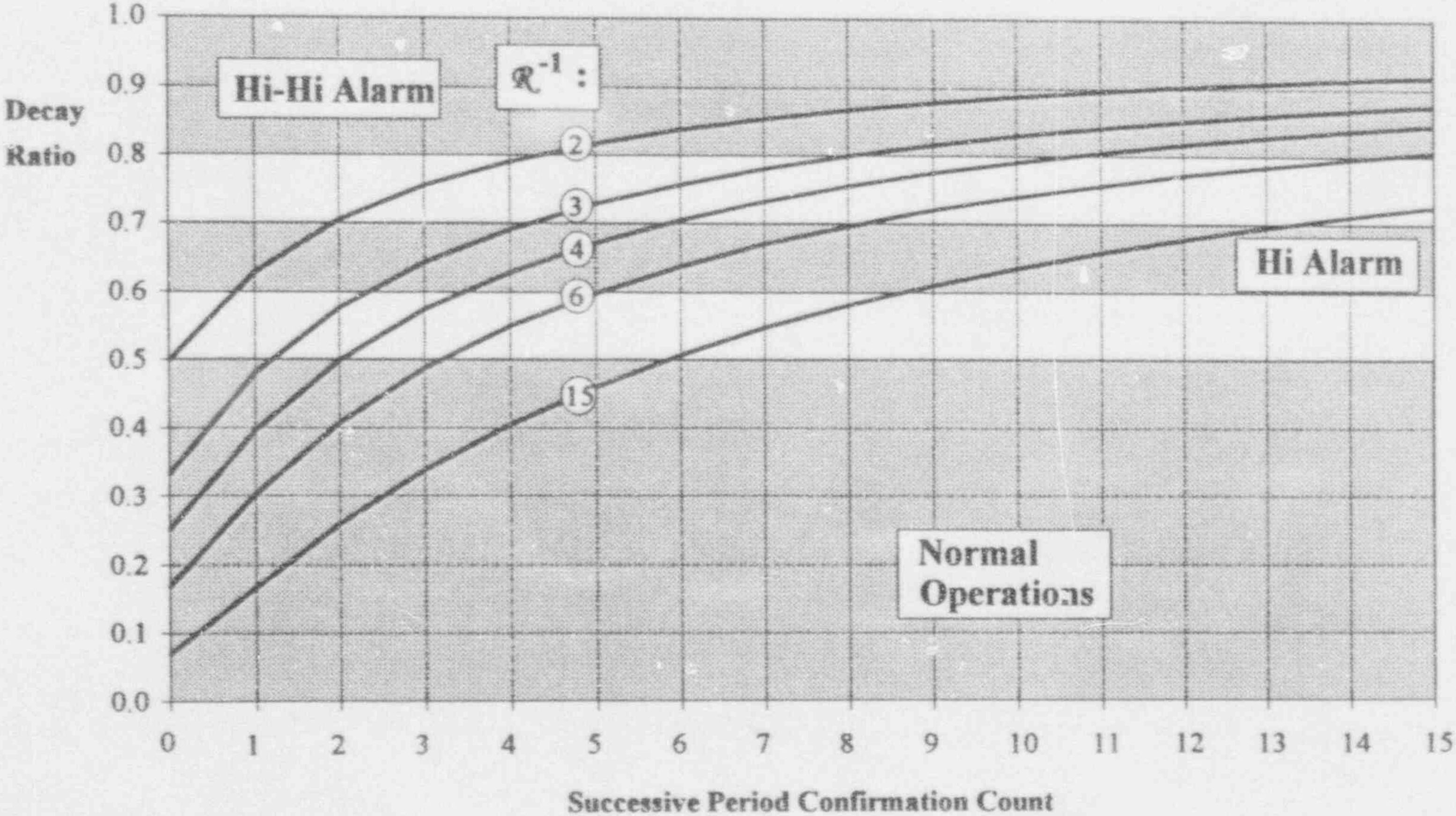


Figure 5-8: PBDS Confirmation Count Range for Alarm Setpoint Determination

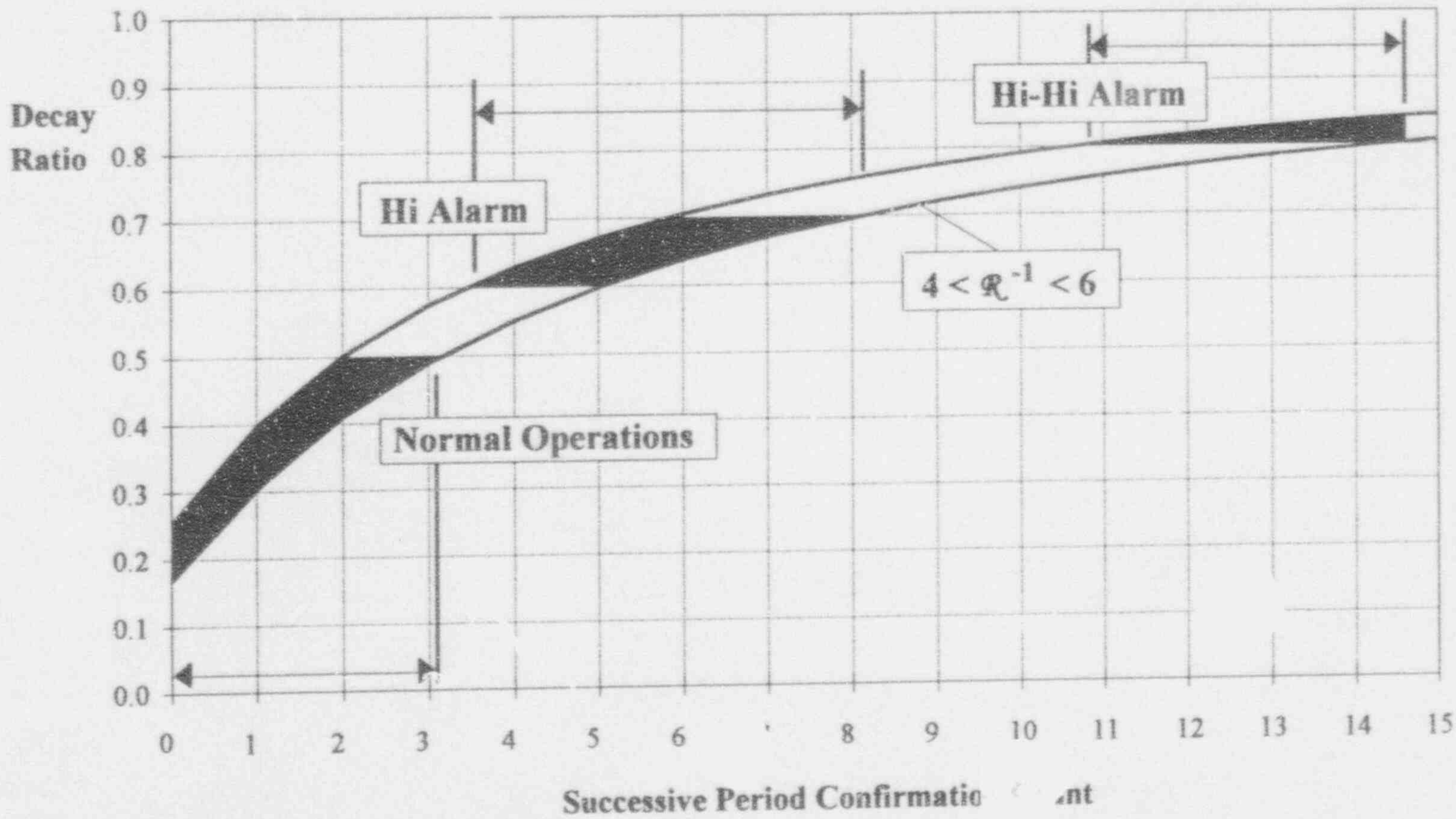


Figure 5-9: PBDS Alarm Setpoints

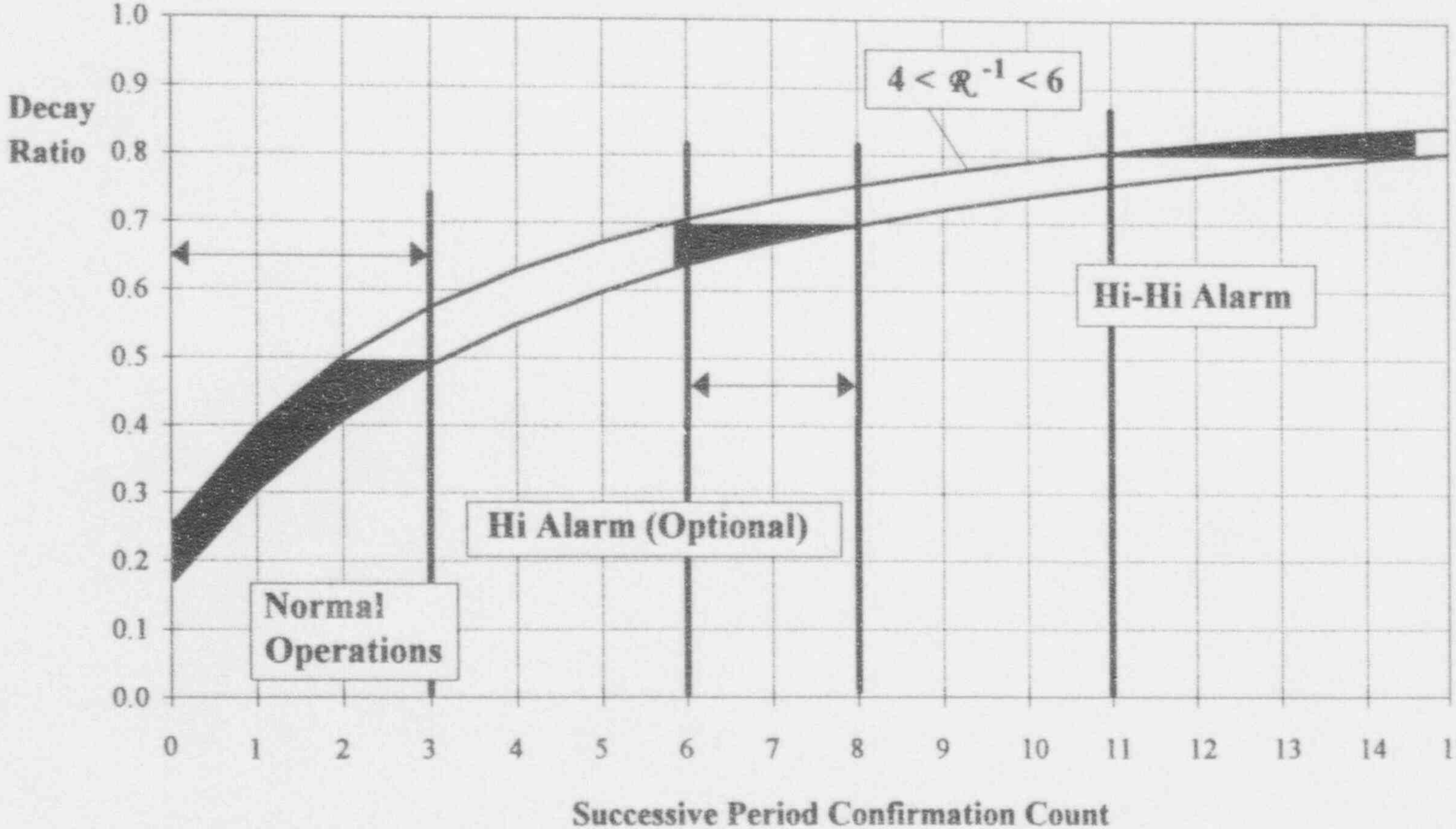


Figure 5-10: PBDS Calibration Process Testing

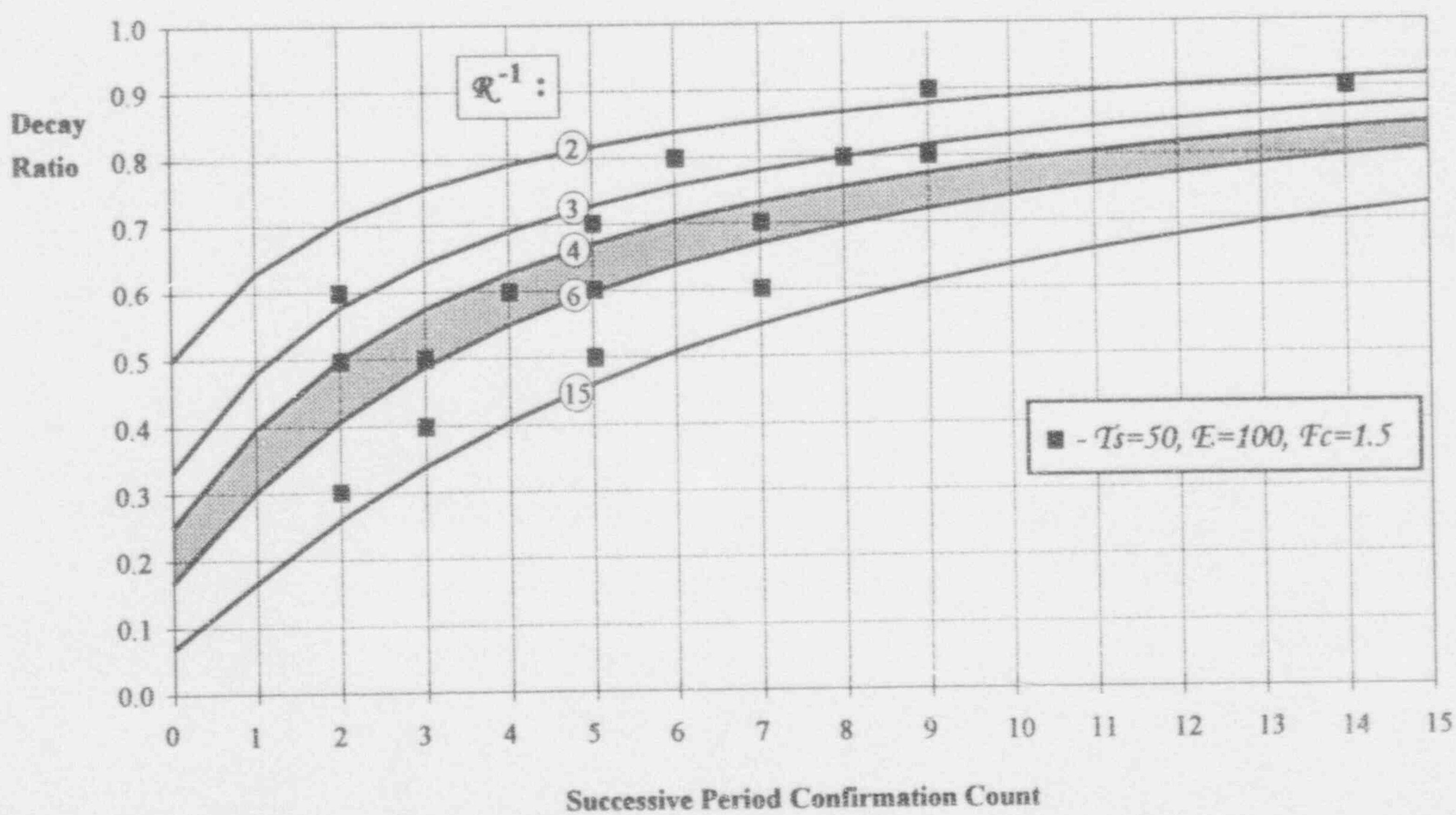


Figure 5-11: PBDS Corner Frequency Calibration Testing

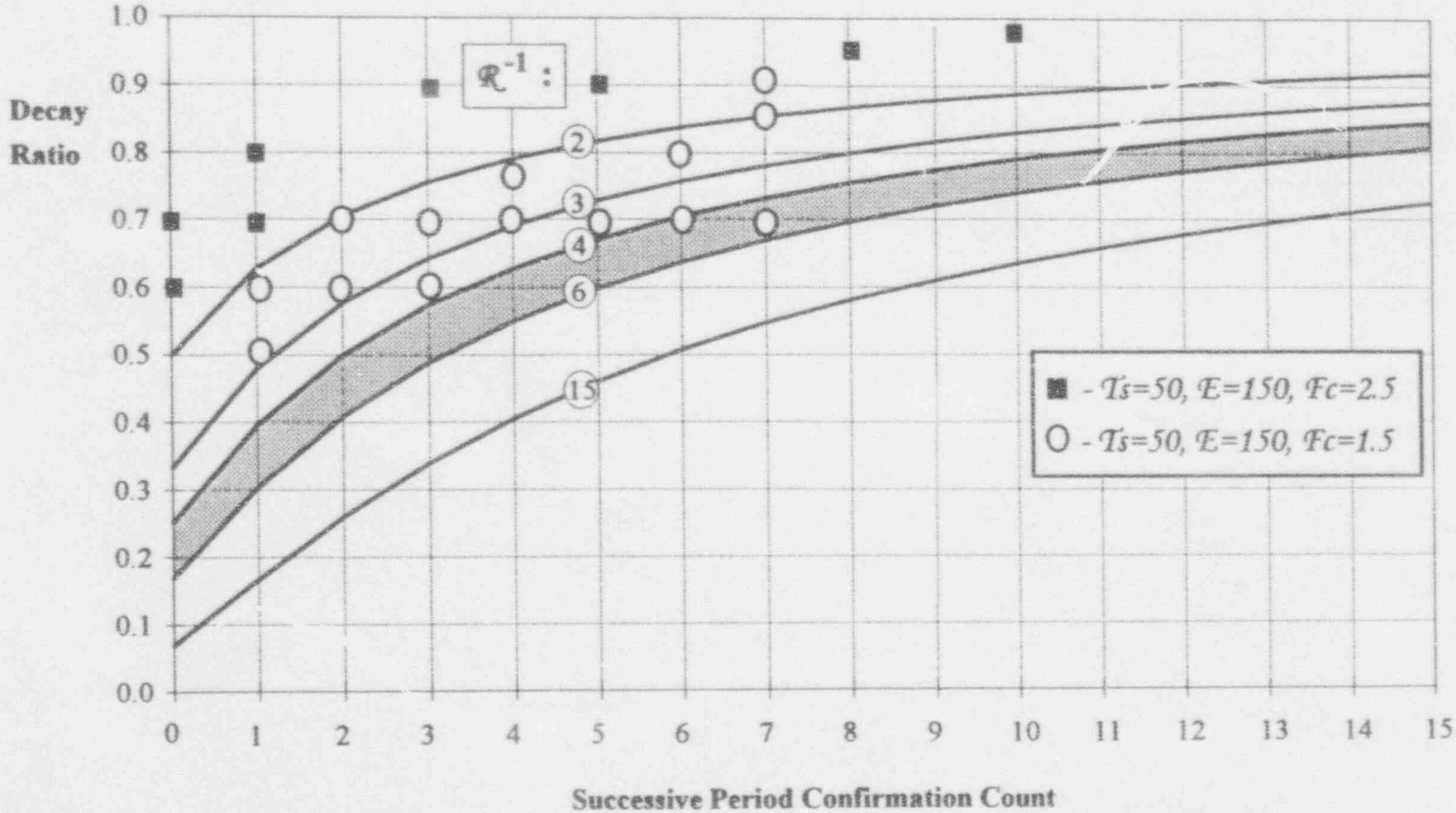


Figure 5-12: PBDS Period Tolerance Calibration Testing

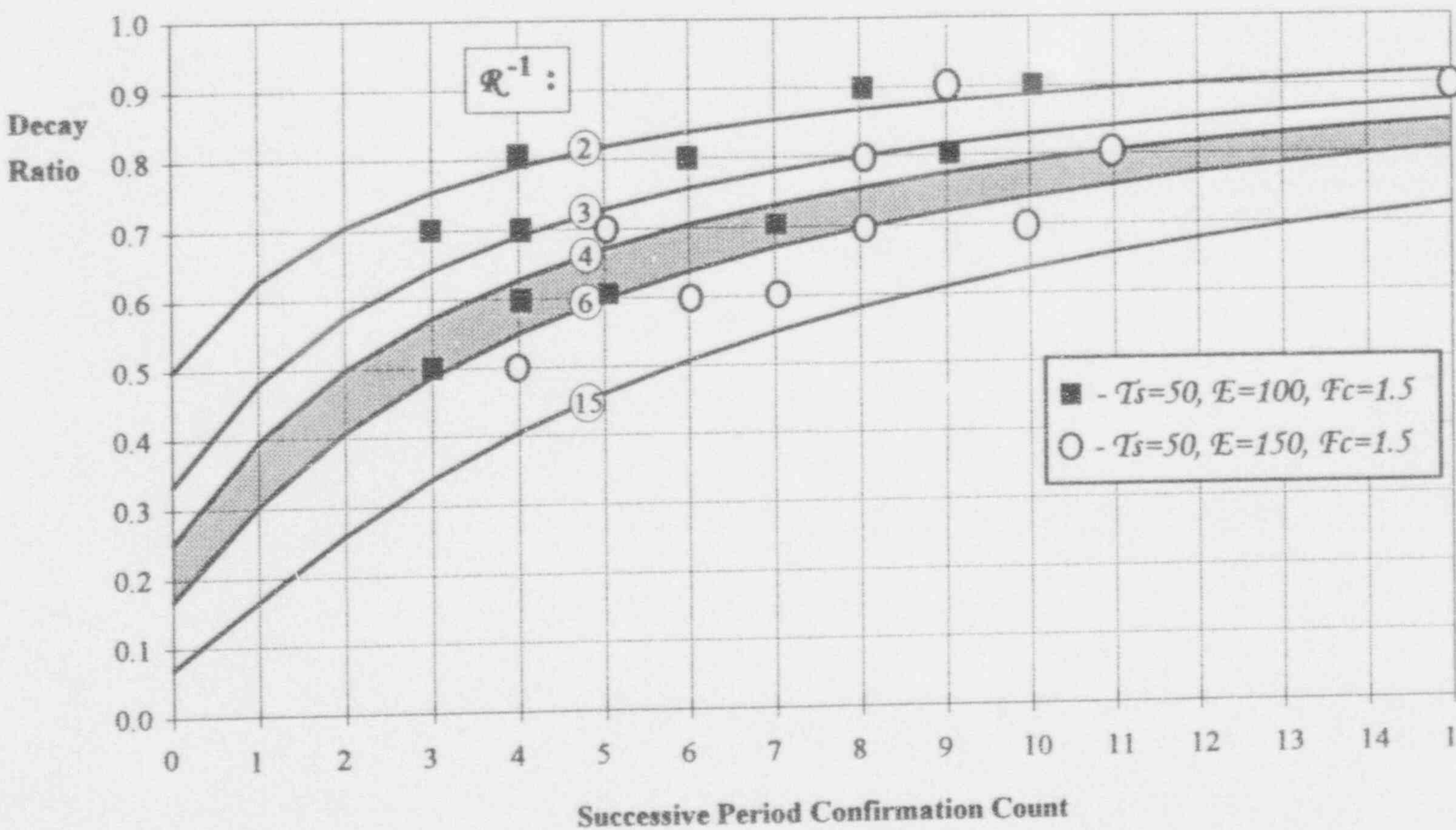


Figure 5-13: Unanticipated Combinations of Events Terminated By Flow-Biased Scram Clamp (Example)

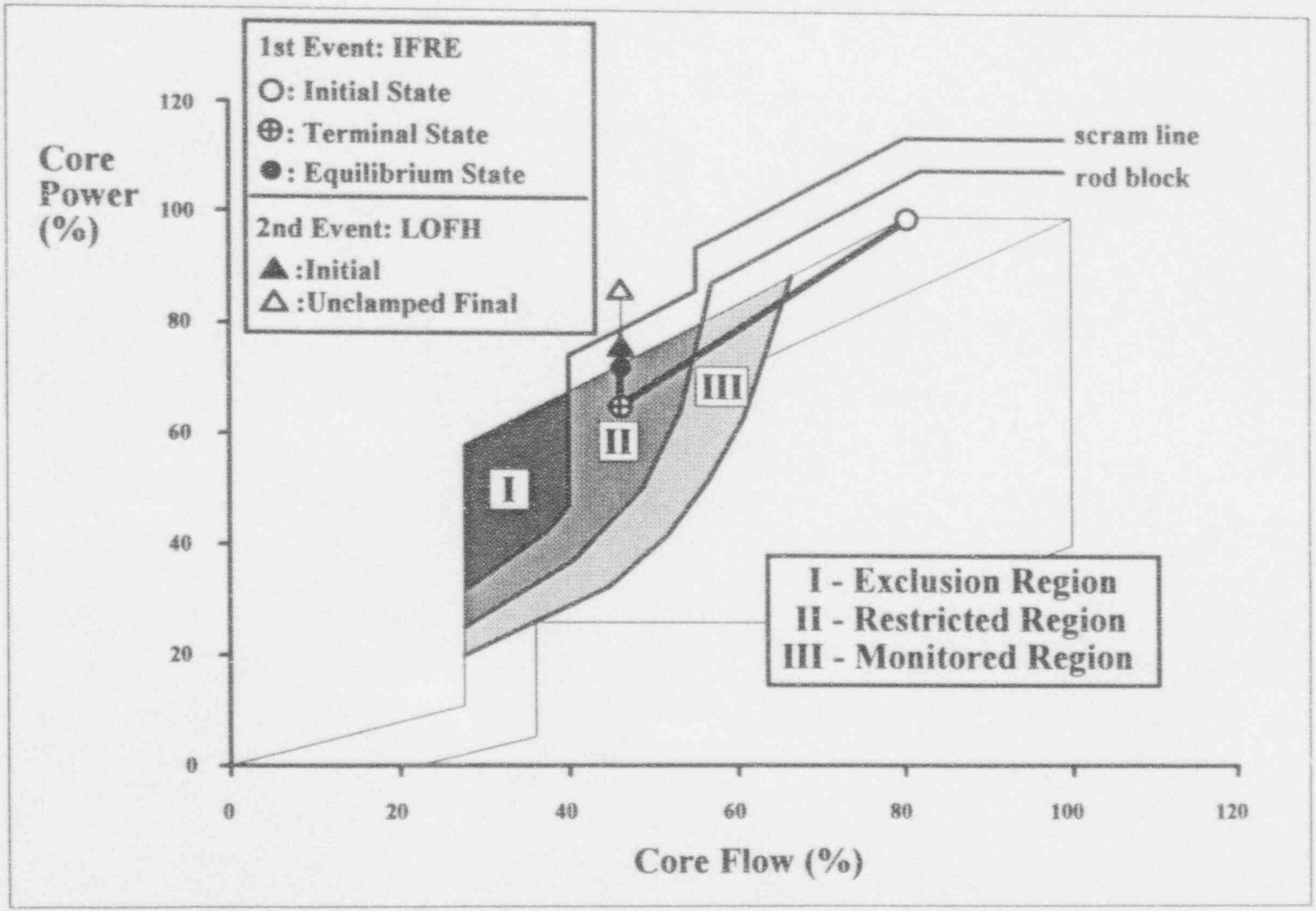
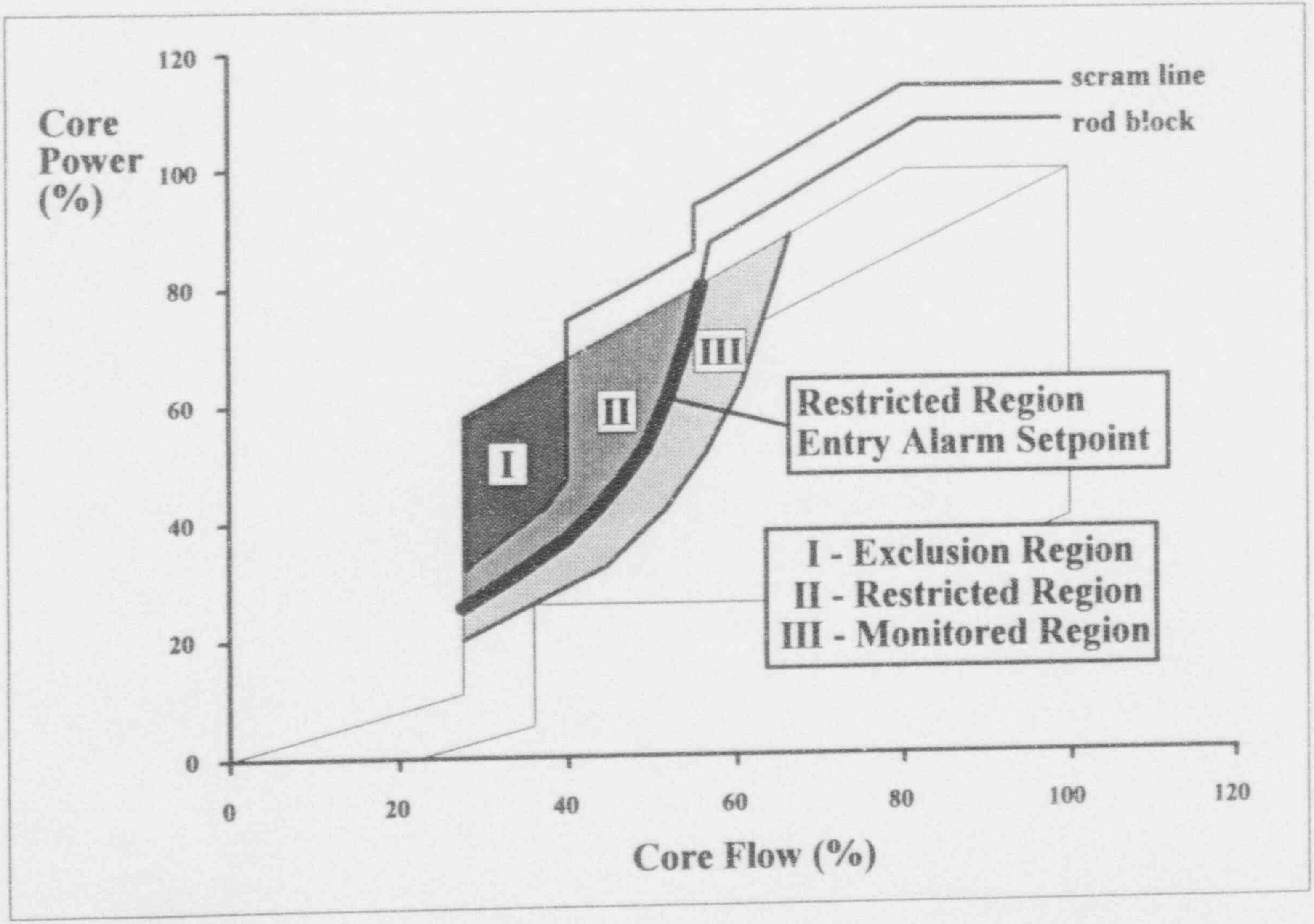


Figure 5-14: Restricted Region Entry Alarm Setpoint



6. DESIGN CONFIGURATION

This section discusses the Enhanced Option I-A design configuration as it relates to the solution analytical methodology. The specific hardware design specifications and details are not discussed in this report. Rather, the purpose of this section is to describe how the analytical models and solution methodology features are translated into engineered features of the BWR Neutron Monitoring System.

6.1 Flow Control Trip Reference

6.1.1 System Design Description

The flow control trip reference card (FCTRC) that exists in BWR neutron monitoring systems (NMS) generates a core flow-biased neutron flux trip setpoint based on a reference reactor recirculation drive flow signal that is representative of core flow. This signal is passed onward to a signal comparator circuit that has access to the average core power (APRM) signal. When the APRM signal reaches the trip reference signal, a reactor scram is initiated by the Reactor Protection System (RPS). his trip function is hexal or octal redundant within the NMS, and is split into two divisions of three or four channels each. The reactor trip logic requires that at least one channel in each division generate a trip signal to cause a scram. This results in a configuration that can be expressed as one-out-of-three (or four), taken twice. This RPS trip logic remains unchanged by the Enhanced Option I-A stability solution.

The flow-biased scram function prior to the implementation of Enhanced Option I-A consists of a linear function of the form:

$$P_{REF} = AW + B \tag{6-1}$$

where:

P_{REF} = trip reference power
W = fraction of rated recirculation drive flow
A = flow-biased scram setpoint slope in core power/flow space
B = flow-biased scram setpoint power axis intercept

The flow-biased neutron flux scram function is not explicitly considered in the plant safety analyses since no AOOs that are analyzed to demonstrate protection of the MCPR safety limit take credit for this feature. Instead, the analyses rely on the high flux trip (120% reactor power) of the NMS to generate a reactor scram. However, the operating domain where all AOOs are initiated is bounded by the flow-biased neutron flux scram function. Therefore, this feature automatically preserves the licensed operating domain.

The Enhanced Option I-A stability solution reconfigures the existing FCTRC design to perform a new safety-related function of enforcing the licensing methodology stability region boundaries. The Exclusion Region boundary is delineated by the core flow-biased neutron flux scram function of the NMS, and the Restricted Region boundary is delineated by the core flow-biased rod block function of the NMS. Both of these new trip reference functions have an associated setpoint setup feature. The introduction of these functions, including the defense-in-depth feature of the flow-biased scram clamp, which limits the power axis intercept of the linear flow-biased trip reference over the Restricted Region, increases the complexity of the FCTRC. For this reason, the existing analog FCTRC is replaced with a digital device where the necessary flow-biased trip reference setpoints are located within a microprocessor.

For each possible operating mode, the new FCTRC uses a set of two digital maps to generate the scram and control rod block setpoints as a function of recirculation drive flow. There are four sets of maps, each of which is associated with one of the following operating modes:

1. Two recirculation loop operation under normal region boundary setpoint conditions.

2. Two recirculation loop operation under region boundary setpoint setup conditions.

3. Single recirculation loop operation under normal region boundary setpoint conditions.

4. Single recirculation loop operation under region boundary setpoint setup conditions.

Figures 6-1 through 6-8 represent the functional form of each setpoint map. Actual plant-specific setpoint maps will be generated as part of the stability solution initial application process. The setpoint maps illustrated in these figures consist of three or four functional zones based on the number of algorithms required to describe the setpoints. Each zone has a unique setpoint algorithm that reflects the analytical basis for that zone's setpoint. The setpoint algorithms for each functional zone is shown in Tables 6-2 through 6-5. Table 6-1 defines the nomenclature for these setpoint algorithms.

The first zone contains the stability region boundary, corresponding to the Exclusion Region or Restricted Region depending on whether the map generates the scram setpoints or the rod block setpoints, and is described by a power function of the form:

$$P_{x-e}^{c-d} = P_z \left(\frac{P_y}{P_z} \right)^{\frac{1}{2}\left[\frac{F-F_z}{F_y-F_z} + \left(\frac{F-F_z}{F_y-F_z} \right)^2 \right]} \tag{6-2}$$

where the notation is described in Table 6-1. F = f(W) is a plant-specific relationship that provides a lower bound of the actual core flow (F) as a function of recirculation drive flow (W). An example of the relationship between core flow and drive flow is shown in Appendix G, Figure G-1. The function, f(W), is selected to bound expected changes in core flow, for a given drive flow, to avoid setpoint changes in future cycles. The function is selected below the actual core flow value for a given drive flow to ensure that actual flow conditions are greater than the derived core flow.

The second zone describes the core flow-biased scram clamp above the Restricted Region. This function is linear, with a power axis intercept value that is set 5% of rated power, plus any signal noise, above the power axis intercept of the highest actual flow-control line in the Restricted Region. The highest flow-control line passes through the rated core power/minimum core flow state point in the licensed operating domain. Zone two also describes the portion of the flow-biased control rod block setup setpoints that lie along the highest licensed flow-control line in the operating domain. The zone two setpoint algorithm is described by a function of the form:

$$P^{c-d}_{x-e} = mW + I^{c-d}_{x-e} \tag{6-3}$$

where P, c, d, x, e, m, W, and I are defined in Table 6-1.

The third zone describes the remainder of the linear flow-biased scram or control rod block setpoint as it currently exists in each plant-specific application. This section maintains the functional form of Equation 6-1 and is provided by Equation 6-3.

The fourth zone describes the neutron flux setpoint clamp for both the scram and rod block functions and has the form:

$$P^{c-d}_{x-e} = I^{c-d}_{x-e} \tag{6-4}$$

where P, c, d, x, e, and I are defined in Table 6-1. Some reactor designs do not have a rod block clamp in zone 4. In this case, the setpoint algorithm for zone 3 is extended to W_{max}, the maximum flow, and there is no zone 4 for the control rod block trip reference.

The Enhanced Option I-A FCTRC is designed to replace the existing FCTRC and is dimensionally, environmentally, and electronically compatible with existing hardware configurations. In addition, all of the Enhanced Option I-A flow-biased trip setpoints will be the same or more conservative than the existing setpoints, and therefore, the Enhanced Option I-A FCTRCs are acceptable substitutes for the existing FCTRCs that perform all functions currently required. This design feature permits direct substitution of either the new or existing FCTRC in the NMS during testing, without adversely affecting operability of any APRM channel.

6.1.2 Core Drive Flow Parameter Acquisition

The FCTRC designed to implement the Enhanced Option I-A stability solution will perform a safety related function, and will be designed and installed as a Class 1E system. However, because this solution will be installed as a backfit to existing plant NMS systems, the new FCTRC must accommodate and account for existing plant interfaces. The most critical interface with existing plants is the recirculation drive flow measurement signal. Existing FCTRCs utilize the reactor recirculation drive flow signal which is not generally Class 1E. The Enhanced Option I-A FCTRC also makes use of this system. Although operational experience demonstrates that the existing signal is highly reliable, the new FCTRC contains features designed to further improve the reliability and quality of the signal.

The recirculation drive flow signal is passed through a conditioning filter. This conditioning filter is used to filter frequency components in the input signal that do not represent true global variations in total core flow. The conditioned input drive flow signal results in a more stable trip reference signal for the FCTRC. Ultimately, this is expected to assure better defined region boundaries that reduce unnecessary challenges to reactor safety systems caused by a spurious scrams from signal noise.

6.1.3 Core Drive Flow Parameter Validation

The FCTRC also performs real time reactor recirculation drive flow signal validation. The purpose of the validation process is to provide adequate assurance that any credible failures in the drive flow signal will be detected and result in a conservative response from the FCTRC. The drive flow signal will be tested for upscale and downscale failures, as well as the fail-as-is condition. Any detected failure of the drive flow signal will cause the FCTRC to generate a failsafe output. This output will cause a reactor scram signal to be generated by the corresponding NMS channel.

6.1.4 Flow-Biased Trip Setpoint Setup Function

Certain reactor maneuvers, such as plant startup, may require entry into the Restricted Region. The Enhanced Option I-A solution methodology permits entry into the Restricted Region when adhering to the stability control. During normal operations, deliberate entry into the Restricted Region is prohibited by the control rod withdraw block function of the FCTRC.

When entry into the Restricted Region is required, the control rod block function is setup to a new setpoint. This manual action, which is permitted after conforming to the FCBB stability limit, is performed by depressing a setup switch located on the new FCTRC. The setup switch changes the control rod block setpoints, as depicted in Figures 6-4 and 6-8. In addition, the lower boundary of the Exclusion Region is setup as shown in Figures 6-3 and 6-7. The setup condition exists until the setpoints are manually setdown, as will be required by Technical Specifications, when the Restricted Region is exited. After the Restricted Region is exited during power ascension, the region boundary setpoints are also automatically setdown when core flow exceeds the value of ($W_{A'_{nom}}$ +5%), as shown in Figure 6-9. Under setup conditions, the control rod block function of the NMS provides the same protection against inadvertent penetration of the setup region boundary as it does for the normal Restricted Region boundary during normal operation.

6.1.5 Single-Loop Operation

The FCTRC is designed to respond to changes in the operating mode of the reactor recirculation system. Depending on whether the reactor is operating with two or one recirculation loop in service, different core flow-biased neutron flux scram and control rod block setpoints must be in place to comply with Technical Specifications. Currently, these differences in setpoints are accommodated simply by altering the power axis intercept (B) of the linear equations that represent the setpoints. Where the existing flow-biased setpoints are not replaced by the methodology of Enhanced Option I-A, this operation is maintained. Figures 6-5 through 6-8 illustrate example setpoint maps for reactor operations in single-loop operation (SLO).

When shifting from one recirculation operating mode to another, a switch located on the new FCTRC is depressed. This manual action adjusts the FCTRC to the proper setpoint maps. Appendix G discusses the validity of the recirculation drive flow signal as a measure of total core flow under SLO operating conditions.

6.1.6 Plant-Specific Application

The generic function and operation of the FCTRC is identical for all plant applications of the Enhanced Option I-A stability solution. However, the exact FCTRC setpoints will reflect plant-specific analysis performed to generate the stability region boundaries. The number of FCTRCs installed for each plant-specific application of the Enhanced Option I-A stability solution corresponds to the number of APRM channels that exist in the plant.

6.2 Period-Based Detection System

The Period-Based Detection System (PBDS) is a defense-in-depth feature designed to provide protection against unanticipated and hypothetical scenarios through early detection of significant reductions in core stability margin. The PBDS utilizes the Period-Based Algorithm described in References 1 and 2. It analyzes pre-selected individual LPRMs in real time, and generates a control room alarm after detection of sufficient successive period confirmations. The PBDS hardware consists of two oscillation detection channels and the associated inputs and outputs.

The PBDS setpoints are designed to respond to actual losses in reactor stability margin. The general PBDS design is common to all plants. Design variations to accommodate plant-specific input and output configurations and unique LPRM noise characteristics do not alter the fundamental architecture of the system.

6.2.1 Period-Based Algorithm

The PBDS utilizes the Period-Based Algorithm (PBA) documented in References 1 and 2, but only includes the successive power oscillation period confirmation count portion of the PBA and not the amplitude setpoint. The amplitude portion of the PBA is useful for discriminating growing power oscillations that occur when the reactor is unstable. Because Enhanced Option I-A is a preventive stability solution, any reactor instability is to be avoided and therefore, the amplitude portion of the PBA is not applicable. LPRM signal averaging, which is used as a reference for the amplitude setpoint component of the PBA in Reference 2, is not needed and, accordingly, is also not included in the PBDS design. A brief summary of the application of the PBA for the PBDS follows.

6.2.1.1 Algorithm Description

The PBA examines individual LPRM signals to determine the elapsed time between successive maxima and successive minima. For an oscillatory signal, these intervals represent an estimate of the oscillation period. The algorithm discriminates periods that are within the range expected for power oscillations induced by reactor instability. The first period that is identified within the expected range is defined as the base period. A subsequent oscillation with a period that is determined to match the base period within a predetermined tolerance constitutes a period confirmation. The base period is updated by a running average of the initial period and all successive confirmed periods. If the period of the current oscillation under consideration does not match the base period within the specified tolerance, the confirmation count is set to zero and the current period is tested to determine if it should be considered a new base period.

The performance of the PBA is determined by the various parameters that are used in conditioning and analyzing the LPRM signals. A complete list of these parameters, including a discussion of their effect on the PBA performance, follows:

1. Sample Interval (t_S)

   The oscillation detection system relies on analog-to-digital (A/D) converters to convert the continuous input voltage signal from the LPRMs into discrete digital values. The rate at which the LPRM signals are sampled is important in determining the ability of the algorithm to recognize instabilities. Sufficient resolution is needed to determine the magnitude and temporal location of the peaks and minima of the oscillation input signal. For the range of expected oscillation periods, typical values of the sample interval should be in the range of 50 to 100 milliseconds, providing a minimum of roughly 15 to 30 data points per oscillation in the expected period range.

2. Minimum Oscillation Period (Tmin)

Each period is tested against the expected range of oscillation periods associated with core thermal-hydraulic instability. If the period is outside the expected range a base period will not be established. The frequency range is expected to be bounded between 0.25 and 0.8 Hz, which results in oscillation periods of 4.0 to 1.2 seconds. Typical values for Tmin (the minimum oscillation period) are in the range of 1.0 to 1.4 seconds.

3. Maximum Oscillation Period (Tmax)

The upper range of expected oscillation period is defined as Tmax. Typical values for Tmax are in the range of 3.3 to 4.0 seconds.

4. Conditioning Filter Order (P_c)

The conditioning filter is used to filter frequency components in the input signal that are higher than the desired frequency range and which could interfere with the algorithm's ability to determine if an instability is occurring. The order or number of poles used in the conditioning filter determines how rapidly the gain falls off beyond the corner frequency and, when combined with the corner frequency, can control the desired filtering effect. Testing has demonstrated that a two-pole filter is sufficient for the objective of the filter.

5. Conditioning Filter Corner Frequency (f_c)

On the basis of the known frequency range of interest, conditioning filters are typically selected with a corner frequency in the range of 1.0 to 3.0 Hz.

6. Period Tolerance (ε)

The period tolerance (ε) is used to determine if two successive periods are sufficiently close to be considered an indication of an instability. It is known from measured plant data that as the reactor becomes unstable, the difference between successive periods decreases; in the limit (i.e., limit cycle oscillations), each successive period has the same value within the resolution of the sampling interval. Large values of ε will result in confirmations even when successive periods show rather large variations. Small values of ε only yield large numbers of successive confirmations when distinct power oscillations have developed.

As a minimum, the period tolerance cannot be set less than the sample interval (the minimum period resolution possible) unless interpolation is used to better resolve actual maxima and minima. Analysis of actual plant data has shown that a period tolerance of approximately 5% of the oscillation period provides a reasonable screen against inappropriate confirmations, while providing early detection of approaching reactor instability. Typical values of the period tolerance are in the range of 50 to 300 milliseconds.

7. Hi-Hi Decay Ratio alarm setpoint (N_{HH})

For each successive period that is within $\pm\varepsilon$ of the current base period, a confirmation occurs. The period confirmation count, N, increases only when successive periods result in confirmations. Whenever any period does not satisfy the period tolerance criterion, the confirmation count is reset to zero. The Hi-Hi Decay Ratio alarm setpoint (N_{HH}) is defined as the number of successive period confirmations that must occur before a manual scram without delay is required. The Hi-Hi Decay Ratio alarm setpoint is expected to be in the range of 10 to 15 successive counts.

8. Hi Decay Ratio alarm setpoint (N_H)

The Hi Decay Ratio alarm setpoint (N_H) is optional. It is defined as the number of successive period confirmations that must occur before an alarm indicating a moderate increase in decay ratio is initiated. Testing shows that during stable operation at low decay ratios, the number of successive period confirmations rarely exceeds five. The Hi Decay Ratio alarm setpoint is expected in the range of 6 to 8 successive period confirmations.

6.2.1.2 Algorithm Testing

A significant amount of testing has been performed using recorded plant data to demonstrate the effectiveness of the PBA and determine the possible range of the algorithm parameters. The testing results are summarized in Reference 2 and are fully applicable to the PBDS. The recorded plant data was chosen to represent a wide range of plant types, operating conditions, expected neutron flux transients and actual reactor instabilities.

The testing with available plant data demonstrated that the PBA performs as expected. For steady-state and transient data examined, the algorithm readily discriminated between the normally occurring neutron flux variations and core power oscillations induced by reactor instability. Considerable flexibility in the choice of algorithm parameter values was found to be available to ensure that the algorithm will provide a similar response when used at various BWRs. For the reactor instability events evaluated, the PBA was demonstrated to have the ability to detect instability induced power oscillations even at very low oscillation magnitudes.

6.2.2 System Design Description

The PBDS hardware configuration consists of two redundant cards that can be installed in existing spare LPRM card slots of the NMS. The input LPRM signals to the PBDS card can be taken from any non-safety related LPRM output signal. The cards provide output signals to support control room alarm indications and operability testing and verification capability.

6.2.2.1 PBDS Card

Although the PBDS is a defense-in-depth system and is not required as part of the licensing basis of Enhanced Option I-A, it is designed to meet Class 1E

Standards (Class 1E associated). A brief summary of the card's primary features follows:

1. The PBDS card receives input from up to 18 individual LPRM signals.

2. The card utilizes the period-based algorithm. The card performs continuous, real time analysis of each individual LPRM signal.

3. The card contains a self test feature to verify that the period-based algorithm is functioning as designed and to validate LPRM signals.

4. The card contains an alarm reset switch.

5. The card provides a Hi Decay Ratio alarm output signal that is actuated when the PBA period count exceeds the Hi Decay Ratio alarm setpoint.

6. The card provides a Hi-Hi Decay Ratio alarm output signal that is actuated when the PBA period count exceeds the Hi-Hi Decay Ratio alarm setpoint.

7. The card provides an INOP output signal that is actuated if the card is not operable.

8. The card provides two analog outputs of the period count. These analog outputs may be used for independent control room indication, data collection, or computer interface.

9. The card contains a status reset switch.

10. The card contains four LEDs to display the number of valid LPRM inputs.

11. The card contains dip switches that provide a selection of period tolerance and corner frequency.

The PBDS is based on six PBA parameters used in the interrogation of the LPRM signals and two alarm setpoints. Four of the PBA parameters are generic and their value is fixed. These parameters are:

1. t_s - Sample interval (seconds),

2. Tmin - Minimum oscillation period (seconds),

3. Tmax - Maximum oscillation period (seconds), and

4. P_c - Order of conditioning filter (number of poles).

The two remaining PBA parameters are tuned subsequent to plant installation of the PBDS card. Therefore, the card design allows post-installation adjustment capability of these parameters to accommodate plant-specific performance. The adjustable parameters are:

1. f_c - Corner frequency for conditioning filter (Hz)

2. ε - Period tolerance (seconds)

6.2.2.2 PBDS Input

The input to the PBDS cards are individual LPRM signals. All LPRM signals fed to a PBDS card are from the same groups of LPRMs. The cards are designed to receive up to 18 LPRM signals. These signals are taken from LPRM levels A, B, and C.

The D-level LPRM signals are not used, since their noise signature is less compatible with the PBA requirements. D-level LPRM signals are expected to exhibit higher noise levels as a result of the higher void fraction at the top of the core and potential bypass voiding at off-rated operating conditions. In addition, because of the longer neutron mean free path at higher elevations in the core, the D-level LPRM signals are more likely to exhibit a more complex oscillation signature.

To provide adequate redundancy for the Hi-Hi Decay Ratio alarm and adequate monitoring of the core during the approach to reactor instability, the PBDS requires at least 8 LPRM input signals for each channel. When less than 8 LPRMs are available, the PBDS channel is considered inoperable.

6.2.2.3 PBDS Output

PBDS card outputs consist of a Hi-Hi Decay Ratio alarm signal, an INOP indication, a Hi Decay Ratio alarm, and two analog outputs of the successive period count. The Hi-Hi Decay Ratio alarm is a required control room alarm. It is used to alert the operator that a manual scram without delay is required. The Hi Decay Ratio alarm control room display is optional and can be used to provide an early indication of a decrease in core stability margin based on plant-specific objectives.

The two analog outputs may be used to validate the Hi-Hi and Hi Decay Ratio alarm indications. The analog output can provide an indication of the period count trend, which is particularly useful in a slow approach to reactor instability. The analog output can also be used for data collection and on-line and off-line evaluation of PBA performance. This feature is especially useful for system calibration to achieve the target period count during startup conditions. Actual use of the analog outputs is not specified as part of the Enhanced Option I-A stability solution.

The INOP indication is used to establish the operational status of the PBDS card. A minimum of one card is required to be operable during operations inside the stability regions. INOP indication for both cards during operations inside the stability regions requires immediate actions as described in Section 3.

6.2.2.4 PBDS Alarm Logic

The Hi-Hi Decay Ratio alarm logic is based, for each PBDS card, on a two-out-of-all-LPRMs logic. If an analog output is observable from the reactor controls in the control room and is operable, verification of the alarm against the PBDS card analog output may be performed without delay prior to the manual scram. Upon receipt of the Hi-Hi Decay Ratio alarm, and confirmation if an analog output is available, the reactor is manually scrammed without delay. The Hi Decay Ratio alarm logic for each PBDS card is based on a one-out-of-all-LPRMs logic. A summary of the PBDS alarm logic is provided in Table 6-8.

6.2.3 Plant-Specific PBDS Application

The implementation of the PBDS for a specific plant requires an appropriate determination of the PBA parameters' values to ensure the adequacy of the PBDS alarm setpoints. The PBA parameters are divided into a set of generic parameters and a set of plant-specific parameters. The generic parameters are common to all plants and are determined based on the BWR thermal-hydraulic instability phenomenon and PBA performance requirements. The plant-specific parameters are determined based on the specific plant neutron noise characteristics to ensure that the PBDS is appropriately calibrated at normal off-rated operating conditions. The values of the generic PBDS parameters that are common to all plants are summarized in Table 6-6.

The target values and calibration ranges for the period tolerance and the filter corner frequency parameters are summarized in Table 6-7. The PBDS card hardware design allows on-line adjustments of these parameters. For each plant, appropriate PBDS performance is established during expected normal off-rated conditions (with the core decay ratio in the low range) by adjusting these parameters to achieve a maximum successive period confirmation count between 2 and 3. The expected successive period confirmation count for different reactor operating conditions is illustrated in Figure 6-10.

Table 6-1: Core Flow-Biased Trip Reference Algorithm Nomenclature

Zone	Algorithm	Drive Flow
1	$P_{x-e}^{c-d} = P_z\left(\frac{P_y}{P_z}\right)^{\frac{1}{2}\left[\frac{F-F_z}{F_y-F_z}-\left(\frac{F-F_z}{F_y-F_z}\right)^2\right]}$ $F = f(W)$	W_f^c
2,3,4	$P_{x-e}^{c-d} = mW + I_{x-e}^{c-d}$	

Zone	Symbol	Value	Meaning
1,2,3,4	x	S	Flow-Biased Scram Trip Reference
1,2,3,4	x	R	Flow-Biased Rod Block Trip Reference
1	c	Z1	Zone 1
2	c	Z2	Zone 2
3	c	Z3	Zone 3
4	c	Z4	Zone 4
1,2,3,4	d	NL	Normal Setpoint
1,2,3,4	d	SU	Setup Setpoint
1,2,3,4	e	1L	Single-Loop Operation
1,2,3,4	e	2L	Two-Loop Operation
1,2	f	40%	40% Core Flow Clamp
1,2	f	S1	Intercept of Nominal Restricted Region Shape Function with Single-Loop Operating Domain Highest Flow-Control Line

Table 6-1: Core Flow-Biased Trip Reference Algorithm Nomenclature (con't)

Zone	Symbol	Value	Meaning
1,2	f	S2	Intercept of Nominal Restricted Region Shape Function with Two-Loop Operating Domain Highest Flow-Control Line
1,2	f	R1	Intercept of Nominal Restricted Region Shape Function with Existing Single-Loop Flow-Biased Rod Block
1,2	f	R2	Intercept of Nominal Restricted Region Shape Function with Existing Two-Loop Flow-Biased Rod Block
3,4	f	CL	High Power Clamp Minimum Flow
3,4	f	Max	Maximum Flow
1	y	A	Intercept of FABLE Procedure Exclusion Region Boundary and High Flow-Control Line
1	y	A'	Intercept of FABLE Procedure Restricted Region Boundary and High Flow-Control Line
1	z	B	Intercept of FABLE Procedure Exclusion Region Boundary and Natural Circulation
1	z	B'	Intercept of FABLE Procedure Restricted Region Boundary and Natural Circulation
2,3	m	m	Flow-Biased Trip Linear Setpoint Slope
2,3,4	I	I	Flow-Biased Trip Linear Setpoint Power Axis Intercept
2,3,4	W	W	Recirculation Drive Flow (% of Rated)
1	F	F(W)	Derived Core Flow (% of Rated)
1,2,3,4	P	P	Core Power (% of Rated)

Table 6-2: Two-Loop Operation, Normal Flow-Biased Trip Reference Summary

Zone	Flow-Biased Scram		Flow-Biased Rod Block	
	Setpoint Algorithm	Flow Range	Setpoint Algorithm	Flow Range
1	$P_{S-2L}^{Z1\text{-}NL} = P_B\left([illegible] \right)^{\frac{1}{[illegible]}\left[\frac{F-F_B}{F_A-F_B}+\left(\frac{F-F_B}{F_A-F_B}\right)^2\right]}$ $F = f(W)$	$0 \le W \le W_{40\%}^{2L}$	$P_{R-2L}^{Z1\text{-}NL} = P_{B'}\left(\frac{P_{A'}}{P_{B'}}\right)^{\frac{1}{2}\left[\frac{F-F_{B'}}{F_{A'}-F_{B'}}+\left(\frac{F-F_{B'}}{F_{A'}-F_{B'}}\right)^2\right]}$ $F = f(W)$	$0 \le W \le W_{R2}^{2L}$
2	$P_{S-2L}^{Z2\text{-}NL} = mW + I_{S-2L}^{Z2-NL}$	$W_{40\%}^{2L} < W \le W_{S2}^{2L}$	N/A	N/A
3	$P_{S-2L}^{Z3\text{-}NL} = mW + I_{S-2L}^{Z3-NL}$	$W_{S2}^{2L} < W \le W_{CL}^{2L}$	$P_{R-2L}^{Z3\text{-}NL} = mW + I_{R-2L}^{Z3-NL}$	$W_{R2}^{2L} < W \le W_{CL}^{2L}$
4	$P_{S-2L}^{Z4\text{-}NL} = I_{S-2L}^{Z4-NL}$	$W_{CL}^{2L} < W \le W_{Max}^{2L}$	$P_{R-2L}^{Z4\text{-}NL} = I_{R-2L}^{Z4-NL}$	$W_{CL}^{2L} < W \le W_{Max}^{2L}$

Table 6-3: Two-Loop Operation, Setup Flow-Biased Trip Reference Summary

Zone	Flow-Biased Scram		Flow-Biased Rod Block	
	Setpoint Algorithm	Flow Range	Setpoint Algorithm	Flow Range
1	$P_{S\text{-}2L}^{Z1\text{-}SU} = 2 \times P_{S\text{-}2L}^{Z1\text{-}NL} - P_{R\text{-}2L}^{Z1\text{-}NL}$	$0 \le W \le W_{40\%}^{2L}$	$P_{R\text{-}2L}^{Z1\text{-}SU} = P_{S\text{-}2L}^{Z1\text{-}NL}$	$0 \le W \le W_{40\%}^{2L}$
2	$P_{S\text{-}2L}^{Z2\text{-}SU} = P_{S\text{-}2L}^{Z2\text{-}NL}$	$W_{40\%}^{2L} < W \le W_{S2}^{2L}$	$P_{R\text{-}2L}^{Z2\text{-}SU} = mW + I_{R\text{-}2L}^{Z2\text{-}SU}$	$W_{40\%}^{2L} < W \le W_{R2}^{2L}$
3	$P_{S\text{-}2L}^{Z3\text{-}SU} = P_{S\text{-}2L}^{Z3\text{-}NL}$	$W_{S2}^{2L} < W \le W_{CL}^{2L}$	$P_{R\text{-}2L}^{Z3\text{-}SU} = P_{R\text{-}2L}^{Z3\text{-}NL}$	$W_{R2}^{2L} < W \le W_{CL}^{2L}$
4	$P_{S\text{-}2L}^{Z4\text{-}SU} = P_{S\text{-}2L}^{Z4\text{-}NL}$	$W_{CL}^{2L} < W \le W_{Max}^{2L}$	$P_{R\text{-}2L}^{Z4\text{-}SU} = P_{R\text{-}2L}^{Z4\text{-}NL}$	$W_{CL}^{2L} < W \le W_{Max}^{2L}$

Table 6-4: Single-Loop Operation, Normal Flow-Biased Trip Reference Summary

Zone	Flow-Biased Scram		Flow-Biased Rod Block	
	Setpoint Algorithm	Flow Range	Setpoint Algorithm	Flow Range
1	$P_{S\text{-}1L}^{Z1\text{-}NL} = P_{S\text{-}2L}^{Z1\text{-}NL}$	$0 \le W \le W_{40\%}^{1L}$	$P_{R\text{-}1L}^{Z1\text{-}NL} = P_{R\text{-}2L}^{Z1\text{-}NL}$	$0 \le W \le W_{R1}^{1L}$
2	$P_{S\text{-}1L}^{Z2\text{-}NL} = mW + I_{S-1L}^{Z2-NL}$	$W_{40\%}^{1L} < W \le W_{S1}^{1L}$	N/A	N/A
3	$P_{S\text{-}1L}^{Z3\text{-}NL} = mW + I_{S-1L}^{Z3-NL}$	$W_{S1}^{1L} < W \le W_{Max}^{1L}$	$P_{R\text{-}1L}^{Z3\text{-}NL} = mW + I_{R-1L}^{Z3-NL}$	$W_{R1}^{1L} < W \le W_{Max}^{1L}$
4	N/A	N/A	N/A	N/A

Table 6-5: Single-Loop Operation, Setup Flow-Biased Trip Reference Summary

Zone	Flow-Biased Scram		Flow-Biased Rod Block	
	Setpoint Algorithm	Flow Range	Setpoint Algorithm	Flow Range
1	$P^{Z1\text{-}SU}_{S\text{-}1L} = P^{Z1\text{-}SU}_{S\text{-}2L}$	$0 \le W \le W^{1L}_{40\%}$	$P^{Z1\text{-}SU}_{R\text{-}1L} = P^{Z1\text{-}NL}_{S\text{-}2L}$	$0 \le W \le W^{1L}_{40\%}$
2	$P^{Z2\text{-}SU}_{S\text{-}1L} = P^{Z2\text{-}NL}_{S\text{-}1L}$	$W^{1L}_{40\%} < W \le W^{1L}_{S1}$	$P^{Z2\text{-}SU}_{R\text{-}1L} = mW + I^{Z2-SU}_{R-1L}$	$W^{1L}_{40\%} < W \le W^{1L}_{R1}$
3	$P^{Z3\text{-}SU}_{S\text{-}1L} = P^{Z3\text{-}NL}_{S\text{-}1L}$	$W^{1L}_{S1} < W \le W^{1L}_{Max}$	$P^{Z3\text{-}SU}_{R\text{-}1L} = P^{Z3\text{-}NL}_{R\text{-}1L}$	$W^{1L}_{R1} < W \le W^{1L}_{Max}$
4	N/A	N/A	N/A	N/A

Table 6-6: PBDS Algorithm Generic Parameters

PBDS Parameter	Value
t_S - Sample Interval (milliseconds)	50
Tmin - Minimum Oscillation Period (seconds)	1.2
Tmax - Maximum Oscillation Period (seconds)	4.0
P_C - Filter Order for Conditioning Filter	2

Table 6-7: PBDS Algorithm Plant-Specific Parameters

PBDS Parameter	Target Value	Expected Calibration Range
ε - Period Tolerance (milliseconds)	150	50 to 300
f_C - Corner Frequency for Conditioning Filter (Hz)	2.0	1.0 - 3.0

Table 6-8: PBDS Alarm Logic

Alarm	Alarm Logic (per channel)
Hi-Hi Decay Ratio (Mandatory)	Two-out-of-all-LPRMs, Once
Hi Decay Ratio (Optional)	One-out-of-all-LPRMs, Once

Figure 6-1: Two-Loop Operation, Normal Flow-Biased Neutron Flux Scram Setpoints

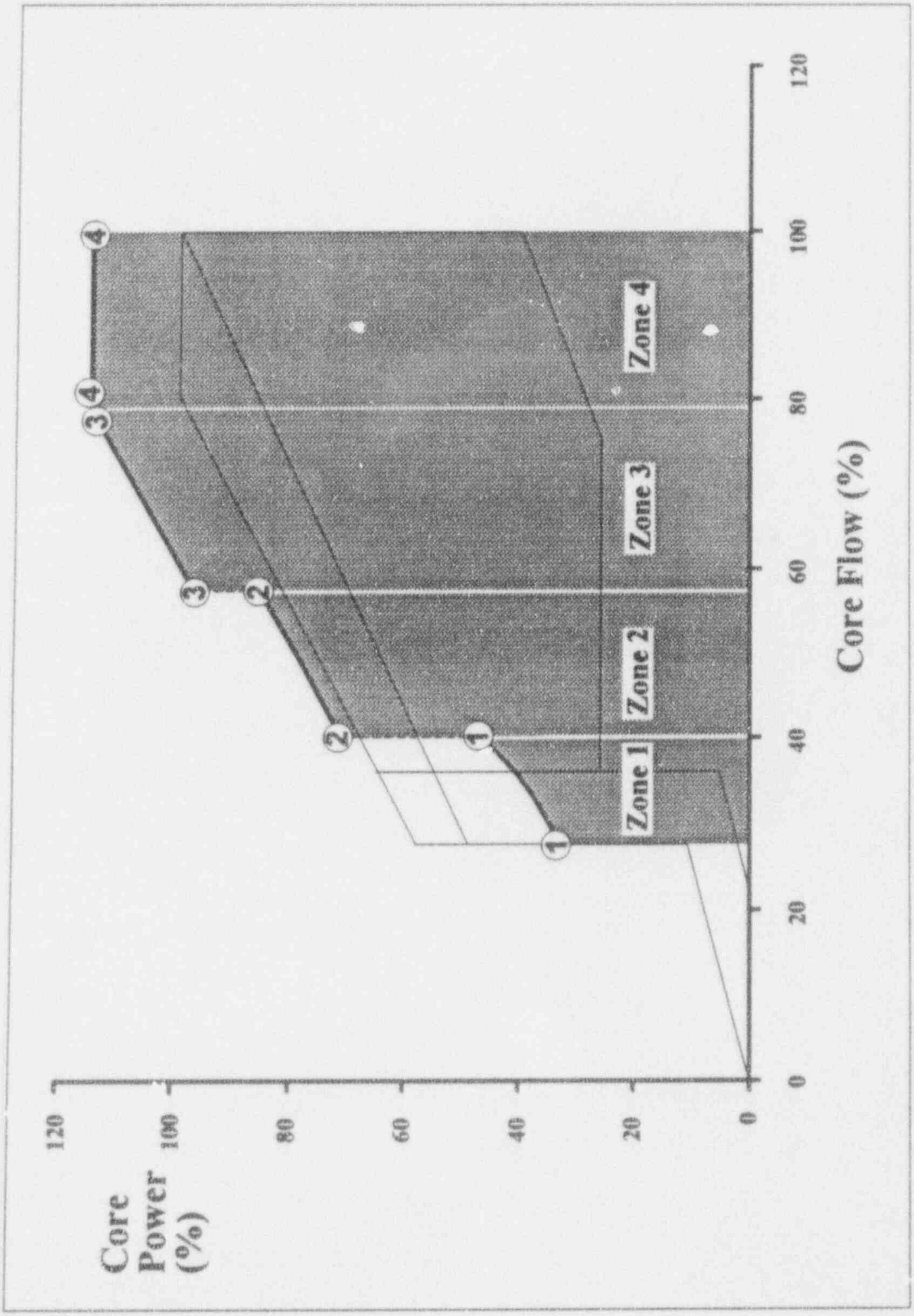


Figure 6-2: Two-Loop Operation, Normal Flow-Biased Neutron Flux Control Rod Block Setpoints

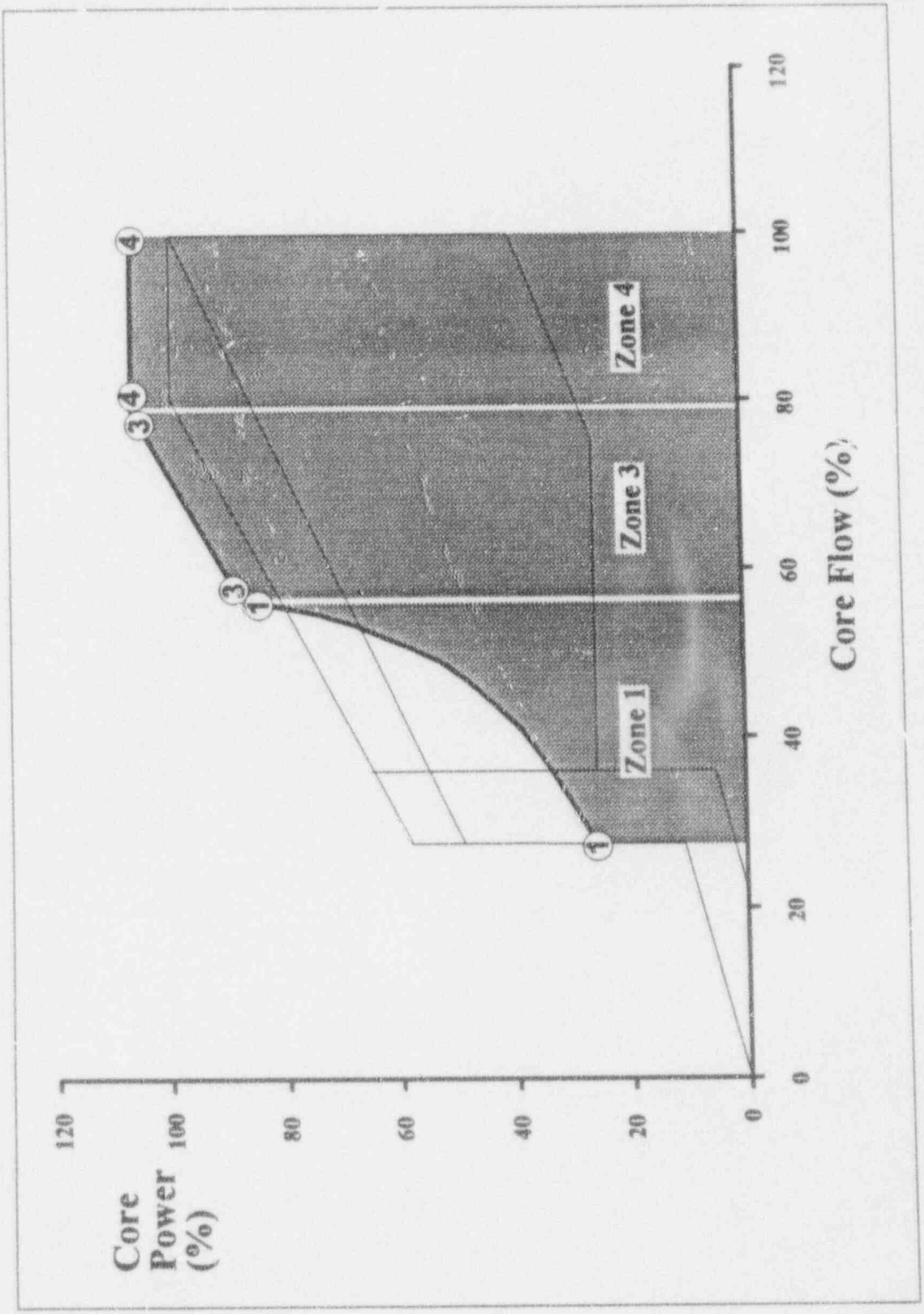


Figure 6-3: Two-Loop Operation, Setup Flow-Biased Neutron Flux Scram Setpoints

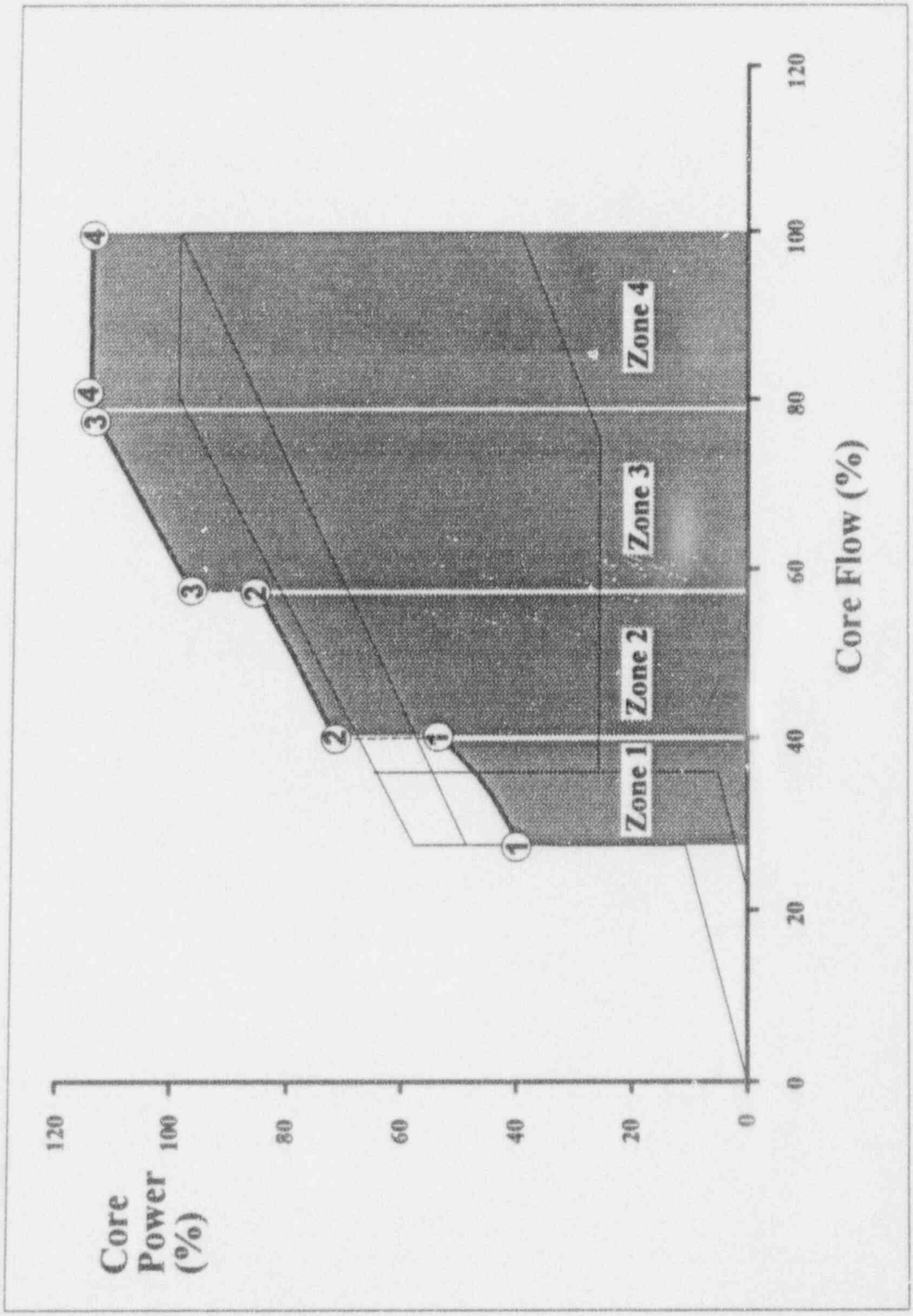


Figure 6-4: Two-Loop Operation, Setup Flow-Biased Neutron Flux Control Rod Block Setpoints

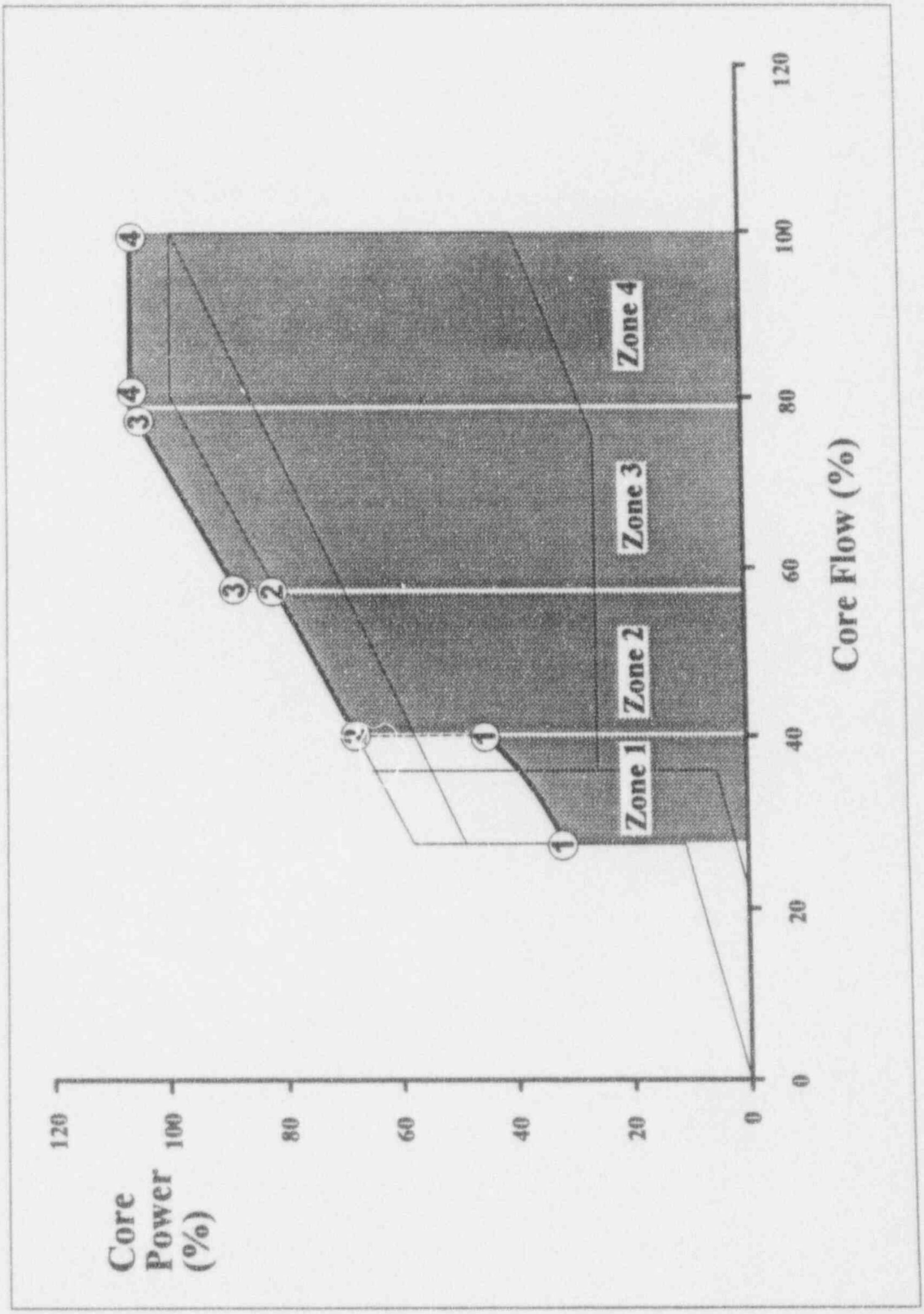


Figure 6-5: Single-Loop Operation, Normal Flow-Biased Neutron Flux Scram Setpoints

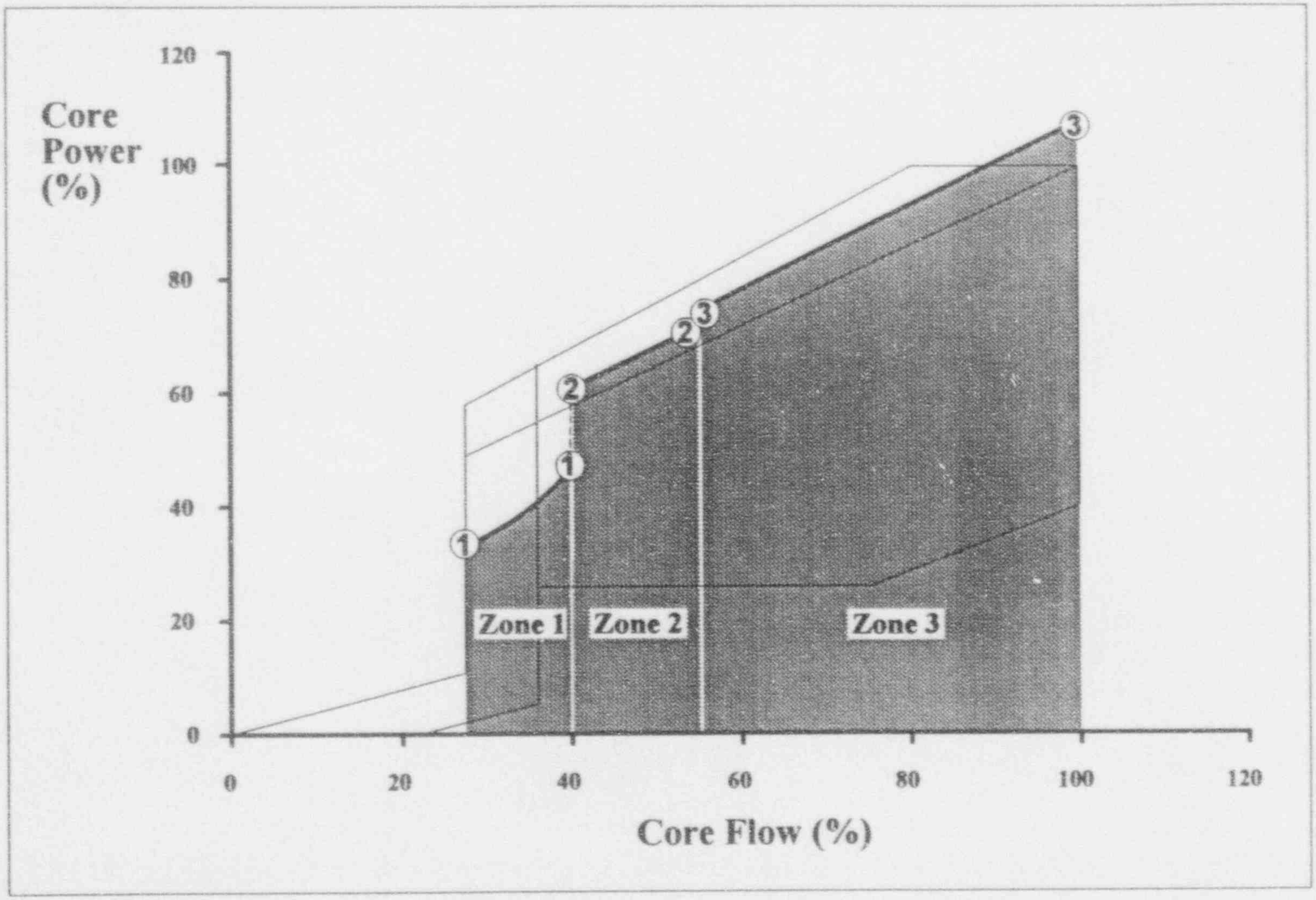


Figure 6-6: Single-Loop Operation, Normal Flow-Biased Neutron Flux Control Rod Block Setpoints

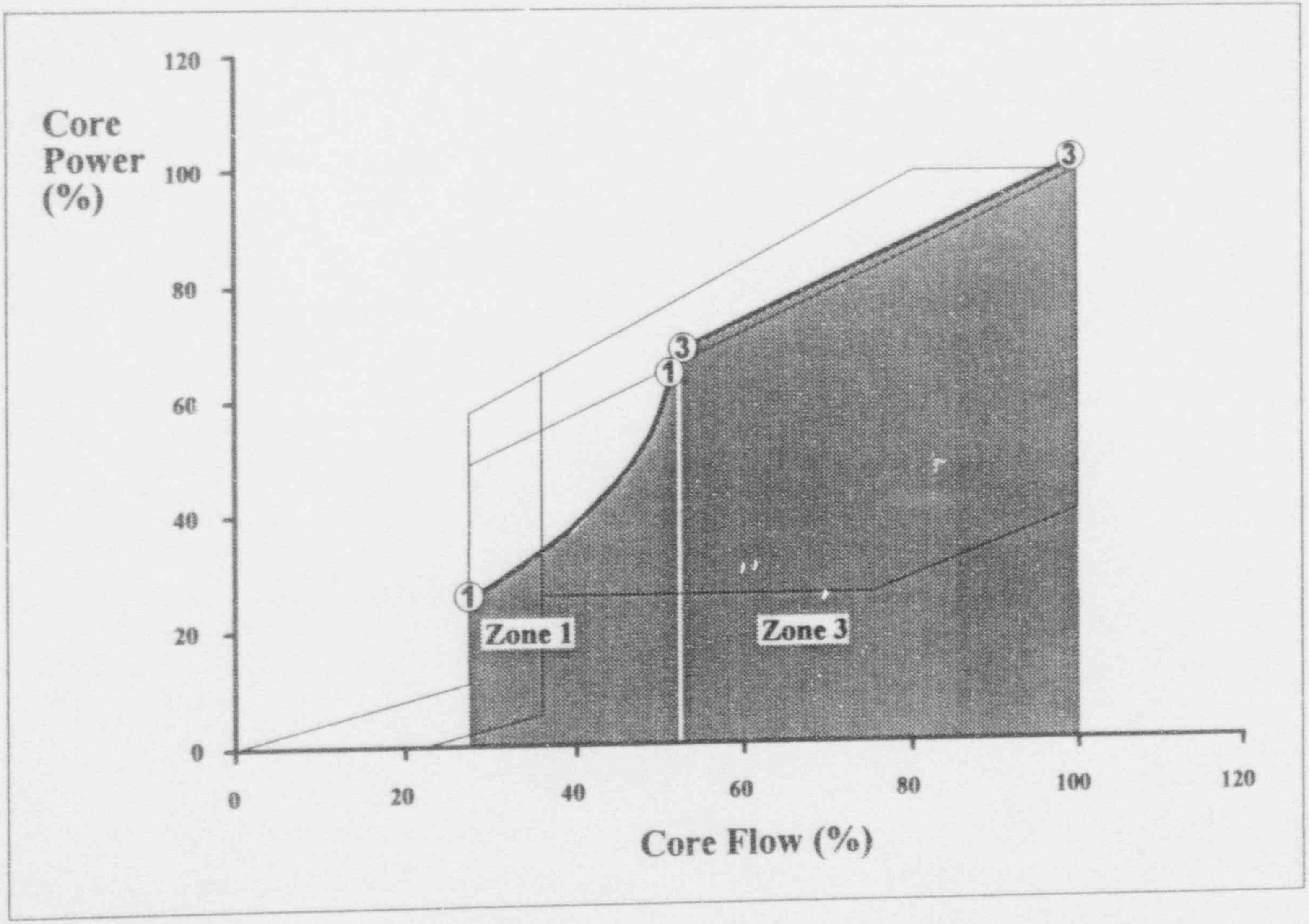


Figure 6-7: Single-Loop Operation, Setup Flow-Biased Neutron Flux Scram Setpoints

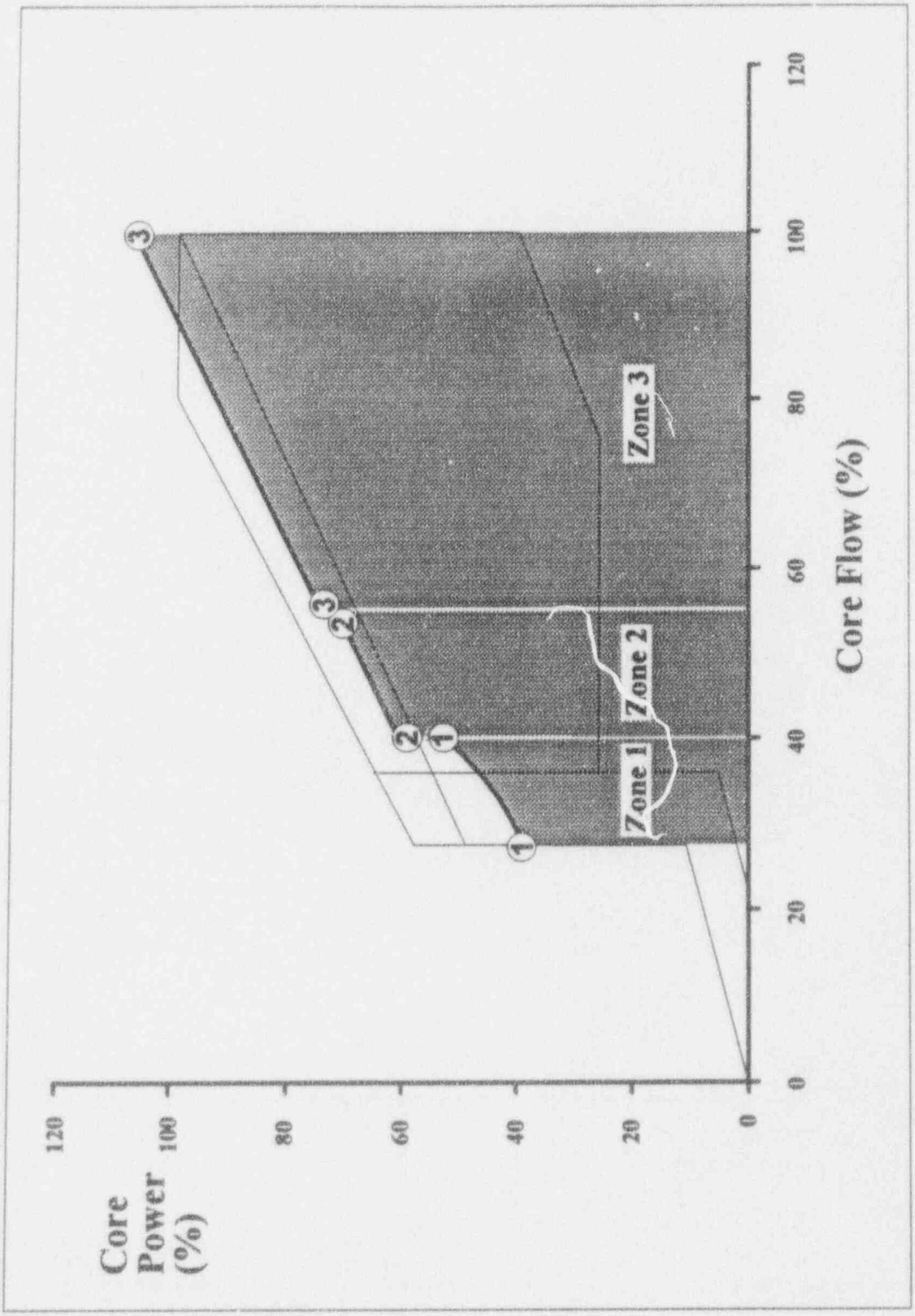


Figure 6-8: Single-Loop Operation, Setup Flow-Biased Neutron Flux Control Rod Block Setpoints

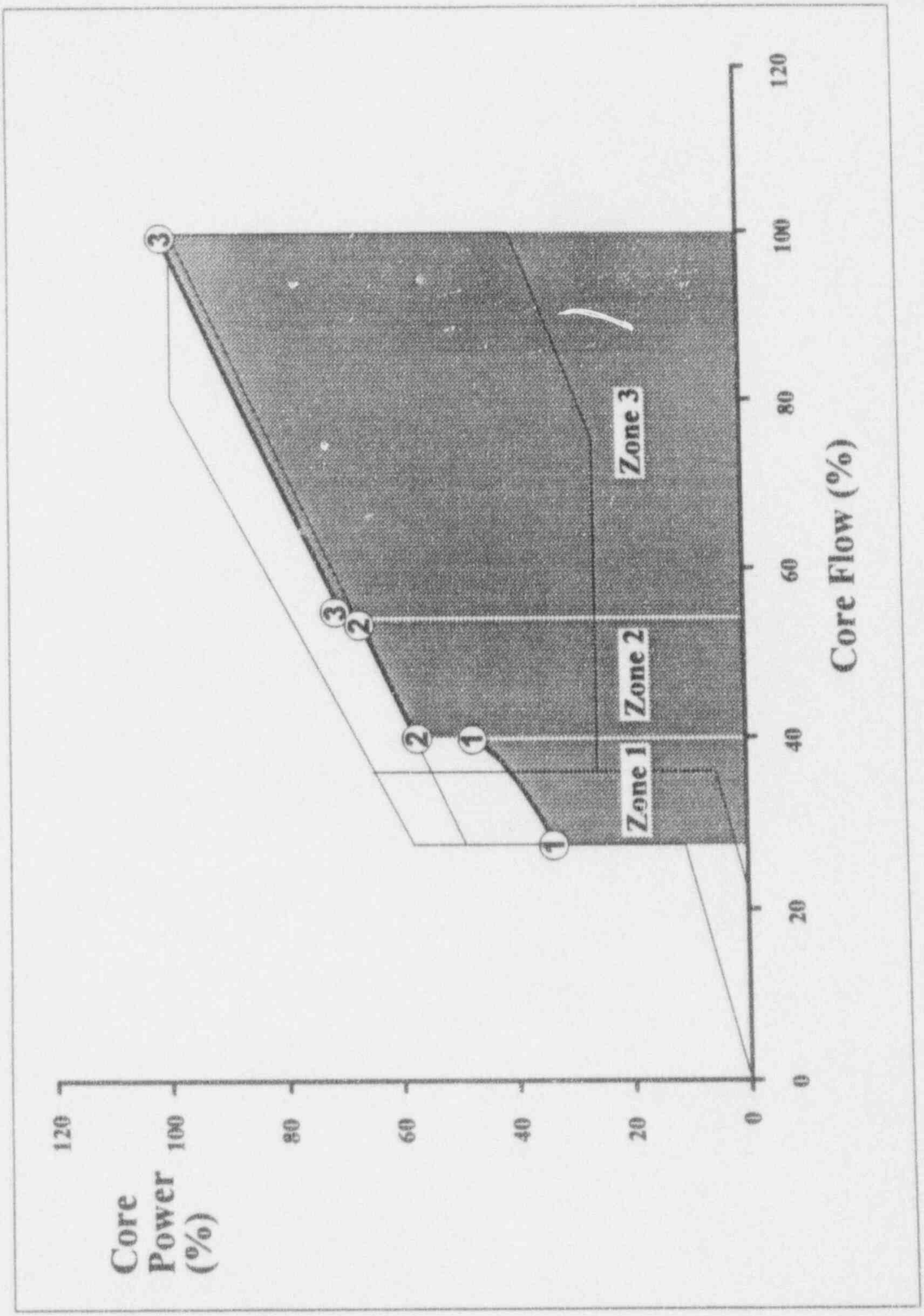


Figure 6-9: Region Boundary Setpoints Automatic Setdown Trigger Setpoint

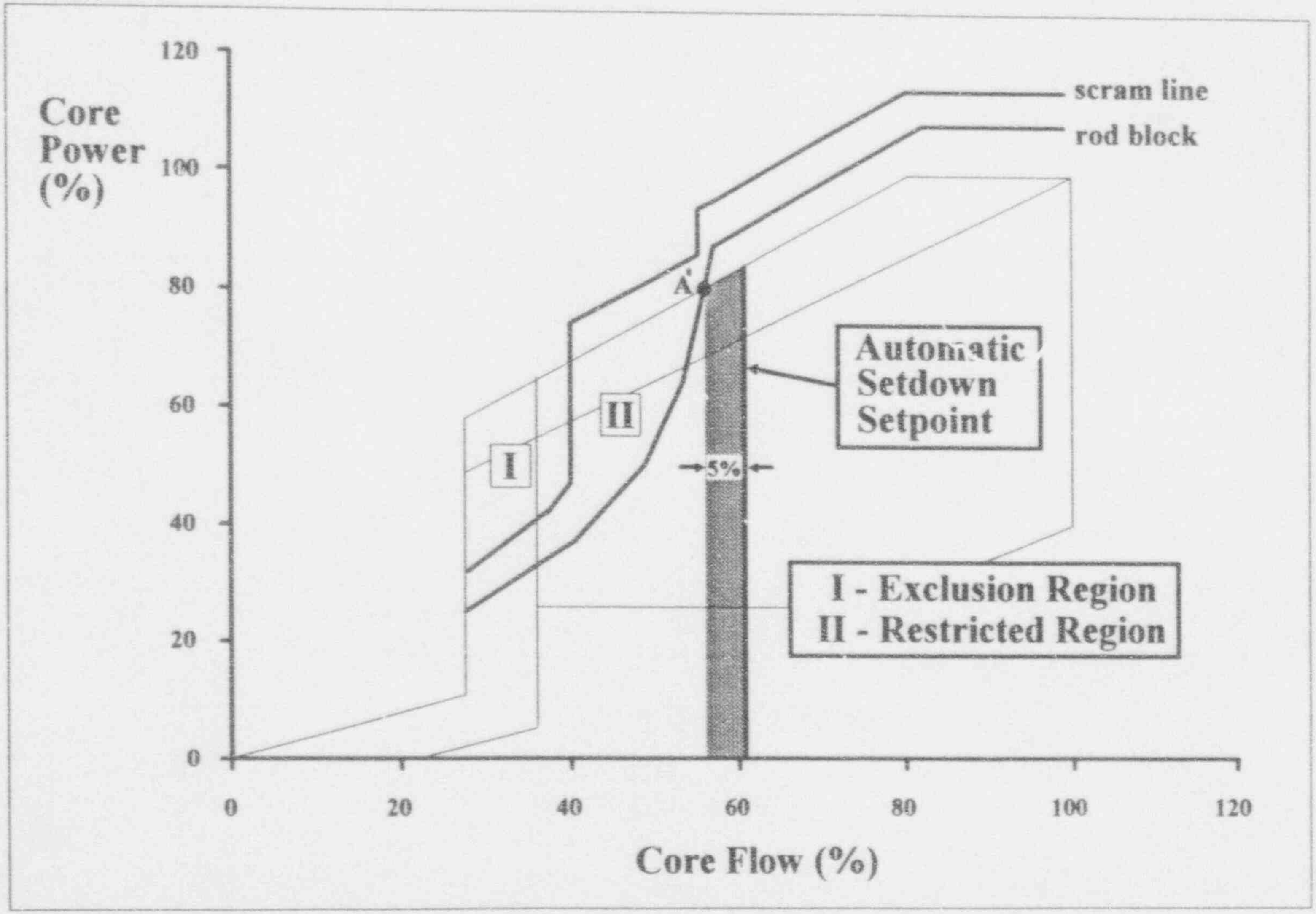
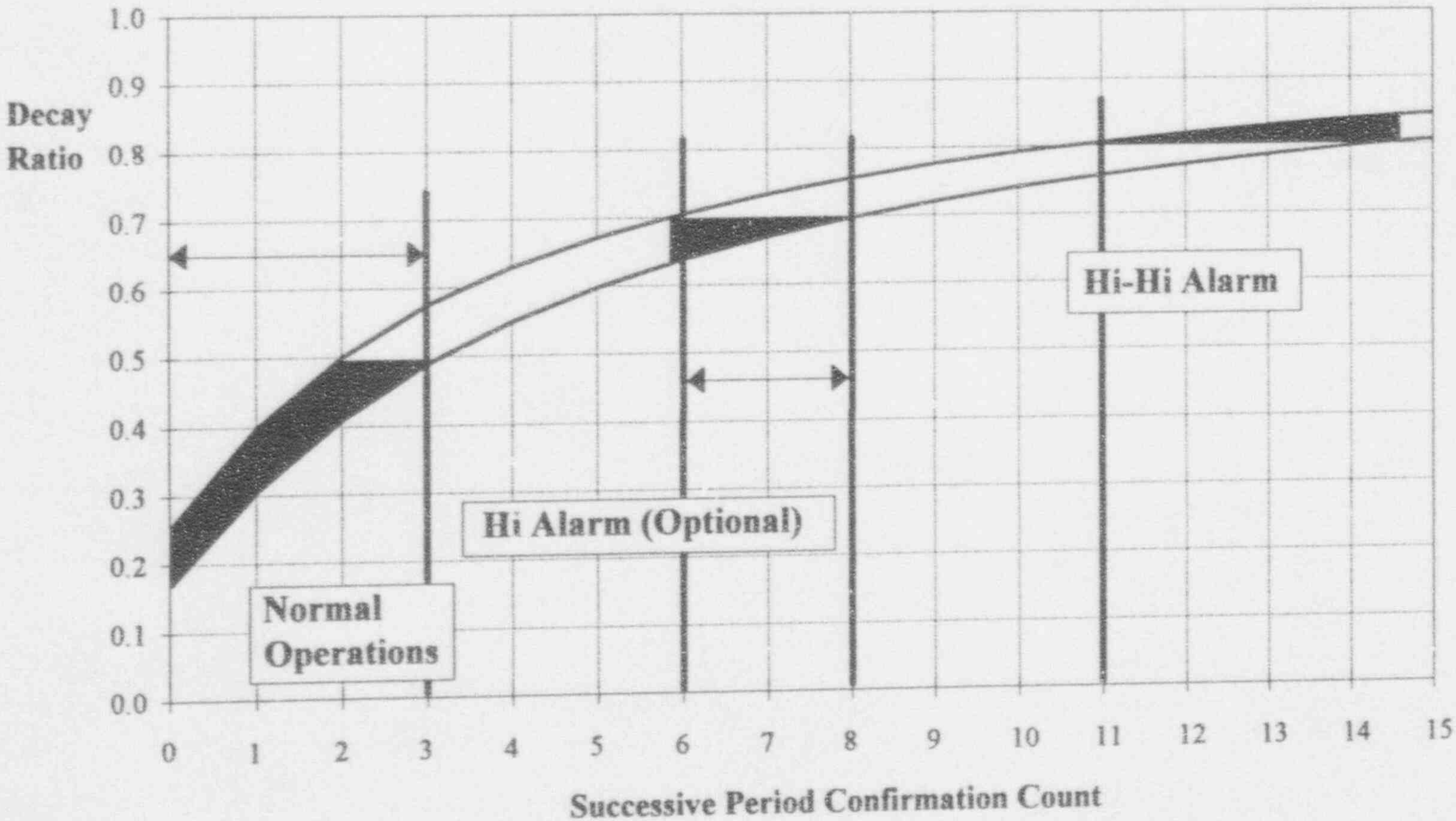


Figure 6-10: PBDS Alarm Setpoints



7. INITIAL APPLICATION PROCESS

The initial application process of Enhanced Option I-A encompasses the generation of new stability region boundaries, the analysis to validate the region boundaries, and specification of performance requirements for stability codes utilized in the process. The initial application process is generic and is used for all plant applications. Application of this process to a specific plant generates unique stability regions that will be documented in a plant-specific licensing submittal.

7.1 Process Overview

The solution initial application process is designed to ensure that the Enhanced Option I-A methodology can be implemented in any domestic BWR using qualified stability analytical tools. Baseline decay ratio calculations are performed with the FABLE/BYPSS stability code, consistent with the methodology documented in References 1 and 2. These calculations form the framework for the generation of the stability region boundaries. A qualified best-estimate stability code is then used to establish plant-specific stability region boundaries based on adjustments to the FABLE analysis. The same best-estimate code is also used to validate the region boundaries at reasonably-limiting steady-state and transient conditions.

7.1.1 Process Objectives

The initial application process is designed to meet these primary objectives:

1. Enhanced Option I-A can be implemented in all domestic BWR designs, by any BWR fuel vendor, and for all fuel cycle designs. The process should allow fuel vendors or utilities to use their own qualified best-estimate stability codes.

2. The process should be generic, prescriptive, and simple in order to reduce the probability for errors and ensure adherence to approved analysis procedures.

3. The process should consider reasonably-limiting steady-state and transient conditions to establish an appropriate degree of conservatism.

4. The process of defining the stability region boundaries should be sufficiently flexible to accommodate future changes in core reload design. This minimizes the potential impact on the region boundaries.

7.1.2 Terminology

The solution initial application process consists of the generation and validation of new stability region boundaries. To facilitate the description of the initial application process, several new terms are introduced:

Standard Cycle (SC)

The SC design consists of a generic fuel assembly design applied to all plants, and plant-specific core designs. The generic fuel assembly design is the only fuel-dependent input to the FABLE analysis. The SC captures all the unique stability-related features of a particular reactor system design that are independent of fuel assembly design, using the FABLE procedure. The SC design is also used in the best-estimate analyses to provide a reference for quantifying boundary setpoint adjustments due to plant-specific fuel and cycle designs relative to the FABLE analysis.

The SC design is completely specified to permit subsequent independent use of the analysis results in other qualified stability methods. The SC design is not expected to change during the plant lifetime, unless major reactor design changes are introduced that significantly affect the plant-specific input to FABLE.

Reference Cycle (RC)

The RC design is used in determining the nominal region boundaries and should envelop anticipated future fuel cycle designs. The RC design process includes allowances for design variations to minimize the need for future region boundary modification. RC decay ratio data used to determine the nominal region boundaries are obtained by adjusting the

FABLE SC decay ratio data by the differences in decay ratio between the SC design and the RC design.

Current Cycle (CC)

The CC design represents the actual fuel cycle design and is used in the region boundary validation process. The validation analysis is performed at the analytical stability region boundaries, which are based on the nominal region boundaries and associated setpoint uncertainties.

Demonstration Validation Matrix (DVM)

The DVM is a set of steady-state and transient conditions used to validate the region boundaries for the demonstration plant.

Initial Validation Matrix (IVM)

The IVM is used to validate new region boundaries for each specific plant application. The IVM is a subset of the DVM where non-limiting DVM state points are excluded.

Best-Estimate Stability Code

A best-estimate stability code performs decay ratio analysis for the RC and SC designs to define the nominal region boundaries. The best-estimate stability code uses the CC design to validate the analytical region boundaries. The best-estimate stability code must meet minimum specified performance requirements, as described in Section 7.4.2, to be qualified for use in generation and validation of the stability region boundaries for Enhanced Option I-A.

7.1.3 Analysis Elements

Elements of the analysis required for the initial application process are illustrated in the process diagram of Figure 7-1. The procedure to generate the nominal region boundaries consists of two major steps. First, the FABLE/BYPSS stability code is applied to the SC design in a manner consistent with the licensing procedure of Reference 1, to generate baseline decay ratio data that reflect the stability performance of the reactor system. This step is performed by GE for all plant-specific applications.

Second, a qualified best-estimate stability code is used to establish a decay ratio bias that accounts for the stability performance of the plant-specific fuel by comparing the stability performance of the RC and SC designs. The decay ratio bias is then applied to the FABLE SC baseline decay ratio data to generate FABLE-based RC decay ratios. The nominal stability region boundaries are established based on the comparison of these RC decay ratios and the boundary generation stability criteria. The best-estimate code analysis can be performed by any stability methodology qualified for Enhanced Option I-A application.

Validation of analytical stability region boundaries is performed with the CC design using a best-estimate stability code at state point conditions defined by the IVM. The CC decay ratio results are compared to the best-estimate code stability criterion to confirm the validity of the region boundaries. The best-estimate code stability criterion defines the instability threshold for applications to the Enhanced Option I-A stability solution.

7.2 Region Boundaries Generation

This section provides a detailed description of the process and basis for the generation of the Enhanced Option I-A stability region boundaries. The SC design, FABLE baseline analysis, RC best-estimate analysis and generation of the final nominal region boundaries are discussed.

7.2.1 Standard Cycle Baseline Stability Analysis

The generation of the SC baseline decay ratio data for any plant selecting long-term solution Enhanced Option I-A is the responsibility of GE. The baseline decay ratio data is generated by the frequency domain code FABLE/BYPSS based on the methodology of References 1 and 2.

7.2.1.1 Standard Cycle Design

The SC consists of a completely-specified fuel and core design that is used to establish FABLE baseline decay ratio data and create a reference for the best-estimate RC stability performance analysis. The application of the FABLE licensing methodology to the SC design should, in general, occur only once per plant-specific application. This approach provides the flexibility to establish nominal stability region boundaries, based on the FABLE baseline decay ratio data, with any qualified best-estimate methodology. In particular, it allows for analysis of cores containing fuel from multiple vendors through use of appropriately-qualified stability methodology.

The SC utilizes a well-established, simple fuel design that facilitates the interface between different stability analysis methods. The fuel design consists of a generic 8x8 fuel assembly with two water rods, typical clad and fuel pellet dimensions and properties, axially-uniform fuel enrichment and gadolinia concentration, and upper and lower natural uranium blankets. The stability performance of the SC fuel design is similar to that of a typical 8x8 fuel design. Fuel designs that are used in the RC may show degraded or improved stability performance relative to the SC design.

The SC core design includes plant-specific inputs that reflect unique aspects of the stability performance of each plant. The SC design approach is generic and consists of a target 18-month equilibrium cycle using a Haling depletion at rated conditions with a one-third-core reload batch size. Actual length of the plant-specific cycle design will vary somewhat from plant to plant.

The SC fuel assembly design parameters are completely specified in Appendix C. The SC fuel assembly axial configuration and 2-D radial fuel pin patterns for the three axial lattice designs are provided. Changes to the parameter list or changes in the values of specific parameters in Appendix C are not anticipated and will be made only to ensure modeling consistency and accuracy. Any changes to the SC design specifications will be documented and justified in a plant-specific licensing submittal.

A set of SC core design parameters is also provided in Appendix C. Values of plant-specific parameters are not specified, however, since the parameter set is plant-unique and will be separately established for each plant. A sample core loading map indicating the equilibrium cycle fuel loading pattern, including load, shuffle, and discharge fuel moves is provided.

The SC fuel and core design is fully specified so that any utility or fuel vendor can perform the required analysis with their qualified methods. The SC design is defined such that the comparison between the SC and RC designs is not affected by the choice of methods. This ensures that differences in decay ratio performance between the two designs are strictly a function of changes in fuel design and not due to differences in the application or choice of methods.

The SC fuel bundle design includes unspecified hydraulic loss coefficients for the water rod inlet, the fuel rod spacers and the out-of-channel bypass inlet. These coefficients are determined by forcing the SC EOEC rated Haling core simulation to match the SC design specifications for the core pressure drop and the active channel, water rods, and out-of-channel bypass flow rates. Values for all core state parameters (e.g., rated core power, rated core flow, rated inlet enthalpy, rated core pressure) are plant-specific.

After the unspecified hydraulic loss coefficients are established, the SC is fully specified. The methods applied to the SC design should be the same as those applied to the RC design when establishing the RC to SC decay ratio bias correction. This is necessary to ensure that the decay ratio bias correction reflects changes in fuel design only, and is not an artifact of the methods used to perform the analysis.

7.2.1.2 FABLE/BYPSS Analysis

The FABLE licensing procedure described in References 1 and 2 is applied to the SC design at generic state points common to all plant applications. Figure 7-2 shows the location of these state points in the operating domain. The state points are specified at pre-determined locations along the natural circulation line and a high flow-control line. The core power and flow coordinates of these state points are listed in Table 7-1.

The state points are selected tc ensure appropriate coverage of the expected ranges of the stability region boundaries. Reactor state points are only analyzed along the edges of the operating domain since a generic shape function is used to define the region boundaries in the interior of the operating domain. The FABLE procedure calculates the core and channel decay ratios at each of the eight state points. These decay ratios are considered conservative based on the overall conservatism of the FABLE methodology as described in Reference 1.

7.2.1.3 Region Generation Stability Criteria

Figure 5-4 of Reference 1 defines the licensing stability criterion for the FABLE methodology. The criterion reflects those reactor conditions considered to be susceptible to the inception of core-wide and regional modes of coupled neutronic/thermal-hydraulic instabilities. The stability criterion is expressed as a function of the core and hot channel decay ratios, and includes an appropriate allowance for method uncertainties. The stability criterion of Reference 1 is used as the basis for establishing the Exclusion Region boundary of Enhanced Option I-A. The concept is expanded and modified to establish the boundaries of the Restricted and Monitored Regions.

The Exclusion Region stability criterion intercepts both the core and channel decay ratio axes at a value of 0.8. The shape of the criterion for combinations of high core and hot channel decay ratios reflects the increased core susceptibility to regional mode oscillations. The Restricted Region stability criterion intercepts the core and channel decay ratio axes at a value of 0.6. A lower decay ratio value is selected since no stability controls are required outside the region. The lower criterion results in a larger stability region and provides additional stability margin in the absence of power distribution restrictions. The use of these criteria provides a method to establish the stability region boundaries, and is not intended to imply that reactor conditions cannot exceed these criteria. The adequacy of the stability regions defined by this process is confirmed during the region boundary validation process.

For the Monitored Region, the decay ratio intercepts are further lowered to core and hot channel decay ratio values of 0.4. A low criterion is selected to ensure that the Monitored Region is sufficiently large to preclude any hypothetical instability events outside the region.

The stability criteria for the Restricted and Monitored Regions are generated by scaling the Exclusion Region criterion and using the decay ratio axis intercepts as defined above. Multipliers of 0.75 and 0.5 are applied to the Exclusion Region stability criterion to generate the Restricted and Monitored Regions criteria, respectively. The boundary generation stability criteria are shown in Figure 7-3 and the criteria coordinates are provided in Table 7-2.

7.2.1.4 Standard Cycle Baseline Decay Ratio

The FABLE analysis results for the SC design along the edges of the operating domain (Figure 7-2) are mapped into core and hot channel decay ratio coordinates. This process is demonstrated in Figure 7-4, where the decay ratio data shown is for illustration purposes only. The decay ratio results in Figure 7-4 are related to the state points in Figure 7-2 as indicated by the full and empty circles and the state point numbers. Figure 7-4 represents the FABLE SC baseline decay ratio data.

7.2.2 Reference Cycle Stability Performance

The RC is a plant-specific design constructed to form a reasonable balance between enveloping potential future design variations and avoiding overly-conservative approaches that would result in unnecessarily large stability regions. The RC and SC designs are analyzed with a best-estimate stability code and the differences are used to establish a best-estimate decay ratio bias between the SC and RC designs.

The primary objectives of the RC design are to capture the stability performance of plant-specific fuel designs and to support a method for ensuring the continued applicability of initial plant-specific region boundaries to future reloads. All fuel cycle designs are evaluated based on a generic fuel depletion scheme that consists of cycle Haling depletions at rated conditions to a nominal end-of-cycle (EOC) exposure. This procedure facilitates a meaningful comparison between the different stability solution design components (SC, RC and CC). The sensitivity of core stability performance to specific operating conditions precludes a practical comparison between the different stability solution design components if arbitrary state conditions and operational patterns are used. Other fuel cycle depletion schemes (e.g., cycle depletions with design control rod patterns) are not used because their application may depend on the fuel cycle design and therefore cannot support a practical generic method of comparing the different solution design components.

The stability region boundaries are generated based on the FABLE SC baseline decay ratio data modified by the appropriate RC-to-SC bias correction. Evaluating the RC design in a manner consistent with the SC cycle depletion process results in a meaningful RC-to-SC bias correction factor that preserves the overall conservatism of the FABLE procedure.

The complexity of the stability phenomena is recognized in the Enhanced Option I-A stability solution. It is addressed by performing evaluation and validation analyses at pre-defined, conservative conditions and by incorporating substantial defense-in-depth protection measures. The generic approach to evaluating the SC and RC cycle designs allows meaningful comparisons that identify and quantify plant-specific fuel and cycle design features affecting the stability performance of the core. Stability performance for other reactor state conditions that may be encountered during operations are addressed by the stability region boundaries validation process and by the robustness of Enhanced Option I-A, including the defense-in-depth features.

7.2.2.1 Reference Cycle Design

The RC design is based on a rated-conditions Haling cycle depletion to end-of-equilibrium-cycle (EOEC). The design may be adjusted to account for possible variations in future fuel and core reloads designs.

The reload review procedure of Section 8 defines criteria used to identify differences in fuel and core design between the CC and the RC design that may significantly affect stability. Exceeding the established criteria requires performance of a region boundary re-validation analysis. To minimize the need for future validation analyses, appropriate allowances may be added to the RC design which anticipate and compensate for potential future design changes that would result in failure to meet the requirements of the reload review criteria. A list of the reload review criteria requirements and possible compensating allowances in the RC design follows. The CC design stability performance is compared to the RC design performance during the reload review process.

Criterion: CC to RC reactor design modifications have no effect on stability.

Compensation: The RC design may include allowances for new reactor design features anticipated in future cycles that can potentially degrade core stability performance.

Criterion: CC to RC fuel and channel mechanical design change have no effect on stability.

Compensation: The RC design may incorporate fuel and channel mechanical design features that are more conservative than the existing design, yielding poorer stability performance. These design features may represent anticipated design changes planned for implementation in future reloads, or reflect arbitrary allowances to address uncertainties in the evolution of fuel and channel designs.

Criterion: CC Haling radial peaking increase over RC by no more than 5%.

Compensation: The RC core loading may be adjusted locally to increase the maximum design radial peaking. This accommodates reload designs that yield increased radial peaking.

Criterion: CC reload batch size change relative to RC by no more than 5% of core size.

Compensation: The RC reload batch size may be set to the nominal batch size anticipated for future reloads.

<u>Criterion:</u> CC to RC full power cycle energy change within 10%.

<u>Compensation:</u> The RC full power cycle energy may be set to the nominal value anticipated for future cycles.

<u>Criterion:</u> Last cycle coastdown change relative to RC no more than 10% of full-power cycle energy.

<u>Compensation:</u> A RC coastdown may be set to the nominal coastdown expected in future cycles.

<u>Criterion:</u> Last cycle energy shorter than RC by no more than 10%.

<u>Compensation:</u> Any cycle energy shortfall or early shutdown is unplanned and therefore cannot be practically incorporated in the RC design.

<u>Criterion:</u> No variation in CC mixed batch reload core relative to RC.

<u>Compensation:</u> The RC design may include allowances that increase radial peaking or incorporate conservative mechanical design features to address transition cycles of mixed fuel assembly cores, including transition cores containing fuel from multiple vendors.

<u>Criterion:</u> CC core loading strategy relative to RC unchanged.

<u>Compensation:</u> Only significant and distinct design changes are considered (e.g., conventional scatter loading versus control cell core). The determination of an optimum core loading for the RC may be impractical. If possible, the more conservative core loading strategy may be used.

Criterion: No new operating modes for CC relative to RC.

Compensation: The RC design may incorporate any new operating modes that can potentially degrade core stability performance and are anticipated in future operations (e.g., feedwater heater out of service).

Criterion: Other differences between CC and RC do not result in equivalent loss of stability margin.

Compensation: An arbitrary bias may be applied for unanticipated future changes.

A list of the reload review criteria used to identify changes in fuel assembly or core design that can affect stability, and a summary of the possible RC design allowances that can be introduced to minimize the need for future analysis, is provided in Table 7-3.

7.2.2.2 Reference Cycle to Standard Cycle Comparison

To generate the RC-to-SC decay ratio bias correction factor, the two designs are analyzed at common state points using a three-dimensional steady-state BWR core simulator and qualified best-estimate stability code. The state points are pre-selected to ensure a fully-specified and generic implementation process and coverage of an appropriate decay ratio range; the decay ratio range should encompass the calculated FABLE SC baseline data and, in effect, envelop the core and hot channel decay ratio range in the boundary generation stability criteria.

The EOC Haling core power shape selected for this comparative analysis is compatible with the FABLE procedure, which also uses an EOC Haling core power shape. The hot channel axial power shape in the RC-to-SC comparative analysis is derived from the core-average EOC Haling simulation, since the best-estimate stability code must utilize a one-dimensional neutronic model that does not allow externally-imposed axial power shapes. The hot channel axial power shape in the FABLE procedure is an imposed, conservative power shape that is not derived from a Haling simulation. The hot channel axial power shape used in the

RC-to-SC design comparison is different than the constant hot channel axial shape of the FABLE procedure. However, this method reflects actual changes in hot channel stability performance. The corresponding effect on the hot channel decay ratio is ultimately applied to the FABLE SC baseline decay ratio data. Therefore, the conservatism of the FABLE procedure is preserved.

The analysis of the RC and SC designs is performed at selected state points along the 100% flow-control line with rated Xenon (equilibrium Xenon at rated conditions), equilibrium feedwater temperature and all control rods out (ARO). Three state points on the 100% flow-control line are analyzed: natural circulation (N/C) core flow, N/C flow + 10% of rated flow, and N/C flow + 25% of rated flow. If the decay ratios obtained at natural circulation are not sufficiently high to envelop the decay ratio range in the boundary generation stability criteria, additional decay ratio data may be obtained by repeating any of the above state points at Xenon-free conditions. The analysis state points are illustrated in Figure 7-5. The optional Xenon-free state points are indicated with dashed symbols and correspond to a higher rod line since ARO conditions are assumed. The SC and RC designs are independently analyzed at each state point. Table 7-4 summarizes the relative coordinates and operational conditions of the analysis state points.

To verify that the selection of the state points is appropriate for the RC-to-SC comparison analysis, the demonstration plant was analyzed at these state conditions with the ODYSY best-estimate stability code (the demonstration plant represents a potential RC design). Table 7-5 provides the core and hot channel decay ratio results, and indicates that a sufficient range of decay ratios are obtained from analysis at the selected state points.

7.2.2.3 Reference Cycle Decay Ratio Bias Correction Factor

The results of the best-estimate code analysis of the RC and SC designs along the 100% flow-control line can be used to generate a decay ratio bias correction between the RC design and the SC design. Figure 7-6 illustrates the comparison of the analysis results in decay ratio space. The bias, ΔDR_{Bias}, for either the core or channel decay ratio at a given state point is defined as:

$$\Delta DR_{Bias} = DR_{RC} - DR_{SC} \qquad (7\text{-}1)$$

where DR_{RC} is the RC decay ratio and DR_{SC} is the SC decay ratio. Since the SC design represents a typical 8x8 fuel design, fuel designs that result in an improved RC design stability performance relative to the SC design will result in a negative ΔDR_{Bias}.

From the ΔDR_{Bias} data, a continuous function, F_{Bias}, expressing the RC-to-SC decay ratio bias as a function of DR_{SC} can be defined to encompass the ΔDR_{Bias} data. The RC bias correction factor is expressed as a function of DR_{SC} to create the proper relationship to the FABLE baseline, which is also a function of the SC design. A decay ratio allowance, ΔDR_{Add}, may be incorporated into the RC bias function, F_{Bias}, as an optional method to accommodate potential stability effects resulting from design changes in future reloads. The RC correction factor bias is therefore given for both core and channel decay ratios by:

$$F_{Bias}(DR_{SC}) = f\left[(DR_{RC} - DR_{SC}), DR_{SC}, \Delta DR_{Add}\right] \tag{7-2}$$

Equation 7-2 represents a plant-specific continuous bias function, which yields the bias correction factor as a function of DR_{SC}. The RC-to-SC bias correction data for core and hot channel decay ratios and the corresponding F_{Bias} functions are plotted against the SC decay ratio in Figure 7-7. This relationship quantifies the stability performance differences between the RC and SC designs for both core and hot channel decay ratios.

Although the example illustrated in Figures 7-6 and 7-7 is based on analysis at three state points, additional state points can be considered if better resolution in the plant-specific fuel related stability performance, obtained from Equation 7-2, is needed. Additional state points to be analyzed should be selected in a manner consistent with the process illustrated in Figure 7-5.

7.2.2.4 Reference Cycle Decay Ratio

The licensing stability performance of the RC design is established by adjusting the FABLE SC baseline decay ratio data using the RC bias correction function, for both core and channel decay ratios. This process can be expressed for both the core and channel decay ratios as:

$$DR_{RC}^{Lic} = DR_{SC}^{Lic} + F_{Bias}(DR_{SC}), \tag{7-3}$$

where $F_{Bias}(DR_{SC})$ is the RC decay ratio bias correction function, DR_{SC}^{Lic} is the FABLE SC baseline decay ratio, and DR_{RC}^{Lic} is the RC decay ratio based on the adjusted FABLE SC baseline decay ratio data.

The application of Equation 7-3 is illustrated in Figure 7-8. The small circles in the figure represent the FABLE SC baseline decay ratio data from Figure 7-4. The state point numbers used for reference are the same as in Figures 7-2 and 7-4. The RC bias correction factor (Figure 7-7 and Equation 7-2) is used in Equation 7-3 to generate the FABLE-based RC core and hot channel decay ratios. The RC bias correction factor is indicated by the arrows in Figure 7-8 and the large circles represent the FABLE-based RC decay ratios. The RC decay ratios illustrated in Figure 7-8 are used to establish the nominal region boundaries.

7.2.3 Nominal Stability Region Boundaries

The nominal stability region boundaries are constructed from the FABLE-based RC decay ratio data, a generic region boundary shape function, and a flow clamp of the Exclusion Region at 40% of rated core flow.

7.2.3.1 Determination of Region Boundaries Intercepts

The state points used in the determination of the FABLE-based RC decay ratios lie along the edges of the operating domain, as shown in Figure 7-2. The RC decay ratios are plotted on core/hot channel decay ratio coordinates with the boundary generation stability criteria in Figure 7-8.

The power and flow coordinates of the intercepts for the three stability region boundaries at the natural circulation line and the high flow-control line are determined from the RC decay ratio data as plotted against the boundary generation stability criteria. The relationship between two adjacent decay ratio data points generated along the natural circulation line or the high flow-control line (illustrated in Figure 7-8) is assumed to be linear. This permits the determination of the region boundary intercepts in power/flow coordinates through linear interpolation.

The first step in determining the nominal region boundaries is to identify the decay ratio coordinates where the lines connecting adjacent analyzed state points intersect the stability criteria. This is illustrated in Figure 7-9 where Point M is the intersection of the line connecting the RC decay ratios for state points along the natural circulation line with the Monitored Region stability criterion. The relative length of the line segments between the adjacent state point decay ratios and the intercept point is computed. This relative length is then applied to the line segment connecting the corresponding natural circulation state points in power/ flow coordinates to determine the location of the Monitored Region boundary intercept. The results of this linear interpolation process for the three stability regions is illustrated in Figure 7-10.

7.2.3.2 Region Boundaries Determination

The stability region boundary generic shape function described in Appendix F is used to determine the shape of the region boundaries in the interior of the licensed operating domain. The generic shape function is applied to the region boundary intercepts for the Exclusion, Restricted and Monitored Regions of Figure 7-10, and expressed as:

$$P = P_y \left(\frac{P_x}{P_y} \right)^{\frac{1}{2}\left[\frac{W - W_y}{W_x - W_y} + \left(\frac{W - W_y}{W_x - W_y} \right)^2 \right]} \tag{7-4}$$

where P_x and W_x are power and flow coordinates of the region boundary intercepts along the high flow-control line and P_y and W_y are the coordinates of the region boundary intercepts along the natural circulation line. The resulting boundaries for the three regions are illustrated in Figure 7-11. The final form of the Exclusion Region boundary is established by applying a clamp at 40% core flow. The final plant-specific nominal region boundaries are illustrated in Figure 7-12.

7.2.4 Setpoint Determination

The stability region setpoints are determined based on the nominal region boundaries, the existing setpoints for core flows above the region boundaries, and the flow-biased neutron flux scram line clamp above the Exclusion Region. The detailed formulation of the setpoints for the flow-biased neutron flux scram line and the control rod block line is provided in Section 6.1.

7.3 Validation of New Region Boundaries

This section describes the analysis required to validate region boundaries for Enhanced Option I-A initial application. This includes a description of the approach employed in the validation process, state point selection, and initial validation demonstration.

7.3.1 Validation Approach

Section 7.2 described the process for establishing nominal region boundaries based on a plant-specific RC design and the generic boundary generation stability criteria. Stability region boundaries are based on the conservative FABLE licensing methodology. The degree of conservatism inherent in the FABLE procedure, and the stability controls required in the Restricted Region, provide sufficient stability margin to explicitly accommodate setpoint uncertainties associated with the location of the region boundaries. The analysis required to validate the region boundaries is performed with a best-estimate stability code at reasonably limiting conditions and demonstrates the applicability of the analytical region boundaries. This process ensures appropriate consideration for setpoint uncertainties.

7.3.1.1 Setpoint Uncertainty

The nominal reactor flow-biased neutron flux scram and control rod block setpoints for Enhanced Option I-A are determined based on the nominal region boundaries. These setpoints are designed to prevent reactor instability during normal reactor operation and anticipated transient conditions. The flow-biased setpoints meet these objectives considering setpoint uncertainties. Therefore, the region boundary validation analysis must explicitly address setpoint uncertainty considerations. Since the stability region boundaries are defined in terms of core power and flow, the setpoint uncertainty is expressed in terms of these variables. The core power and flow uncertainties are plant-specific and are applied to the plant-specific nominal region boundaries to define analytical region boundaries.

The analytical boundaries are used in the region boundary validation analysis to ensure that the overall conservatism of each region boundary is maintained. Given the shape of the stability region boundaries near the intercepts with the natural circulation line and the high flow-control line, a change in core flow along a region boundary near the natural circulation line or in core power along a region boundary near the high flow-control line results in a small changes in decay ratio. The core power uncertainty is therefore only applied to analysis along the natural circulation line, and the core flow uncertainty only to analysis along the high flow-control line.

An illustration of nominal and analytical region boundaries for the Exclusion and Restricted Regions is provided in Figure 7-13. Since setpoint uncertainty is plant-specific, the details of the setpoint uncertainty determination will be performed separately for each plant application.

For the demonstration plant analysis, conservative values of setpoint uncertainty of approximately 10% are selected. An off-rated core power uncertainty of 3% of rated power and a core flow uncertainty of 5% of rated flow are used for the specific demonstration analysis conditions. The two-loop setpoint uncertainties are applicable to SLO as discussed in Appendix G.

7.3.1.2 Analysis Approach

The validation analysis is performed by a best-estimate stability code, which meets the requirements of Section 7.4, at reasonably-limiting state points including steady-state and transient conditions. The reactor conditions for all state points are analyzed with a three-dimensional steady-state BWR core simulator. For transients, the validation of the region boundaries is conducted only for the final conditions of the transient. Therefore, a steady-state neutronic calculation can be performed to obtain the final state point conditions required for input to the best-estimate stability code.

7.3.2 State Points Selection

The initial validation matrix (IVM) defines the state points and conditions utilized in the process of validating new region boundaries. The IVM is a subset of the demonstration validation matrix (DVM) created for the demonstration plant which excludes state points demonstrated to be non-limiting in terms of stability performance.

7.3.2.1 Demonstration Plant Analysis Review

Extensive analysis using the ODYSY best-estimate stability code was performed to confirm the validity of the stability region boundaries generated for the demonstration plant as documented in Appendix E. This analysis was performed to investigate the feasibility of the region boundary validation process, and is not a part of the normal process for validating region boundaries. The demonstration plant validation analysis clearly establishes that steady-state and transient conditions with stability controls in place exhibit non-limiting stability performance and are bounded by state conditions with no stability controls. These conclusions are used in defining the IVM analysis scope.

The approach used for validating the demonstration plant region boundaries was different than the approach defined for validating region boundaries using the IVM in that it involved various iterative searches for worst-case stability conditions. The validation analysis for the demonstration plant is performed at the analytical stability region boundaries, with a stability control of a 4.0 foot core average bulk coolant boiling boundary imposed in the Restricted Region. Xenon-free conditions are assumed at low-flow startup conditions, and equilibrium Xenon at higher power conditions. Control rod pattern searches for worst-case axial power shapes are performed outside the Restricted Region.

The analyzed state conditions include steady-state conditions, IFREs from rated power, and the LOFH events. A minimum set of ten state conditions defines the DVM. The location of the DVM state points in the demonstration plant operating domain are shown in Figure 7-14. The DVM includes steady-state points and LOFH events at the four intercepts of the Exclusion and Restricted analytical boundaries with the edges of the licensed operating domain. Two IFREs from rated power with minimum (W_{min}) and with rated flow conditions (H0 and H1) to the Exclusion Region analytical boundary (W_{fr}) are also considered. Additional state points analyzed as part of the demonstration plant validation effort (not shown in Figure 7-14) are presented in Appendix E.

The DVM analysis results are shown in Figure 7-15. The bold symbols in Figure 7-15 represent state points analyzed with stability controls in place and the light symbols represent analysis performed with no stability controls. Steady-state points and LOFH initial points on the Restricted Region analytical boundary are analyzed with no stability controls. Steady-state points and LOFH initial points on the Exclusion Region analytical boundary are analyzed with stability controls applied. The IFREs termination points on the Exclusion Region analytical boundary are analyzed with no stability controls.

The analysis results demonstrate that stability-controlled reactor state conditions are well-bounded by the state conditions analyzed with no controls applied, and result in hot channel decay ratios of approximately zero. In particular, the decay ratios for steady-state and LOFH events analyzed at Point A_{fr} are completely suppressed with a core decay ratio of approximately 0.2 and a hot channel decay ratio of zero. This result is supported by additional analyses performed in the Restricted Region at high flow-control lines. It is concluded that at increased forced flow conditions with the core boiling boundary at 4.0 feet or above, the core decay ratios are very low (approximately 0.1 to 0.2) and hot channel decay ratios are approximately zero.

7.3.2.2 Non-Limiting State Points

Based on the DVM analysis results, all state conditions analyzed with stability controls in place on the Exclusion Region analytical boundary exhibits non-limiting stability performance. Therefore, the steady-state and the LOFH events analyzed at state Points A_{fr} and B are non-limiting (See Figure 7-15). Although the steady-state condition at state Point B is not limiting, it is retained in the IVM to validate the effectiveness of the stability controls. Table 7-6 describes the DVM state points excluded from the IVM.

7.3.2.3 Initial Validation Matrix

The initial and final analysis conditions for the IVM state points constitute a generic, prescriptive and reasonably limiting process for validating stability region boundaries. The IVM is a subset of the full DVM, excluding the non-limiting state points. The IVM state points are shown in Figure 7-16 and listed in Table 7-7.

The IVM is applied to the current cycle (CC) design at the analytical region boundaries. Cycle Haling depletions to end-of-full-power conditions are performed. Two Haling depletions are required, the first to Point H0 in Figure 7-16 with rated core flow and power, and the second to Point H1 with minimum core flow (W_{min}) at rated power. FWHOOS should be considered in the Haling depletions, if applicable.

The Haling depletion is uniquely defined and provides a meaningful comparison tool for future reload validation analyses, which are also performed based on a cycle Haling depletion. The Haling axial power shape is typically very flat and is associated with low core boiling boundary. The EOC conditions also result in a highly-negative void coefficient. Overall, the choice of EOC Haling core conditions is conservative.

Xenon-free conditions are assumed at natural circulation conditions, an assumption consistent with the low Xenon concentrations associated with reactor start-up. Xenon-free conditions are identified as conservative based on the analysis supporting the core boiling boundary control. Additionally, the Xenon-free condition results in a hot channel axial power shape that is highly bottom-peaked, and therefore conservative for channel decay ratio calculations. For analysis along the high flow-control line, equilibrium Xenon concentrations are assumed, consistent with anticipated reactor operations.

Feedwater temperature changes during the LOFH event are assumed to follow observed patterns. The combination of reasonably limiting axial power shape (i.e., Haling) and end-of-cycle exposure (i.e., void reactivity coefficient), coupled with the expected change in feedwater temperature, results in reasonably limiting conditions for the LOFH event. A generic 60°F equivalent-rated feedwater temperature change during LOFH events is utilized.

The analysis of IFREs at the event terminal point assumes that immediate post-flow reduction conditions include rated Xenon and rated feedwater temperature. The Haling axial power shape and end-of-cycle exposure result in reasonably limiting conditions for IFREs. Since the feedwater temperature transient following an IFRE is slow (i.e., several minutes) and significant defense-in-depth features exist (e.g., Restricted Region Entry alarm, PBDS, required operator actions), analysis of the event terminal point with equilibrium feedwater temperature is not required.

Control rod patterns at off-rated conditions are prescribed for the different state point conditions. In general, a radially uniform deep rod pattern at notch position 00 or 08 is prescribed. A single control rod per core quadrant may be adjusted to ensure criticality. Inside the Restricted Region, a radially-uniform shallow rod pattern that achieves the target boiling boundary limit is prescribed. For state points with no stability controls, minimum shallow control rod insertion for thermal limits control is prescribed.

The initial conditions for IVM analysis are summarized in Table 7-8. For each state point, the table specifies the Haling depletion (i.e., H0 or H1), feedwater temperature, Xenon concentration, and control rod pattern. The final conditions for the IVM analysis are summarized in Table 7-9. Steady-state conditions remain unchanged. The final conditions for transient events include specified feedwater temperature, Xenon concentration and core flow conditions.

7.3.3 Validation of Region Boundaries

The results of the IVM state point analysis are decay ratio values which quantify the susceptability to fundamental (core wide) and first order azimuthal harmonic (regional) modes of reactor instability. These values are compared against the corresponding best-estimate code boundary validation stability criterion. If all analysis results satisfy the criterion, then the IVM analysis validates the new region boundaries.

7.3.4 Initial Application Validation Demonstration

The IVM was applied to the demonstration plant to confirm the analysis process. In this case, the IVM was based on the same equilibrium cycle design that was used to generate the demonstration plant region boundaries. The IVM analysis was performed with the ODYSY best-estimate stability code. In plant-specific applications of the Enhanced Option I-A initial application process, it is expected that the Current Cycle design will be bounded by the Reference Cycle design.

The results of the demonstration-plant IVM analysis are shown in Figure 7-17. The IVM results indicate that the validation process is challenging, especially at Point B' (highest decay ratio for steady-state and LOFH). The steady-state point at Point B with stability controls in place is non-limiting as expected. The IFRE results at immediate post-event conditions show considerable stability margin. Additional information is provided in Appendix E, including a full set of IVM results for the demonstration plant with FWHOOS.

7.4 Stability Codes Requirements

7.4.1 FABLE/BYPSS Procedure

The FABLE/BYPSS procedure used to perform the Standard Cycle baseline stability analysis is the same as that described in NEDO-31960 (Reference 1). The procedure uses a combination of nominal and conservative inputs and a specified bias correction to calculate state points along the natural circulation line and a high flow-control line. The inputs which are specified by the procedure include void coefficient, thermal-hydraulic data, axial power shapes, radial power distribution, pellet-clad gap conductance, recirculation loop time constant and resistance, plant heat balance data, fuel physical parameters, and material properties. The specified procedure and the corresponding stability criteria result in an overall conservative method for calculating the Standard Cycle decay ratios.

7.4.2 Best-Estimate Method Requirements

The Initial Application and Reload Review procedures for Enhanced Option I-A are designed to utilize any best-estimate stability methodology which has been qualified for Enhanced Option I-A application. The best-estimate frequency domain stability code ODYSY, which is proprietary to GE, is described in Supplement 1 to this report. Its description is provided since it has been utilized in performance of the stability control studies and demonstration plant analyses. However, plant-specific application of Enhanced Option I-A may use any method which has been qualified for this application.

The characteristics of a best-estimate stability code are defined below so that the types of analysis necessary to be performed in initial application and reload reviews for Enhanced Option I-A will be adequately modeled.

7.4.2.1 Core Thermal-Hydraulics Model

A one-dimensional thermal-hydraulic model is required. Minimum key features required include:

(1) Multiple channel types with independent geometry and axial power distributions,
(2) Separate bypass region,
(3) Mass, energy, and momentum equations,
(4) Hydraulic and heat transfer calculations for single-phase liquid, two-phase subcooled boiling, and two-phase nucleate (saturated) boiling using best-estimate thermal-hydraulic correlations,
(5) Direct heating of moderator,
(6) Local pressure losses due to spacers and upper tie plate,
(7) Unheated fuel bundle components modeled (above active length), and
(8) Transient redistribution of channel flows due to channel coupling.

7.4.2.2 Kinetics Model

An accurate nuclear model that accounts for nodal reactivity perturbations is required. It must include the reactivity effects of different axial void distributions and fuel temperatures in each channel type. Kinetics parameters are collapsed from a three-dimensional BWR simulator output at specific plant conditions. Minimum key features must include:

(1) One-group diffusion theory,
(2) One-dimensional variation of neutron flux,
(3) One-dimensional void reactivity feedback,
(4) Six delayed neutron groups,
(5) Doppler reactivity feedback,
(6) Control rod dependent properties,
(7) Bypass reactivity effects,
(8) Modeling of axial and radial variations in control fraction and power shape; each axial node must have a unique set of collapsed parameters based on fuel type, control fraction, and power distribution, and
(9) Coefficients that describe the effects of fuel temperature and moderator density changes on the kinetics parameters generated for each channel group.

7.4.2.3 Ex-Core Model

A lumped parameter nodal model is required which includes the following components: vessel steam dome, separator, upper plenum, recirculation system, and lower plenum. The recirculation system and lower plenum must include the momentum equation of the recirculation model for the steady-state pressure drop balance and the dynamics calculation. Minimum key features must include:

(1) Fundamental modeling of steam separators, downcomer, recirculation system, and lower plenum,
(2) Modeling of forward and reverse flow, and
(3) Recirculation system model includes external pumps with jet pumps.

7.4.2.4 Fuel Heat Transfer Model

A one-dimensional radial conduction heat transfer model is required. Minimum key features must include:

(1) Multiple bundle types,
(2) Each bundle represented by an average fuel rod for heat transfer,
(3) One-dimensional radial heat transfer (no axial heat transfer or radiation heat transfer models required),
(4) Axially varying power generation rate,
(5) Radially varying power generation within the fuel pellet,
(6) Temperature dependent thermal properties for fuel,
(7) Surface heat transfer coefficient dependent upon moderator conditions, and
(8) Direct energy deposition to fluid.

7.4.3 Best-Estimate Methodology Uncertainty

The best-estimate methodology must be qualified against stability data as part of the demonstration that it will accurately calculate decay ratios. The accuracy of the model in predicting tested state points is used to determine a statistical uncertainty in the decay ratio predictions. The statistical uncertainty is explicitly incorporated into the stability criterion of the vendor-specific best-estimate methodology.

For example, the FABLE/BYPSS methodology was qualified against test data using nominal inputs to represent the plant operating conditions. The qualification testing showed a conservative bias in the decay ratio calculations and a bias correction was developed. With the bias correction applied, the decay ratios calculated by the procedure are accurate to a statistical uncertainty standard deviation of 0.08. The application procedure uses a model uncertainty of 0.2, corresponding to greater than two standard deviations, when calculating the core and channel decay ratios. This results in the FABLE/BYPSS stability criterion shown in the Exclusion Region column of Table 7-2, and plotted for the Exclusion Region on Figure 7-3.

A similar derivation must be performed for the vendor's qualified best-estimate stability code. The resulting stability criterion will be used to determine if the validation results calculated with the best-estimate code are acceptable.

Table 7-1: FABLE SC Analysis State Points

State Point	Core Flow (% of Rated)	Core Power (% of Rated)
1	30	15
2	30	25
3	30	35
4	30	45
5	35	62
6	45	71
7	55	80
8	70	91

Table 7-2: Boundary Generation Stability Criteria Coordinates

Exclusion Region		Restricted Region		Monitored Region	
DR_{ch}	DR_{core}	DR_{ch}	DR_{core}	DR_{ch}	DR_{core}
0.00	0.80	0.00	0.60	0.00	0.40
0.56	0.80	0.42	0.60	0.28	0.40
0.58	0.70	0.43	0.52	0.29	0.35
0.60	0.60	0.45	0.45	0.30	0.30
0.63	0.50	0.47	0.37	0.31	0.25
0.67	0.40	0.50	0.30	0.33	0.20
0.72	0.30	0.54	0.22	0.36	0.15
0.79	0.20	0.59	0.15	0.39	0.10
0.80	0.19	0.60	0.14	0.40	0.09
0.80	0.00	0.60	0.00	0.40	0.00

Table 7-3: Reference Cycle Design Considerations

Reload Review Procedure Design Change Criteria	**Possible RC Design Features**
CC to RC reactor design modifications have no effect on stability	Reactor design features planned for future cycles
CC to RC fuel and channel mechanical design changes have no effect on stability	Fuel and channel design features anticipated in future cycles or arbitrary design features to address uncertainties in fuel design evolution
CC Haling radial peaking increase over RC by no more than 5%	Local fuel bundle shuffle increases radial peaking to accommodate expected increased in peaking in reload designs
CC reload batch size change relative to RC by no more than 5% of core size	Nominal batch size anticipated for future reloads
CC to RC full power cycle energy change within 10%	Nominal cycle energy anticipated for future cycles
Last cycle coastdown change relative to RC no more than 10% of full-power cycle energy	Nominal coastdown anticipated for future cycles
Last cycle energy shorter than RC by no more than 10%	Not applicable
No variation in CC mixed batch reload core relative to RC design	Allowance for transition cycles including radial peaking and mechanical design features
CC core loading strategy relative to RC unchanged	Conservative core loading, if practical
No new operating modes for CC relative to RC	New operating modes if degrade stability performance
Other differences between CC and RC do not result in equivalent loss of stability margin	Arbitrary bias to accommodate unanticipated future changes

Table 7-4: Analysis Conditions for Standard to Reference Cycle Comparison

<table>
<tr><th>Core Flow</th><th>State Point Conditions</th><th>Optional Conditions</th></tr>
<tr><td>Natural circulation (N/C)</td><td rowspan="3">EOC Haling 100% flow-control line:
• Equilibrium feedwater temperature
• All rods out
• Rated Xenon</td><td rowspan="3">From EOC Haling 100% flow-control line:
• Equilibrium feedwater temperature
• All rods out
• Xenon-free</td></tr>
<tr><td>N/C + 10% of rated</td></tr>
<tr><td>N/C + 25% of rated</td></tr>
</table>

Table 7-5: Reference to Standard Cycle Comparison State Point Demonstration

Core Flow (% of Rated)	Xenon Conditions	Core Decay Ratio	Channel Decay Ratio
55	Rated	0.31	0.06
40	Rated	0.47	0.33
30	Rated	0.77	0.37
30	Free	1.33	1.28

Table 7-6: DVM State Points Not Included in IVM

Category	Analytical State Point	Justification
Steady-State	A_{fr}	Required controls and forced flow result in DR≈0.1/0.2
Flow Events	None	N/A
LOFH Events	A_{fr}, B	Control required at initial condition, result bounded by LOFH at A', B'

Table 7-7: Initial Validation Matrix State Points

Category	Analytical State Point
Steady-State	B A' B'
Flow Events	From H1 and H0 to exclusion region analytical boundary
LOFH Events	A' B'

Table 7-8: Initial Validation Initial State Conditions

Category	Analytical State Point	Initial State Conditions
All	All	EOC Haling. FWHOOS if applicable.
Steady-State	B	Based on H0 depletion. Xenon-free. Radially uniform control rods: deep at 00/08 with one rod per core quadrant adjusted for criticality, shallow for Z_{bb} control.
	A'	Based on H1 depletion. Equilibrium Xenon. Deep control rods at 00/08, with one rod per core quadrant adjusted for criticality, to compensate for lower Xenon.
	B'	Based on H0 depletion. Xenon-free. Radially uniform control rods: deep at 00/08 with one rod per core quadrant adjusted for criticality, shallow to control thermal limits.
Flow Events	H1	EOC Haling to H1.
	H0	EOC Haling to H0.
LOFH Events	A'	Same as A' steady-state.
	B'	Same as B' steady-state.

Table 7-9: Initial Validation Final State Conditions

Category	Analytical State Point	Final State Conditions
Steady-State	B A' B'	Same as initial state conditions.
Flow Events	H1 H0	W_{fr} core flow. Initial T_{fw}. Initial Xenon.
LOFH Events	A' B'	T_{fw} - 60°F equivalent rated. Initial Xenon.

Figure 7-1: Initial Application Analysis Elements

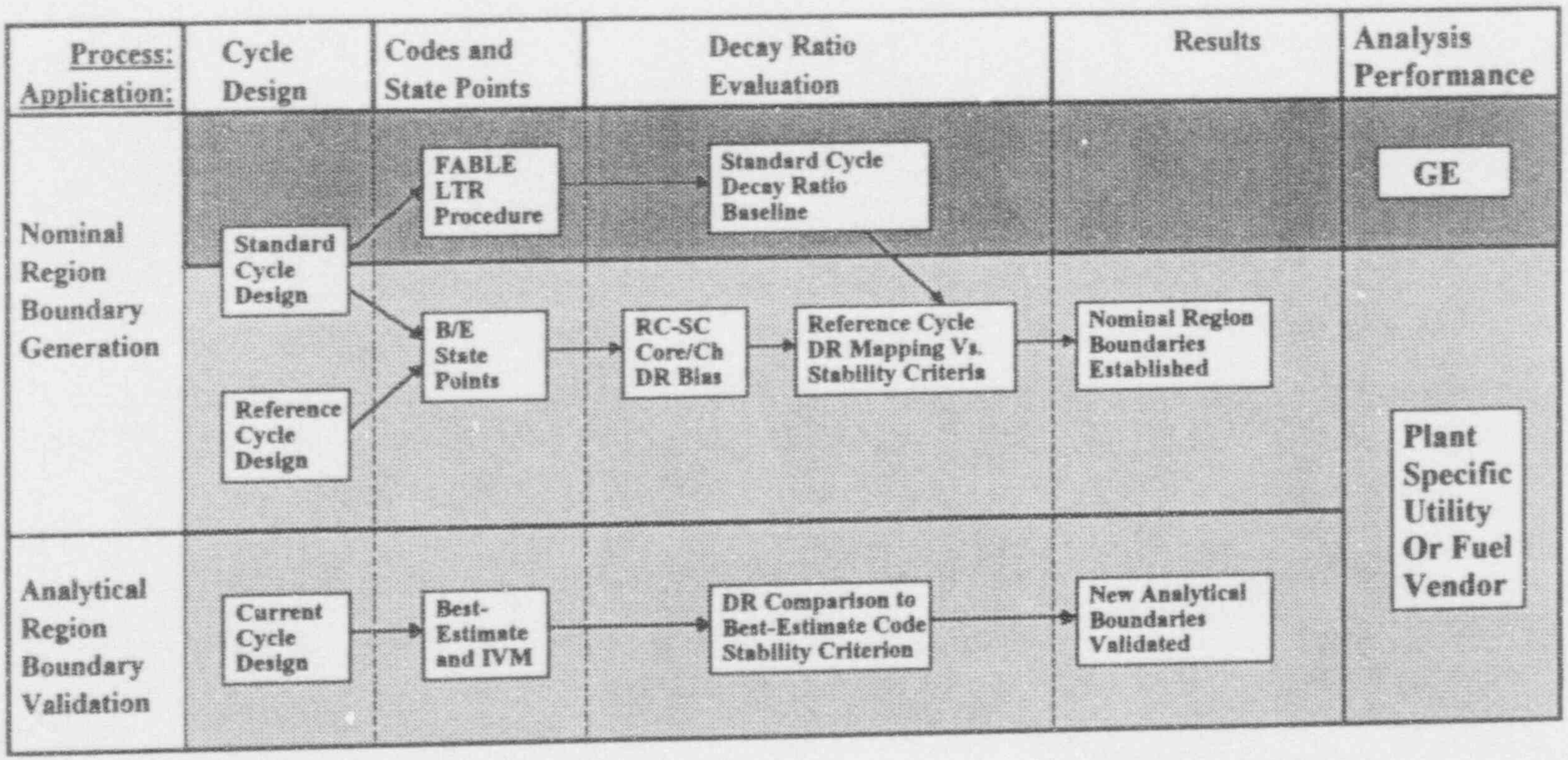


Figure 7-2: FABLE Procedure State Points

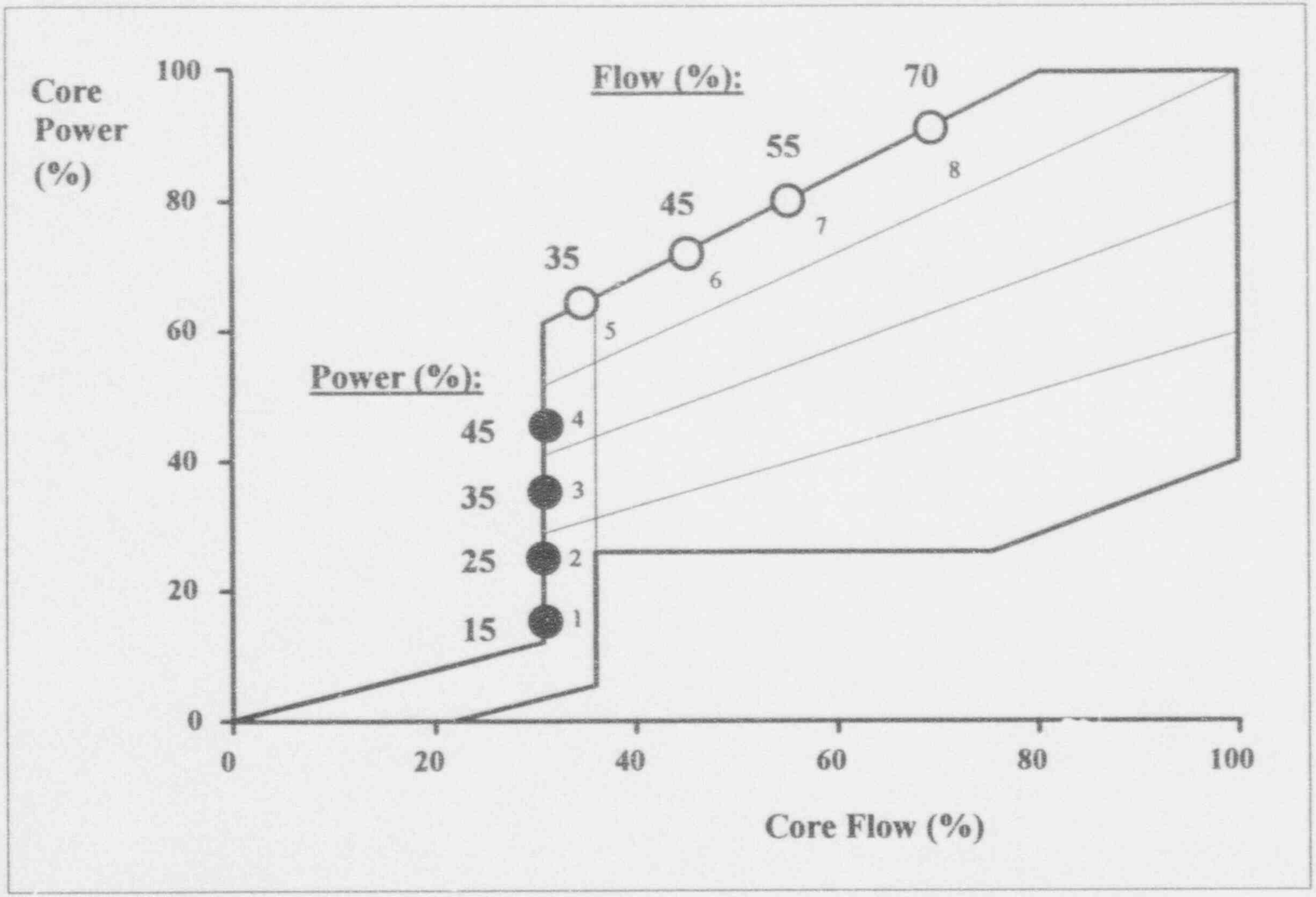


Figure 7-3: Boundary Generation Stability Criteria

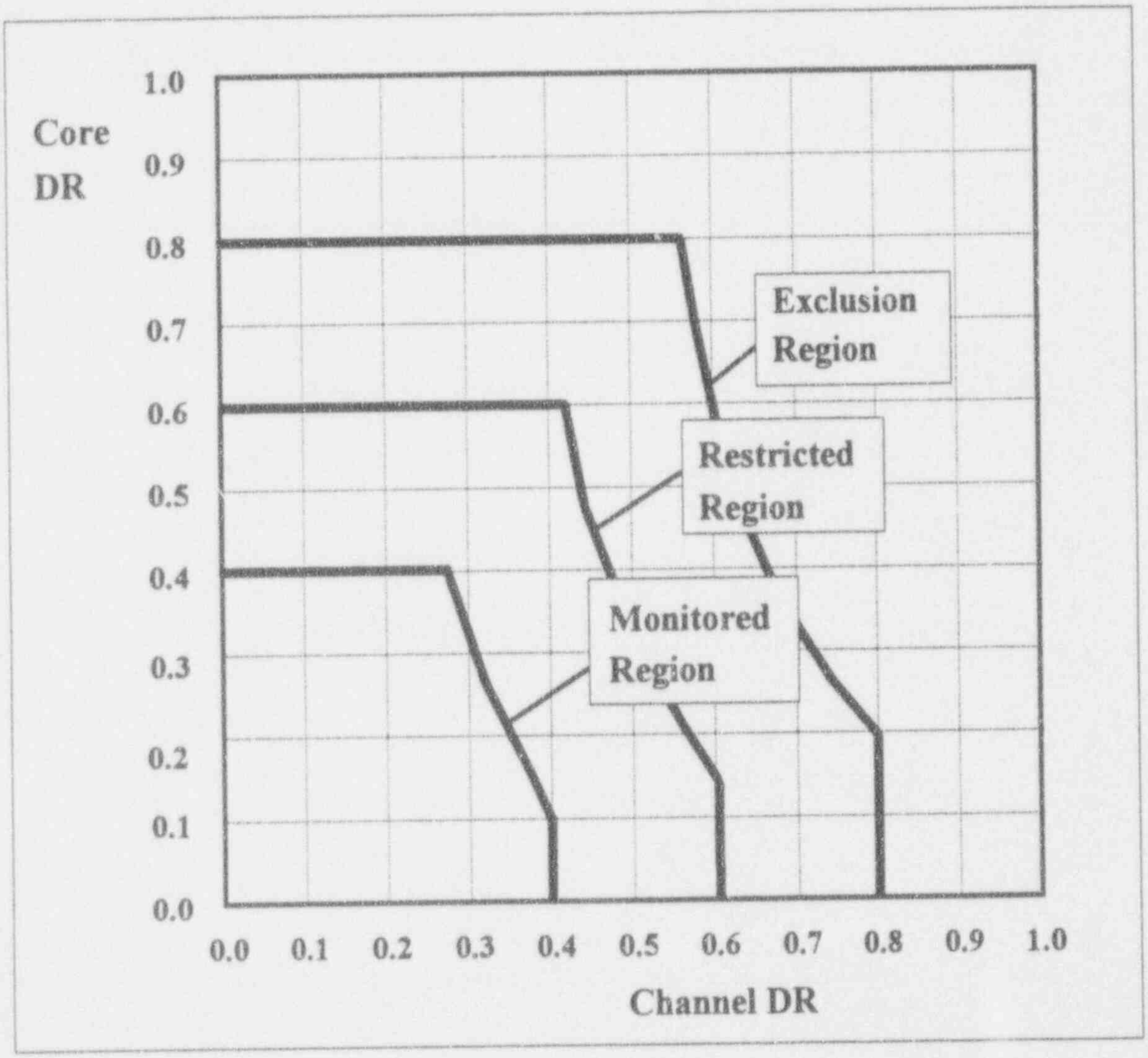


Figure 7-4: FABLE Standard Cycle Decay Ratio Baseline (example)

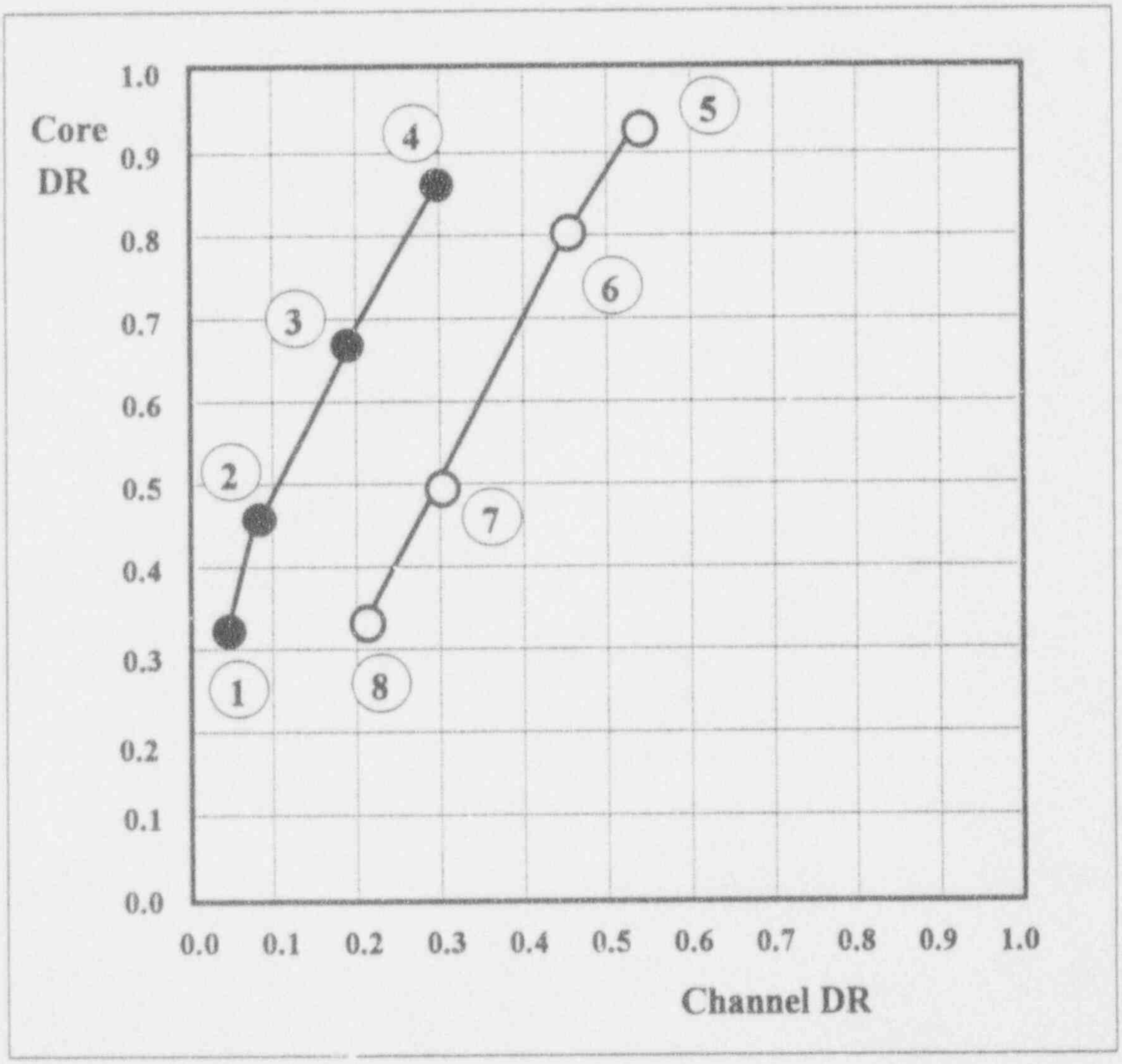


Figure 7-5: Reference Cycle vs. Standard Cycle Stability Performance

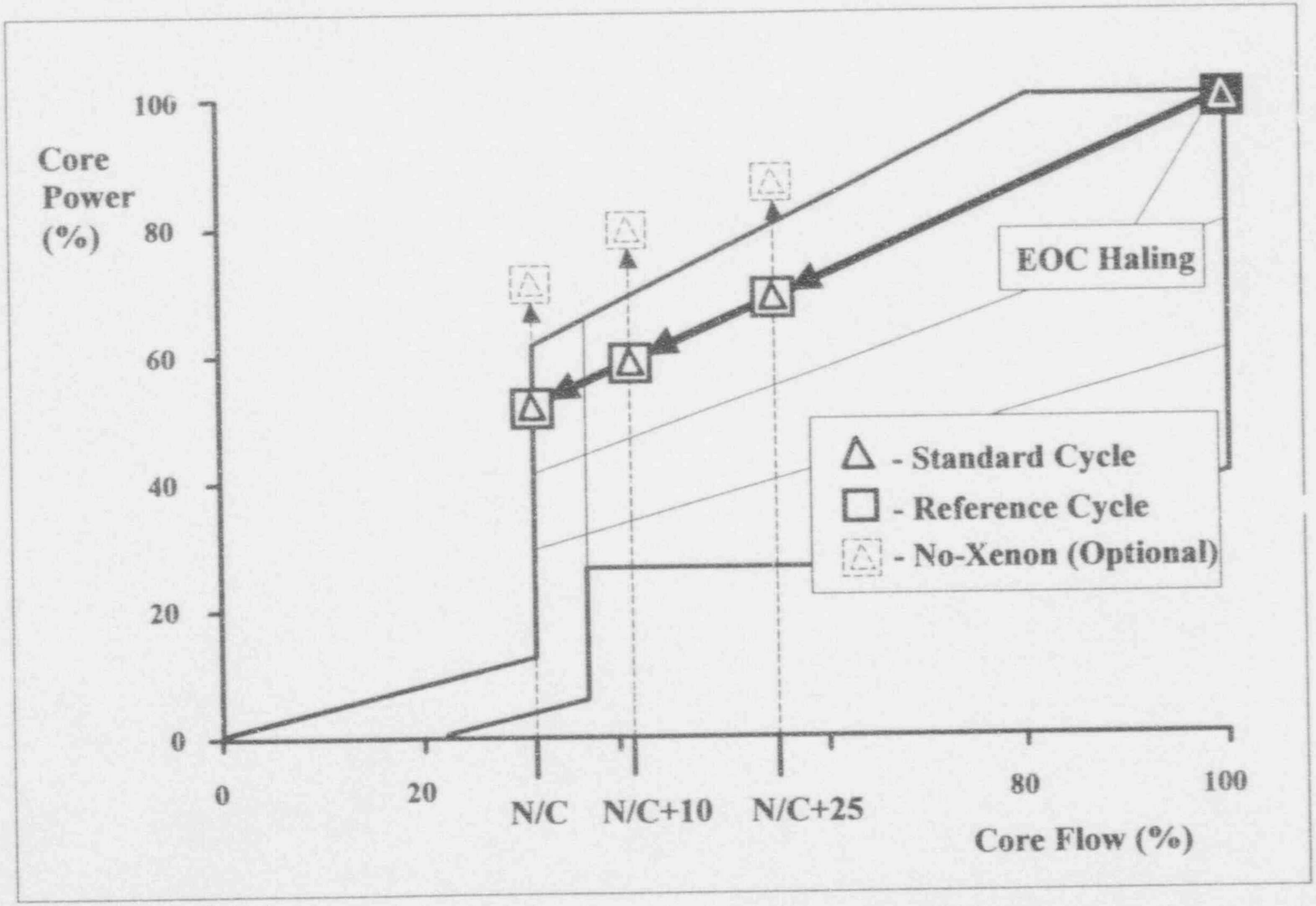


Figure 7-6: Reference Cycle vs. Standard Cycle Decay Ratio (example)

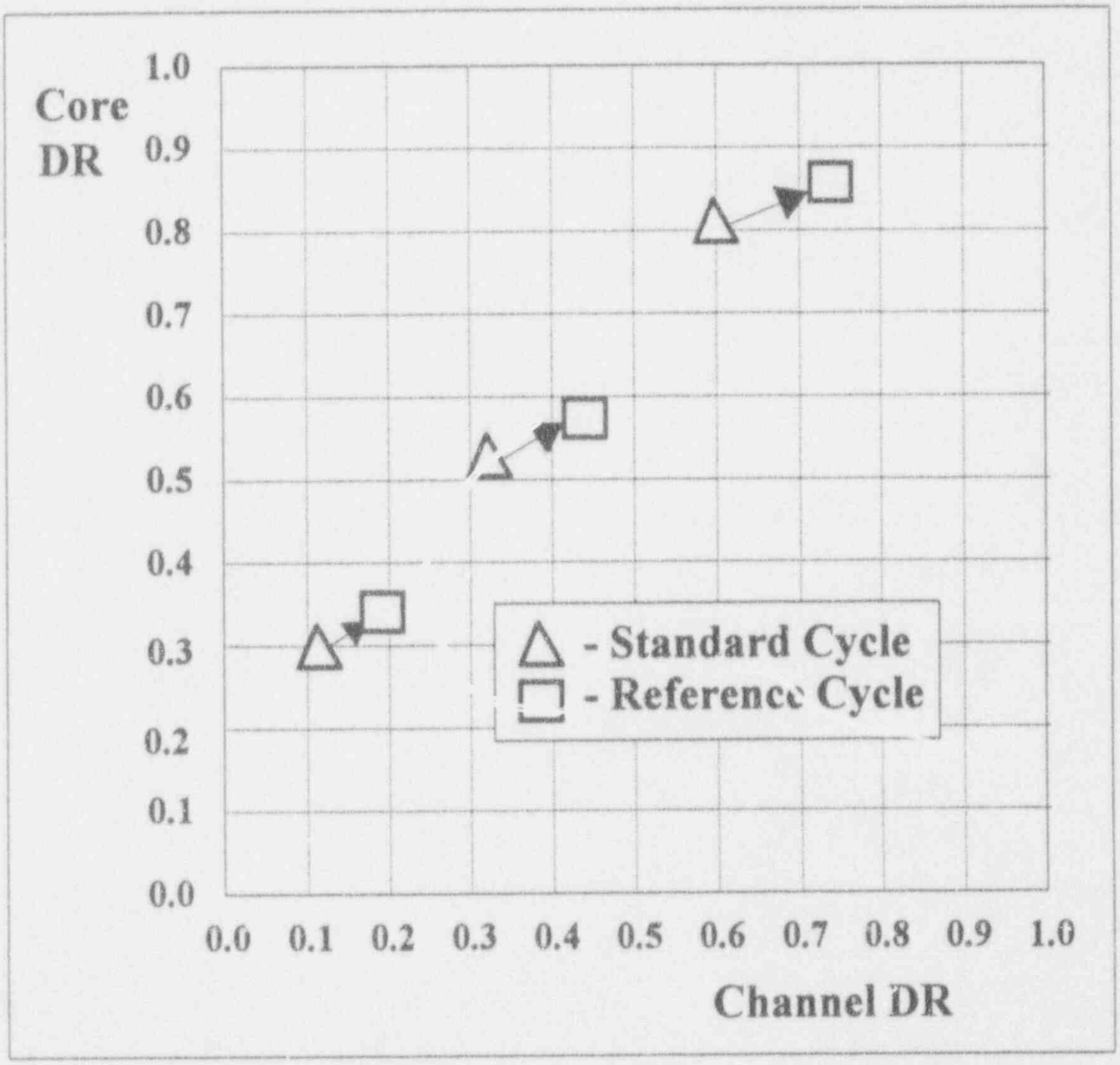


Figure 7-7: Plant-Specific Reference Cycle Decay Ratio Bias (example)

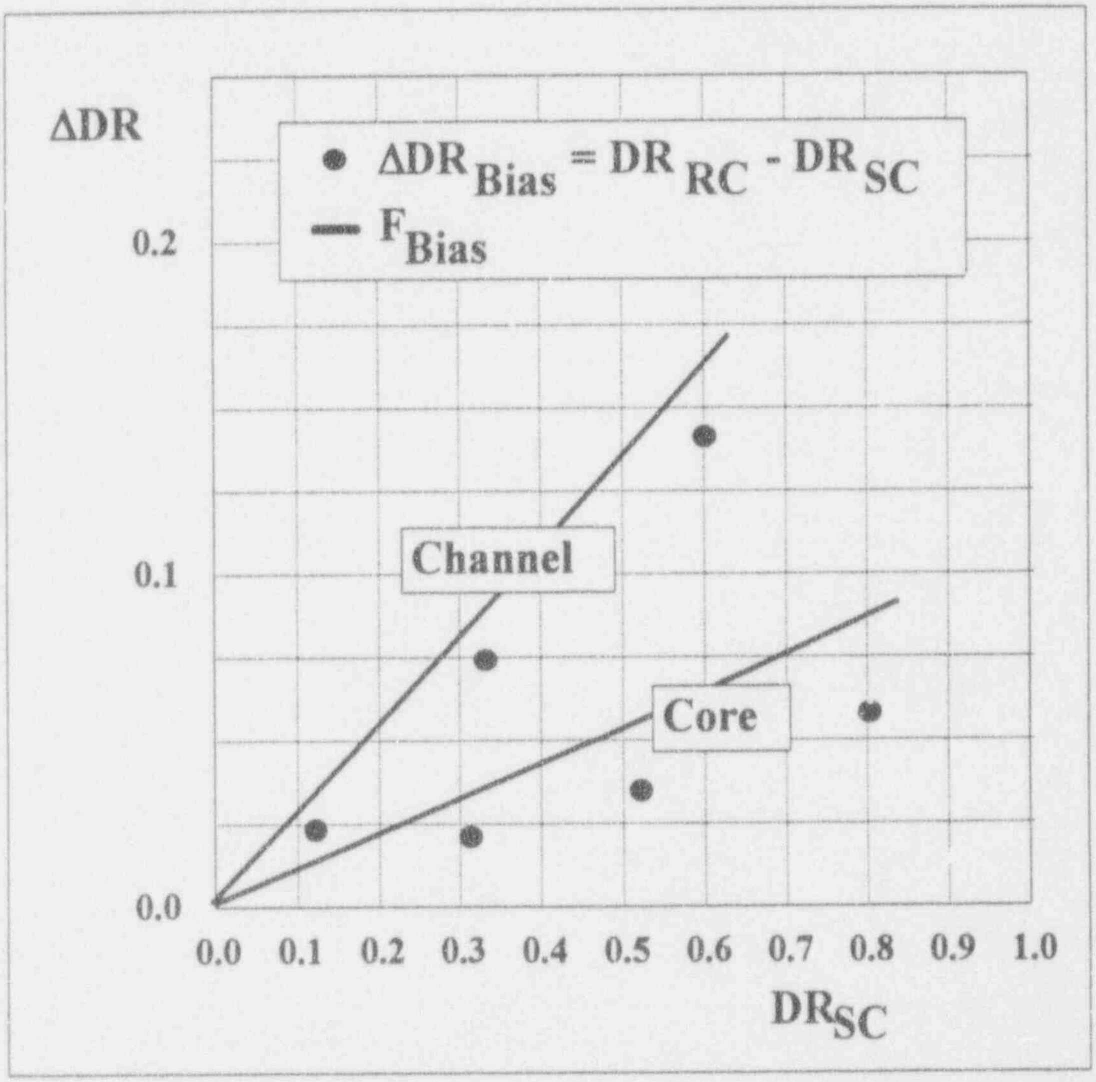


Figure 7-8: Reference Cycle FABLE-Based Decay Ratio (example)

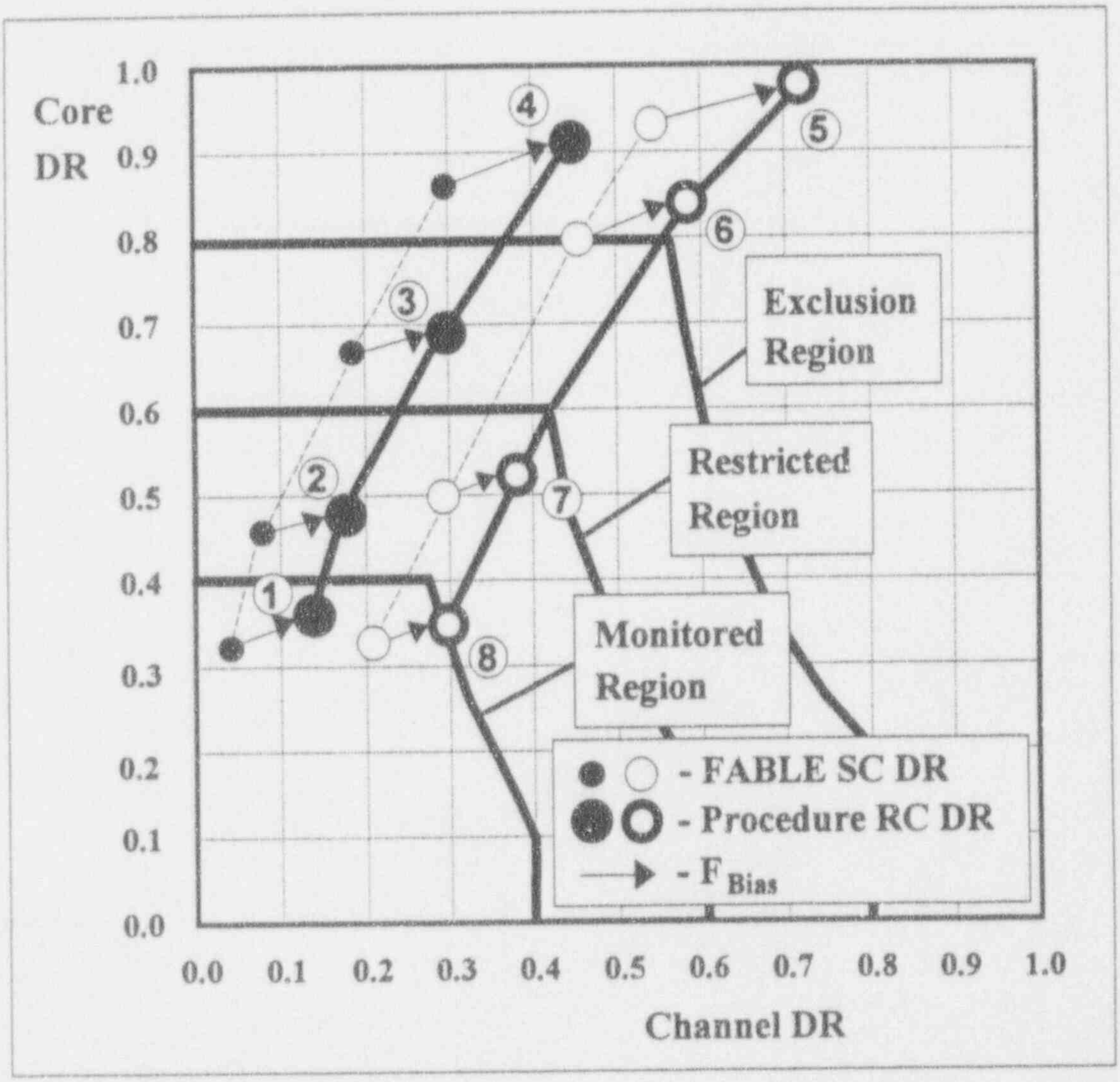


Figure 7-9: Stability Region Intercepts Determination (example)

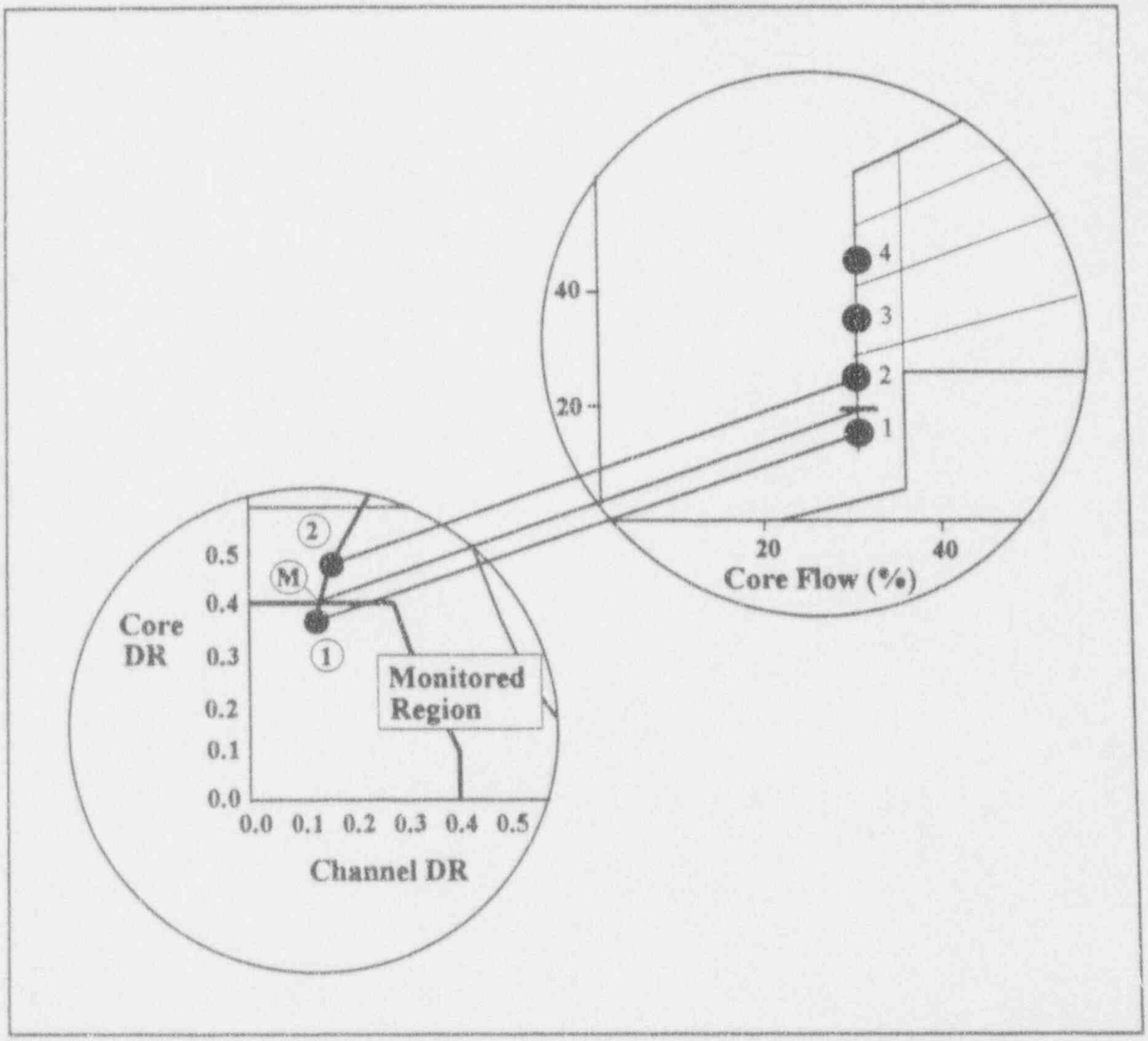


Figure 7-10: Region Boundaries Intercepts (example)

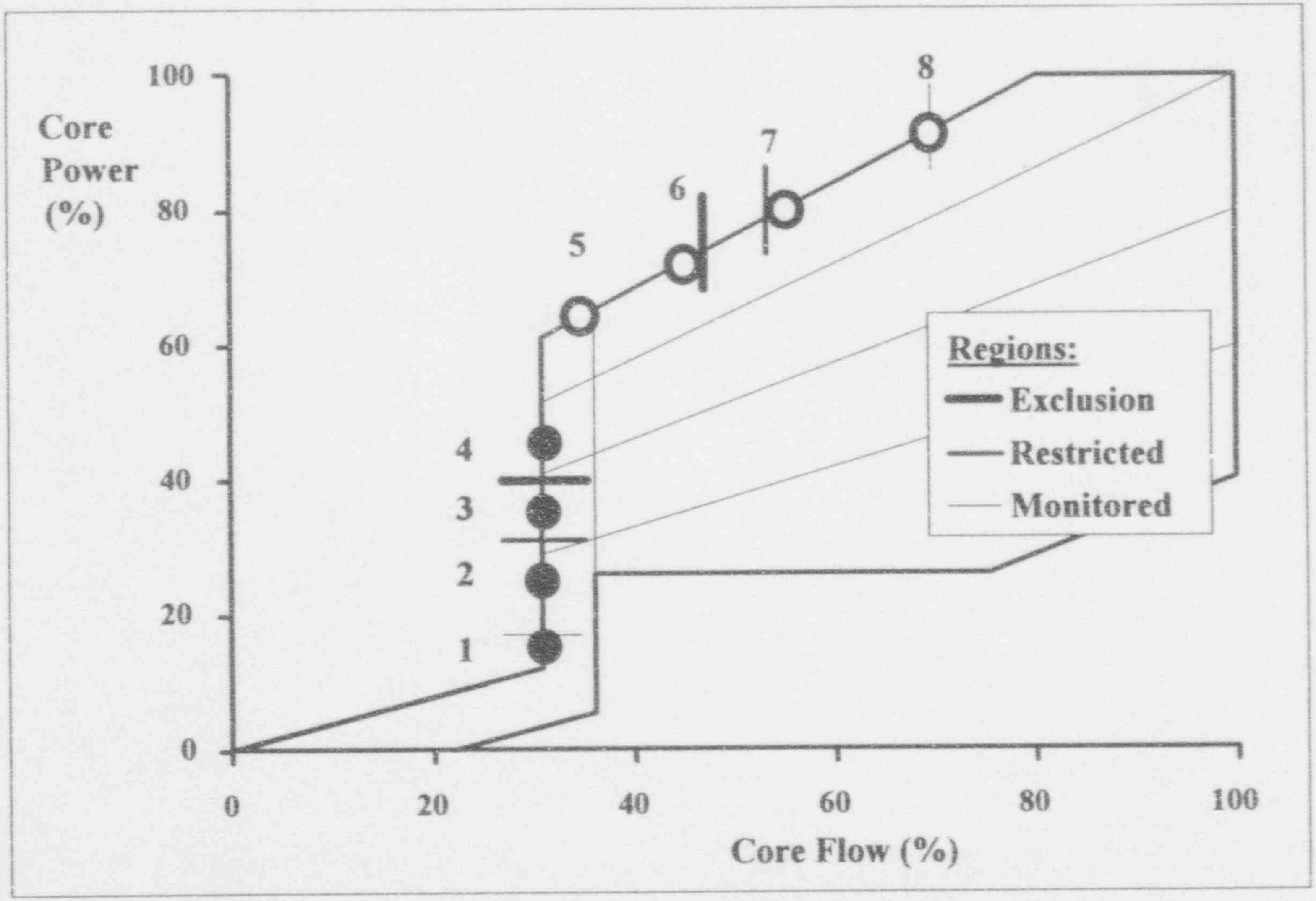


Figure 7-11: Region Boundaries Generic Shape Function (example)

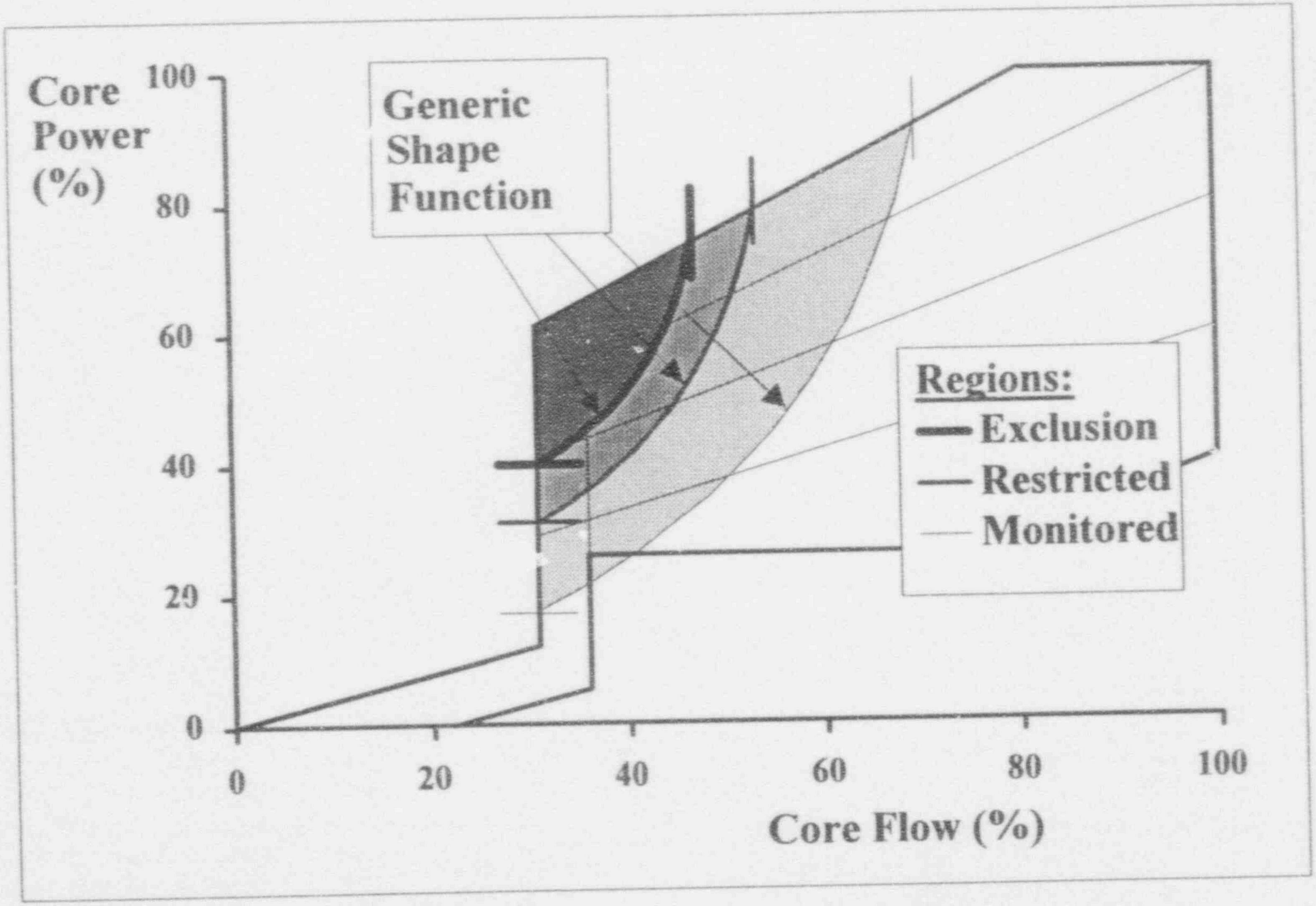


Figure 7-12: Final Nominal Region Boundaries (example)

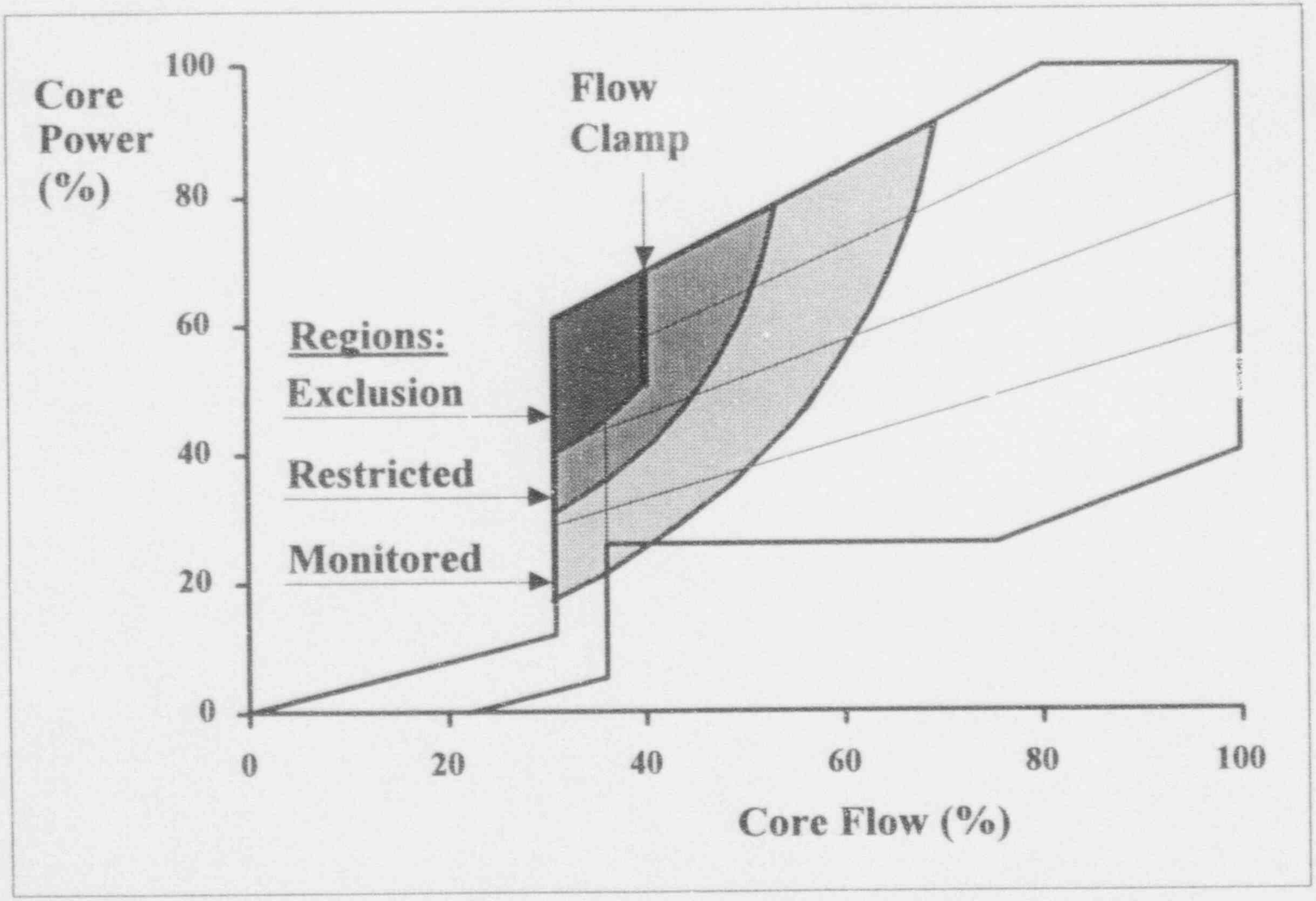


Figure 7-13: Analytical Region Boundaries (example)

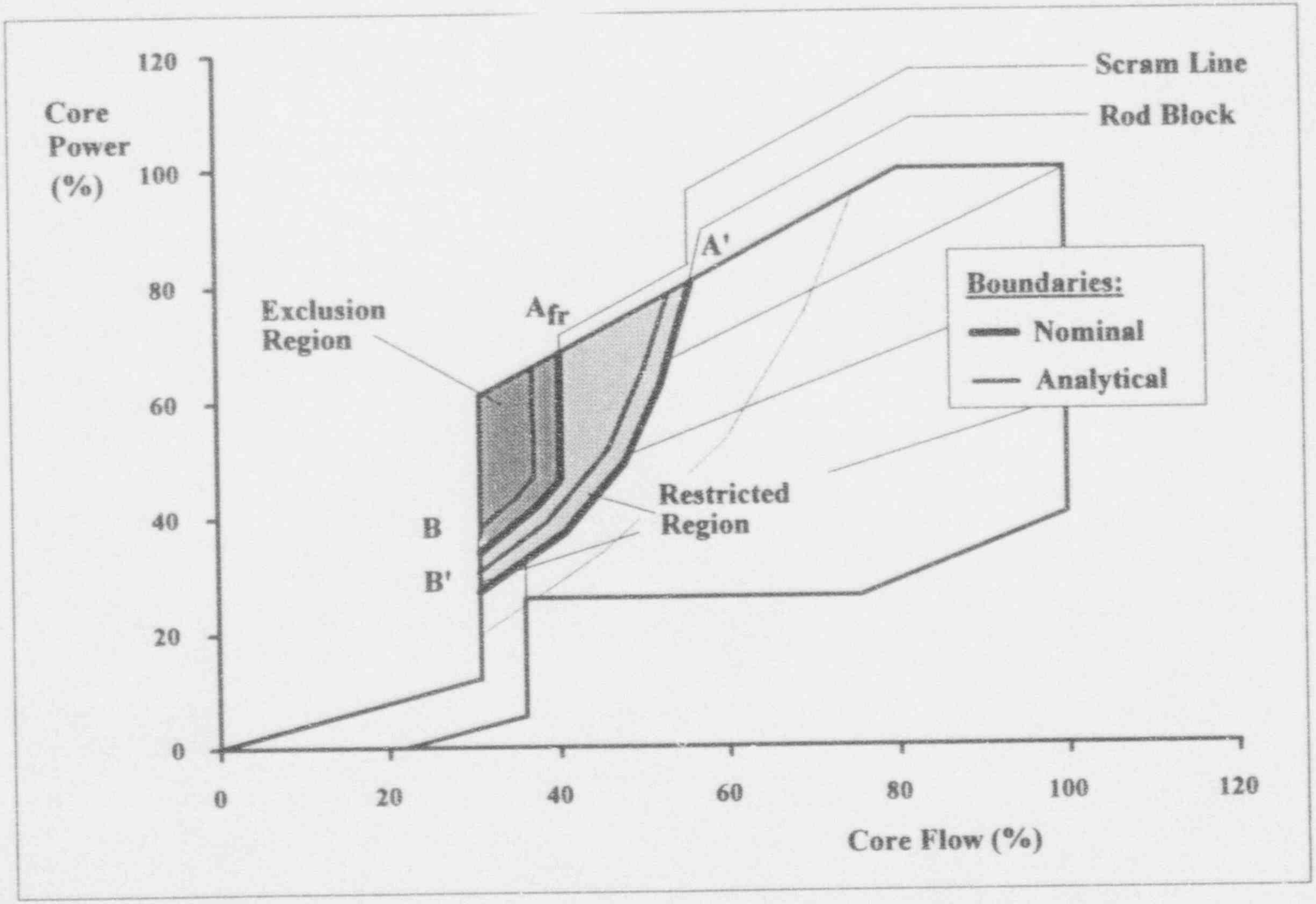


Figure 7-14: Demonstration Plant Validation State Points

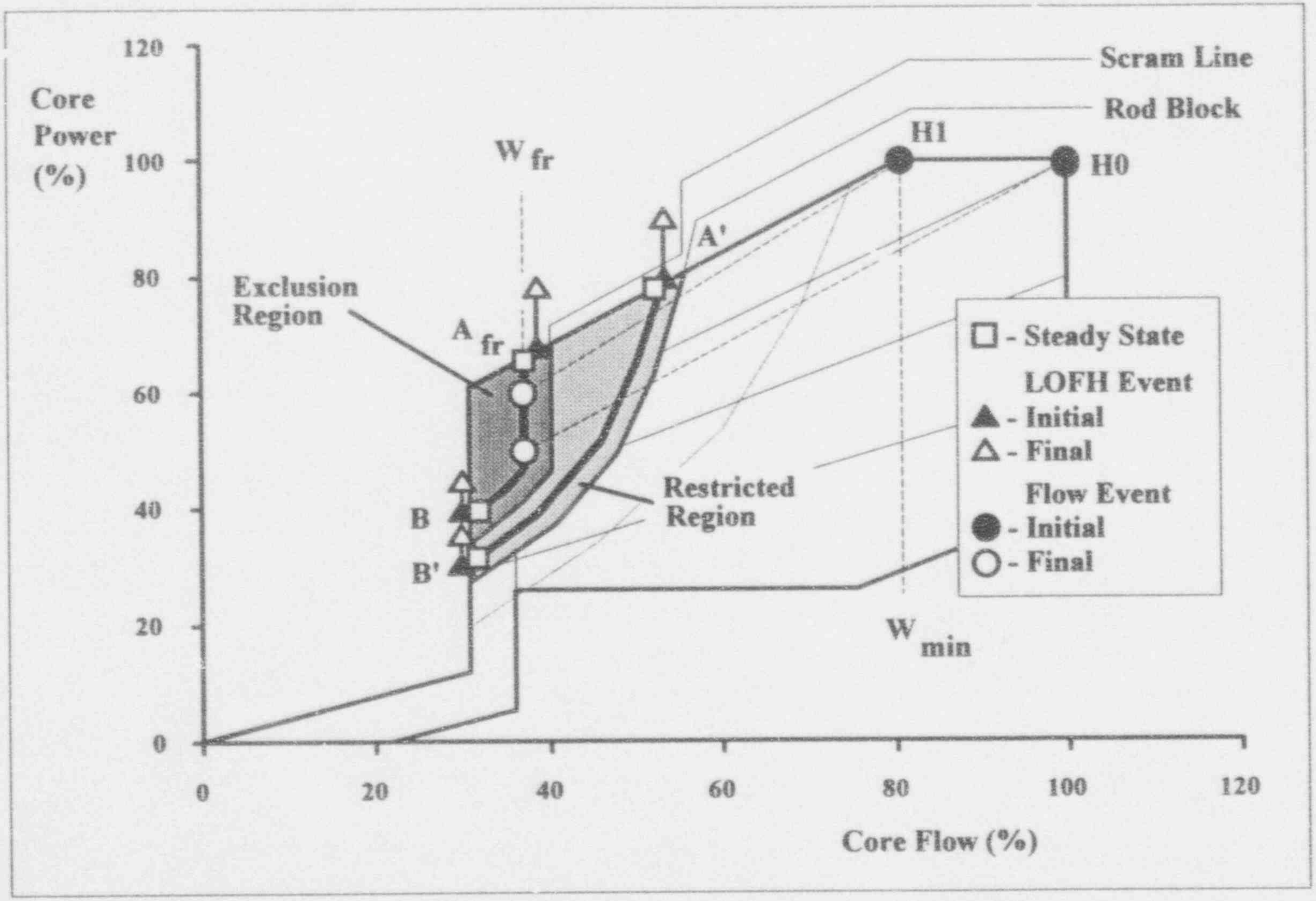


Figure 7-15: Demonstration Plant Validation Result Summary

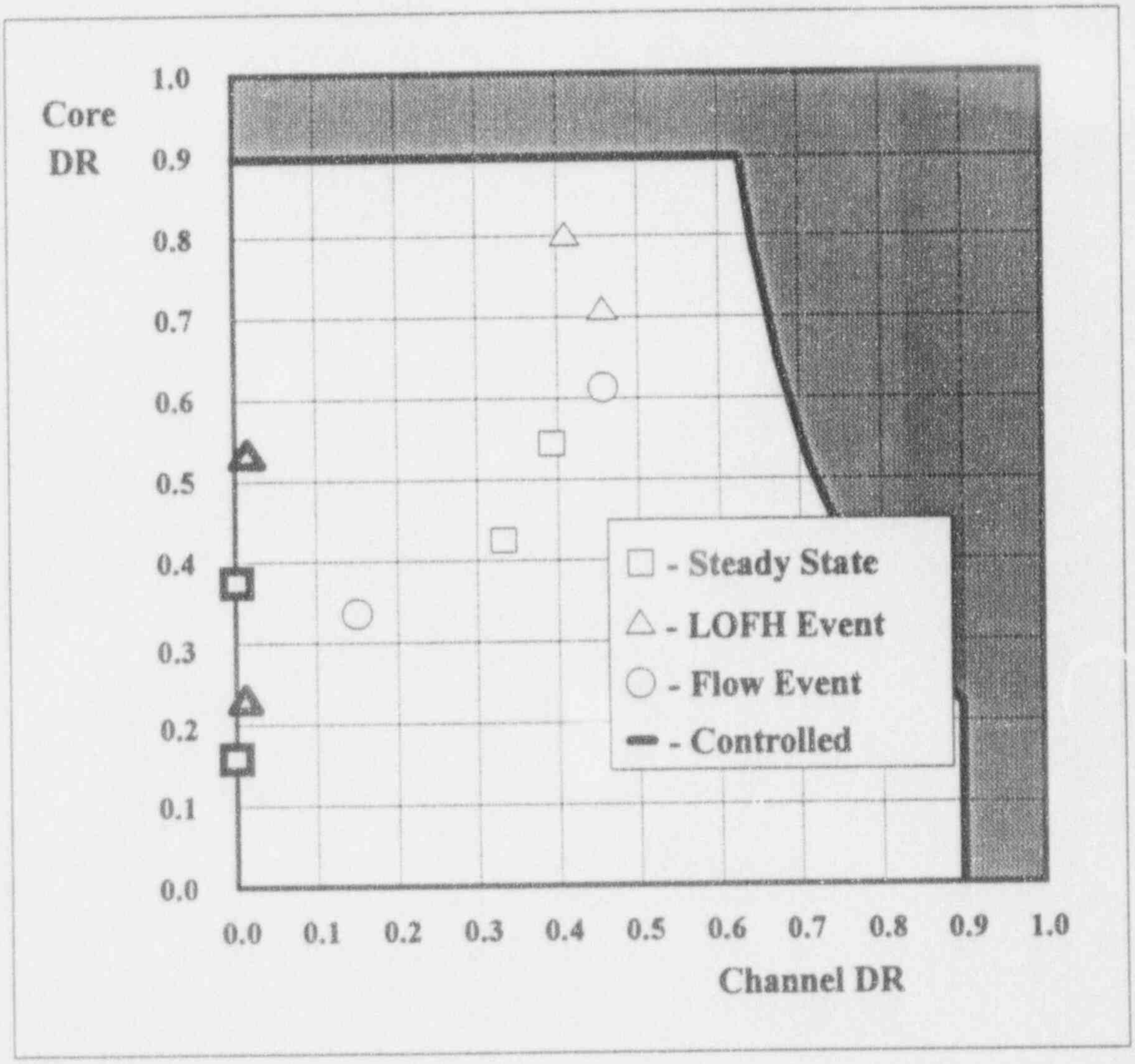


Figure 7-16: Initial Validation Matrix State Point Conditions

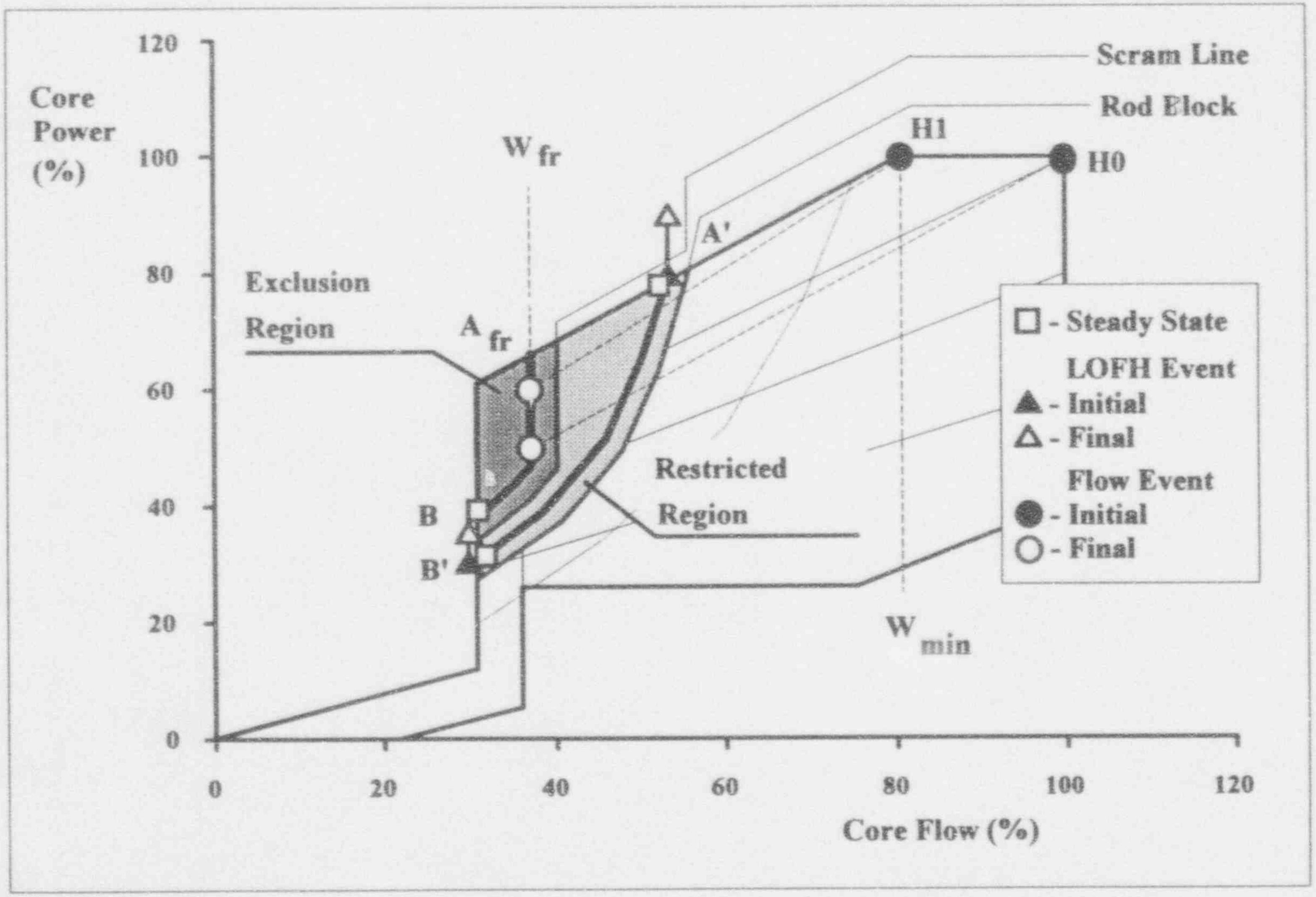
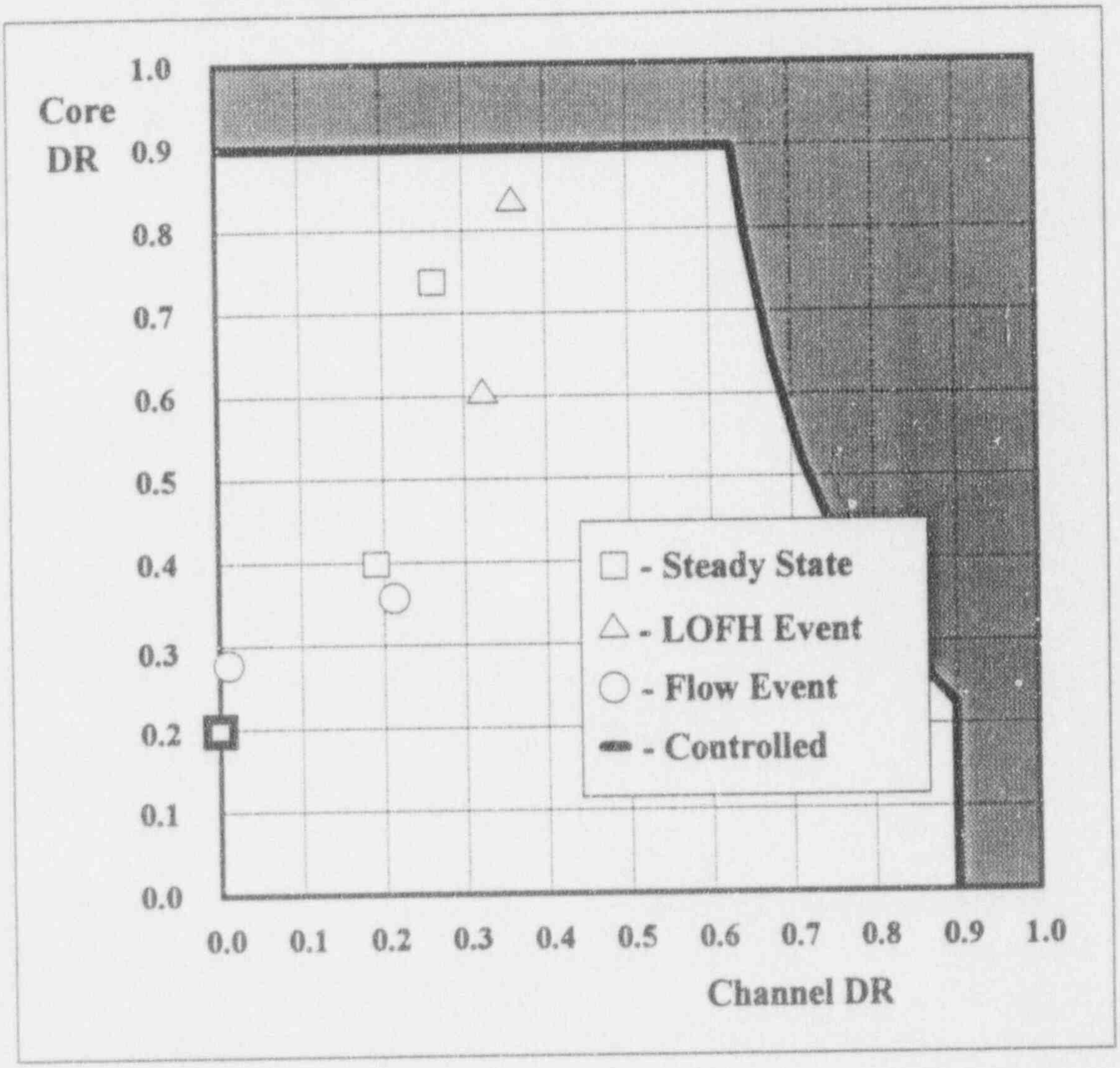


Figure 7-17: Initial Validation Matrix Demonstration Summary



8. RELOAD REVIEW PROCESS

The reload review process for Enhanced Option I-A establishes the continued applicability of existing region boundaries for each new fuel cycle. All changes to reactor, core, and fuel designs are assessed to establish the scope of any stability reload analysis. The reload review process is generic and is applicable to all plant-specific applications of the Enhanced Option I-A stability solution.

8.1 Process Overview

The primary objectives of the reload review process, consistent with the objectives of the initial application process, are:

1. Enhanced Option I-A can be implemented in all domestic BWR designs, by any BWR fuel vendor, and for all fuel cycle designs. The process should allow fuel vendors or utilities to use their own qualified best-estimate stability codes.

2. The process should be generic, prescriptive and simple in order to reduce the probability for errors and ensure adherence to approved analysis procedures.

3. The scope of the reload analysis should depend on the extent to which reload-related changes affect core stability performance.

4. The process should consider reasonably-limiting steady-state and transient conditions to establish an appropriate degree of conservatism.

In addition to the concepts associated with the initial application process, a Reload Validation Matrix (RVM) is employed in the reload review process. The RVM is used to validate the existing region boundaries of each specific plant application of Enhanced Option I-A. The RVM is a subset of the initial validation matrix (IVM), where IVM state points identified as exhibiting non-limiting stability performance during the initial application process are excluded.

8.1.1 Process Elements

The reload review process is illustrated in the flow diagram of Figure 8-1. All changes introduced in the Current Cycle (CC) design relative to the Reference Cycle (RC) design are assessed to determine their impact on the stability region boundaries. The reload review process either confirms the continued applicability of the existing region boundaries or results in new region boundaries. Three types of stability related design changes are evaluated.

First, changes in reactor system design sufficient to invalidate the Standard Cycle (SC) design require a full-scope initial application reanalysis. This reanalysis consists of generating new region boundaries using both FABLE analysis of the new SC design and a best-estimate stability code analysis of the new RC design. A new IVM region boundary validation analysis is also required. Less significant changes in reactor system design that are sufficient to invalidate the RC design require a new region boundary generation analysis based on a revised RC design and an IVM validation analysis. These analysis elements are a subset of those shown in Figure 7-1 since the SC design remains valid and therefore FABLE reanalysis is not necessary. Details of these analysis elements are discussed in Section 7 and are summarized in Figure 8-2.

Second, the introduction of a different fuel design sufficient to invalidate the RC design requires a new region boundary generation analysis based on a revised RC design and an IVM validation analysis. The required analysis elements are a subset of those shown in Figure 7-1 since the FABLE analysis is not required, and are summarized in Figure 8-2. Details of these analysis elements are discussed in Section 7.

Finally, the introduction of a change in fuel cycle design that may have a significant effect on core stability performance requires a RVM analysis to validate existing region boundaries. The elements of a RVM analysis are summarized in Figure 8-3.

If no significant changes in reactor, fuel or cycle design are introduced, reload analysis is not required and the existing region boundaries are applicable for the CC design.

The region boundary reload validation analysis of a CC design is performed using a qualified best-estimate stability code, at state conditions specified by the RVM, to confirm the validity of the region boundaries. The decay ratio results, which quantify the susceptability to core wide and regional mode reactor instability, are compared to the boundary validation stability criterion, which defines the instability threshold for applications of the best-estimate stability code to Enhanced Option I-A. The validation analysis can be performed by any best-estimate stability code qualified for Enhanced Option I-A.

8.1.2 Reload Time-Line

The plant-specific application of Enhanced Option I-A requires a full-scope initial application analysis for the first cycle utilizing the Enhanced Option I-A stability solution. The scope of analysis required for subsequent cycles depends on the outcome of the reload review process. Figure 8-4 illustrates a reload time-line as a function of successive fuel cycles. This example is not rigorous and is only presented to outline the reload review process.

The FABLE SC baseline decay ratio is established for the initial cycle (dark-shaded area). A best-estimate stability code is used to establish the region boundaries by adding a RC-to-SC bias correction factor (light-shaded area) to the SC baseline. The new region boundaries are validated by applying the IVM to the initial cycle actual core design (i.e., CC design). All subsequent cycles require a RVM region boundary validation analysis unless no significant changes are introduced into the CC designs relative to the RC design. If a reactor system design change or a new fuel design invalidating the RC design is introduced, a new RC design and new region boundaries are required; the FABLE SC decay ratio baseline is not changed, however. An IVM region boundary validation analysis is again required for the new region boundaries. RVM region boundary validation analysis is performed for subsequent cycles as necessary. Finally, for reactor system design changes invalidating the SC design, a full scope initial application is performed, including a FABLE base-line analysis of the modified SC design, a new RC design and subsequent analysis to generate new region boundaries. IVM region boundary validation analysis is also required. The IVM analysis is performed for the CC design that incorporates the modified reactor design. RVM region boundary validation analysis is performed for subsequent cycles as necessary.

8.2 Reload Review Checklist

The nominal region boundaries for each plant are based on the plant-specific RC design. Since the CC designs, representing actual reload designs, are expected to vary from cycle to cycle, criteria are established to assess the associated variations in stability performance relative to the RC design. The scope of the reload analysis depends on the degree to which changes introduced in the reload cycle CC design relative to the RC design affect core stability performance.

8.2.1 Reactor Design Modifications

Reactor design modifications are fuel-independent. Those modifications that may affect core stability performance are evaluated against screening criteria to confirm the continued applicability of the SC and RC designs. Since most reactor design modifications are compatible with existing system design bases, reactor modifications invalidating the SC and RC designs and requiring new region boundaries are unlikely.

When necessary, the continued applicability of the existing region boundaries is assessed by reanalyzing the RC design, incorporating the changes associated with the new reactor configuration, and comparing the results to the current RC design analysis. For the comparison to be meaningful, the reanalysis of the modified RC design must follow the process outlined in Section 7. The analysis of the modified RC is performed at the predetermined state points along the 100% flow-control line as illustrated in Figure 8-5 and summarized in Table 7-4, starting from an end-of-equilibrium-cycle (EOEC) full-power Haling depletion.

Both the core and channel decay ratios of the modified RC design, DR_{RC}^{Mod}, are compared to the existing RC design decay ratio data, DR_{RC}. The differences between these two sets of decay ratio values provide a direct measure of the effect of the reactor system design changes on core stability performance.

For major reactor modifications having significant effects on the stability performance of the reactor, a reanalysis of the FABLE SC baseline, with the reactor design modifications incorporated, is appropriate. If the criterion:

$$DR_{RC}^{Mod} \leq DR_{RC} + 0.3 \tag{8-1}$$

for either core or hot channel decay ratio is not met, a FABLE baseline reanalysis of the modified SC design, including the reactor design change, is required. The 0.3 factor in Equation 8-1 represents a change in decay ratio that is selected to identify significant changes in reactor stability performance which result from major reactor system modifications. The range of the DR_{RC} values used in Equation 8-1 is limited to the range used in establishing the stability region boundaries. The revised SC decay ratio baseline can be established using methods equivalent to the FABLE methodology, if they are appropriately qualified.

If the criterion of Equation 8-1 is met, the FABLE SC baseline remains applicable and a second criterion is then applied to establish the continued applicability of the existing RC design. Since the decay ratio of the RC design, used in establishing the existing region boundaries, is given by $DR_{RC} + \Delta DR_{Add}$ (see Section 7), the second criterion is:

$$DR_{RC}^{Mod} \leq DR_{RC} + \Delta DR_{Add} , \tag{8-2}$$

where the decay ratio is evaluated at the commonly analyzed state points and for both core and hot channel decay ratios. If the criterion of Equation 8-2 is not met, the modified reactor design invalidates the RC design and the existing region boundaries.

A new RC design that incorporates the modified reactor design and new region boundaries which are based on the redefined F_{Bias} function (illustrated in Figure 8-2) is then required. The existing best-estimate DR_{SC} analysis is unchanged since the SC design remains the same. A FABLE SC design evaluation is not necessary since the criterion of Equation 8-1 is satisfied.

If the criterion of Equation 8-2 is met, the existing RC design and stability region boundaries remain applicable and a third criterion is then applied to establish whether the reactor design modification has any negative impact on stability performance. This is accomplished by a direct comparison of the modified and existing RC designs at all analyzed state points and for both core and hot channel decay ratios:

$$DR_{RC}^{Mod} \leq DR_{RC}. \tag{8-3}$$

If the criterion of Equation 8-3 is not met, indicating worse stability performance due to the reactor design change, a RVM analysis is required. In the unlikely situation where the RVM analysis fails to validate the existing region boundaries, a revised RC design, new region boundaries and IVM analysis are required.

If the criterion of Equation 8-3 is met, the reactor design modification has no negative effect on stability performance and no further analysis is required. For reactor design modifications that have no effect on core stability performance, the criterion of Equation 8-3 is satisfied by definition, and analysis of the modified RC design is not necessary. The reload review criteria for reactor design modifications and the analysis requirements are summarized in Table 8-1.

8.2.2 New Fuel Designs

New fuel mechanical designs, including fuel channel designs, can affect the stability performance of the reactor core. New mechanical design features that may affect stability performance are evaluated against screening criteria to confirm the continued applicability of the RC design. Since new fuel designs are typically thermal-hydraulically, neutronically, and mechanically compatible with existing fuel designs in the core, it is unlikely that the introduction of new fuel designs will invalidate the existing RC design and therefore require new region boundaries.

Similar to the approach described for assessing the impact of reactor design modifications, the continued applicability of the existing stability region boundaries is determined by reanalyzing the RC design, incorporating the new fuel design changes, and comparing the results to the existing RC design analysis. The analysis of the modified RC design is performed at the predetermined state points along the 100% flow-control line as illustrated in Figure 8-5 and summarized in Table 7-4, starting from a EOEC full-power Haling depletion.

Both the core and hot channel decay ratios of the modified RC design, DR_{RC}^{Mod}, are compared to the existing RC design decay ratio data, DR_{RC}. The differences between these two sets of decay ratio values provide a direct measure of the effect of the new fuel design features on core stability performance. The process of comparing the two RC designs is similar to the process described for reactor system design modifications with the exception of the FABLE procedure. The FABLE SC design evaluation is not necessary for a new fuel design since the SC definition does not include plant-specific fuel design parameters and is therefore unaffected by the change.

Similar to the reactor design modification, the criterion of Equation 8-2 is used to establish the continued applicability of the existing RC design. If the criterion is not met, the new fuel design invalidates the existing RC design, and therefore, the existing region boundaries. A new RC design that incorporates the new fuel design and region boundaries which are based on the re-defined F_{Bias} function (illustrated in Figure 8-2) is required.

If the criterion of Equation 8-2 is met, the existing RC design and stability region boundaries remain applicable and a second criterion is then applied to establish whether the new fuel design has any negative impact on stability performance. As for the reactor design modification, this criterion is given by Equation 8-3. If the criterion of Equation 8-3 is not met, indicating a poorer stability performance due to the new fuel design, a RVM analysis is required. In the unlikely situation where the RVM analysis fails to validate the existing region boundaries, a revised RC design, new region boundaries and IVM analysis are required.

If the criterion of Equation 8-3 is met, the new fuel design has no negative effect on stability performance and no further analysis is required. Similar to reactor design modifications, new fuel design changes that have no effect on core stability performance satisfy the criterion of Equation 8-3 by definition, and analysis of the modified RC design is not necessary. The reload review criteria for the new fuel design and the analysis requirements are summarized in Table 8-1.

8.2.3 Fuel Cycle Changes

Variations in fuel cycle designs, excluding reactor design modifications and fuel mechanical designs changes, are expected to occur every reload. These variations result from changes in fuel cycle design objectives, design optimization based on cycle energy and reactor operation considerations, actual operating history of previous cycles, changes to operating strategies, mixed core evolutions, and the introduction of new operating modes. Some of these variations are minor and have a negligible effect on core stability performance. Others may have a considerable effect on core stability performance and therefore must be identified by the reload review process. To assess the stability performance of reload cycles relative to the RC design, a screening criteria is established to quantify the variations between designs.

The fuel cycle evaluation criteria include an extensive listing of the important indicators of potential changes to core stability performance. Other comparison criteria not explicitly included in the list, which are identified during the reload review process as having a potential impact on stability, should be evaluated. The criteria are based on variations observed between typical reload designs. Meeting the criteria implies cycle-to-cycle core design and operation within the very narrow range of the stability indicators. When the criteria are just exceeded, the variations in the corresponding stability indicators are within a typical range for core reload design and operation and have insignificant effects on stability performance.

If the criteria are not fully met, a RVM analysis as described by Figure 8-3 is required to validate the continued applicability of existing region boundaries. If the RVM analysis fails to qualify the continued applicability of the existing region boundaries, a new RC design incorporating the fuel cycle changes and region boundary generation analysis, and an IVM validation analysis as described by Figure 8-2, are required.

The reload review criteria for fuel cycle design changes and the associated analysis requirements are listed in Table 8-1. A discussion of the fuel design criteria follows:

1. <u>CC Haling radial peaking increase over RC by no more than 5%</u>

   A large increase in radial peaking can have a significant effect on stability performance. The CC design radial peaking is limited to 5% above the radial peaking in the RC design.

2. <u>CC reload batch size change relative to RC by no more than 5% of core size</u>

   A large change in reload batch size can significantly alter core stability performance throughout the operating cycle. This criterion restricts the variations in CC reload batch size, relative to the RC design, to no more than 5% of the total number of assemblies in the core. As an example, for core sizes of 800 and 400 assemblies, the reload batch size is limited to ±40 and ±20 assemblies, respectively, relative to the RC batch size.

3. <u>CC to RC full-power cycle energy change within 10%</u>

   A large change in full-power cycle energy can significantly alter the core stability performance throughout the operating cycle. A CC variation of no more than 10% of full power cycle length relative to the RC design is required. As an example, for a cycle length of 400 effective full power days (EFPD), the change is limited to ±40 EFPD.

4. <u>Last cycle coastdown change relative to RC no more than 10% of full-power cycle energy</u>

   The stability performance of a CC design can be significantly affected by a large change in the cycle energy of the previous cycle. An EOC coastdown that differs from the RC design can result in this energy change. The change in previous cycle energy is limited to 10% of full power cycle energy of the RC design.

5. <u>Last cycle energy shorter than RC by no more than 10%</u>

   This criterion addresses cycle energies for the previous cycle that fall short of the RC design either due to a forced early shutdown or a design-related energy shortfall. As in criterion 4, the change in previous cycle energy is limited to 10% of full power cycle energy of the RC design.

6. <u>No variation in CC mixed batch reload core relative to RC design</u>

   Cores containing mixed batch fuel of differing mechanical designs are susceptible to transitions in stability performance. A reload batch design that results in a new combination of core fuel designs or a reload batch inserted into a mixed core violates this criterion. Mixed reload batch cores, therefore, always require region boundary validation analysis.

7. <u>CC core loading strategy relative to RC unchanged</u>

   A significant change in core loading strategy can affect stability performance. An example of a change in core loading strategy is a move from a conventional scatter loading plan to a control cell core.

8. <u>No new operating modes for CC relative to RC</u>

   New operating modes such as FWHOOS can have an important effect on core stability performance. Introduction or elimination of any new operating modes affecting core thermal-hydraulic or neutronic performance violates this criterion.

9. <u>Other differences between CC and RC do not result in equivalent loss of stability margin</u>

   This criterion addresses changes in any indicators of stability performance not covered by the other criteria that may be identified during the reload review process.

If any of the above criteria are not met, a RVM region boundary reload validation analysis of the CC design is required. Since the RVM analysis is performed for CC designs which are simulated using full-power EOC Haling depletion, meaningful trends in stability performance between reload cycles can be established. Actual operating conditions and cycle design conditions differing from EOC Haling conditions may yield different stability performance. These conditions, however, are addressed by the choice of analysis conditions for the RVM and the overall robustness of the stability protection afforded by Enhanced Option I-A. Appropriate choices for plant-specific RC design parameters will minimize the need for future region boundary generation or reload validation analyses, since the effect of anticipated changes in future fuel designs on core stability performance is small.

8.3 Validation of Existing Region Boundaries

This section describes the analysis required to validate existing region boundaries. The validation process is based on RVM analysis and is summarized in Figure 8-3.

8.3.1 Reload Validation Approach

Validation analysis of existing region boundaries is required if certain screening criteria are not met (Table 8-1). The validation analysis is performed with a qualified best-estimate stability code that meets the requirements of Section 7.4. The analysis is performed at reasonably-limiting conditions to demonstrate the continued applicability of existing region boundaries. The RVM analysis addresses setpoint uncertainties similar to the initial region boundary validation (IVM) analysis, since it is performed at the analytical region boundaries.

8.3.2 State Points Selection

The RVM defines the state points at which the CC is analyzed to validate existing region boundaries. The RVM is a subset of the IVM used in a plant-specific initial application, excluding the state points demonstrated to exhibit non-limiting stability performance for the specific plant.

8.3.2.1 Non-Limiting State Points

Based on the DVM analysis described in Section 7.3, all state conditions analyzed with stability controls in place on the Exclusion Region analytical boundary exhibit non-limiting stability performance. The IVM analysis retains one steady-state point with stability controls in place (Point B, Figure 7-16) to validate the effectiveness of the Enhanced Option I-A stability controls. Unless demonstrated otherwise in the plant-specific IVM analysis, Point B with controls in place is assumed to be bounded by Point B' with no controls in place and is excluded from the RVM. In addition, only the more limiting of the two state points used for the intermediate flow reduction and LOFH events in the IVM analysis are selected for the RVM. Table 8-2 describes the state points excluded from the IVM.

8.3.2.2 Reload Validation Matrix

The RVM state points consist of the IVM state points excluding the non-limiting state points identified in Table 8-2. The RVM state points are shown on the power-flow map of Figure 8-6 and are described in Table 8-3.

The initial and final analysis conditions for the RVM state points constitute a generic, prescriptive and reasonably-limiting analytical path. The RVM is applied to the CC design at the analytical region boundaries (see Section 7.3 for a discussion of setpoint uncertainty). A Haling cycle depletion to end-of-full-power at Point H1 is performed for the analysis at Point A'. A Haling cycle depletion to end-of-full-power at Point H0 is performed for the analysis at Point B'. The flow event is initiated from the more limiting of Points H1 or H0 and the LOFH event is performed at the more limiting of Points A' or B' as established by the IVM analysis results. FWHOOS should be considered for the Haling depletion, if applicable.

The initial and final conditions for the different state points, including cycle depletion method, Xenon conditions, feedwater temperature for the LOFH and flow events and control rod positions are the same as for the IVM analysis described in Section 7.3. The initial conditions for the RVM analysis are summarized in Table 8-4 and the final conditions in Table 8-5.

8.3.3 Validation of Region Boundaries

The results of the RVM state point analysis are decay ratio values which quantify the susceptability to fundamental (core wide) and first order azimuthal harmonic (regional) modes of reactor instability. These values are compared against the corresponding best-estimate code boundary validation stability criterion. If all analysis results satisfy the criterion, then the RVM analysis validates the existing region boundaries.

Table 8-1: Reload Review Criteria and Analysis Requirements

Design Change	Reload Review Criteria	Analysis Required if Criteria Not Met
A. Reactor	1. Reactor design modification does not invalidate SC design $DR_{RC}^{Mod} \leq DR_{RC} + 0.3$	• FABLE procedure • New RC design • New region boundaries • IVM
	2. Reactor design modification does not invalidate RC design $DR_{RC}^{Mod} \leq DR_{RC} + \Delta DR_{Add}$	• New RC design • New region boundaries • IVM
	3. Reactor design modification has no negative effect on stability $DR_{RC}^{Mod} \leq DR_{RC}$	• RVM If RVM fails: • New RC design • New region boundaries • IVM
B. New Fuel	1. New fuel design does not invalidate RC $DR_{RC}^{Mod} \leq DR_{RC} + \Delta DR_{Add}$	• New RC design • New region boundaries • IVM
	2. New fuel design has no negative effect on stability $DR_{RC}^{Mod} \leq DR_{RC}$	• RVM If RVM fails: • New RC design • New region boundaries • IVM
C. Fuel Cycle	1. CC Haling radial peaking increase over RC by no more than 5% 2. CC reload batch size change relative to RC no more than 5% of core size 3. CC to RC full power cycle energy change within 10% 4. Last cycle coastdown change relative to RC no more than 10% of full power cycle energy 5. Last cycle energy shorter than RC by no more than 10% 6. No variations in CC mixed batch reload core relative to RC 7. CC core loading strategy relative to RC unchanged 8. No new operating mode for CC relative to RC 9. Other difference between CC and RC do not result in equivalent loss of stability margin	• RVM If RVM fails: • New RC design • New region boundaries • IVM

Table 8-2: IVM State Points Not Included in RVM

Category	State Point	Justification
Steady-State	B	Control required, bounded by B'.
Flow Events	H0 or H1	IVM established flow-control line impact. Only limiting event for RVM.
LOFH Events	A' or B'	IVM Established state point impact. Only limiting LOFH for RVM.

Table 8-3: Reload Validation Matrix State Points

Category	State Point
Steady-State	A' and B'.
Flow Events	Limiting event based on IVM: From H0 or H1 to Exclusion Region analytical boundary.
LOFH Events	Limiting event based on IVM: A' or B'.

Table 8-4: Reload Validation Initial State Conditions

Category	State Point	Initial State Conditions
All	All	EOC Haling. FWHOOS if applicable.
Steady-State	A'	Based on H1 depletion. Equilibrium Xenon. Deep control rods at 00/08, with one rod per core quadrant adjusted for criticality, to compensate for lower Xenon.
	B'	Based on HO depletion. Xenon-free. Radially uniform control rods: Deep at 00/08 with one rod per core quadrant adjusted for criticality, shallow to control thermal limits.
Flow Events	H0 or H1	EOC Haling to H0 or H1.
LOFH Events	A' or B'	Same as A' or B' Steady-State.

Table 8-5: Reload Validation Final State Conditions

Category	Initial State Point	Final State Conditions
Steady-State	A' B'	Same as initial conditions.
Flow Events	H1 or H0	W_{fr}. Initial T_{fw}. Initial Xenon.
LOFH Events	A' or B'	T_{fw} - 60°F equivalent rated. Initial Xenon.

Figure 8-1: Reload Review Process

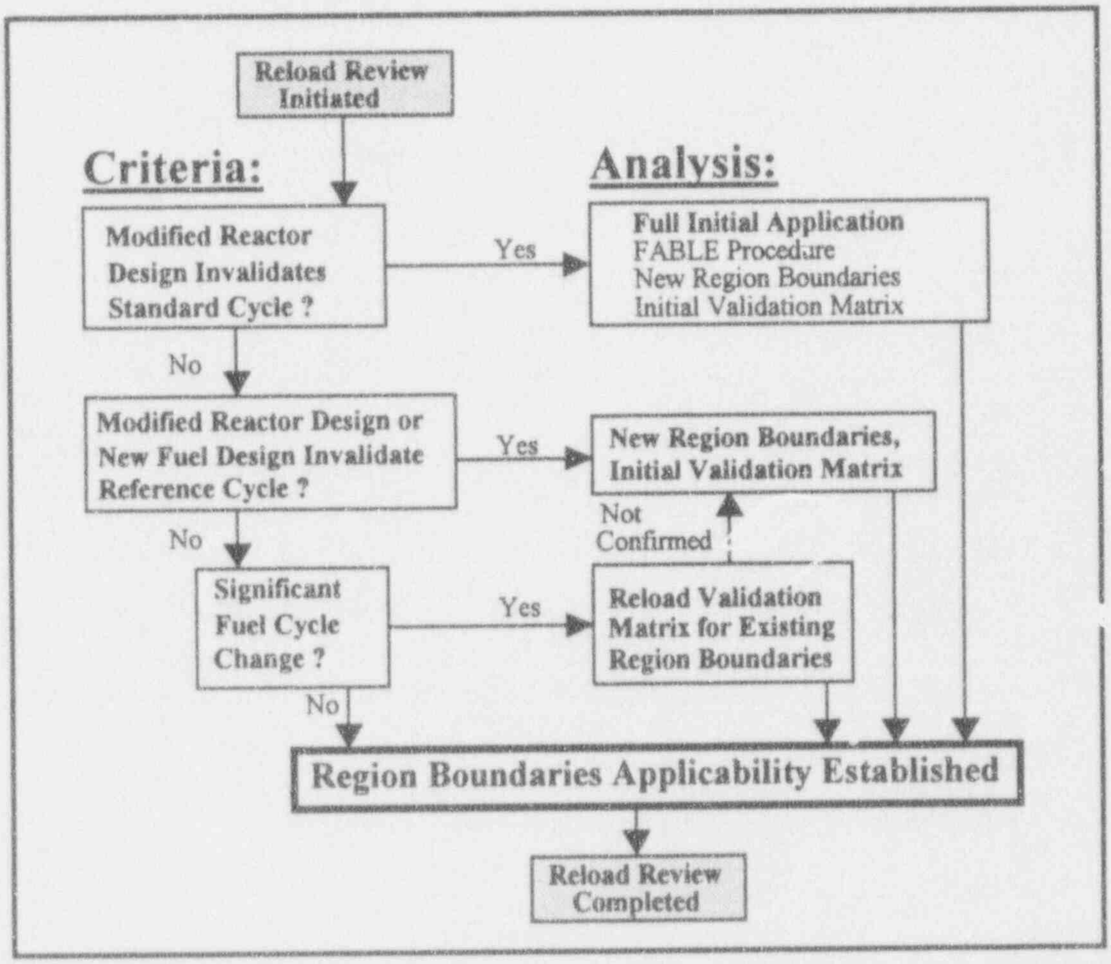


Figure 8-2: New Reference Cycle Analysis Elements

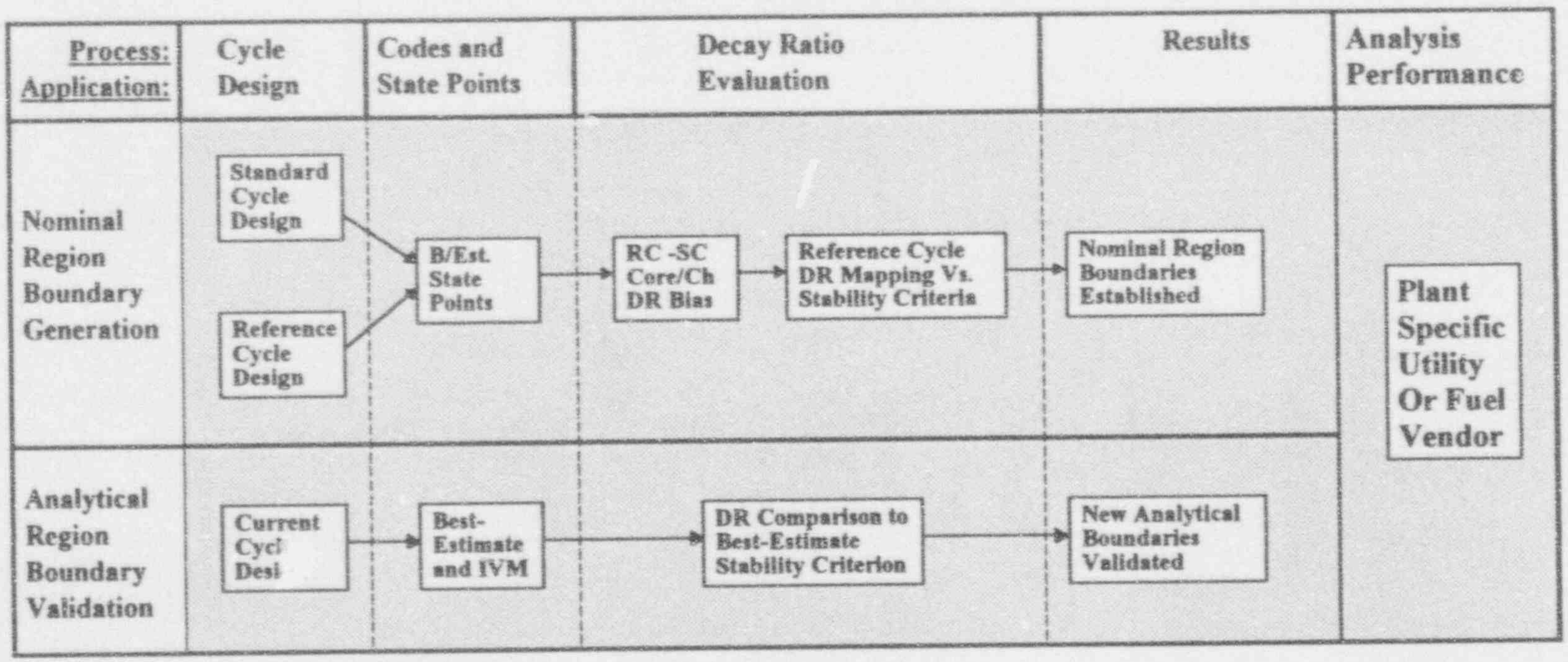


Figure 8-3: Reload Validation Analysis Elements

Process: / Application:	Cycle Design	Codes and State Points	Decay Ratio Evaluation	Results	Analysis Performance
Analytical Region Boundary Reload Validation	Current Cycle Design →	Best-Estimate and RVM →	DR Comparison to Best-Estimate Stability Criterion →	Existing Analytical Boundaries Validated	Plant-Specific Utility or Fuel Vendor

Figure 8-4: Reload Time-Line

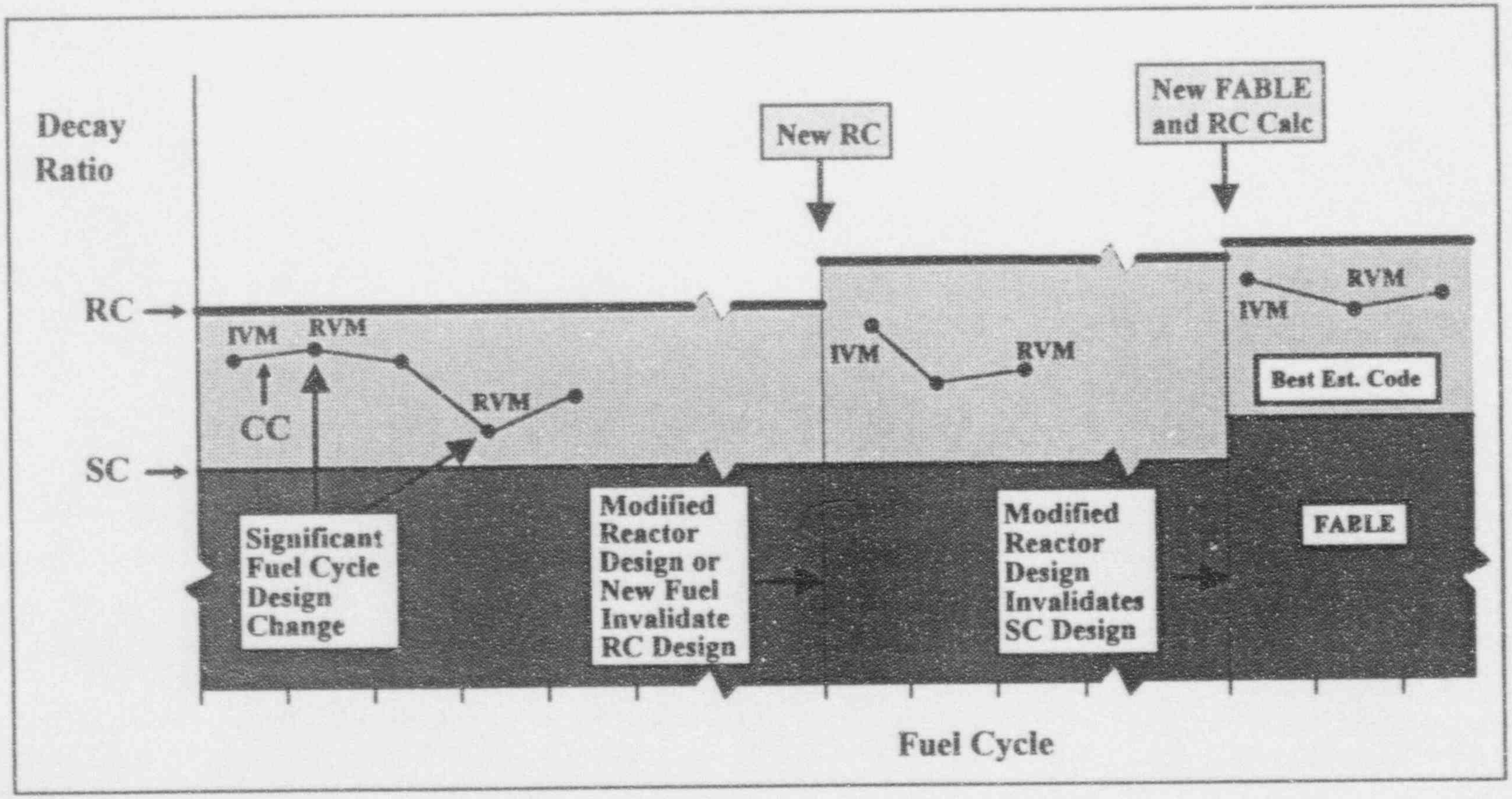


Figure 8-5: Modified Reference Cycle Stability Performance

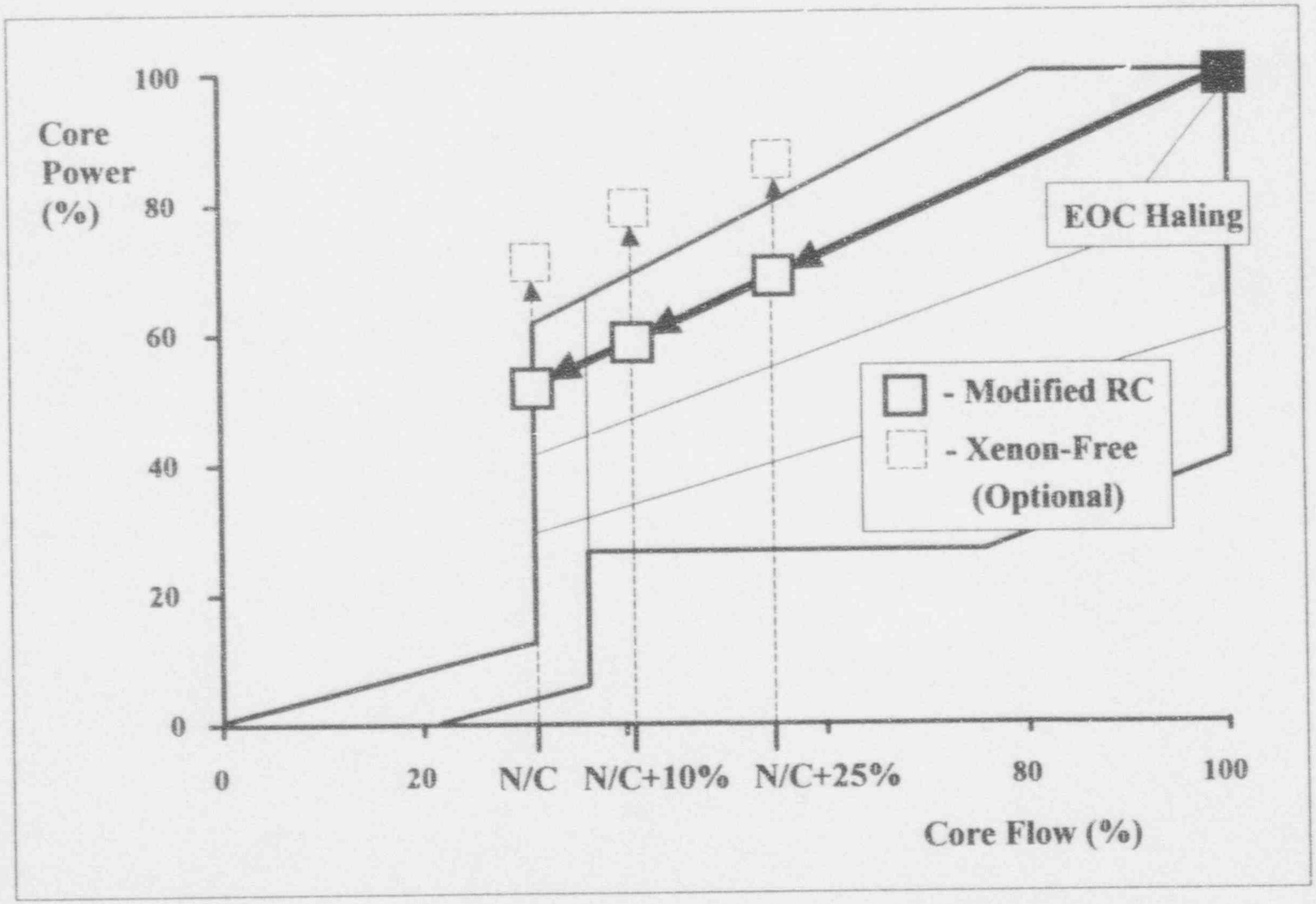
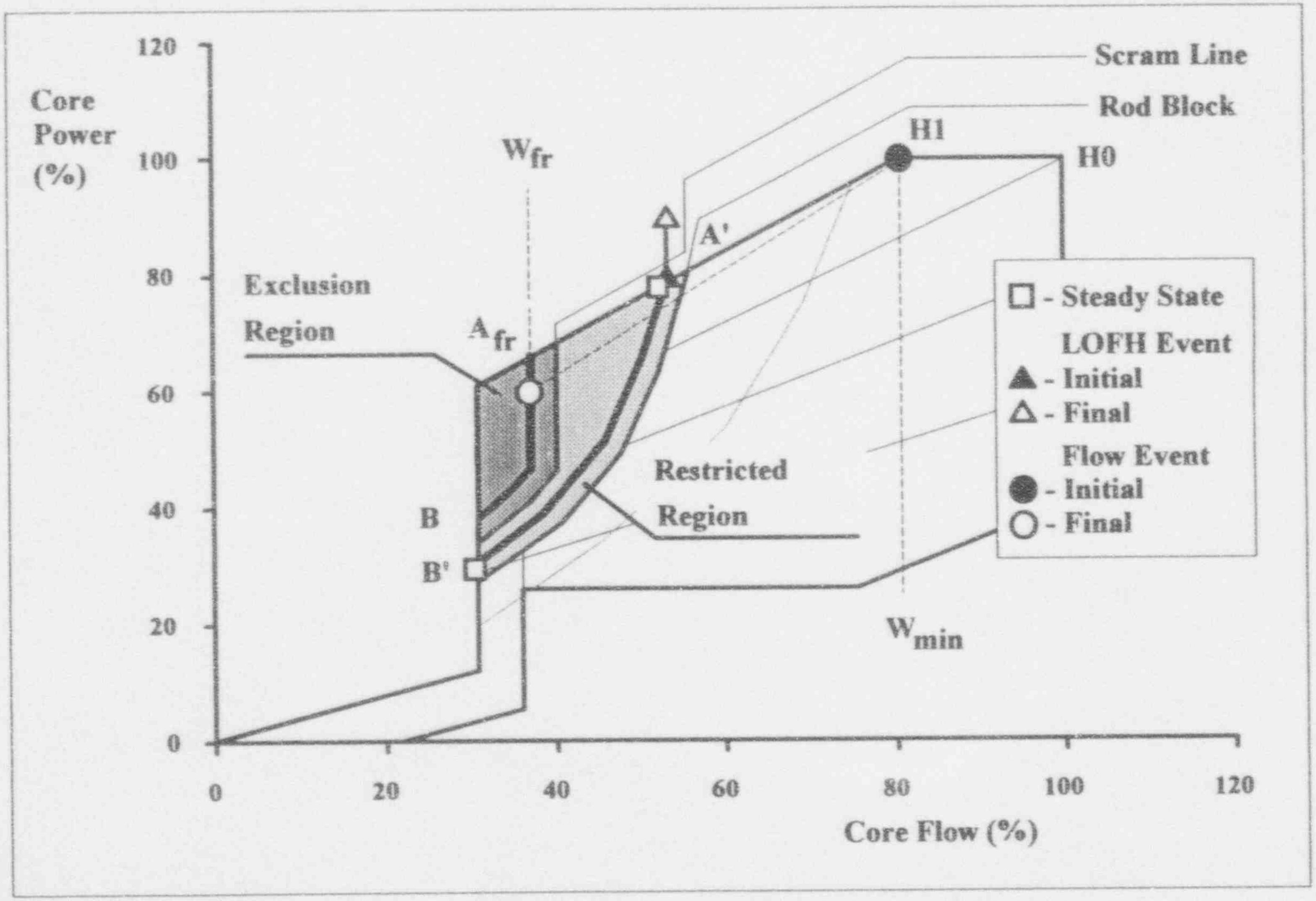


Figure 8-6: Example Reload Validation Matrix State Point Conditions



9. REACTOR STABILITY CONTROL

9.1 Stability Control Introduction

A simple, effective, power distribution control - limiting the core average bulk coolant saturation elevation above a predetermined axial plane - has been developed that provides the means to reliably influence the stability of a reactor. The stability control is an integral part of the licensing methodology for Enhanced Option I-A, and is designed for easy use by operators during reactor maneuvering. These features of the stability control are necessary to maintain consistency with the stability solution design philosophy.

The relative insensitivity of the stability control to variations in all major state parameters affecting reactor stability is demonstrated. This assures that core stability can be directly influenced by the control without concern for variations in these state parameters. Implementation of this stability control generates significant axial heterogeneities within the core. Proper analysis of these conditions requires special consideration of reactor kinetics and core thermal-hydraulics models. The control is designed for use during quasi-steady-state conditions of controlled reactor maneuvering. Changes in reactor state conditions resulting from transients can defeat the stability control and are therefore addressed by other features of Enhanced Option I-A.

This stability control, its phenomenological basis, sensitivity to relevant reactor parameters, and method of implementation are described in this section.

9.2 Stability Control Formulation

9.2.1 Background

The BWR core consists of a large number of fuel assemblies, exhibiting radially independent hydraulic behavior, that are coupled at their inlet and exit via the reactor upper and lower plenums. The presence of boiling within fuel channels makes them susceptible to reactor coolant density-wave instabilities.

Pressure perturbations at the core inlet cause flow disturbances that travel up the fuel channels as time-varying coolant density waves. These waves result in local deviations from the steady-state axial pressure drop distribution. The local pressure drop in a fuel assembly is highly dependent on void fraction. Since the coolant voiding increases axially with greater core elevation, the highest void fraction is found at the channel outlet.

The effect of density waves on total channel pressure drop is therefore effectively delayed in time - the void sweeping time - until the perturbation is felt at the channel exit. When the channel pressure drop time delay (phase lag) nears 180° out of phase with the channel inlet flow variations, the fuel assembly can become thermal-hydraulically unstable. Thus, the thermal-hydraulic stability margin of a fuel channel is dependent on the phase lag caused by void sweeping time, and the gain which is dependent on the channel void distribution.

An additional complexity is introduced in BWR stability because of the reactor power dependency on coolant density. Local void reactivity (ρ_v) responds to the time-varying density wave described above. The reactivity change affects local neutron flux (ϕ_{dV}), and is manifested after a time delay (fuel thermal time constant) as changes in fuel cladding surface heat flux and ultimately in local coolant voiding. This mechanism can also provide positive feedback to density wave oscillations. The neutronic feedback gain is dependent on the fuel thermal time constant and on the local void fraction.

For point kinetics models, void reactivity is related to void fraction and local neutron flux by:

$$\Delta\rho_V \propto \int_V \phi_{dV}^2 \times \left(\frac{d\rho}{d\nu}\right) \times \Delta\nu \times dV \qquad (9\text{-}1)$$

The flux-squared dependency of ρ_v reflects the feedback relationship of the relatively high power fuel bundles on core stability, which increases non-linearly with power.

The two feedback mechanisms, thermal-hydraulic and neutronic, are coupled in a BWR core and can generate power oscillations in both core flow and thermal power. These oscillations can affect margins to fuel thermal safety limits. In addition, reactor instabilities can occur even when neither feedback mechanism alone is sufficient to generate reactor instability. The feedback mechanisms described above are illustrated in Figure 9-1.

9.2.2 Parameters Affecting Stability

Predicting and controlling reactor stability in an operational setting, where the fuel and core designs are fixed, is difficult. Commonly used operational parameters for measuring core thermal-hydraulic and neutronic behavior do not provide sufficient insight into the basic mechanics of reactor stability. Thus, a more fundamental approach is taken to permit development of a functional stability control.

A broad assessment of the stability issue provides the impetus for the development of such a stability control. Coupled neutronic/thermal-hydraulic instability is a phenomenon only found in boiling water reactors. This is because only BWRs have significant bulk coolant boiling in the core during normal reactor operations. This can be restated as the following:

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

$$DR_{core} = f\{\text{void distribution}\}, \quad (9\text{-}2)$$

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. This relationship is illustrated in Figure 9-2.

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

$$DR_{core} = f\{AP_i, RP_j, P, W, DHS\} \tag{9-3}$$

where:

AP_i = axial power shape,
RP_j = radial power shape,
P = core thermal power,
W = core flow, and
DHS = core inlet subcooling.

After differentiating:

$$d(DR_{core}) = \left(\frac{\partial DR_{core}}{\partial AP_i}\right)d(AP_i) + \left(\frac{\partial DR_{core}}{\partial RP_j}\right)d(RP_j) + \left(\frac{\partial DR_{core}}{\partial P}\right)d(P) + \left(\frac{\partial DR_{core}}{\partial W}\right)d(W) + \left(\frac{\partial DR_{core}}{\partial DHS}\right)d(DHS) \tag{9-4}$$

The usefulness of Equation 9-4 is limited, however. First, although the behavior of all terms except:

$$\left(\frac{\partial DR_{core}}{\partial AP_i}\right)$$

is generally understood, it is difficult to establish the partial derivatives for reasonable changes in the variables. This is due to the interdependency of these five parameters in an operational environment. During reactor startup, for example, the core radial power shape is constantly changing in response to control rod withdrawals executed to increase reactor power. This interdependence must be recognized in the development of a successful stability control.

Second, no unique relationship between AP_i and DR_{core} has been demonstrated. Analysis demonstrates that sole examination of the axial power shape cannot provide effective and reliable control of reactor stability. For example, consider the two cases depicted in Figures 9-3 and 9-4.

These figures show two hypothetical reactor states that differ only in their axial power shapes and core inlet subcooling (RP_j, P, and W remain constant). When core inlet subcooling (DHS) is low and bulk coolant saturates at elevation 'a', $DR_{Shape\ 1} < DR_{Shape\ 2}$. However, when core inlet subcooling is high and bulk coolant saturates at elevation 'b', then $DR_{Shape\ 1} > DR_{Shape\ 2}$. This example illustrates the difficulty of determining relative reactor stability margins based on changes in axial power shape alone.

The development of a simple, reliable stability control based on a direct independent assessment of each parameter in Equation 9-3 is therefore not feasible. The variables are either interdependent, or their influence on DR_{core} cannot be resolved. To formulate the stability control, the observation that the voided region of the core determines reactor stability must be revisited.

9.2.3 Axial Power Shape Effects

To simplify the discussion, a radial collapse of the core, as depicted in Figure 9-5, will be initially assumed.

For an average fuel channel, Equation 9-3 is simplified to:

$$DR_{core} = f\{AP_i, P, W, DHS\} \qquad (9\text{-}5)$$

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.

These separated regions can be directly related to the feedback mechanisms driving reactor instability, described in Section 9.2.1. Figure 9-6 illustrates the relationship between the separated regions of the core, and the stability feedback mechanisms.

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 (Equation 9-1), $AP_i^{2\phi}$ can have a significant impact on stability margin if axial flux peaks high in the voided region of the core. These concepts, as illustrated in Figure 9-6, motivate the following:

<u>*Critical Observation 2*</u>: ***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.***

$$DR_{core} = f\left\{\frac{L_{2\phi}}{L_{1\phi} + L_{2\phi}}, AP_i^{2\phi}\right\}, \tag{9-6}$$

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:

$$Z_{bb} = f\{AP_i, P, W, DHS\} \tag{9-7}$$

The issue of how AP_i is related to DR_{core} is now resolved. 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:

$$\rho_v = f\{AP_i, i > Z_{bb}\} \tag{9-8}$$

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:

$$\sum_{i=1}^{Z_{bb}} AP_i = C\frac{W \times DHS}{P} \tag{9-9}$$

where C is a constant (see Section 9.3 for derivation of this equation).

Variations in each parameter of Equation 9-9 result in an appropriate change in the core average boiling boundary as tabulated in Table 9-1.

9.2.4 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 (Figure 9-5). 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 power, f, required for coolant saturation in an average channel can be written as follows:

$$f = \frac{DHS \times \frac{W}{N}}{\frac{P}{N}}, \tag{9-10}$$

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:

$$f = \frac{DHS \times \overline{w}}{\overline{p}}. \tag{9-11}$$

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

$$f_{ch} = \frac{DHS_{ch} \times w_{ch}}{p_{ch}} \tag{9-12}$$

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

$$\begin{aligned} DHS_{ch} &= DHS, \\ p_{ch} &= RP_j^{ch} \times \overline{p}, \text{ and} \\ w_{ch} &\cong \overline{w}, \end{aligned} \tag{9-13}$$

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:

$$f_{ch} \cong \frac{f}{RP_j^{ch}}. \tag{9-14}$$

As discussed in Section 9.2.3, the axial power shape also 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, to a large extent, power sharing among adjacent fuel assemblies ameliorates these effects. Reducing the average power at the core bottom will limit the length of the hot channel two-phase column length.

The influence of the high power fuel bundles on the stability of the entire reactor core can be disproportionally large, as noted in Section 9.2.1. Therefore, an effective stability control must limit the hot channel decay ratio, DR_{ch}.

9.2.5 Identification of Stability Control

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.

<u>Critical Observation 3</u>: ***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.***

<u>Critical Observation 4</u>: ***When Z_{bb} is sufficiently high, then variations in all parameters that affect stability will produce only second order effects on DR_{core} and may be ignored if existing fuel thermal limits are not exceeded.***

Observation 4 permits the terms of Equation 9-4 to be evaluated at Z_{bb}:

$$\nabla DR_{core}\big|_{Z_{bb}^{high}} \approx 0 \qquad (9\text{-}15)$$

Reactor stability is assured with a high boiling boundary primarily because of the consequences of a short two-phase column on the thermal-hydraulic and neutronic feedback mechanisms. The effect of variations in the two-phase axial power shape cannot render the core unstable at sufficiently high boiling boundaries. The Z_{bb} concept also addresses the interdependence of the important parameters affecting stability. For a constant boiling boundary, a change in one stability parameter forces a compensating change in the others (Equation 9-9).

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. Because of power sharing among fuel bundles, this low average power in the core bottom limits the hot channel two-phase column length and therefore maintains its relative stability. Unusual control rod configurations that support sufficient power sharing among a group of adjacent high power bundles could potentially threaten core stability, even at high core average boiling boundaries. However, this situation, where a highly skewed power shape exists, is not compatible with maintenance of existing fuel thermal limits while operating with the stability control in place.

9.3 Derivation of Stability Control Limit

A stability control limit is only useful operationally if adherence to the limit can be determined using currently defined core parameters, and can be accomplished during necessary reactor maneuvers. The Z_{bb} stability control, which only utilizes core average parameters and obviates the need for radial constraints, will be employed to define the stability limit.

The Z_{bb} stability control is based on consideration of bulk coolant saturation in the active core flow region. The out-of-channel bypass region and the in-channel bypass pathways (e.g., water rods or box) are excluded from the total core flow. These bypass flows are expected to remain subcooled and do not contribute to the two-phase thermal-hydraulic or neutronic feedback mechanisms. If bypass voiding is present, the core average bulk boiling in the active flow region is elevated, thereby shortening the two-phase column length in the active region. It is, therefore, appropriate and conservative to assume no bypass voiding.

Stability controls are required in the Restricted Region (Figure 9-26), and therefore, determination of bulk coolant saturation is only necessary in this region. The total bypass flow in the Restricted Region can vary from a few percent of core flow at natural circulation to approximately 10% of core flow at the high flow corner of the region. To simplify the formulation of the stability control limit, and since variations in total bypass flow are restricted to a tight range, a single bypass value is selected for use in establishing the stability control limit. This value is representative of the maximum bypass flow value in the Restricted Region (i.e., 10% of core flow) to ensure that actual operating conditions with lower total bypass flow will result in a higher core average bulk saturation elevation.

The total bypass flow value selected is conservative for startup operating conditions near the low flow corner of the Restricted Region since the actual total bypass flow is well below 10% of core flow. For operating conditions near the high corner of the Restricted Region, it has been demonstrated that the core and hot channel decay ratios are very low with stability controls in place and, therefore, use of a representative value of the total bypass flow in this region is appropriate. 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:

$$f = F_{af} \frac{W \times DHS}{P} \qquad (9\text{-}16)$$

where:

F_{af} = active core flow fraction for Restricted Region (0.9),

W = core flow rate,

DHS = core inlet subcooling, and

P = total core thermal power,

or following unit conversion:

$$f = \frac{0.293 \times F_{af} \times W \times DHS}{P}, \qquad (9\text{-}17)$$

where:

W in 10^6 lb_m/hr,

DHS in Btu/lb_m, and

P in MW_t.

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:

$$\frac{1}{n}\sum_{i=1}^{n} AP_i = 1.0. \qquad (9\text{-}18)$$

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}):

$$\frac{1}{n}\sum_{i=1}^{Z_{bb}} AP_i \equiv 0.293\frac{F_{af} \times W \times DHS}{P} \qquad (9\text{-}19)$$

The relationship of the core average boiling boundary to all core average parameters that are important to stability, is illustrated in Figure 9-7.

To control the core average boiling boundary during reactor operations, the boiling boundary (Z_{bb}) can 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, $\overline{Z}_{bb}$:

$$Z_{bb} \geq \overline{Z}_{bb}. \qquad (9\text{-}20)$$

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:

$$\frac{1}{n}\sum_{i=1}^{Z_{bb}} AP_i \leq 0.293\frac{F_{af} \times W \times DHS}{P}. \tag{9-21}$$

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:

$$FCBB = \frac{\frac{1}{n}\sum_{i=1}^{Z_{bb}} AP_i}{0.293\frac{F_{af} \times W \times DHS}{P}} \tag{9-22}$$

where $F_{af} = 0.9$. This normalized limit should satisfy the condition:

$$FCBB \leq 1.0. \tag{9-23}$$

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 powerful mechanism for operational control of reactor stability. Its strength is derived from two significant features. First, the control explicitly incorporates all core parameters that have a significant influence on stability. This means that the stability control can be reliably and effectively used by itself, without concern for changes in other parameters. Second, the stability control is readily derived from core average parameters normally available to a reactor operator. The normalized stability control limit, FCBB, can easily be incorporated into core monitoring software for automatic display to reactor operators. These two features permit quick and efficient evaluation of core stability during reactor maneuvering.

For example, the change in Z_{bb}, caused by the repositioning of control rods, is reflected in Figure 9-8. 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 9.2.3.

9.4 Stability Control Analysis

9.4.1 Background

BWR stability analysis is 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 is 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.

The analysis has been performed for a BWR/6 design. Due to the generic nature of the physics and gross reactor systems design of BWRs, similar results are expected for all designs. Sensitivity analysis performed for a BWR/3 design (Appendix E) supports this conclusion.

A typical core power and flow operating map for the analyzed reactor design is shown in Figure 9-9. The shaded area in the figure is representative of the operating domain region susceptible to reactor instabilities without stability control in place. Its shape is consistent with the influence of core power and flow on reactor stability.

The calculational procedure described in Section 7 is used for establishing the boundary of the Exclusion Region. The procedure uses the stability criterion that is shown in Figure 9-10. This stability criterion is used to account for susceptibility to the fundamental and higher order harmonic modes of power oscillations, based on calculated values for core and hot channel decay ratios.

The core decay ratio (DR_{core}) is related to the fundamental or core-wide mode of oscillation. The hot channel decay ratio (DR_{ch}) is used to expand the criterion to include consideration of the higher order modes, or regional modes, of oscillation. The criterion incorporates uncertainties in methods for predicting reactor instabilities based on the core and channel decay ratio parameters. Any reactor state conditions above the stability criterion (shaded area) are considered unstable.

The boiling boundary limit, $\overline{Z}_{bb}$, is established based on analysis performed just outside the Enhanced Option I-A Exclusion Region, as illustrated in Figure 9-9, since entry into that region is prohibited.

9.4.2 Methods

The stability analysis is performed using a qualified 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 the 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.

9.4.3 Scope of Analysis

A demonstration of the stability control concept is accomplished by varying Z_{bb} at a given core power and flow state point near the Exclusion Region boundary. The core average boiling boundary is moved upward in the core using an incremental insertion of shallow (shaping) control rods . A target limit, $\overline{Z}_{bb}$, is 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 is performed at the target $\overline{Z}_{bb}$. The objective of the sensitivity analysis is to demonstrate that, for the target $\overline{Z}_{bb}$, core stability performance is insensitive to variations in these parameters and conditions.

The sensitivity analysis is performed near the Exclusion Region boundary. The analysis considers previously identified parameters that affect core void distribution. Core power and flow define the instability region boundary, and therefore remain constant in this analysis. The parameters for which sensitivity analysis is performed include:

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

In addition, special conditions are 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 is performed, in general, at reasonably limiting conditions. This is 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 are 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.

9.4.4 Target Boiling Boundary Height

Analysis is first performed to characterize and quantify the relationship between stability performance, as calculated by the core decay ratio (DR_{core}), the hot channel decay ratio (DR_{ch}), and the core average boiling boundary (Z_{bb}). The shaping control rods are incrementally and uniformly inserted while maintaining the same reactor state point conditions. This is done to establish a Z_{bb} of 2.0, 3.0 and 4.0 feet, successively.

Figure 9-11 depicts the calculated DR_{core} and DR_{ch} as a function of Z_{bb}. Figure 9-12 provides a comparison of the calculated decay ratio values to the stability criterion.

The results of Figure 9-11 are consistent with the earlier discussion of the effect of Z_{bb} on stability performance. The DR_{core} decreases as boiling boundary is raised, implying a reduction in the probability for core-wide mode of oscillations. The DR_{ch}, which is only affected by thermal-hydraulic considerations, is completely suppressed with a high Z_{bb} (short two-phase column). It follows that higher order modes of oscillation are not compatible with high Z_{bb}.

Figure 9-11 supports a selection of 4.0 feet for the core boiling boundary limit, $\overline{Z}_{bb}$. This value has been demonstrated to provide significant stability margin when applied outside the Exclusion Region. The target $\overline{Z}_{bb}$ value of 4.0 feet is assumed in the following sensitivity analyses.

9.4.5 Axial Flux Shape

The effect of neutronic feedback on reactor stability for high Z_{bb} core conditions is investigated by varying the core average axial flux shape above a fixed 4.0 foot boiling boundary. Control rod adjustments are used to significantly shift the axial flux above Z_{bb} toward the top of the core. These adjustments do not represent realistic operating practices; however, they do represent limiting extremes of axial flux shapes. The variation in the axial flux shape is depicted in Figure 9-13.

Figure 9-14 shows the decay ratio results. As expected, the top-peak axial flux shape results in a higher DR_{core} due to the increased neutronic feedback in the two-phase region. However, DR_{core} is demonstrated to be relatively insensitive for this extreme axial shape when Z_{bb} is maintained at 4.0 feet. As expected, the DR_{ch} is completely suppressed and is not affected by the axial shape change.

9.4.6 Radial Flux Peaking

The effect of radial flux peaking on reactor stability at high Z_{bb} conditions is investigated by varying the control rod pattern to achieve higher radial flux peaking. The changes to the control rod pattern used to support this sensitivity do not represent normal operating conditions but are required to achieve the necessary power shapes.

Figure 9-15 depicts the DR_{core} and DR_{ch} as a function of the radial flux peaking. Figure 9-16 provides a comparison of the calculated decay ratio values to the stability criterion. Both DR_{core} and DR_{ch} are insensitive to significant changes in radial flux peaking at high Z_{bb}. The small decrease in DR_{core} for higher radial flux peaking values is attributed to the compensating effect of axial power redistribution in the two-phase region. Power is shifted down towards Z_{bb} for higher radial peaking, which decreases neutronic feedback.

9.4.7 Feedwater Temperature

The effect of variations in core inlet subcooling on reactor stability at high Z_{bb} conditions is investigated by varying the feedwater temperature (T_{fw}). Control rod positions are adjusted to maintain a constant core average boiling boundary of 4.0 feet with the same core power. This sensitivity addresses operations with reduced T_{fw}, including feedwater heater out of service operations (FWHOOS).

Figure 9-17 depicts the DR_{core} and DR_{ch} as a function of T_{fw}. Figure 9-18 provides a comparison of the calculated decay ratio values to the stability criterion. T_{fw} was varied over a wide range at the analyzed state point. The T_{fw} range in Figure 9-17 is presented in terms of the equivalent change of T_{fw} at rated conditions.

For the analyzed range of T_{fw}, both DR_{core} and DR_{ch} show very small sensitivity. This is expected since Z_{bb} is unchanged, which prevents any significant change in the void distribution. The slight increase in decay ratio with reduced T_{fw} may be attributed in part to the increased rod density that is introduced for reactivity compensation. This sensitivity supports the conclusion that an explicit control of T_{fw} with high Z_{bb} is not necessary.

9.4.8 Hot Radial Region

A sensitivity analysis has been performed to assess the effectiveness of a high core average boiling boundary to control extreme regional power peaking effects on core stability. This sensitivity addresses the concern that even though the core average boiling boundary may be high, isolated regions in the core could be associated with a low local boiling boundary. This may cause the entire core to be more unstable. The sensitivity was performed by incrementally withdrawing control rods from the center of the core to produce an increasingly 'hot' radial region as shown in Figure 9-19. Control rods fully inserted are denoted in the figure by 00 and rods completely withdrawn (notch position 48) are not shown. Figures 9-20 and 9-21 show the core average and the hot channel axial power shapes for the analyzed cases.

The same core power was maintained for the four analyzed cases by appropriate control rod adjustments. Z_{bb} was maintained at 4.0 feet where possible; however, for the most extreme regional peaking configuration, the 4.0 foot $\overline{Z}_{bb}$ target could not be met. The variations in control rod patterns used in this sensitivity do not represent normal operations, and are solely designed to investigate the boiling boundary concept under extreme conditions.

Figure 9-22 depicts the DR_{core} and DR_{ch} as a function of MCPR. The shaded area represents conditions where the MCPR operating limit is exceeded. Figure 9-23 compares the calculated decay ratio values to the stability criterion. The MCPR operating limit is exceeded well before both DR_{core} and DR_{ch} approach the stability criterion. Moreover, the stability criterion is exceeded only after the core average boiling boundary target of 4.0 feet can no longer be maintained.

Under these extreme conditions, adequate stability margin is ensured by compliance with a high (4.0 feet) core average boiling boundary and conformance to existing fuel operating limits. Therefore, the core average boiling boundary stability control, in conjunction with existing thermal limits, eliminates the need for any local or regional stability controls.

9.4.9 Cycle Depletion

The effect of cycle depletion on stability performance at high Z_{bb} conditions is investigated by adjustments to the control rod pattern at selected cycle exposures, (beginning of cycle (BOC), middle (MOC) and end of cycle (EOC)), to achieve high Z_{bb}. Figure 9-24 depicts DR_{core} and DR_{ch} as a function of cycle exposure. Figure 9-25 provides a comparison of the calculated decay ratio values to the stability criterion. This sensitivity demonstrates applicability of the Z_{bb} control concept throughout the fuel operating cycle, including spectral shift operations, and under Xenon-free conditions. For the conditions assumed, the highest Z_{bb} achieved at MOC is 3.8 feet.

9.4.10 Analysis Summary

The analysis demonstrates that an effective stability control is achieved by a single control, $\overline{Z}_{bb}$, which 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 Exclusion Region boundary.

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.

Another outcome of the boiling boundary concept, as supported by this analysis, is the ability to define conditions that are limiting for core stability. This capability can be used in identifying and validating the stable region of the operating domain. These conditions include low core average boiling boundary, top-peaked axial flux shape in the core two-phase region, high radial peaking, negative void coefficient, and Xenon-free conditions. The low core average boiling boundary can be achieved by increasing core power, lowering core flow, decreasing feedwater temperature, and minimizing shallow control rod insertion.

9.5 Stability Control Implementation and Plant Experience

9.5.1 Implementation

The presence of conditions conducive to reactor instability outside the Exclusion Region is possible if stability controls are not implemented inside the Restricted Region (Figure 9-26). Examples of such conditions include low feedwater temperature, unfavorable Xenon conditions and skewed axial and radial flux distributions. The stability control, Fraction of Core Boiling Boundary (FCBB), has been analytically demonstrated to significantly increase the margin to reactor instability for operating inside the Restricted Region. In effect, state points at the boundary of the Exclusion Region that satisfy the FCBB limit will result in reactor conditions well within the stability criterion of Figure 9-10.

The target elevation of the core average boiling boundary, $\overline{Z}_{bb}$, is used to define the operating limit FCBB. From Equation 9-22, with $\overline{Z}_{bb} = 4.0$ feet and $F_{af} = 0.9$, the FCBB limit is:

$$FCBB = \frac{\frac{1}{n}\sum_{i=1}^{\overline{Z}_{bb}=4.0'} AP_i}{0.264\frac{W \times DHS}{P}} \qquad (9\text{-}24)$$

where: $\sum_{i=1}^{n} AP_i = n$,

W in $10^6\ lb_m/hr$,

DHS in Btu/lb_m, and

P in MW_t.

The FCBB limit is normalized such that conformance to $\overline{Z}_{bb}$ during controlled reactor maneuvers is ensured if FCBB does not exceed 1.0.

As an example, for a core model consisting of 25 nodes, each 6.0 inches high, FCBB requires:

$$\frac{(AP_1 + AP_2 + ... + AP_8)/25}{0.264 \times W \times DHS/P} \leq 1.0, \qquad (9\text{-}25)$$

where: $\sum_{i=1}^{25} AP_i = 25$,

W in $10^6\ lb_m/hr$,

DHS in Btu/lb_m, and
P in MW_t.

If during controlled reactor operations FCBB exceeds 1.0, the boiling boundary is below $\overline{Z}_{bb}$ and corrective action is needed. The most effective way to decrease FCBB is by insertion of shaping control rods. These rods will suppress the power at the bottom of the core and shift the boiling boundary upward (Figure 9-8).

The FCBB limit in conjunction with other fuel operating limits, has been demonstrated to provide adequate protection from reactor instabilities. However, as a matter of good operating practice, non-uniform control rod patterns should be avoided. This includes control rods in deep position for reactivity control, as well as shallow position for Z_{bb} control. Control rod dispersion that is radially non-uniform may lead to situations where small regions in the core become neutronically decoupled, exhibiting a low Z_{bb} and potentially reducing the stability margin of the reactor.

Inserting control rods used for reactivity control as far as possible into the core can also increase the stability margin of the reactor. This insertion minimizes the power generated at the core top, which weakens the neutronic feedback.

In summary, control rod distribution patterns that are radially uniform in the core should be used for both the shaping and the reactivity control rods. The shaping control rod inventory should be set to achieve the target $\overline{Z}_{bb}$. The reactivity control rod inventory should be set to minimize the power peaking in the top of the core. Placing control rods at other intermediate positions should be avoided to the extent practicable.

As the reactor startup is initiated, Z_{bb} is at the top of the core, where bulk saturation is first achieved. Subsequently, Z_{bb} is moved downward in the core as control rods are being withdrawn and reactor power increases. As the rated operating condition is approached, the bulk saturation elevation in the core is lowered and Z_{bb} settles below $\overline{Z}_{bb}$. Reactor instability is not a concern, however, because of the high core flow rate.

Since $\overline{Z}_{bb}$ is initially very high in the core, the startup path can be planned such that Z_{bb} will not fall below $\overline{Z}_{bb}$ prior to maneuvering through the controlled region. This strategy will eliminate unnecessary and untimely control rod maneuvers to satisf the FCBB limit. Upon exiting the Restrict Region, the shaping control rods can be withdrawn to achieve the target rod pattern for rated conditions, allowing Z_{bb} to fall below $\overline{Z}_{bb}$.

9.5.2 Plant Experience

Analysis of the core average boiling boundary control's effects on reactor stability performance has been performed for realistic reactor conditions and with control rod pattern consistent with operational practices (Section 9.4.4). The results of this analysis suggest that a core average boiling boundary limit of 4.0 feet is not only an effective control, but that it is also feasible. This conclusion is also supported by actual plant data.

Although the stability control is compatible with all fuel and reactor technical limitations, administrative conflicts may exist at certain BWR plants. Implementation of this control at such plants would necessitate resolution of those administrative conflicts.

Startup data from a U.S. BWR plant was evaluated to assess the implementation feasibility of a $\overline{Z}_{bb}$ limit of 4.0 feet. The data was selected at the most challenging core power and flow state point along the startup path. This state point is achieved during a required reactor recirculation pump upshift from slow to high speed at minimum core flow conditions. A summary of selected actual conditions from one operating cycle is provided in Table 9-2.

The table represents operating state points with different Xenon conditions (shutdown duration prior to startup), cycle exposures and control rod patterns. The secondary control rods remain inserted early in the startup and are typically withdrawn prior to achieving the final rod pattern at rated power. The purpose of the secondary rods is power shaping during the startup to compensate for non-equilibrium Xenon conditions. They may be withdrawn before or after the recirculation pump upshift. As expected, the Z_{bb} values in Table 9-2 are directly related to the use of the shaping secondary rods. No correlation is observed (or expected) in relation to Xenon condition or cycle exposure.

The operating conditions shown in Table 9-2 were achieved without any consideration of Z_{bb}. They represent actual operating conditions for the fuel cycle. Moreover, additional evaluation based on actual reactor operating conditions demonstrated that Z_{bb} values of 5.0 feet, at varying cycle conditions from beginning to end of cycle, are achievable. Thus, a $\overline{Z}_{bb}$ limit of 4.0 feet can be operationally consistent with typical plant operations near the region susceptible to reactor instability.

Details of Startup Conditions 1 and 3 in Table 9-2 are provided to demonstrate the difference between low and high Z_{bb} startups. Figure 9-27 depicts the core average boiling boundary, Z_{bb}, of Startup Condition 3 as a function of the actual startup path. The operating map is shown as a reference on the core power and flow plane.

As expected, Z_{bb} starts high in the core when power is low, and decreases to about 2.0 feet when the 100% recirculation flow-control-line is reached. The secondary control rods are withdrawn early in the startup path, which results in a Z_{bb} of 2.5 feet at the recirculation pump upshift conditions. The corresponding control rod pattern, with quarter-core symmetry, is shown in Figure 9-28. The axial power shape, with the actual Z_{bb} indicated, is shown in Figure 9-29. In this case Z_{bb}, is below the target $\overline{Z}_{bb}$ limit of 4.0 feet.

In contrast, Figure 9-30 depicts the core average boiling boundary, Z_{bb}, of Startup Condition 1 as a function of the actual startup path. Here, the secondary rods are withdrawn late in the startup path. This results in a Z_{bb} value over 4.0 feet at the recirculation pump upshift conditions.

The control rod pattern is shown in Figure 9-31 and the axial power shape in Figure 9-32. In this case, Z_{bb} is above the target $\overline{Z}_{bb}$ limit of 4.0 feet.

The FCBB limit, with $\overline{Z}_{bb}$ set at 4.0 feet, has been implemented successfully in an operating U.S. BWR. Implementation of the FCBB limit did not result in any significant additional burden to the operating staff. Significant stability margin was created and maintained throughout the reactor startup path, without any need for reliance on a stability monitoring system, on-line instability predictions, or pre-startup analysis.

9.6 Stability Control Conclusions

Operational experience has clearly demonstrated the need for an effective stability control that can readily be applied to reactor operations. The core average boiling boundary control fulfills this need.

BWR stability performance is dominated by the core void distribution. All global core parameters must be considered in defining the location of the bulk coolant saturation elevation that marks the beginning of the voided core region. However, the two-phase column length and neutron flux shape in the two-phase region of a core are the major factors influencing reactor stability. The two-phase column length determines the void sweeping time and therefore the pressure drop phase lag. It also limits the core void fraction that controls the thermal-hydraulic and neutronic feedback gains.

The core average boiling boundary provides a convenient parameter for expressing the relative lengths of the single-phase and two-phase columns in a reactor core. When defined using core average parameters, the equation incorporates all the factors important to reactor stability for a radially collapsed core. The effects of varying radial core power shapes can be controlled through use of the boiling boundary parameter in conjunction with existing fuel thermal limits.

When the core average boiling boundary is raised sufficiently, the core remains very stable during reactor maneuvering in the Restricted Region of the licensed operating domain. In addition, variations in all parameters affecting stability become secondary and may be ignored. Use of the stability control can guide plant operations in a practical manner, to assure adequate stability margins during reactor maneuvering. The ability to utilize this stability control has been demonstrated analytically, and verified during reactor startups with a large BWR.

Table 9-1: 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

Table 9-2: BWR Startup Conditions

	Exposure (GWD/MT)	Shutdown (days)	Secondary Rods	Z_{bb} (ft)
1.	0.1	> 14	used	4.3
2.	1.4	< 1	none	2.8
3.	2.0	> 9	none	2.5
4.	5.4	> 1	used	4.8

Figure 9-1: Neutronic and Thermal-Hydraulic Feedback Mechanisms

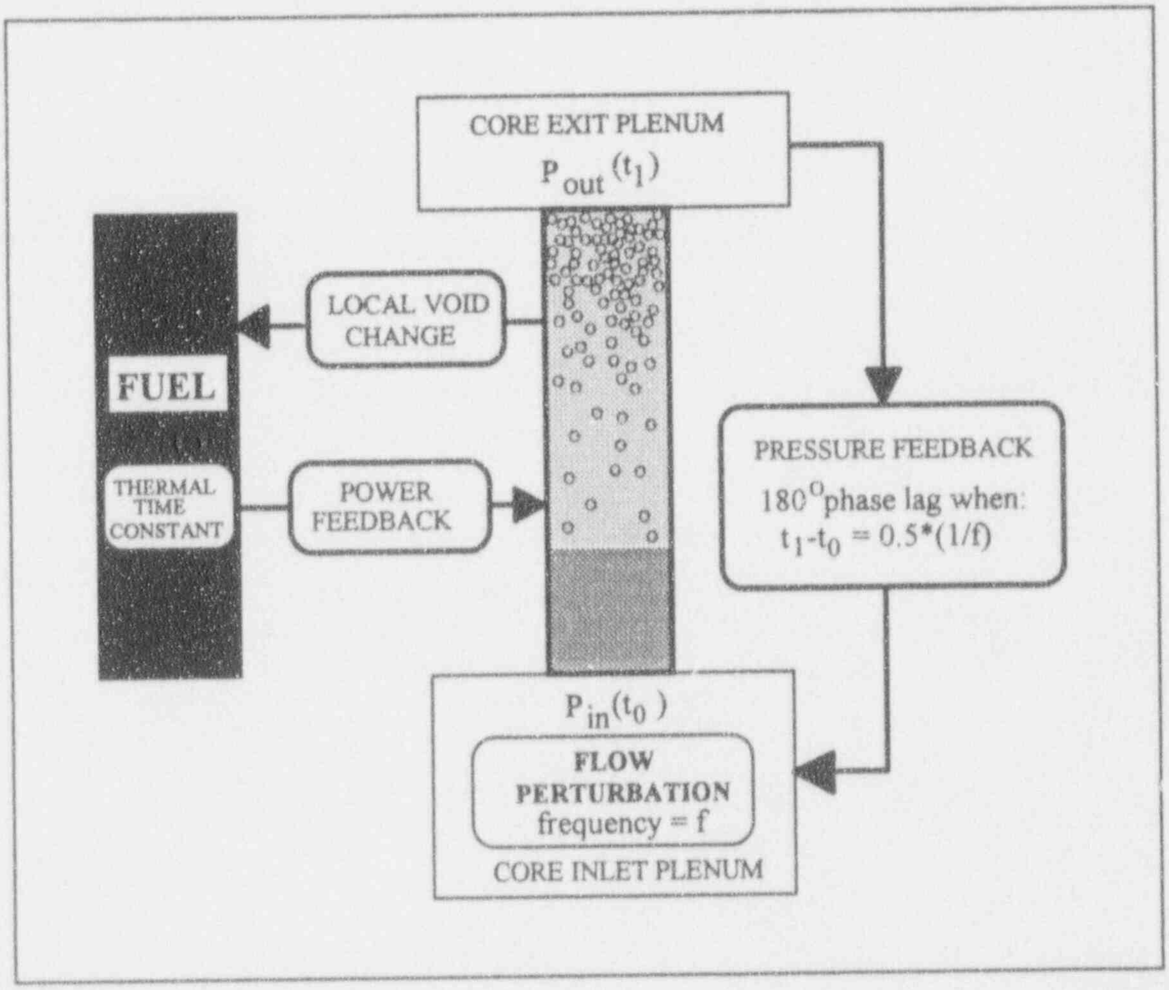


Figure 9-2: Parameters Affecting Core Stability

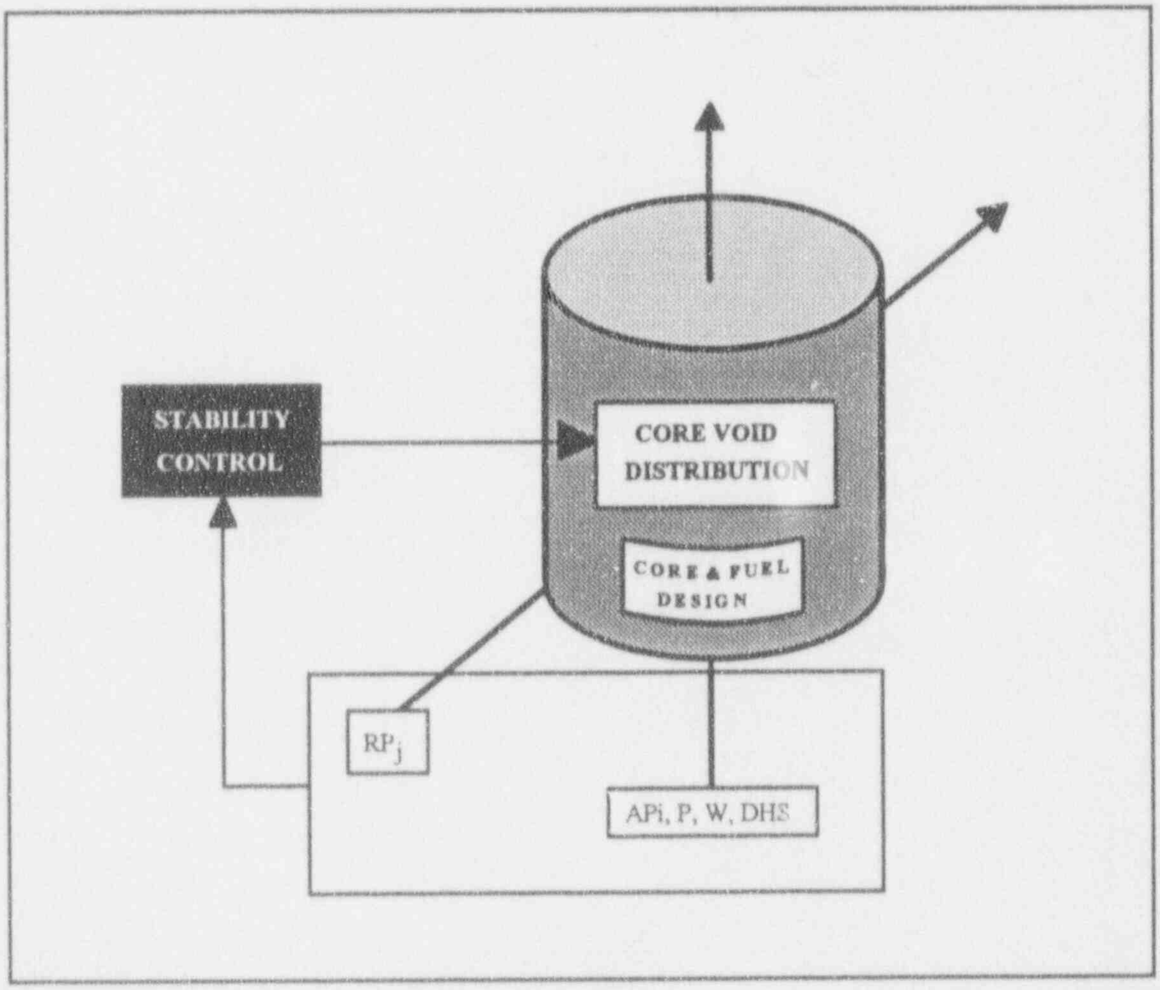


Figure 9-3: Bottom-Peaked Average Axial Power

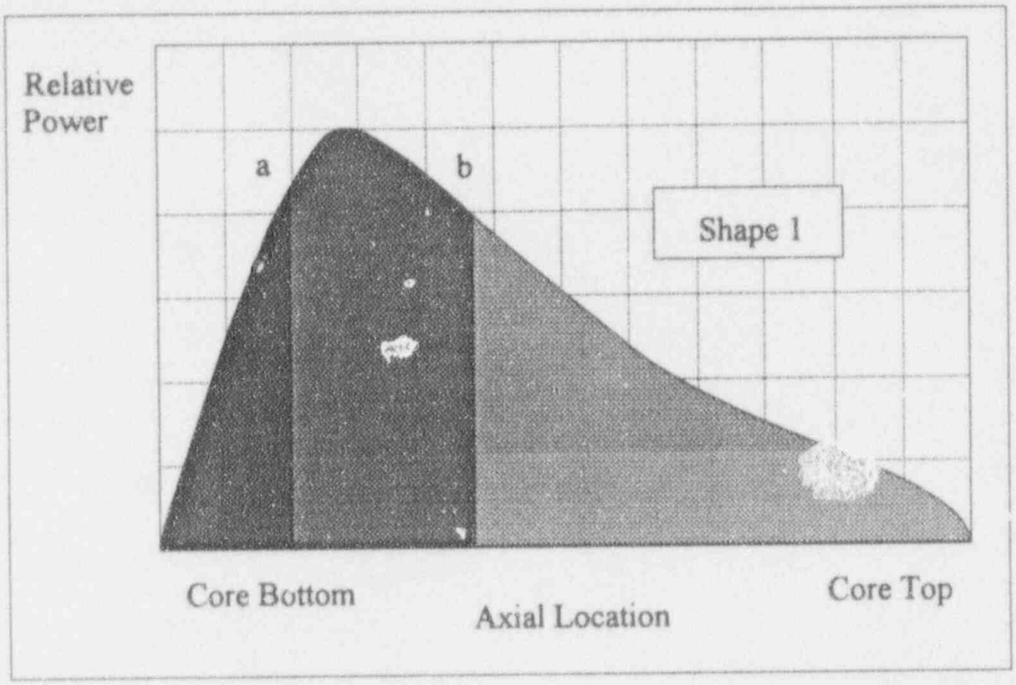


Figure 9-4: Top-Peaked Average Axial Power

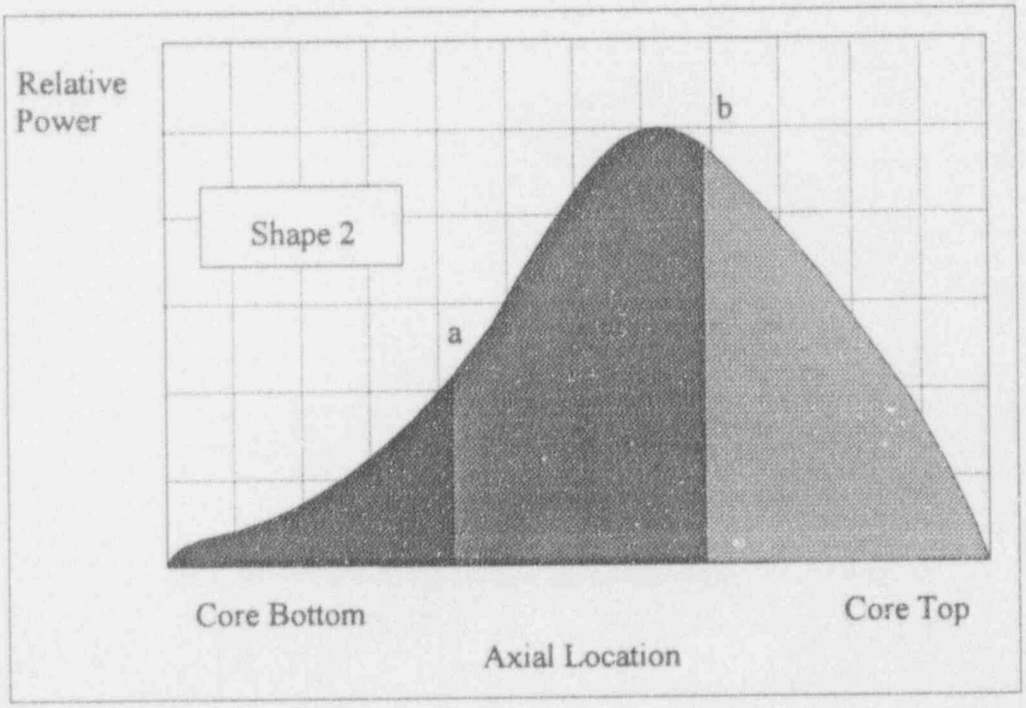


Figure 9-5: Core Radial Collapse

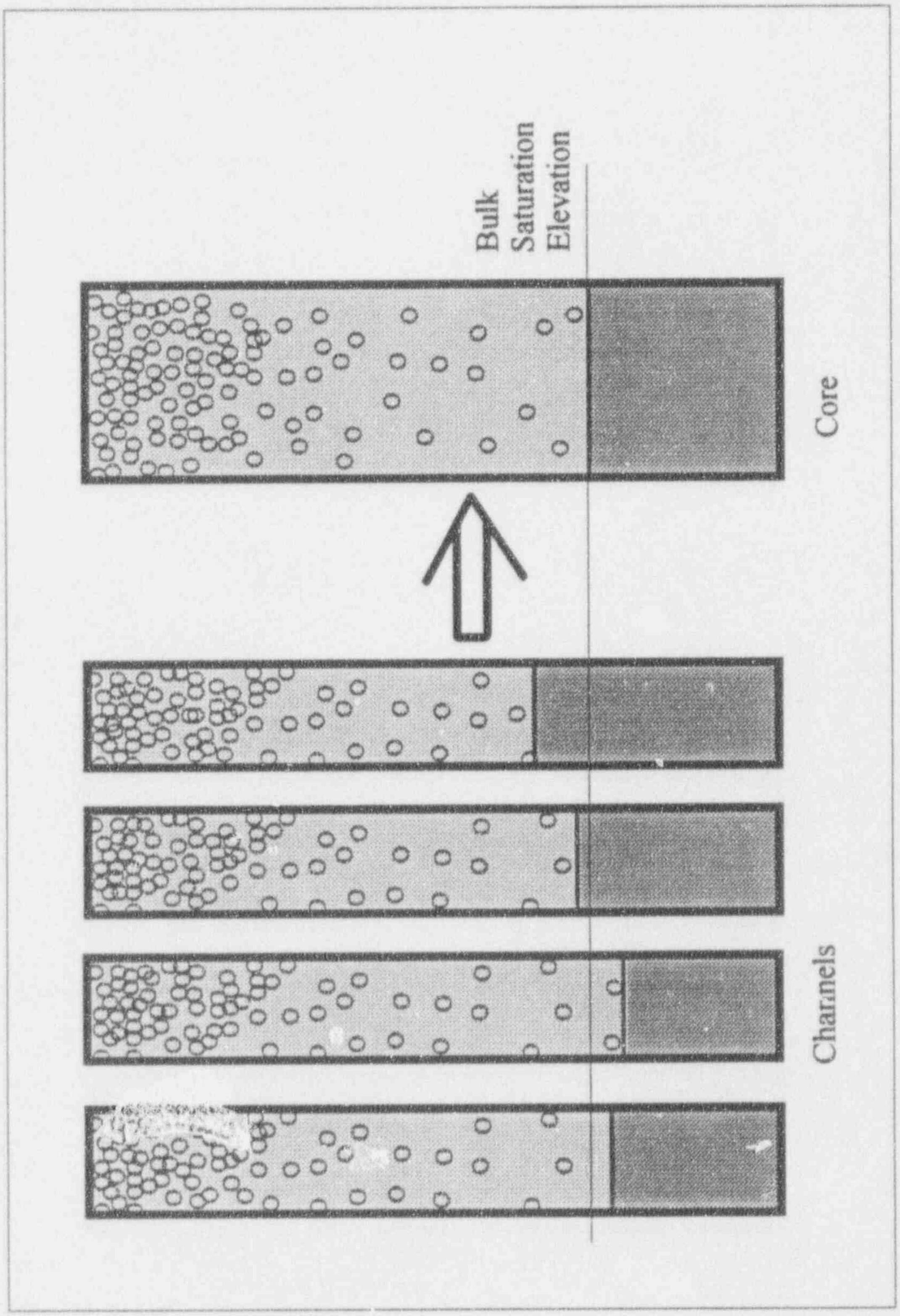


Figure 9-6: Factors Affecting Stability Feedback Mechanisms

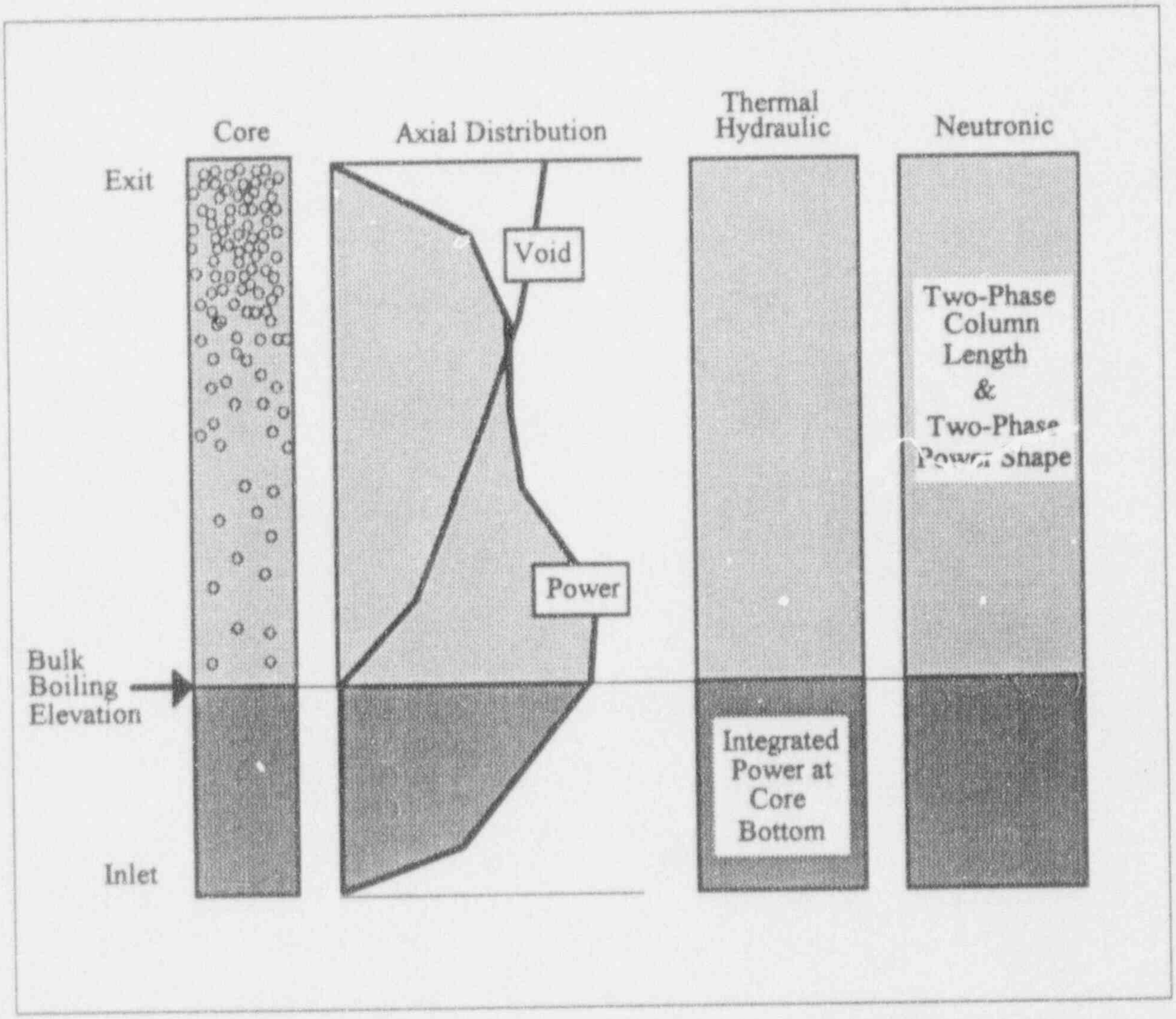


Figure 9-7: Core Average Boiling Boundary

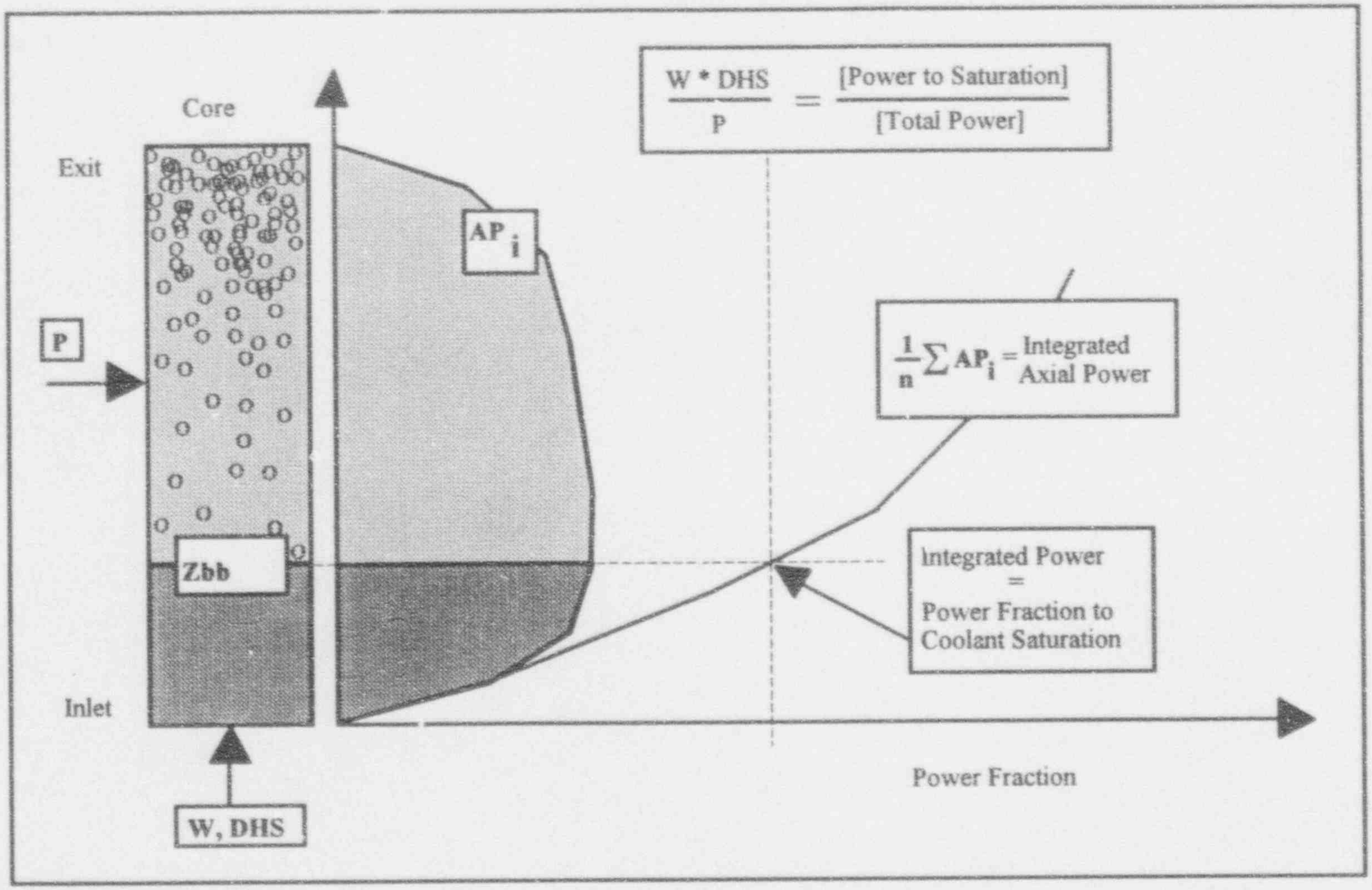


Figure 9-8: Stability Evaluation of Changes to Core Power Shapes

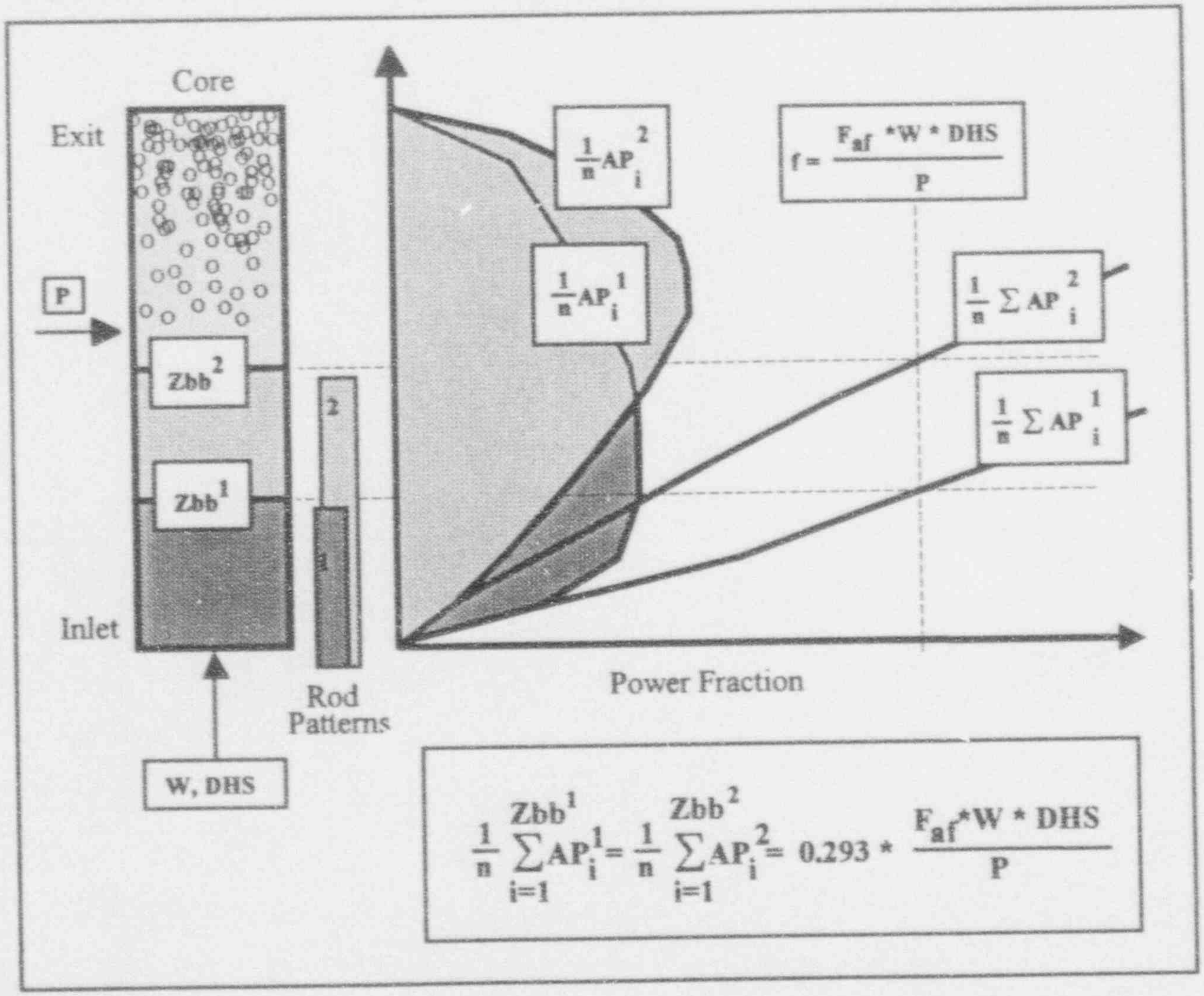


Figure 9-9: Operating Region Susceptible to Oscillations

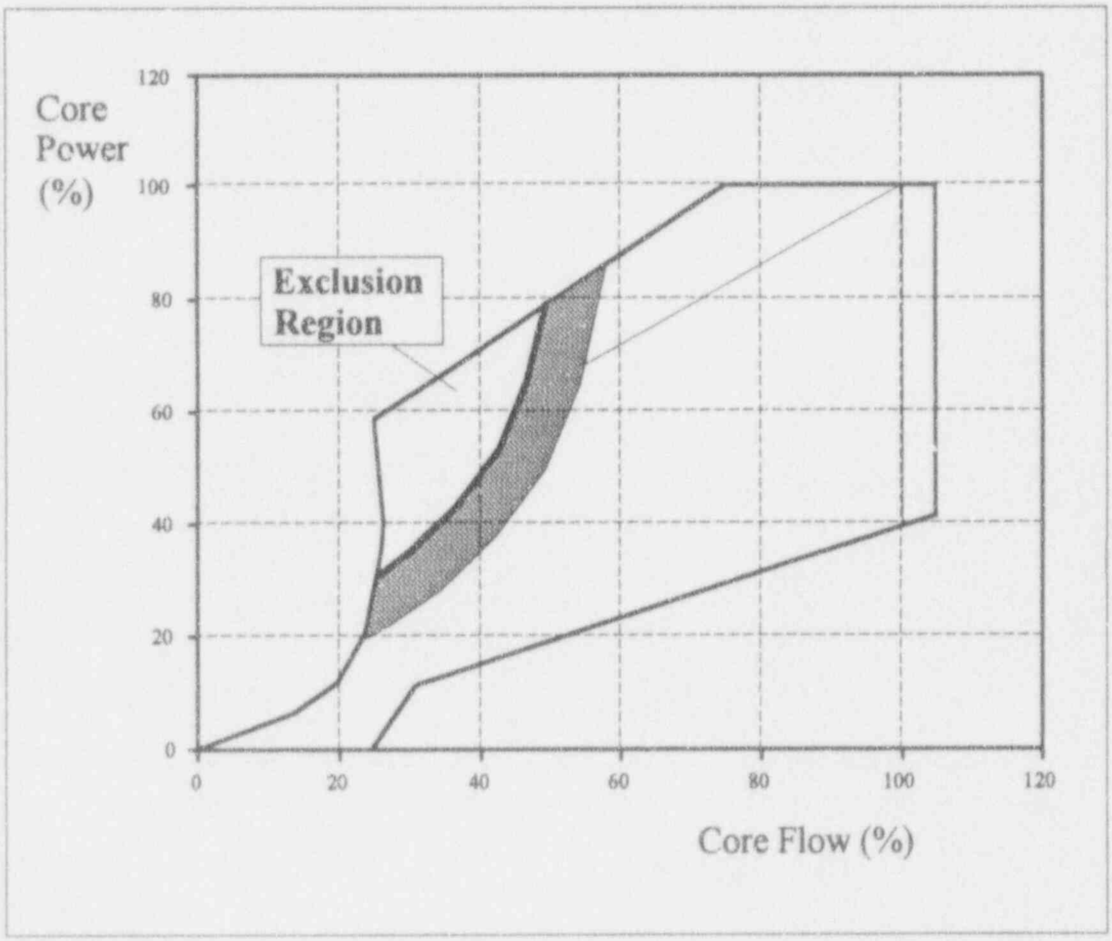


Figure 9-10: Exclusion Region Boundary Generation Stability Criterion

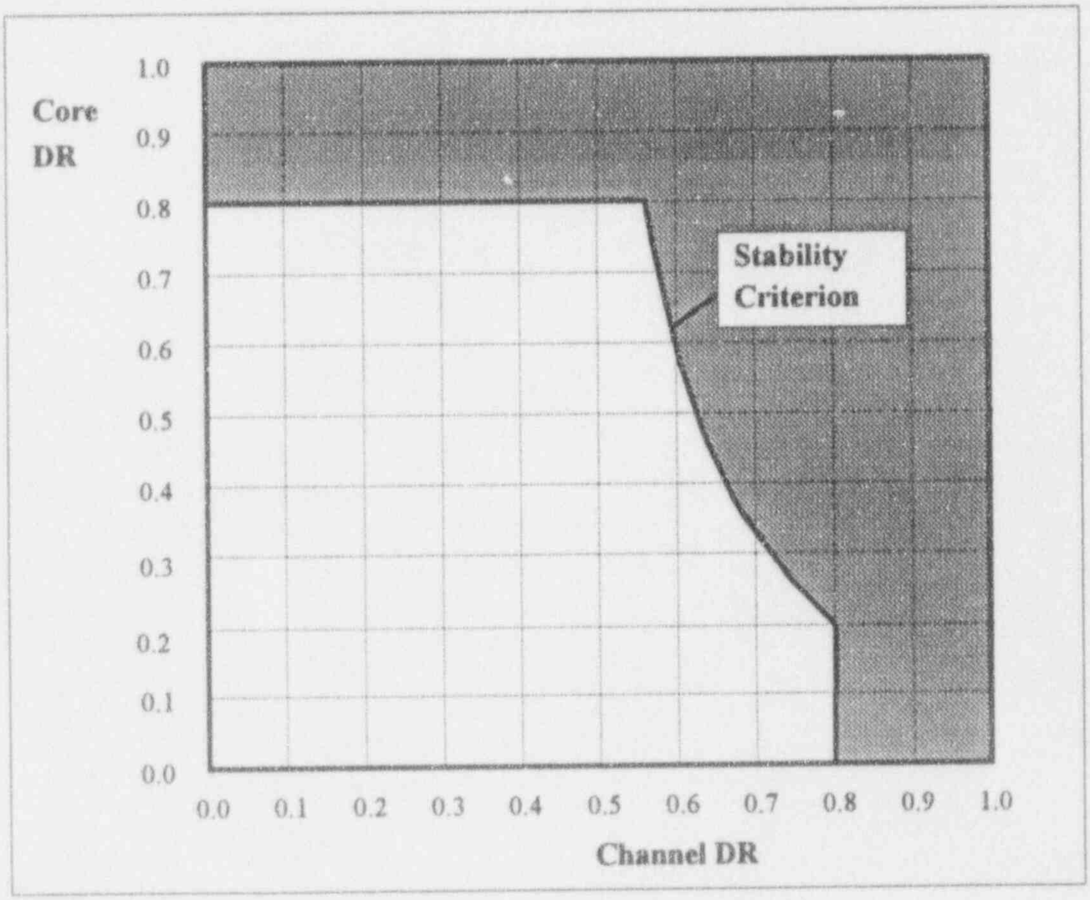


Figure 9-11: Core Stability versus Z_{bb}

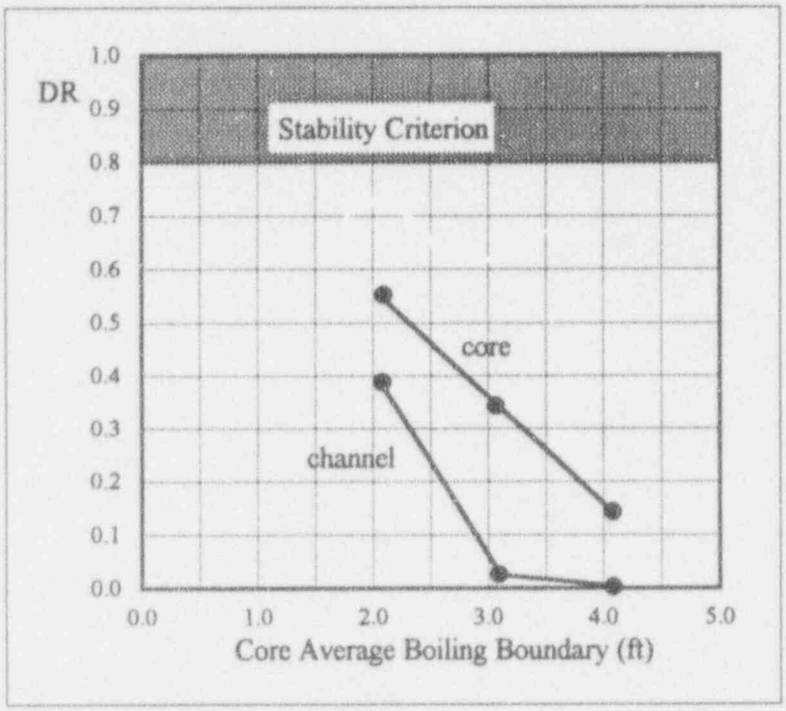


Figure 9-12: Z_{bb} versus Stability Criterion

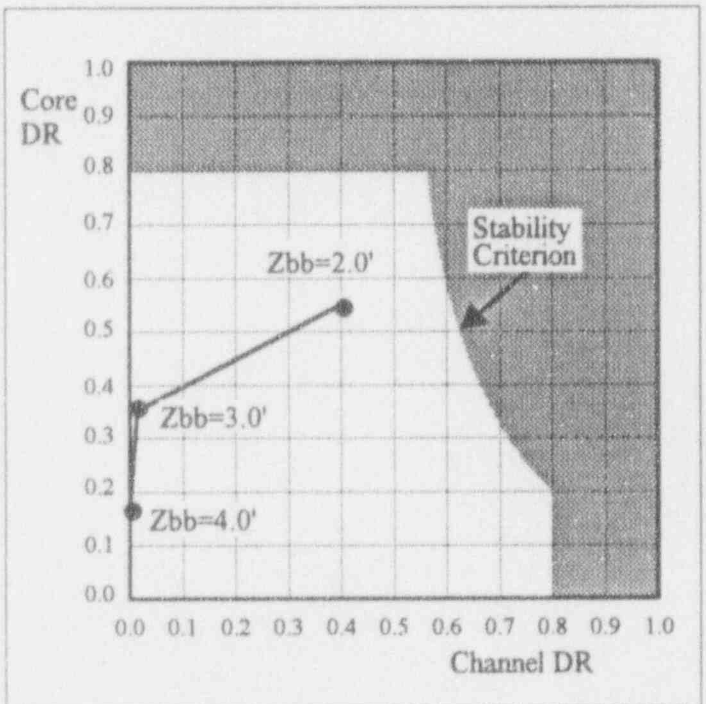


Figure 9-13: Two-Phase Column Axial Flux Shapes

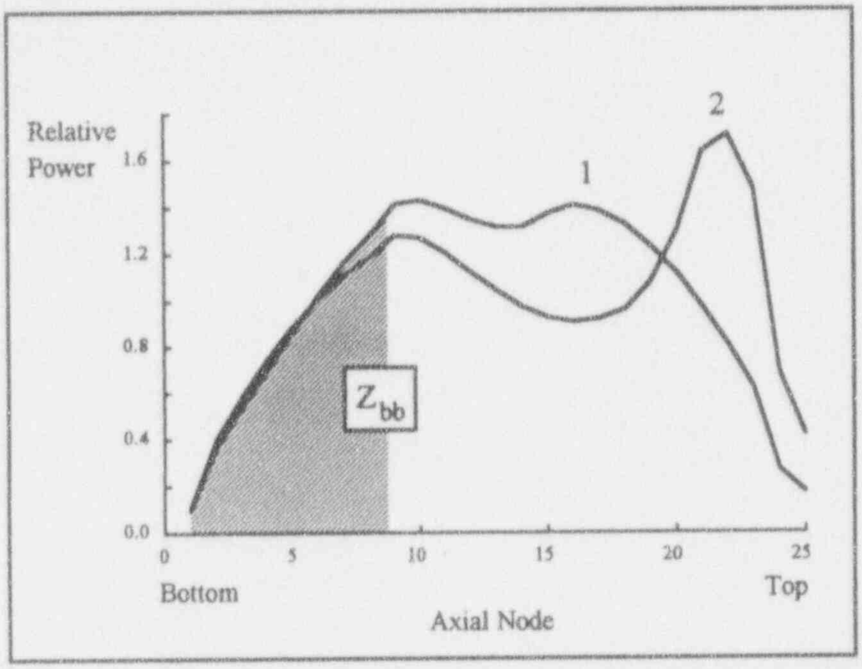


Figure 9-14: Two-Phase Axial Flux Shape versus Stability Criterion

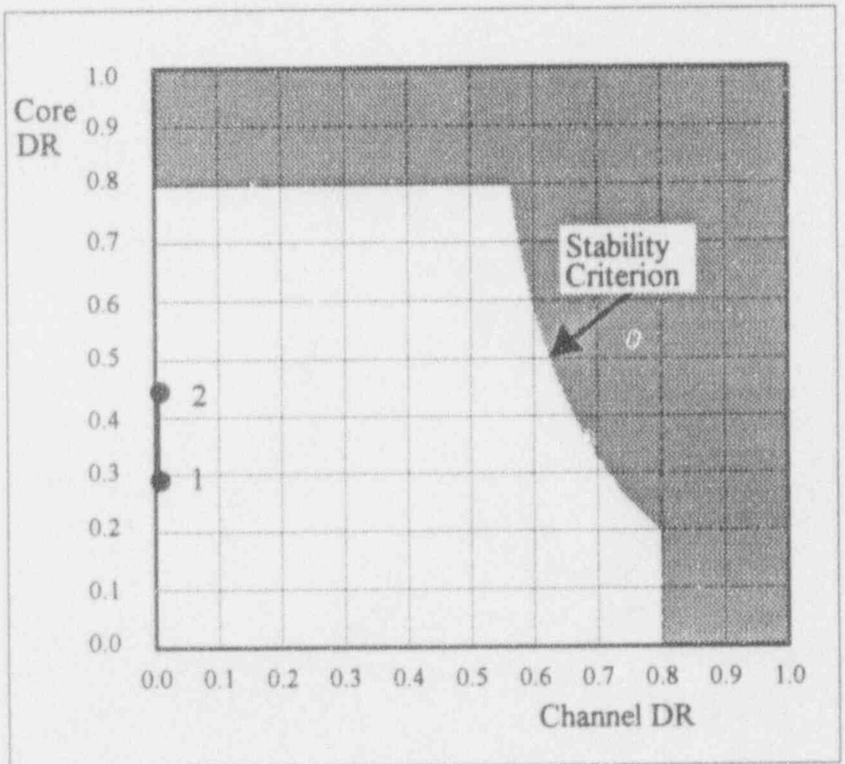


Figure 9-15: Radial Flux Peaking versus Stability at high Z_{bb}

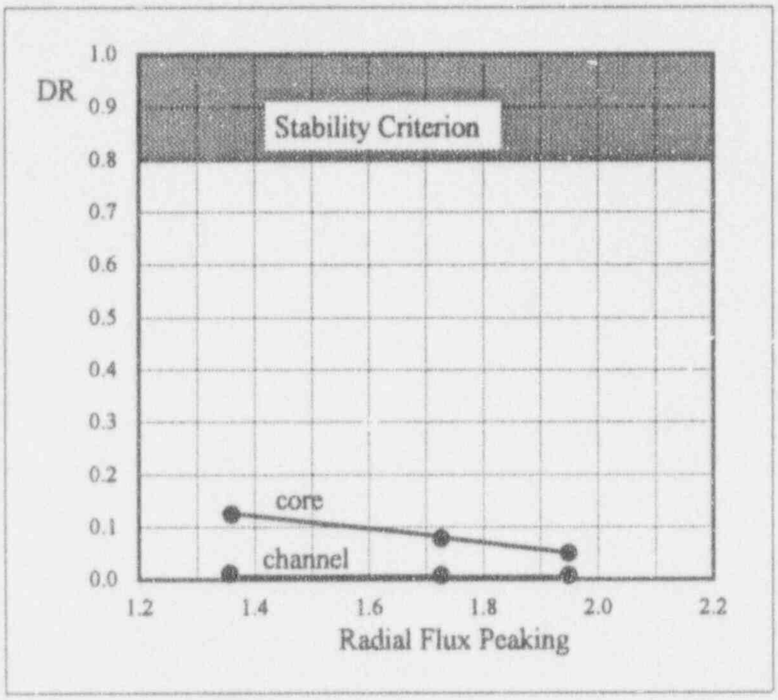


Figure 9-16: Radial Flux Peaking versus Stability Criterion

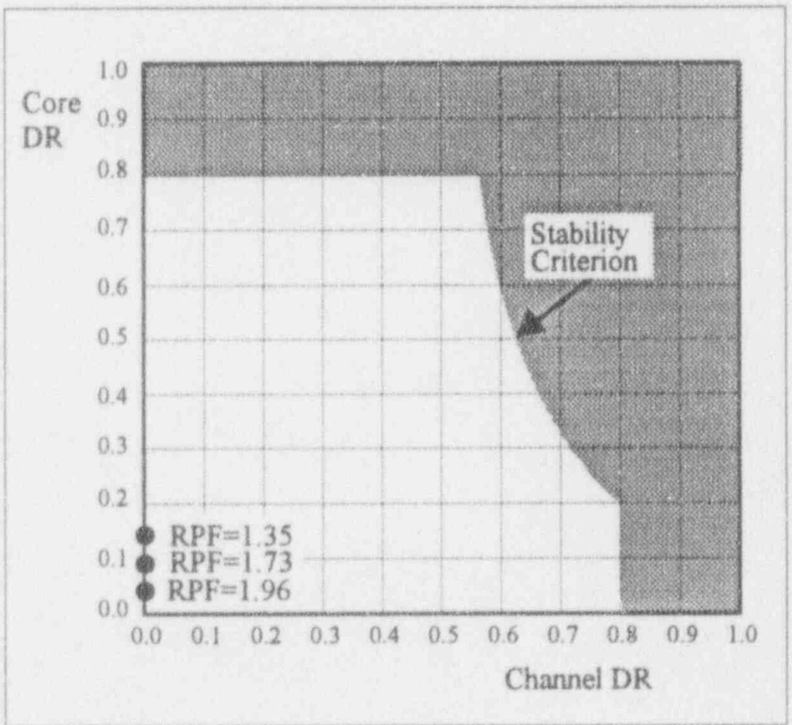


Figure 9-17: Feedwater Temperature versus Core Stability

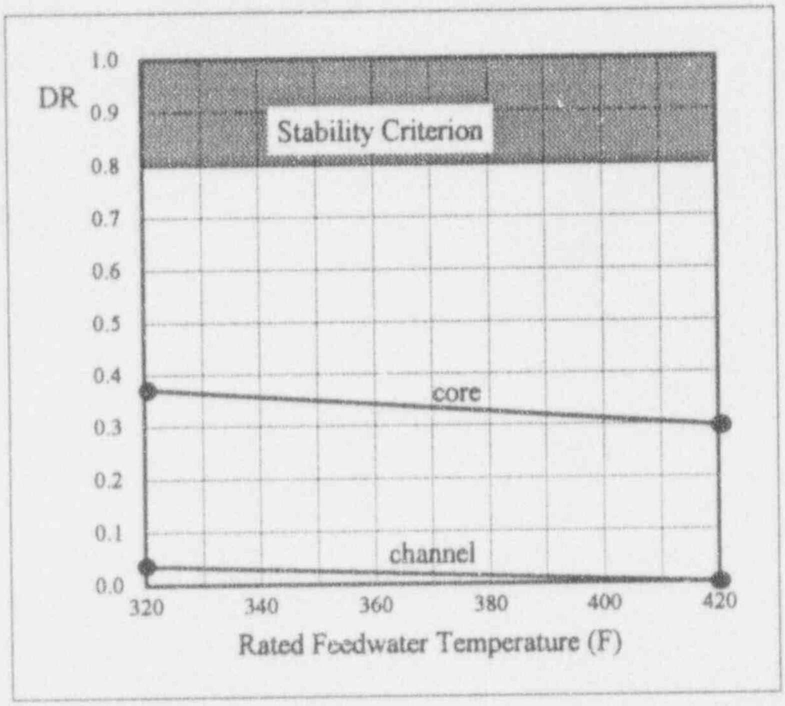


Figure 9-18: Feedwater Temperature versus Stability Criterion

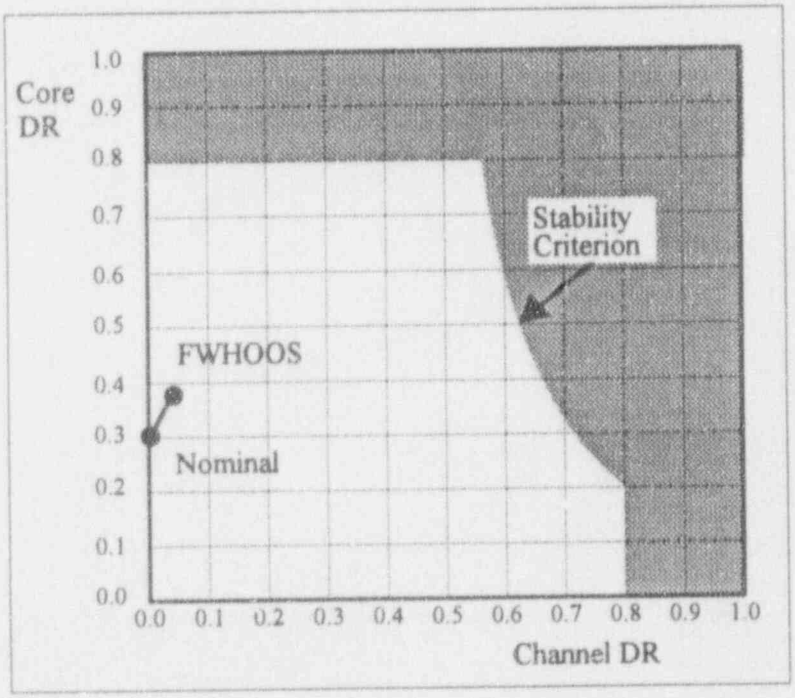


Figure 9-19: Hot Region Control Rod Patterns

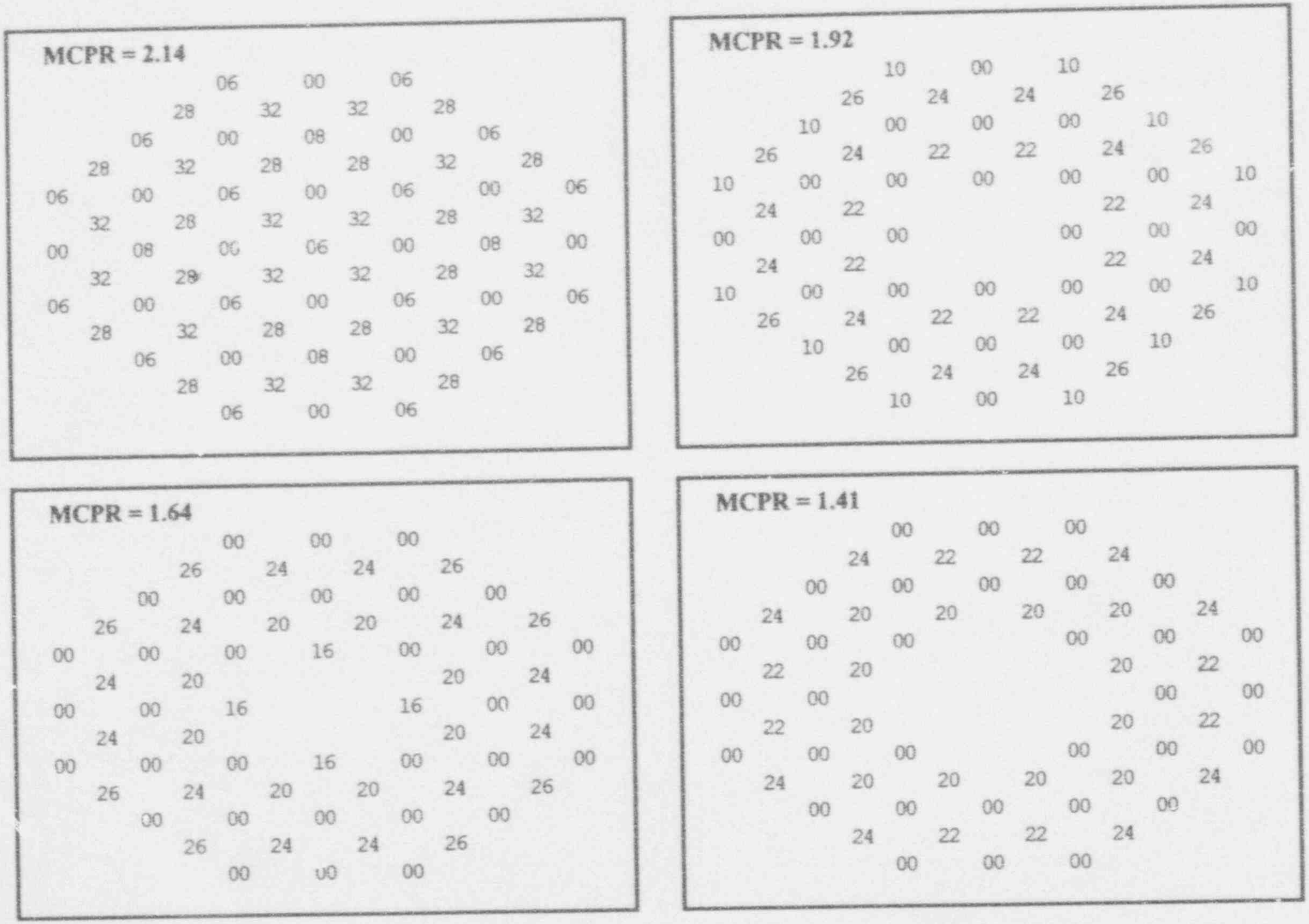


Figure 9-20: Hot Region Core Average Axial Power Shapes

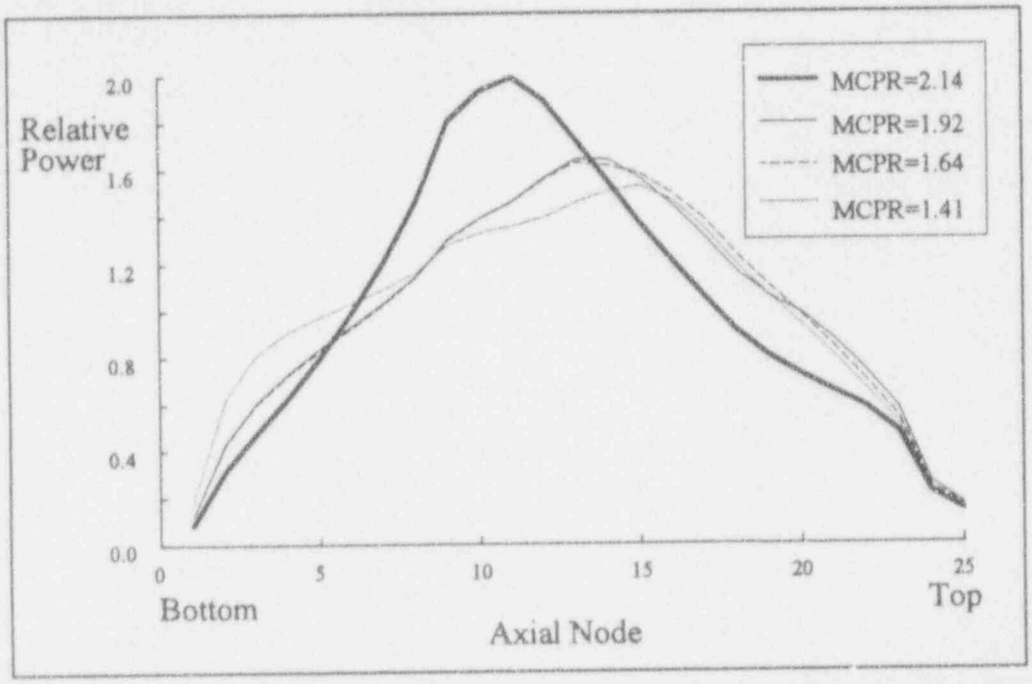


Figure 9-21: Hot Channel Axial Power Shapes

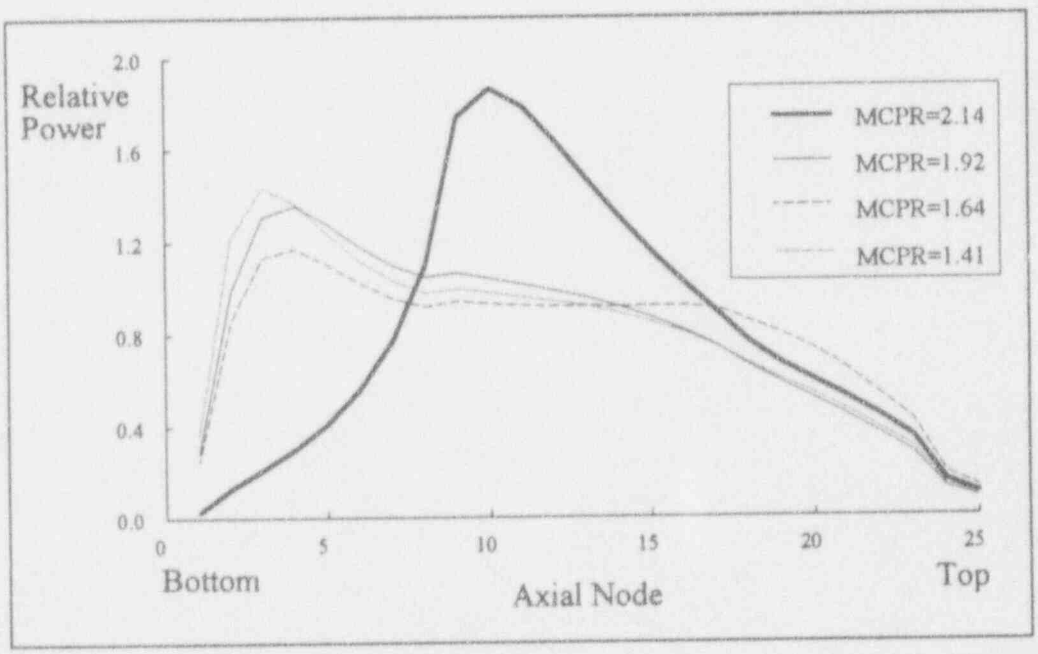


Figure 9-22: Hot Region MCPR versus Core Stability

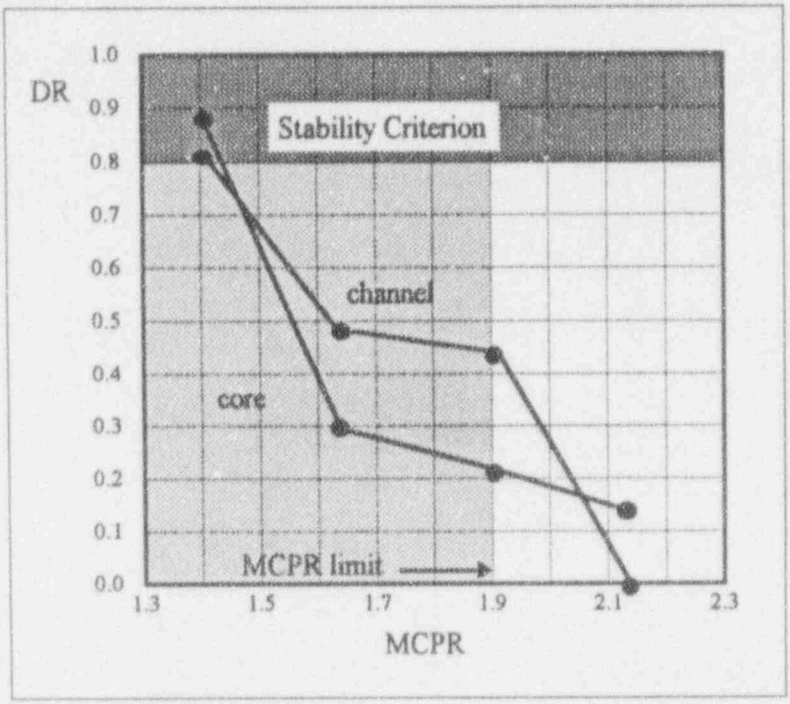


Figure 9-23: Hot Region MCPR versus Stability Criterion

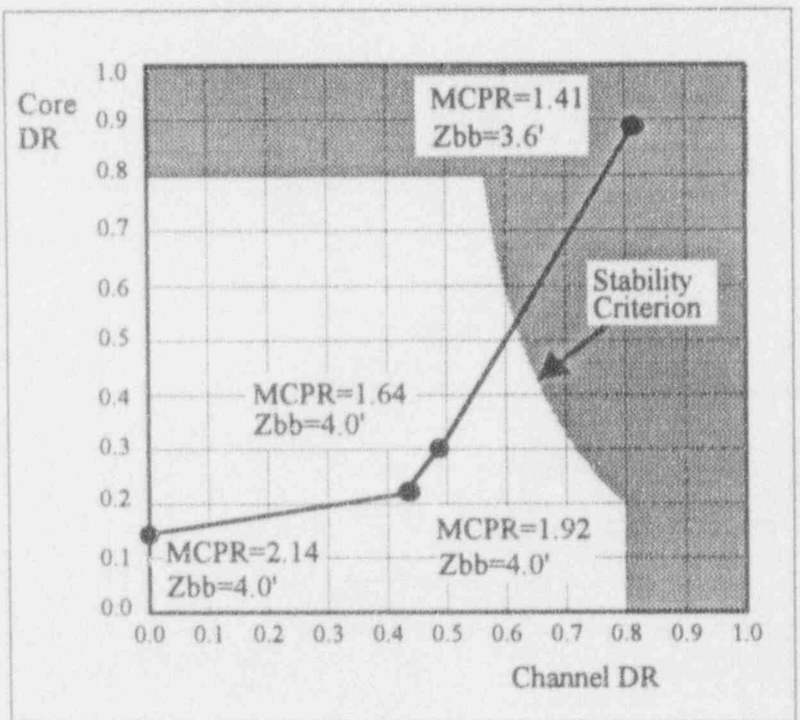


Figure 9-24: Cycle Depletion versus Core Stability

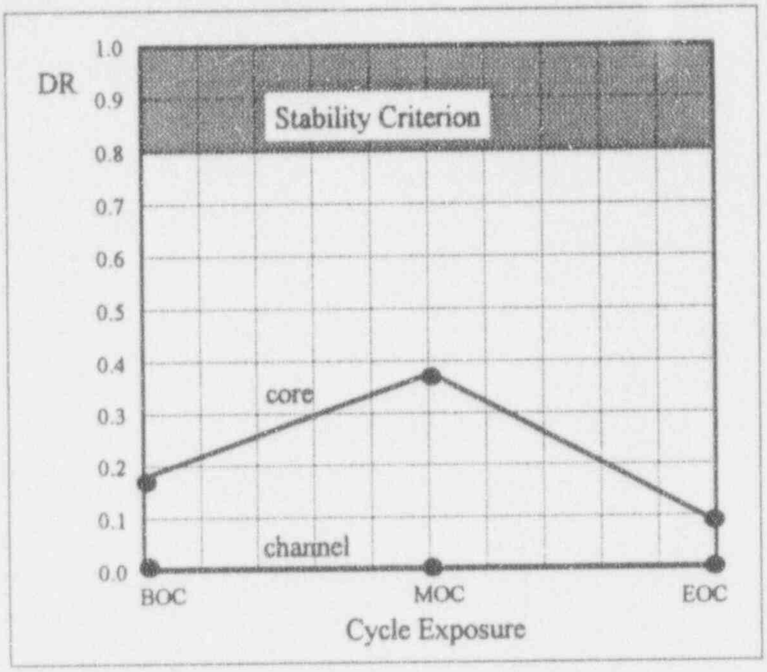


Figure 9-25: Cycle Depletion versus Stability Criterion

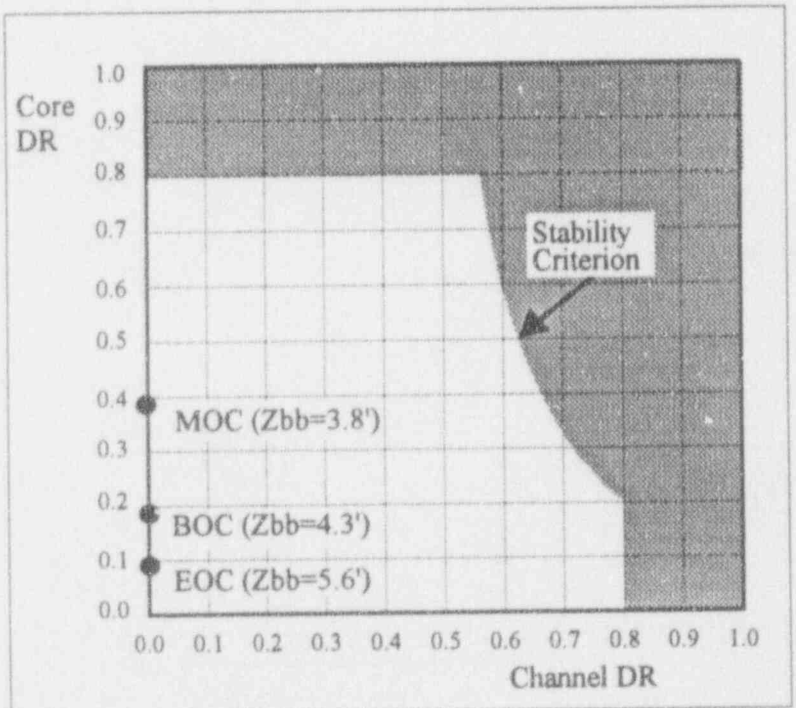


Figure 9-26: Operating Stability Regions

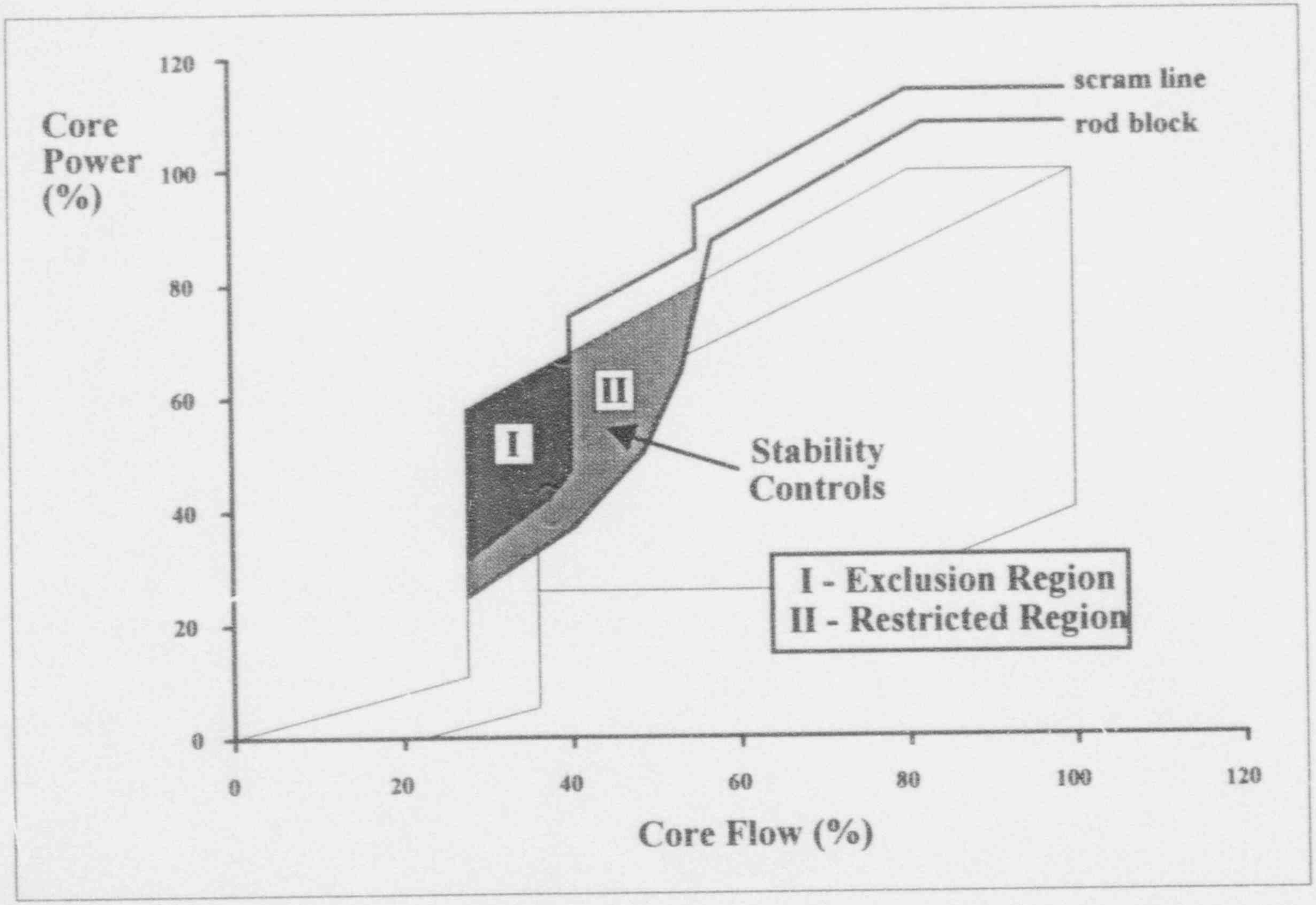


Figure 9-27: Low Z_{bb} Startup

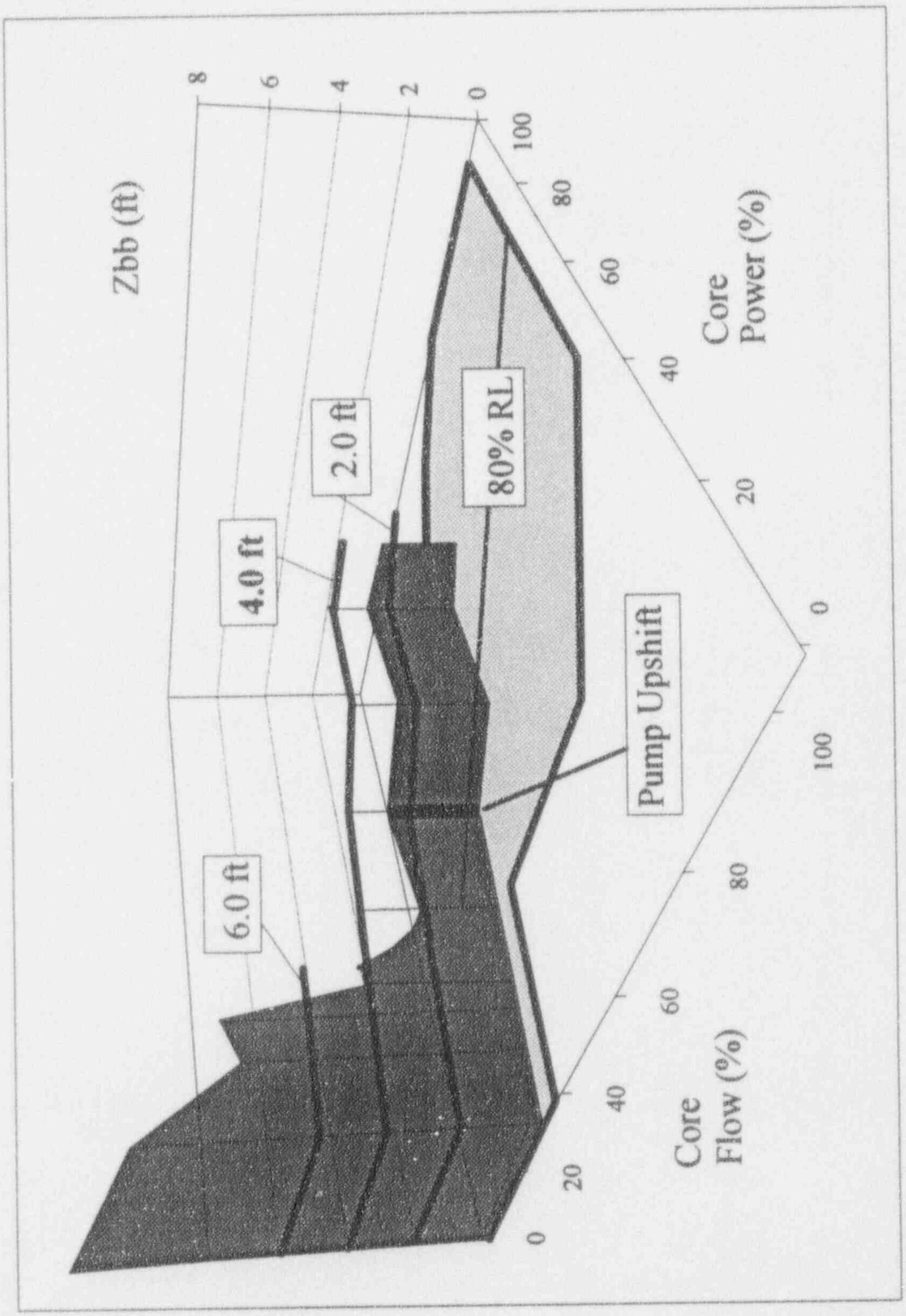


Figure 9-28: Low Z_{bb} Control Rod Pattern

Control Rod Insertion Fractions

				——	——	——	——
		——	——	0.38	——	0.75	——
	——	——	——	——	——	——	——
	——	0.83	——	1.00	——	1.00	——
——	——	——	——	——	——	——	——
——	——	0.92	——	0.92	——	0.63	——
——	——	——	——	——	——	——	——
0.75	——	0.92	——	0.63	——	0.92	——

—— = full out

Figure 9-29: Low Z_{bb} Axial Shape

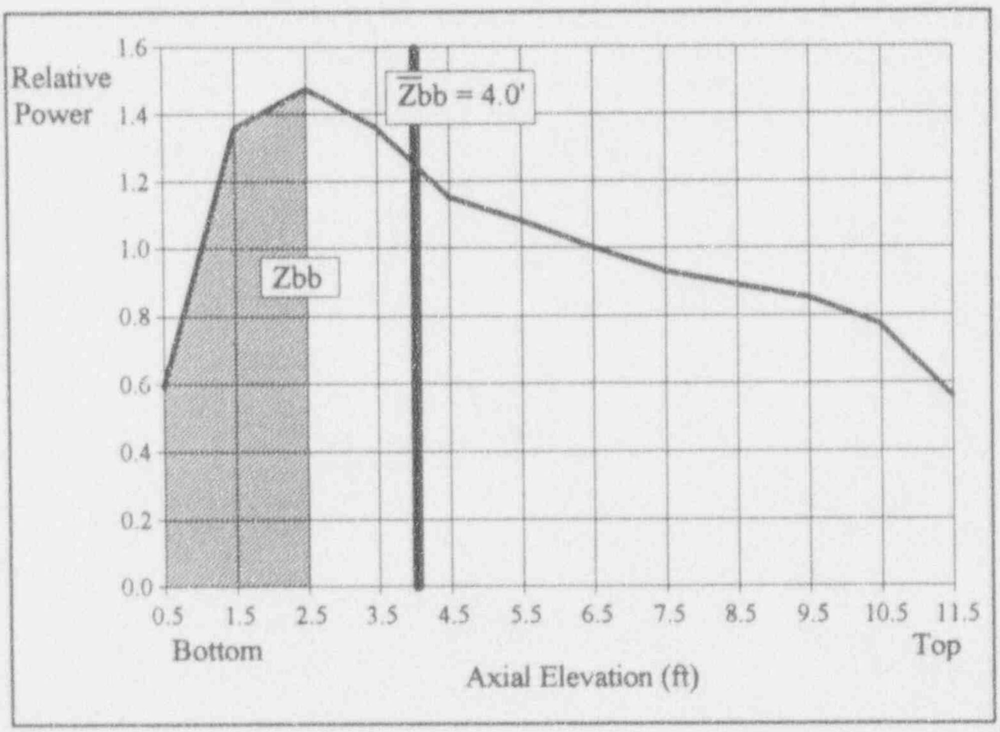


Figure 9-30: High Z_{bb} Startup

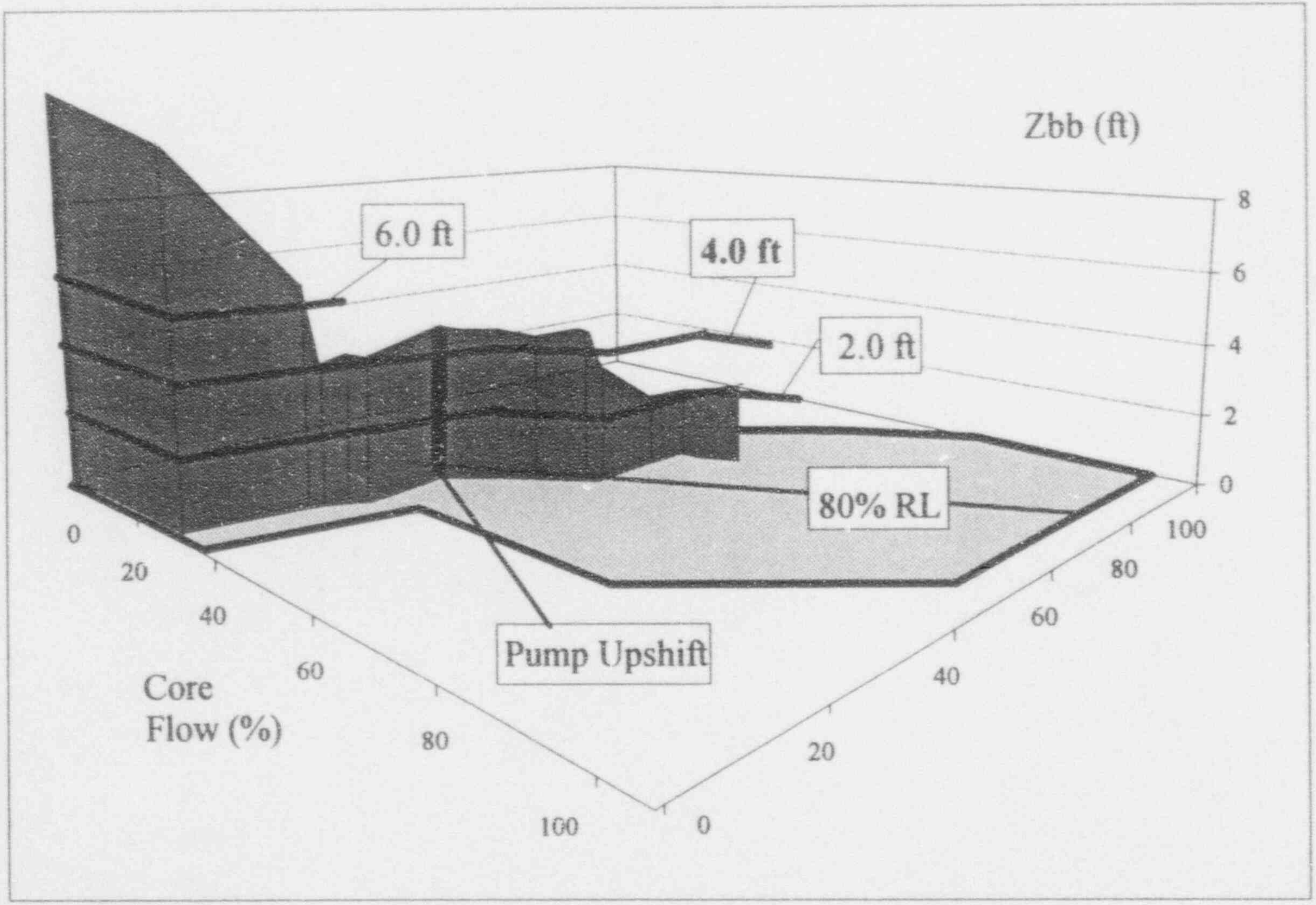


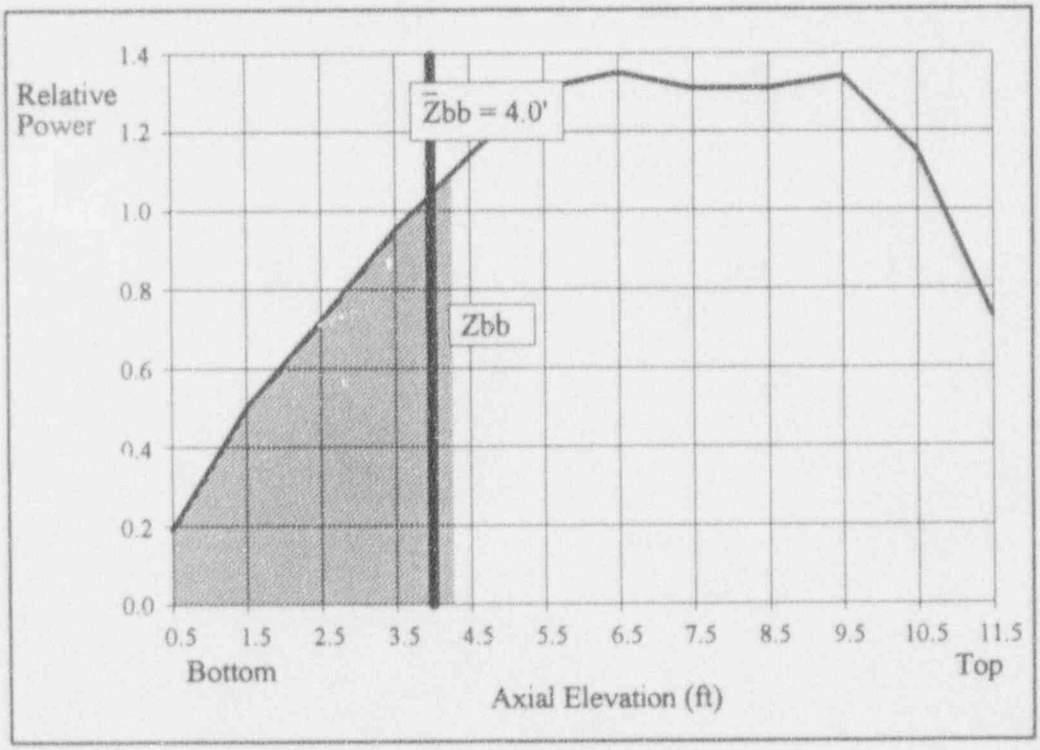
Figure 9-31: High Z_{bb} Control Rod Pattern

Control Rod Insertion Fractions

				----	----	----	----
			----	----	0.75	----	1.00
	----	0.13	----	0.63	----	0.38	----
	----	----	0.75	----	1.00	----	0.75
----	----	0.63	----	0.25	----	0.50	----
----	0.75	----	1.00	----	0.75	----	1.00
----	----	0.38	----	0.50	----	----	----
----	1.00	----	0.75	----	1.00	----	0.75

---- = full out

Figure 9-32: High Z_{bb} Axial Shape



10. REFERENCES

1. Licensing Topical Report, BWR Owner's Group Long-Term Stability Solutions Licensing Methodology, NEDO-31960, GE Nuclear Energy, June 1991.

2. Licensing Topical Report, BWR Owner's Group Long-Term Stability Solutions Licensing Methodology (Supplement 1), NEDO-31960 Supplement 1, GE Nuclear Energy, March 1992.

3. Letter, Ashok C. Thadani (USNRC) to L.A. England (BWROG), Acceptance for Referencing of Topical Reports NEDO-31960 and NEDO-31960 Supplement 1, "BWR Owner's Group Long-Term Stability Solutions Licensing Methodology" (TAC NO. M75928), July 12, 1993.

APPENDIX A: INITIAL APPLICATION PROCEDURE

This appendix contains the procedure for implementing a plant-specific initial application for Enhanced Option I-A, as described in Section 7. The initial application procedure can be divided into two parts: (1) generation of the plant-specific stability region boundaries, and (2) validation of the region boundaries.

A.1 Region Boundaries Generation

A.1.1 Standard Cycle Baseline

A.1.1.1 Standard Cycle Design

Complete a plant-specific Standard Cycle (SC) design by determining the plant-specific values of the parameters listed in Appendix C, Table C-1. This set of core design parameters, together with the generic fuel assembly design described in Appendix C, Table C-2 and Figures C-1 through C-4 constitutes a completely-specified fuel and core design necessary for the region boundary generation procedure.

A.1.1.2 FABLE SC Analysis

Apply the FABLE/BYPSS licensing procedure to the SC design generated in Step A.1.1.1 for the state points listed in Table A-1. The results of the analysis, DR_{SC}^{Lic}, are one pair of core and hot channel decay ratios for each of the eight state points listed in Table A-1.

A.1.2 Reference Cycle Performance

A.1.2.1 Reference Cycle Design

Prepare a plant-specific Reference Cycle (RC) design based on end-of-equilibrium-cycle (EOEC) Haling depletion conditions. The RC should be designed to envelope anticipated future cycle designs. Make appropriate adjustments, as described in Table A-2, to account for possible variations in future fuel and core designs to minimize the need for future changes in the region boundaries or region validation analyses. Avoid adjusting the RC design such that unwarranted conservatism is introduced, resulting in overly large stability regions.

A.1.2.2 Best-Estimate SC Analysis

Adjust loss coefficients, listed in Appendix C, Tables C-1 and C-2, to match the EOEC Haling core pressure drop and flow path splits established in Step A.1.1.1. Use a qualified best-estimate stability code to analyze the SC design generated in Step A.1.1.1 at the state point conditions listed in Table A-3. The results of the analysis are one pair of core and hot channel decay ratios for each of the three state points listed in Table A-3.

If the decay ratio range does not span the range of the FABLE SC decay ratio data generated in Step A.1.1.2, perform additional analysis at the optional Xenon-free state point conditions listed in Table A-3.

A.1.2.3 Best-Estimate RC Analysis

Use the qualified best-estimate stability code to analyze the design generated in Step A.1.2.1 at the same state point conditions analyzed in Step A.1.2.2. The results of the analysis are one pair of core and hot channel decay ratios for each of the state points analyzed.

A.1.2.4 RC Decay Ratio Bias

Determine a bias function $F_{Bias}(DR_{SC})$ as a function of SC decay ratio for both core decay ratio and hot channel decay ratio. Compute the decay ratio difference, ΔDR_{Bias}, between the results of the best-estimate SC and RC analyses obtained from Steps A.1.2.2 and A.1.2.3 using Equation A-1,

$$\Delta DR_{Bias}(DR_{SC}) = DR_{RC} - DR_{SC} + DR_{Add} \qquad \text{(A-1)}$$

where DR_{Add} is an arbitrary decay ratio allowance which may be used to address potential stability trends resulting from design changes in future reloads.

Generate two continuous $F_{Bias}(DR_{SC})$ functions, one for core decay ratio and one for hot channel decay ratio, relating RC-to-SC differences as a function of SC decay ratio. The function encompasses the decay ratio values calculated by Equation A-1.

If additional resolution of the $\Delta DR_{Bias}(DR_{SC})$ values for core decay ratio or hot channel decay ratio is desired for the generation of the F_{Bias} function, the SC and RC designs may be analyzed with the qualified best-estimate stability code at additional state points consistent with the state points and conditions listed in Table A-3.

A.1.2.5 FABLE-Based RC Decay Ratio

Determine the FABLE-based stability performance of the RC design using the FABLE SC decay ratio data generated in Step A.1.1.2, DR_{SC}^{Lic}, and the F_{Bias} function obtained in Step A.1.2.4,

$$DR_{RC}^{Lic} = DR_{SC}^{Lic} + F_{Bias}(DR_{SC}) \tag{A-2}$$

The result, DR_{RC}^{Lic}, is one pair of core and hot channel decay ratios for the RC design at each of the eight state points listed in Table A-1 based on the FABLE SC decay ratio results of Step A.1.1.2.

A.1.3 Nominal Region Boundaries

A.1.3.1 Decay Ratio Mapping

Plot the RC natural circulation and high flow-control line decay ratio data, DR_{RC}^{Lic}, obtained in Step A.1.2.5 onto a map of core versus hot channel decay ratio, which includes the boundary generation stability criteria listed in Table A-4.

A.1.3.2 Boundary Generation Stability Criteria Intercepts

Identify the decay ratio coordinates of the intersections of the lines connecting state points along the natural circulation line and the high flow-control line with the three boundary generation stability criteria. For each stability criterion intercept, interpolate to find the relative location between the two adjacent analysis state points.

A.1.3.3 Region Boundary Intercepts

Based on the interpolated values obtained in Step A.1.3.2, determine the corresponding coordinates along the natural circulation line and the high flow-control line that lie between the same analysis state points in the power/flow map.

A.1.3.4 Region Boundaries Determination

Apply the generic region boundary shape function, Equation A-3, to the region boundary intercepts identified in Step A.1.3.3, for each of the three regions,

$$P = P_y \left(\frac{P_x}{P_y} \right)^{\frac{1}{2}\left[\frac{W - W_y}{W_x - W_y} + \left(\frac{W - W_y}{W_x - W_y} \right)^2 \right]} \qquad \text{(A-3)}$$

where P_x and W_x are core power and flow coordinates of the high flow-control line region boundary intercepts (e.g., A, A') and P_y and W_y are the coordinates of the region boundary intercepts along the natural circulation line (e.g., B, B').

A.1.3.5 Final Exclusion Region Boundary

Apply a 40% rated core flow clamp to the Exclusion Region boundary obtained in Step A.1.3.4 to obtain the final form of the Exclusion Region boundary.

A.1.3.6 Nominal Setpoints Determination

Determine the stability region setpoints based on the nominal region boundaries, the existing setpoints for core flows above the stability region boundaries, and the flow-biased neutron flux scram clamp above the Exclusion Region. The detailed formulation of the setpoints for the flow-biased neutron flux scram line and the control rod-block line is provided in Section 6.1.

A.2 Validation of New Region Boundaries

A.2.1 Generation of Analytical Boundaries

A.2.1.1 Setpoint Uncertainty Quantification

Quantify plant-specific uncertainties in core power for the nominal region boundaries of the Exclusion and Restricted Regions on the natural circulation line. Quantify plant-specific uncertainties in core flow for the region intercepts on the high flow-control line. The region intercepts are obtained from Steps A.1.3.3 and A.1.3.5.

A.2.1.2 Analytical Region Boundary Intercepts

Add the core power uncertainty obtained in Step A.2.1.1 to the Exclusion and Restricted Region boundary power intercepts with the natural circulation line obtained in Step A.1.3.3. Subtract the core flow uncertainty obtained in Step A.2.1.1 from the Restricted Region boundary flow and Exclusion Region flow clamp intercepts with the high flow-control line obtained in Steps A.1.3.3 and A.1.3.5, respectively.

A.2.2 Initial Validation Matrix Analysis

A.2.2.1 Current Cycle Design

Prepare a plant- and cycle-specific Current Cycle (CC) design.

A.2.2.2 State Point Conditions Simulation

Use a qualified three-dimensional steady-state BWR core simulator to determine the initial and final conditions for all state points listed in Tables A-5, A-6 and A-7. Perform the analysis at the analytical state points of Step A.2.1.2.

A.2.2.3 Steady-State Analysis

Use the qualified best-estimate stability code to analyze the CC design at the steady-state points listed in Table A-5 at the conditions listed in Table A-6. The results of the analysis are decay ratio values that quantify susceptability to core wide and regional modes of reactor instability at each of the three steady-state points listed in Table A-5.

A.2.2.4 LOFH Events Analysis

Use the qualified best-estimate stability code to analyze the CC design at the terminal points of the LOFH events listed in Table A-5 at the event termination conditions listed in Table A-7. The results of the analysis are decay ratio values that quantify susceptability to core wide and regional modes of reactor instability at each of the two LOFH termination points.

A.2.2.5 Intermediate Flow reduction Events Analysis

Use the qualified best-estimate stability code to analyze the CC design at the terminal points of the intermediate flow reduction events listed in Table A-5 at the event termination conditions listed in Table A-7. The results of the analysis are decay ratio values that quantify susceptability to core wide and regional modes of reactor instability at each of the two flow event termination points.

A.2.2.6 Validation of Region Boundaries

Compare the decay ratio results obtained in Steps A.2.2.3 through A.2.2.5 against the boundary validation stability criterion of the qualified best-estimate stability code. Determine whether all IVM points meet the criterion. If the criterion is not met, adjust the RC design in Step A.1.2.1 and repeat all subsequent steps.

Table A-1: FABLE SC Analysis State Points

State Point	Core Flow (% of Rated)	Core Power (% of Rated)
1	30	15
2	30	25
3	30	35
4	30	45
5	35	62
6	45	71
7	55	80
8	70	91

Table A-2: Reference Cycle Design Considerations

Reload Review Procedure Design Change Criteria	Possible RC Design Features
CC to RC reactor design modifications have no effect on stability	Reactor design features planned for future cycles
CC to RC fuel and channel mechanical design changes have no effect on stability	Fuel and channel design features anticipated in future cycles or arbitrary design features to address uncertainties in fuel design evolution
CC Haling radial peaking increase over RC by no more than 5%	Local fuel bundle shuffle increases radial peaking to accommodate expected increased in peaking in reload designs
CC reload batch size change relative to RC by no more than 5% of core size	Nominal batch size anticipated for future reloads
CC to RC full power cycle energy change within 10%	Nominal cycle energy anticipated for future cycles
Last cycle coastdown change relative to RC no more than 10% of full-power cycle energy	Nominal coastdown anticipated for future cycles
Last cycle energy shorter than RC by no more than 10%	Not applicable
No variation in CC mixed batch reload core relative to RC design	Allowance for transition cycles including radial peaking and mechanical design features
CC core loading strategy relative to RC unchanged	Conservative core loading, if practical
No new operating modes for CC relative to RC	New operating modes if degrade stability performance
Other differences between CC and RC do not result in equivalent loss of stability margin	Arbitrary bias to accommodate unanticipated future changes

Table A-3: Analysis Conditions for Standard to Reference Cycle Comparison

<table>
<tr><th>Core Flow</th><th>State Point Conditions</th><th>Optional Conditions</th></tr>
<tr><td>Natural circulation (N/C)</td><td rowspan="3">EOC Haling 100% flow-control line:
• Equilibrium feedwater temperature
• All rods out
• Rated Xenon</td><td rowspan="3">From EOC Haling 100% flow-control line:
• Equilibrium feedwater temperature
• All rods out
• Xenon-free</td></tr>
<tr><td>N/C + 10% of rated</td></tr>
<tr><td>N/C + 25% of rated</td></tr>
</table>

Table A-4: Boundary Generation Stability Criteria Coordinates

Exclusion Region		Restricted Region		Monitored Region	
DR_{ch}	DR_{core}	DR_{ch}	DR_{core}	DR_{ch}	DR_{core}
0.00	0.80	0.00	0.60	0.00	0.40
0.56	0.80	0.42	0.60	0.28	0.40
0.58	0.70	0.43	0.52	0.29	0.35
0.60	0.60	0.45	0.45	0.30	0.30
0.63	0.50	0.47	0.37	0.31	0.25
0.67	0.40	0.50	0.30	0.33	0.20
0.72	0.30	0.54	0.22	0.36	0.15
0.79	0.20	0.59	0.15	0.39	0.10
0.80	0.19	0.60	0.14	0.40	0.09
0.80	0.00	0.60	0.00	0.40	0.00

Table A-5: Initial Validation Matrix State Points

Category	Analytical State Point
Steady-State	B A' B'
Flow Events	From H1 and H0 to exclusion region analytical boundary
LOFH Events	A' B'

Table A-6: Initial Validation Initial State Conditions

Category	Analytical State Point	Initial State Conditions
All	All	EOC Haling. FWHOOS if applicable.
Steady-State	B	Based on H0 depletion. Xenon-free. Radially uniform control rods: deep at 00/08 with one rod per core quadrant adjusted for criticality, shallow for Z_{bb} control.
	A'	Based on H1 depletion. Equilibrium Xenon. Deep control rods at 00/08, with one rod per core quadrant adjusted for criticality, to compensate for lower Xenon.
	B'	Based on H0 depletion. Xenon-free. Radially uniform control rods: deep at 00/08 with one rod per core quadrant adjusted for criticality, shallow to control thermal limits.
Flow Events	H1	EOC Haling to H1.
	H0	EOC Haling to H0.
LOFH Events	A'	Same as A' steady-state.
	B'	Same as B' steady-state.

Table A-7: Initial Validation Final State Conditions

Category	Analytical State Point	Final State Conditions
Steady-State	B A' B'	Same as initial conditions.
Flow Events	H1 H0	W_{fr} core flow. Initial T_{fw}. Initial Xenon.
LOFH Events	A' B'	T_{fw} - 60°F equivalent rated. initial Xenon.

APPENDIX B: RELOAD REVIEW PROCEDURE

This appendix contains the procedure for implementing a plant-specific reload review for Enhanced Option I-A as described in Section 8. The reload review process can be divided into three parts: (1) assessment of reactor design changes; (2) assessment of fuel assembly design changes; and (3) assessment of fuel cycle design changes. These assessments constitute the validation of existing region boundaries for the reload cycle. Together with the initial application procedure of Appendix A, Appendix B prescribes all the analysis requirements necessary for the reload review procedure.

B.1 Reactor Design Modifications

B.1.1 Determination if Stability Performance Affected

Assess reactor design modifications in the reload Current Cycle (CC) design against Criteria A.1, A.2 and A.3 of Table B-1.

Situation 1: Reactor design modifications are unrelated to core stability performance; the assessment of the reload CC reactor design modifications is complete. No reload analysis is required due to the reactor design modifications. Proceed to Section B.2.

Situation 2: Reactor design change can potentially affect core stability performance; the effects on stability performance must be quantified. Proceed to Section B.1.2.

B.1.2 Quantification of Effect on Stability Performance

B.1.2.1 Modified RC Design

Generate a modified Reference Cycle (RC) design based on the existing plant-specific RC design, incorporating the changes associated with the reactor design modification. The modified RC design is based on end-of-equilibrium-cycle (EOEC) Haling depletion conditions.

B.1.2.2 Best-Estimate Modified RC Analysis

Use a qualified best-estimate stability code to analyze the modified RC design generated in Step B.1.2.1 at the state point conditions listed in Table B-2 that were used in the existing RC design analysis. (The analysis for the existing RC design includes the core and hot channel decay ratios, DR_{RC}, and the decay ratio allowance, ΔDR_{Add}.) The results of the analysis are one pair of core and hot channel decay ratios (DR_{RC}^{Mod}) for each of the state points analyzed.

B.1.3 Region Boundaries Assessment

Situation 1: $DR_{RC}^{Mod} > DR_{RC} + 0.3$ (B-1)

The existing SC design is not bounding; follow the full procedure of Appendix A to perform a new region boundary generation and IVM validation analysis, including a FABLE analysis of a new SC design incorporating the reactor design modifications.

Situation 2: $DR_{RC}^{Mod} > DR_{RC} + \Delta DR_{Add}$, and

$DR_{RC}^{Mod} \leq DR_{RC} + 0.3$ (B-2)

The existing RC design is not bounding; follow the procedure described in Appendix A beginning with Section A.1.2 to perform a new region boundary generation and IVM validation analysis for the new RC design incorporating the reactor design modifications.

Situation 3: $DR_{RC}^{Mod} > DR_{RC}$, and

$DR_{RC}^{Mod} \leq DR_{RC} + \Delta DR_{Add}$, and

$DR_{RC}^{Mod} \leq DR_{RC} + 0.3$ (B-3)

The existing RC design is bounding; perform a RVM analysis to validate the existing region boundaries using the procedure established in Section B.4. The RVM analysis is performed for the reload CC design incorporating the reactor design modifications.

Situation 3.1: RVM analysis confirms existing region boundaries; proceed to Section B.2.

Situation 3.2: RVM analysis does not confirm existing region boundaries; the existing RC design is not bounding. Follow the procedure described in Appendix A beginning with Section A.1.2 to revise the RC design by incorporating the reactor design modifications, generate new region boundaries and perform an IVM validation analysis.

Situation 4: $$DR_{RC}^{Mod} \leq DR_{RC} \tag{B-4}$$

The reactor stability performance is not affected; the reactor design modification does not degrade stability performance. Region boundary validation is not required.

B.2 New Fuel Design

B.2.1 Determination if Stability Performance Affected

Assess the fuel design changes in the reload CC design against criteria B.1 and B.2 of Table B-1.

Situation 1: Fuel design changes are unrelated to core stability performance; the assessment of the reload CC reactor design modifications is complete. No reload analysis is required due to the fuel design changes. Proceed to Section B.3.

Situation 2: Design changes can potentially affect core stability performance; the effects on stability performance must be quantified. Proceed to Section B.2.2.

B.2.2 Quantification of Effect on Stability Performance

B.2.2.1 Modified RC Design

Generate a modified RC design based on the existing plant-specific RC design, incorporating the changes associated with the new fuel design. The modified RC design is based on EOEC Haling depletion conditions.

B.2.2.2 Best-Estimate Modified RC Analysis

Use a qualified best-estimate stability code to analyze the modified RC design generated in Step B.2.2.1 at the state point conditions listed in Table B-2 that were used in the existing RC design analysis. (The analysis for the existing RC design includes the core and hot channel decay ratios, DR_{RC}, and the decay ratio allowance, ΔDR_{Add}.) The results of the analysis are one pair of core and hot channel decay ratios (DR_{RC}^{Mod}) for each of the state points analyzed.

B.2.3 Region Boundaries Assessment

Situation 1: $DR_{RC}^{Mod} > DR_{RC} + \Delta DR_{Add}$ (B-5)

The existing RC design is not bounding; follow the procedure described in Appendix A beginning with Section A.1.2 to perform a new region boundary generation and IVM validation analysis for the new RC design incorporating the new fuel design.

Situation 2: $DR_{RC}^{Mod} > DR_{RC}$, and

$DR_{RC}^{Mod} \leq DR_{RC} + \Delta DR_{Add}$. (B-6)

The existing RC design is bounding; perform a RVM analysis to validate the existing region boundaries using the procedure established in Section B.4. The RVM analysis is performed for the reload CC design incorporating the fuel design changes.

Situation 2.1: RVM analysis confirms existing region boundaries; proceed to Section B.3.

Situation 2.2: RVM analysis does not confirm existing region boundaries; the existing RC design is not bounding. Follow the procedure described in Appendix A beginning with Section A.1.2 to revise the RC design by incorporating the new fuel design, generate new region boundaries and perform a IVM validation analysis.

Situation 3: $$DR_{RC}^{Mod} \leq DR_{RC} \tag{B-7}$$

The stability performance is not affected; the new fuel design does not degrade stability performance. Region boundary validation is not required.

B.3 New Fuel Cycle Design

B.3.1 Effect on Stability Performance

B.3.1.1 Reload Review Criteria

Determine if the fuel cycle design changes associated with the reload CC design violate any of the reload review criteria, C.1 through C.9, of Table B-1.

B.3.2 Region Boundary Assessment

Situation 1: Reload review criteria met; the stability performance is not affected. Assessment of the new fuel cycle is complete. No analysis is necessary to validate the existing region boundaries for the reload CC design.

Situation 2: Reload review criteria failed; validate the existing region boundaries. Follow the procedure of Section B.4 below for performing a RVM region boundary validation analysis of the existing region boundaries for the reload CC design.

Situation 2.1: RVM analysis confirms existing region boundaries; reload review procedure for Enhanced Option I-A is complete.

Situation 2.2: RVM analysis does not validate existing region boundaries; the existing RC design is not bounding. Follow the procedure described in Appendix A beginning with Section A.1.2 to revise the RC design by incorporating the fuel cycle design changes, generate new region boundaries and perform an IVM validation analysis.

B.4 Validation of Existing Region Boundaries

B.4.1 State Point Selection

B.4.1.1 Non-Limiting Transient Conditions

Use the results of the existing IVM analysis to identify the more limiting of the two flow events and the two LOFH events.

B.4.2 RVM Analysis

B.4.2.1 Reload Cycle Design

Prepare a plant- and cycle-specific CC design for the reload cycle.

B.4.2.2 State Point Conditions Simulation

Use a qualified three-dimensional steady-state BWR core simulator to determine the initial and final conditions for the state points listed in Tables B-3, B-4 and B-5. Perform the analysis at the analytical state points used for the existing IVM analysis.

B.4.2.3 Steady-State Analysis

Use the qualified best-estimate stability code to analyze the reload CC design at the steady-state points listed in Table B-3 at the conditions listed in Table B-4. The results of the analysis are decay ratio values that quantify susceptability to core wide and regional modes of reactor instability at each of the steady-state points analyzed.

B.4.2.4 Flow Events Analysis

Use the qualified best-estimate stability code to analyze the CC design for the limiting conditions, as established in Step B.4.1.1, of the flow events listed in Table B-3 at the event termination conditions listed in Table B-5. The results of the analysis are decay ratio values that quantify susceptability to core wide and regional modes of reactor instability at the limiting termination point.

B.4.2.5 LOFH Analysis

Use the qualified best-estimate stability code to analyze the CC design for the limiting conditions, as established in Step B.4.1.1, of the LOFH events listed in Table B-3 at the event termination conditions listed in Table B-5. The results of the analysis are decay ratio values that quantify susceptability to core wide and regional modes of reactor instability at the limiting termination point.

B.4.2.6 Validation of Region Boundaries

Compare the decay ratio results obtained in Steps B.4.2.3 through B.4.2.5 against the boundary validation stability criterion of the qualified best-estimate stability code. Determine whether all RVM points meet the criterion. If the criterion is not met, follow the procedure described in Appendix A beginning with Section A.1.2 and repeat all subsequent steps.

Table B-1: Reload Review Criteria and Analysis Requirements

Design Change	Reload Review Criteria	Analysis Required if Criteria Not Met
A. Reactor	1. Reactor design modification does not invalidate SC design $DR_{RC}^{Mod} \le DR_{RC} + 0.3$	• FABLE procedure • New RC design • New region boundaries • IVM
	2. Reactor design modification does not invalidate RC design $DR_{RC}^{Mod} \le DR_{RC} + \Delta DR_{Add}$	• New RC design • New region boundaries • IVM
	3. Reactor design modification has no negative effect on stability $DR_{RC}^{Mod} \le DR_{RC}$	• RVM If RVM fails: • New RC design • New region boundaries • IVM
B. New Fuel	1. New fuel design does not invalidate RC $DR_{RC}^{Mod} \le DR_{RC} + \Delta DR_{Add}$	• New RC design • New region boundaries • IVM
	2. New fuel design has no negative effect on stability $DR_{RC}^{Mod} \le DR_{RC}$	• RVM If RVM fails: • New RC design • New region boundaries • IVM
C. Fuel Cycle	1. CC Haling radial peaking increase over RC by no more than 5% 2. CC reload batch size change relative to RC no more than 5% of core size 3. CC to RC full power cycle energy change within 10% 4. Last cycle coastdown change relative to RC no more than 10% of full power cycle energy 5. Last cycle energy shorter than RC by no more than 10% 6. No variations in CC mixed batch reload core relative to RC 7. CC core loading strategy relative to RC unchanged 8. No new operating mode for CC relative to RC 9. Other difference between CC and RC do not result in equivalent loss of stability margin	• RVM If RVM fails: • New RC design • New region boundaries • IVM

Table B-2: Modified Reference Cycle Analysis Conditions

Core Flow	State Point Conditions	Optional Conditions
Natural circulation (N/C)	EOC Haling 100% flow-control line: • Equilibrium feedwater temperature • All rods out • Rated Xenon	From EOC Haling 100% flow-control line: • Equilibrium feedwater temperature • All rods out • Xenon-free
N/C + 10% of rated		
N/C + 25% of rated		

Table B-3: Reload Validation Matrix State Points

Category	State Point
Steady-State	A' and B'.
Flow Events	Limiting event based on IVM: From H0 or H1 to Exclusion Region analytical boundary.
LOFH Events	Limiting event based on IVM: A' or B'.

Table B-4: Reload Validation Initial State Conditions

Category	State Point	Initial State Conditions
All	All	EOC Haling. FWHOOS if applicable.
Steady-State	A'	Based on H1 depletion. Equilibrium Xenon. Deep control rods at 00/08, with one rod per core quadrant adjusted for criticality, to compensate for lower Xenon.
	B'	Based on HO depletion. Xenon-free. Radially uniform control rods: Deep at 00/08 with one rod per core quadrant adjusted for criticality, shallow to control thermal limits.
Flow Events	H0 or H1	EOC Haling to H0 or H1.
LOFH Events	A' or B'	Same as A' or B' Steady-State.

Table B-5: Reload Validation Final State Conditions

Category	Initial State Point	Final State Conditions
Steady-State	A' B'	Same as initial conditions.
Flow Events	H1 or H0	W_{fr} core flow. Initial T_{fw}. Initial Xenon.
LOFH Events	A' or B'	T_{fw} - 60°F equivalent rated. Initial Xenon.

APPENDIX C: STANDARD CYCLE SPECIFICATIONS

The Standard Cycle (SC) design consists of a completely-specified fuel and core design used to establish FABLE base-line decay ratio data and as a reference for the best-estimate RC performance analysis. The SC design consists of a generic fuel assembly design for all plant applications, and plant-specific core designs defined through a set of generic core design parameters. The common fuel assembly design is the only fuel-dependent input to the FABLE analysis. The SC design is completely specified for use in any qualified best-estimate stability code.

The SC fuel design is a generic 8x8 fuel assembly with two water rods, typical clad and fuel pellet dimensions and properties, axially-uniform fuel enrichment and gadolinia concentration, and upper and lower natural uranium blankets.

The SC core design consists of plant-specific inputs for each plant application. The design approach is generic, however, and consists of a target 18-month equilibrium cycle using a Haling depletion at rated conditions with a one-third-core reload batch size. It is expected that actual length of the plant-specific cycle design will vary somewhat from plant to plant.

The SC design is completely specified in Tables C-1 and C-2. Table C-2 provides a complete list of the SC fuel assembly design parameters and their values. Changes to the parameter list or changes in the values of specific parameters are not anticipated and will be made only for modeling consistency and accuracy. Any such changes to the SC design specifications will be documented and justified in plant-specific Enhanced Option I-A licensing submittal. Figure C-1 shows the axial fuel assembly configuration. Figures C-2 through C-4 provide the two-dimensional radial fuel pin patterns for the three lattice designs.

Table C-1 provides a list of the SC core design parameters. Values for plant-specific parameters are not provided and will be established separately for each plant. Figure C-5 is an example core loading map specifying the equilibrium cycle fuel loading pattern, including load, shuffle, and discharge fuel moves.

The SC fuel and core design is fully specified so that any utility or fuel vendor can perform the required initial application and reload review analysis with their qualified methods. The SC design is defined such that the comparison between the SC and RC designs is not affected by the choice of qualified stability methods. This ensures that differences in decay ratio performance between the two designs are strictly a function of changes in fuel design and not due to differences in the application or choice of methods.

The SC fuel bundle design includes unspecified hydraulic loss coefficients for the water rod inlet, the fuel rod spacers and the out-of-channel bypass inlet. These coefficients are determined by forcing the SC EOEC rated Haling core simulation to match the SC design specifications for the core pressure drop and the active channel, water rods, and out-of-channel bypass flow rates. Values for all core state parameters (e.g., rated core power, rated core flow, rated inlet enthalpy, rated core pressure) are plant-specific.

After the unspecified hydraulic loss coefficients are established, the SC is fully specified. The methods applied to the SC design should be the same as those applied to the RC design when establishing the RC to SC decay ratio bias correction. This is necessary to ensure that the decay ratio bias correction reflects changes in fuel design only, and is not an artifact of the methods used to perform the analysis.

Table C-1: Standard Cycle Core Design Parameter List

Parameter	Value[1]
Cycle Depletion Method	Equilibrium Cycle Haling
Cycle Depletion Operating Conditions	Rated Power and Flow
Full Power Cycle Length	MWd/MTU
2-D Core Loading Map (quarter core symmetry/top left) - containing information on discharge, load, and previous location (same loading map applied to all cycles)	Figure C-5 (example)
Number of Fuel Assemblies in Core	unitless
Pressure Drop Loss Coefficients for Inlet Orifice:	
Central	unitless
Peripheral	unitless
Reference Flow Area for Inlet Orifice	10 in^2
Pressure Drop Loss Coefficients for Bypass Region:	
Inlet	Adjustable[2]
Friction and Local Losses excluding Inlet	0.0
Rated Core Pressure	psia
Rated Thermal Power	MWt
Rated Core Flow	Mlb/hr
Haling Rated Total Bypass Flow[3]	Mlb/hr
Haling Rated Total Water Rod Flow	Mlb/hr
Haling Rated Core Pressure Drop[4]	psi
Rated Core Inlet Enthalpy	Btu/lb

1. All values are plant-specific.
2. Adjusted to achieve Haling rated total bypass flow.
3. Includes all bypass flow paths, e.g., lower tie plate holes, channel/lower tie plate interface, all other leakage paths in fuel support structure and core plate, but excludes water rods.
4. Upstream of inlet orifice to upper tie plate exit.

Table C-2: Standard Cycle Bundle Design Parameter List

Parameter	Value[1]
Bundle Configuration	Figure C-1
2-D Bundle Lattice Maps	Figures C-2, C-3, C-4
Pellet Diameter	0.411 in
Fuel Rod Outer Diameter	0.483 in
Fuel Rod Wall Thickness	0.032 in
Fuel Rod Pitch	0.636 in
Water Rod Outer Diameter	0.591 in
Water Rod Wall Thickness	0.030 in
Channel Thickness	0.120 in
Channel Inside Width	5.215 in
Channel Inside Corner Radius	0.380 in
Assembly Pitch	6.0 in
Fuel Active Length: UO_2 Rods Gd_2O_3 Rods	150.0 in 144.0 in
Fuel Unheated, Rodded Length	12.0 in
Channel Length - Bottom of Active Fuel to Top of Channel	164.3 in
Fuel Pellet Density	96.0 %TD
Fuel Pellet Densification	0.0 %TD
Fuel Stack Density[2]	95.0 %TD
Gd_2O_3 Concentration in Gadolinia Rods	4.0 wt %
Fuel Plenum Volume: UO_2 Rods Gd_2O_3 Rods	1.2 in^3 2.0 in^3
Fuel Rod Helium Prepressurization Level	5 atm
Fuel Rod Helium Prepressurization Temperature	70 °F
Fuel Rod Fill Gas Type	He
Fuel Pellet Surface Roughness	30 μin AA
Fuel Cladding Inner Surface Roughness	20 μin AA
Initial U-234 Concentration in Fuel	0.0
Initial U-236 Concentration in Fuel	0.0

Table C-2: Standard Cycle Bundle Design Parameter List (continued)

Parameter	Value[1]
Cladding Material[3]	Zirc-2
Channel Material	Zirc-4
Impurity Level in All Materials	0 ppm
Single-Phase Irreversible Pressure Drop Loss Coefficients:	
Lower Tie Plate	3.44
Spacer	Adjustable[6]
Upper Tie Plate[4]	0.40
Reference Flow Area for Loss Coefficients	10 in^2
Number of Spacers[5]	7
Location of Spacers above Bottom of Active Fuel (inches)	18.9, 39.2, 59.3, 79.4, 99.6, 119.7, 139.8
Water Rod Loss Coefficients:	
Inlet K/A[2]	Adjustable[7]
Outlet K/A[2]	85 in^{-4}
Mean Axial Elevation of Water Rod Holes relative to Bottom of Active Fuel:	
Inlet	0.9 in
Outlet	158.1 in

1. All values are for generic application.
2. Includes chamfer, dishing, chipping, cracking, etc., but is not smeared.
3. Cladding is fully annealed.
4. Area expansion "reversible" velocity head effects not included.
5. Spacer and instrument tube not explicitly modeled in lattice calculations.
6. Adjusted to achieve Haling rated core pressure drop (see Table C-1).
7. Adjusted to achieve Haling rated total water rod flow (see Table C-1).

Figure C-1: Standard Cycle Bundle Axial Configuration

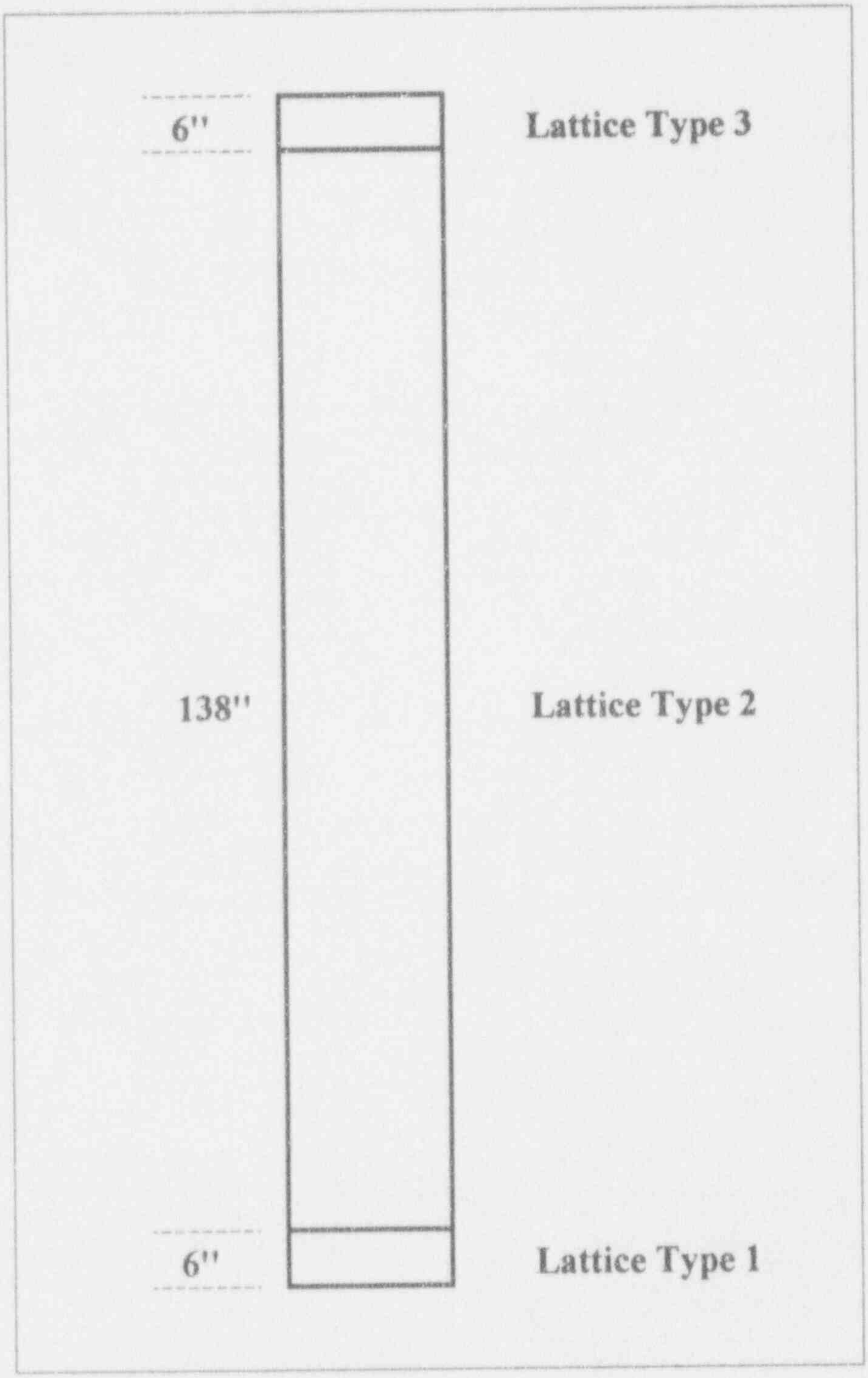


Figure C-2: Standard Cycle 2-D Bundle Map Lattice Type 1

	A	B	C	D	E	F	G	H
1	13	13	13	13	13	13	13	13
2	13	13	13	13	13	13	13	13
3	13	13	13	13	13	13	13	13
4	13	13	13	11	13	13	13	13
5	13	13	13	13	11	13	13	13
6	13	13	13	13	13	13	13	13
7	13	13	13	13	13	13	13	13
8	13	13	13	13	13	13	13	13

Rod ID	Rod Type	U-235 w%	Gad w%	No. of Rods
13	UO2	0.711	0.00	62
11	H2O	0.000	0.00	2

Figure C-3: Standard Cycle 2-D Bundle Map Lattice Type 2

	A	B	C	D	E	F	G	H
1	1	2	3	4	5	5	4	6
2	2	3	7	8	8	12	10	4
3	3	7	8	8	8	8	12	5
4	4	8	8	11	8	8	8	9
5	5	8	8	8	11	8	8	9
6	5	12	8	8	8	8	12	5
7	4	10	12	8	8	12	10	4
8	6	4	5	9	9	5	4	6

Rod ID	Rod Type	U-235 w%	Gad w%	No. of Rods
1	UO2	2.00	0.00	1
2	UO2	2.40	0.00	2
3	UO2	2.80	0.00	3
4	UO2	3.00	0.00	8
5	UO2	3.40	0.00	8
6	UO2	2.20	0.00	3
7	UO2	3.80	0.00	2
8	UO2	3.95	0.00	22
9	UO2	3.60	0.00	4
10	UO2	3.20	0.00	3
11	H2O	0.00	0.00	2
12	GD2O3	3.60	4.00	6

Figure C-4: Standard Cycle 2-D Bundle Map Lattice Type 3

	A	B	C	D	E	F	G	H
1	13	13	13	13	13	13	13	13
2	13	13	13	13	13	14	13	13
3	13	13	13	13	13	13	14	13
4	13	13	13	11	13	13	13	13
5	13	13	13	13	11	13	13	13
6	13	14	13	13	13	13	14	13
7	13	13	14	13	13	14	13	13
8	13	13	13	13	13	13	13	13

Rod ID	Rod Type	U-235 w%	Gad w%	No. of Rods
13	UO2	0.711	0.00	56
11	H2O	0.000	0.00	2
14	VOID	0.000	0.00	6

Figure C-5: Standard Cycle Core Loading Map (example)

D = Discharge Location
F = Fresh Fuel Location
I J = Previous Cycle Coordinates

J/I	1	2	3	4	5	6	7	8	9	10	11	12	13	14
1											D 12 6	D 4 12	D 4 14	D 6 7
2								D 1 12	D 12 4	D 6 8	8 9	12 11	11 8	8 13
3							D 6 12	13 6	7 12	10 13	9 12	F	11 12	2 13
4						D 14 6	12 7	5 9	F	13 14	F	6 6	F	10 4
5					D 8 10	3 9	13 8	F	12 3	F	5 14	F	4 13	F
6				D 6 14	13 2	12 13	9 13	11 13	F	9 6	9 3	5 7	F	10 3
7			D 7 10	9 8	12 9	13 9	D 11 5	F	8 7	3 10	D 9 7	F	10 5	D 9 9
8		D 11 6	14 5	9 5	F	13 11	F	D 11 9	F	D 12 2	F	D 5 11	F	D 7 13
9		D 10 7	8 11	F	3 12	F	7 8	F	12 5	F	4 11	F	4 9	F
10		D 8 6	14 9	14 13	F	6 9	3 11	13 3	F	D 8 3	D 6 10	D 2 11	F	D 6 5
11	D 6 11	6 13	9 14	F	5 12	7 5	D 7 9	F	11 4	D 9 11	D 5 6	F	8 5	D 5 13
12	3 14	13 12	F	2 12	F	11 3	F	D 13 5	F	D 11 2	F	D 4 7	F	D 7 4
13	D 14 4	9 10	13 10	F	13 4	F	5 10	F	9 4	F	5 8	F	D 3 8	F
14	D 7 6	10 9	2 14	4 10	F	3 13	D 4 8	D 10 6	F	D 8 4	D 13 7	D 14 2	F	D 14 3

APPENDIX D: CONCEPTUAL TECHNICAL SPECIFICATIONS

D.1 Introduction

Implementation of the Enhanced Option I-A stability solution results in the addition of several new Technical Specification requirements. The purpose of this appendix is to conceptually describe how the relevant features of the Enhanced Option I-A stability solution will be reflected in Technical Specifications. The Conceptual Technical Specifications are presented in the improved Technical Specification format. They have not yet been optimized and are included for illustration only.

Actual plant revised Technical Specifications will be included with plant-specific documentation associated with implementing the Enhanced Option I-A stability solution. Plant-specific applications will include a **license condition** for plant procedures to require operator action to immediately place the mode switch in the shutdown position upon PBDS Hi-Hi Decay Ratio alarm initiation.

The Technical Specification modifications will include the following:

A. Definitions of new concepts introduced as a result of adding Enhanced Option I-A.

B. Boundary setpoints for the Exclusion and Restricted Regions, including normal and setup setpoints for both one-loop and two-loop recirculation system operation.

C. Actions associated with operation in or near the stability region boundaries, including setup and setdown of the flow-biased APRM flux scram and control rod block setpoints.

D. Requirements for determining and complying with the Fraction of Core Boiling Boundary (FCBB) stability control, including required actions when FCBB limits are not met.

E. Instrumentation requirements for the Period-Based Detection System (PBDS) and the subsystem that provides the flow-biased APRM flux scram reference and control rod block reference to the neutron monitoring system.

D.2 Summary of Conceptual Requirements

D.2.1 Definitions

Definitions of new concepts introduced to the Technical Specifications by the Enhanced Option I-A stability solution will be included in Section 1.1, Definitions. These will include "Controlled Operation in Restricted Region", "Uncontrolled Entry to Restricted Region", and "Uncontrolled Operation in the Restricted Region".

D.2.2 Reactivity Control Systems

One new subsection will be added to Technical Specification Section 3.1, Reactivity Control Systems, to address the FCBB.

The Limiting Condition for Operation (LCO) for the FCBB section will specify that FCBB must be less than or equal to 1.0 during controlled operation in the Restricted Region. If the LCO is not met, FCBB must be restored to within limits or the Restricted Region must be exited. Surveillance requirements will verify that FCBB is met upon controlled entry to the Restricted Region and periodically during controlled operation inside the Restricted Region. Additional requirements are specified for uncontrolled entry and operation in the Restricted Region.

D.2.3 Instrumentation

The existing sections of the Technical Specification which address Reactor Protection System (RPS) instruments and control rod block instruments will be updated as a result of the new flow-biased setpoints of the Enhanced Option I-A stability solution. In addition, a new section will be added to the Instrumentation section to address the Period-Based Detection System (PBDS) instrumentation.

The APRM flow-biased trip reference is already included in the RPS Instrumentation section of the Technical Specifications. The allowable values for the flow-biased trip setpoints will be identified in the associated instrument table, by reference to the Core Operating Limits Report (COLR). The table will require setpoints for the nominal trip line and the setup trip line, for both one-loop and two-loop recirculation system operation. The LCO and specified actions associated with the instrument channels providing the trip reference will not change from current Technical Specifications. Surveillance requirements will require periodic checks, functional testing, and calibration.

The flow-biased control rod block trip reference must be added to the existing Technical Specification for control rod block instrumentation. The allowable values for the flow-biased rod block setpoints will be identified in the associated instrument table, by reference to the COLR. The table will require setpoints for the nominal rod block and the setup rod block, for both one-loop and two-loop recirculation system operation. The LCO and specified actions associated with the instrument channels providing the control rod block will require an exit from the Monitored Region if the required number of channels are not operable, and restoration of the control rod block function. Otherwise, the reactor must be shutdown. In addition the FCBB limit must be enforced until the control rod block function is returned to the normal setpoint. Surveillance requirements will require periodic functional testing and calibration. Verifying that the control rod block alarm is operable will be required as part of the functional testing.

The LCO for the PBDS instrument will require that one channel of the PBDS be operable whenever the plant is operating in the Restricted or Monitored Regions. If the PBDS becomes inoperable, the regions must be exited. If the PBDS is inoperable when there is an uncontrolled entry to the Restricted Region, an immediate manual scram is required. Surveillance requirements will require periodic functional testing and calibration. Verifying that the required number of LPRMs are operable will be required as part of the functional testing. Verifying PBDS period count sensitivity will be required as part of the calibration procedure.

D.2.4 Reactor Coolant System

The existing section of the Technical Specification which addresses operating recirculation loops will be updated to reflect the correct reference to the APRM flow-biased flux trip and flow-biased control rod block setpoints for single-loop operation. In addition, all references to existing stability-related surveillance requirements and LCO actions will be deleted.

D.3 Conceptual Outline of Revised Technical Specifications Associated with Stability Solution Enhanced Option I-A

Conceptual Technical Specification additions and changes that result from implementing the Enhanced Option I-A stability solution are provided for illustration only in the Improved Technical Specification format. Sections included are as follows:

1.1 Definitions

3.1.10 Fraction of Core Boiling Boundary

3.3.1.1 Reactor Protection System Instrumentation

3.3.1.3 Period-Based Detection System Instrumentation

3.3.2.1 Control Rod Block Instrumentation

3.4.1 Recirculation Loops Operating

The conceptual outline of the revised Technical Specifications associated with the Enhanced Option I-A stability solution follows.

Conceptual Outline of Revised Technical Specifications Associated With the Enhanced Option I-A Stability Solution

1.1 DEFINITIONS

The following definitions are to be added:

CONTROLLED OPERATION IN RESTRICTED REGION	CONTROLLED OPERATION IN RESTRICTED REGION shall be intentional, planned reactor power increases or core flow decreases that place or maintain the core average power and flow conditions within the Restricted Region of the licensed operating domain.
UNCONTROLLED ENTRY TO RESTRICTED REGION	UNCONTROLLED ENTRY TO RESTRICTED REGION shall be any unplanned or unintentional change in reactor power or core flow which results in entry to the Restricted Region.
UNCONTROLLED OPERATION IN RESTRICTED REGION	UNCONTROLLED OPERATION IN RESTRICTED REGION shall be any unplanned or unintentional change in reactor power or core flow that occurs completely within the Restricted Region.

3.1 REACTIVITY CONTROL SYSTEMS

3.1.10 Fraction of Core Boiling Boundary

LCO 3.1.10 Fraction of Core Boiling Boundary (FCBB) shall be less than or equal to 1.0.

APPLICABILITY: During operation in the Restricted Region.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Following UNCONTROLLED ENTRY TO RESTRICTED REGION	A.1 Initiate action to exit the Restricted Region	Immediately
B. During UNCONTROLLED OPERATION IN RESTRICTED REGION	B.1 Initiate action to exit the Restricted Region	Immediately
C. FCBB greater than 1.0 during operation in the Restricted Region.	C.1 Restore FCBB to less than or equal to 1.0.	15 minutes
D. Required Action and associated Completion Time of Condition C not met.	D.1 Exit the Restricted Region.	1 hour

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.1.10.1 Verify FCBB less than or equal to 1.0.	Prior to recirculation pump upshift operation AND 24 hours

3.3 INSTRUMENTATION

3.3.1.1 Reactor Protection System Instrumentation

[Reviewer's Note: There is no change to the LCO, Applicability, Surveillance Requirements or Actions specification of the Standard Improved Technical Specifications. The LCO and Applicability section are repeated for convenience. Changes to Table 3.3.1.1-1 are specified below. Verifying the automatic setdown setpoints will be included in the CHANNEL CALIBRATION.]

LCO 3.3.1.1 The RPS instrumentation for each Function in Table 3.3.1.1-1 shall be OPERABLE.

APPLICABILITY: According to Table 3.3.1.1-1.

Update the following in Table 3.3.1.1-1:

Table 3.3.1.1-1

REACTOR PROTECTION SYSTEM INSTRUMENTATION

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITION REFERENCED FROM REQUIRED ACTION D.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
2. Average Power Range Monitors (APRM):					
a. Two-loop operation	1 (b)	[3]	G	SR 3.3.1.1.1 SR 3.3.1.1.9 SR 3.3.1.1.13	As specified in COLR
b. Two-loop operation, Setup	(a)	[3]	G	SR 3.3.1.1.1 SR 3.3.1.1.9 SR 3.3.1.1.13	As specified in COLR
c. Single-loop operation	1 (b)	[3]	G	SR 3.3.1.1.1 SR 3.3.1.1.9 SR 3.3.1.1.13	As specified in COLR
d. Single-loop operation, Setup	(a)	[3]	G	SR 3.3.1.1.1 SR3.3.1.1.9 SR 3.3.1.1.13	As specified in COLR

(a) During CONTROLLED OPERATION IN RESTRICTED REGION.
(b) Not required during CONTROLLED OPERATION IN RESTRICTED REGION.

[Reviewers Note: Allowable Values in the COLR will be those specified in Table 6-2, 6-3, 6-4, and 6-5 of the Enhanced Option I-A LTR (NEDO-32339). Entry to the Applicable Modes or Other Specified Conditions will be allowed for a specified time period to perform the CHANNEL CALIBRATION if required by the instrument design.]

3.3 INSTRUMENTATION

3.3.1.3 Period-Based Detection System Instrumentation

LCO 3.3.1.3 One channel of the Period-Based Detection System (PBDS) instrumentation shall be OPERABLE.

APPLICABILITY: During operation in the Restricted Region.
During operation in the Monitored Region.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One required channel of the PBDS inoperable during UNCONTROLLED ENTRY TO RESTRICTED REGION.	A.1 Place the mode switch in the shutdown position.	Immediately
B. One required channel of the PBDS inoperable during CONTROLLED OPERATION IN RESTRICTED REGION.	B.1 Restore the PBDS to OPERABLE status.	15 minutes
C. Required Action and associated Completion Time of Condition B not met.	C.1 Exit the Restricted Region.	1 hour
D. One required channel of the PBDS inoperable during operation in the Monitored Region.	D.1 ---------NOTE--------- Not applicable for 6 hours when FCBB < 1.0 for the purpose of reactor shutdown. ------------------------- Restore the PBDS to OPERABLE status.	15 minutes

E. Required Action and associated Completion Time of Condition D not met.	E.1 ---------NOTE---------- Not applicable for 6 hours when FCBB < 1.0 for the purpose of reactor shutdown. -------------------------- Exit the Monitored Region.	 1 hour

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.3.1.3.1	Perform a CHANNEL FUNCTIONAL TEST.	7 days during operation in the Restricted Region and operation in the Monitored Region <u>AND</u> [92] days
SR 3.3.1.3.2	Perform a CHANNEL CALIBRATION.	[18] months

[Reviewer's Note: Verifying that the required number of LPRM inputs are OPERABLE will be included in the CHANNEL FUNCTIONAL TEST. Verifying PBDS period count sensitivity will be included in the CHANNEL CALIBRATION.]

3.3 INSTRUMENTATION

3.3.2.1 Control Rod Block Instrumentation

[Reviewer's Note: There is no change to the LCO and Applicability sections of the Standard Improved Technical Specifications. The LCO and Applicability section are repeated for convenience. Additions to the Actions, Surveillance Requirements, and Table 3.3.2.1-1 are specified below.]

LCO: The control rod block instrumentation channels for each Function in Table 3.3.2.1-1 shall be OPERABLE.

APPLICABILITY: According to Table 3.3.2.1-1.

Add as follows:

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. One required APRM channel inoperable.	D.1 Restore required APRM channels to OPERABLE status.	7 days
E. Required Action and associated Completion Time of Condition D not met.	E.1 Place channel in trip.	1 hour
D. Two or more required APRM channels inoperable.	F.1 Place channel in trip.	1 hour

Add as follows:

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.3.2.1.x	Perform a CHANNEL FUNCTIONAL TEST.	[92] days
SR 3.3.2.1.y	Perform a CHANNEL CALIBRATION.	[18] months

[Reviewer's Note: Verifying the flow-biased control rod block alarm is OPERABLE will be included in the CHANNEL FUNCTIONAL TEST. Verifying automatic setdown setpoints will be included in the CHANNEL CALIBRATION.]

Add as follows:

Table 3.3.2.1-1

CONTROL ROD BLOCK INSTRUMENTATION

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
3. APRM Control Rod Block				
a. Two-loop operation, flow-biased rod block.	1 (b)	[6]	SR 3.3.2.1.x SR 3.3.2.1.y	As specified in COLR
b. Two-loop operation, setup flow-biased rod block.	(a),(c)	[6]	SR 3.3.2.1.x SR 3.3.2.1.y	As specified in COLR
c. Single-loop operation, flow-biased rod block.	1 (b)	[6]	SR 3.3.2.1.x SR 3.3.2.1.y	As specified in COLR
d. Single-loop operation, setup flow-biased rod block.	(a),(c)	[6]	SR 3.3.2.1.x SR 3.3.2.1.y	As specified in COLR

(a) During CONTROLLED OPERATION IN RESTRICTED REGION.
(b) Not required during CONTROLLED OPERATION IN RESTRICTED REGION.
(c) FCBB ≤ 1.0

[Reviewer's Note: Allowable Values in the COLR will be those specified in Table 6-2, 6-3, 6-4, and 6-5 of the Enhanced Option I-A LTR (NEDO-32339). Entry to the Applicable Modes or Other Specified Conditions will be allowed for a specified time period to perform the CHANNEL CALIBRATION if required by the instrument design.]

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.1 Recirculation Loops Operating

[Reviewer's Note: The only change to the RCS section is the LCO for Recirculation Loops Operating. The unchanged sections are not repeated herein.]

LCO 3.4.1 Two recirculation loops with matched flows shall be in operation,

<u>OR</u>

One recirculation loop may be in operation provided the following limits are applied when the associated LCO is applicable:

a. [*no change*]

b. [*no change*]

c. LCO 3.3.1.1, "Reactor Protection System (RPS) Instrumentation," Function 2.c (Average Power Range Monitor Single-loop operation), or Function 2.d (Average Power Range Monitor Single-loop operation, Setup) is applied; and

d. LCO 3.3.2.1, "Control Rod Block Instrumentation," Function 2.c (APRM Control Rod Block Single-loop operation, flow-biased rod block) or Function 2.d (APRM Control Rod Block Single-loop operation, setup flow-biased rod block) is applied.

APPENDIX E: DEMONSTRATION PLANT ANALYSIS

This appendix presents stability analysis that has been performed for a demonstration plant to validate the methodology defining Enhanced Option I-A. The Perry plant was chosen as the demonstration plant and both Cycle 2 and a hypothetical equilibrium fuel cycle containing all GE 8x8 fuel were used in the evaluation. These are the same fuel cycles that were analyzed and presented in Reference 1. In this regard, the basis for the analysis presented below is consistent with that previously presented.

The analysis consists of feasibility studies performed for the demonstration plant to support the basis for the generic aspects of Enhanced Option I-A. It includes generation of the boundaries for the Exclusion Region, the Restricted Region and the Monitored Region with FABLE/BYPSS and validation of the boundaries with the best-estimate frequency domain stability code ODYSY. In addition, a demonstration of the initial application validation process and supporting results for the stability controls analysis are presented.

E.1 Demonstration Plant Feasibility Studies

Nominal region boundaries were determined for the equilibrium fuel cycle using the data presented in Reference 1 and the results of FABLE/BYPSS calculations performed at additional state points on the power/flow operating map. The demonstration analysis supports the selection of the defined set of power/flow state points used in the Enhanced Option I-A initial application procedure.

Feasibility studies were performed with the best-estimate stability code ODYSY to validate the nominal region boundaries of the demonstration plant. A Demonstration Validation Matrix (DVM) was developed which provides the basis for the selection of the analysis required in the Initial Validation Matrix (IVM) and the Reload Validation Matrix (RVM). The IVM and RVM are used in the initial application and reload review procedures. The analysis performed to define the DVM involved various iterative searches for worst-case stability conditions. Both the DVM and additional validation analysis are presented below.

E.1.1 Region Boundaries

The Region Boundary Definition Procedure presented in Reference 1 is expanded to define the nominal boundaries of the Exclusion Region, the Restricted Region and the Monitored Region. To determine the region boundaries of the demonstration plant, FABLE/BYPSS calculations were performed over a range of power/flow conditions to determine a line of constant stability margin as defined by the boundary generation stability criteria. A different criterion was applied to each region. Inputs and calculational procedures were chosen to collectively provide conservative results. Note that for this feasibility study the stability regions are defined based on Perry equilibrium cycle; the initial application procedure, including the Standard and Reference Cycle designs, was not exercised.

The points analyzed to determine the nominal region boundaries for the equilibrium cycle of the demonstration plant are shown in Figure E-1. Points 1, 2, 3, 7, 8 and 9 were presented in Reference 1 and the remainder are additional points at higher core flows along a high rod line (Points 4, 5 and 6) and at lower powers along the natural circulation line (Points 10 and 11). The additional points yield lower decay ratios that are required to define the Restricted and Monitored Regions.

The hot channel and core decay ratios computed at each point are shown in Table E-1. The decay ratios are compared to the boundary generation stability criteria in Figure E-2, where the numbered points correspond to the points in Figure E-1 and Table E-1. The intersection of the Exclusion Region with the high rod line is determined by interpolation between Points 2 and 3 and the intersection with the natural circulation line is determined by interpolation between Points 8 and 9. Likewise, the intercepts for the Restricted Region are determined by interpolation between Points 3 and 4 and Points 9 and 10. The intercepts for the Monitored Region are determined by extrapolation beyond Point 6 and interpolation between Points 10 and 11. Boundary intercepts are normally determined by interpolation, but extrapolation beyond Point 6 is acceptable because of its close proximity to the Monitored Region stability criterion.

The region boundaries are then determined by applying the generic shape function (Appendix F) to the intercepts of each boundary. The resulting region boundaries are shown in Figure E-3. The generic shape function was developed based on the existing Exclusion Region boundary database for several plants and the Exclusion Region presented in Figure E-3 shows good agreement with the one presented in Reference 1, Figure 5-19, for the Perry equilibrium cycle. The Exclusion Region is then clamped at 40% of rated core flow as discussed in Section 4.3.4 of this report. The final nominal region boundaries are shown in Figure E-4.

E.1.2 Region Boundary Validation

The boundaries of the Exclusion Region and the Restricted Region are an integral part of the licensing basis for Enhanced Option I-A and are validated with the best-estimate stability code ODYSY. The Monitored Region is part of the defense-in-depth of the solution and validation of its boundary is not necessary.

Validation analysis is performed at the analytical boundary of each region which takes into account setpoint uncertainty of the nominal boundaries. For the demonstration analysis, the setpoint uncertainty is assumed to be ~10% of measured signal, which at the region boundaries corresponds to 5% of rated core flow at high rod lines and 3% of rated core power at natural circulation. This magnitude of uncertainty is conservative relative to actual plant uncertainties. The 5% flow uncertainty is applied to the analytical boundary at the high rod line and the 3% power uncertainty is applied to natural circulation. This yields a more restrictive boundary at which to perform the validation analysis. The nominal and analytical boundaries for the demonstration plant are shown in Figure E-5 and the boundary intercepts are given in Table E-2.

Points A_{fr} and B in Figure E-5 and Table E-2 correspond to the endpoints of the Exclusion Region boundary at the high rod line and natural circulation, respectively, and Points A' and B' correspond to the endpoints of the Restricted Region boundary at the same positions. (Point A in Table E-2 corresponds to the intersection of the generic shape function with the high rod line and validation analysis performed at Point A is presented in Section E.1.4.) Demonstration of acceptable stability performance at the analytical region boundaries for steady-state and transient conditions validates the use of the nominal region boundaries for setpoint definition.

E.1.3 Demonstration Validation Matrix

Validation analysis was performed for the demonstration plant with the best-estimate stability code ODYSY at the analytical boundary of the Exclusion and Restricted Regions for steady-state conditions and for limiting transients. The limiting transients are Loss of Feedwater Heating (LOFH) and intermediate flow reduction from rated power. Iterative searches for worst-case stability conditions were made, considering

- core power,
- core flow,
- setpoint uncertainty,
- void coefficient,
- cycle exposure,
- axial power shape,
- radial peaking,
- Xenon conditions,
- feedwater heater out-of-service,
- thermal limits,
- realistic control rod patterns, and
- core average boiling boundary

to demonstrate the feasibility of the region boundary validation process and to establish the scope of the validation analysis.

The DVM is defined by a minimum set of ten state conditions. The locations of the DVM state points on the core power and flow operating domain are shown in Figure E-6. The DVM includes steady-state points and LOFH events at the four intercepts of the Exclusion and Restricted analytical boundaries on the edge of the operating domain and two intermediate flow reduction events from rated power at minimum and rated flow conditions (H0 and H1) to the Exclusion Region analytical boundary. Additional state points analyzed as part of the demonstration plant validation effort are discussed in Section E.1.4.

A detailed result summary of the feasibility analysis for the DVM is provided in Table E-3 and the decay ratio results are shown relative to the ODYSY boundary validation stability criterion in Figure E-7. The discussion of the DVM below is broken up into Exclusion Region boundary validation analysis in which stability controls are required and Restricted Region boundary validation analysis in which no stability controls are assumed.

E.1.3.1 Exclusion Region Boundary Validation

Validation analysis was performed for the Exclusion Region boundary of the demonstration plant for steady-state conditions, LOFH events, and flow reduction events.

For steady-state conditions, the analysis was performed at Points A_{fr} and B on the analytical boundary. These are the highest power and lowest flow conditions attainable outside of the Exclusion Region, taking into account setpoint uncertainty.

The analysis was performed at the middle-of-equilibrium-cycle with the largest excess reactivity, which is the most challenging cycle conditions to achieve a high boiling boundary. A four-foot core average boiling boundary is required for steady-state operation at Points A_{fr} and B and this was achieved for Point A_{fr} with shallow control rod insertion. The maximum achievable boiling boundary for Point B was 3.8 feet, as noted in Table E-3, resulting from the no-Xenon and the middle-of-cycle exposure conditions assumed. The shallow rods had to be inserted to medium-depth to compensate for the large excess reactivity and, as a result, the core average axial power shape flattened out and lowered the boiling boundary. At other less limiting exposures, a four-foot boiling boundary is attainable and the resulting core decay ratio is lower. Thus, the core decay ratio result at Point B is conservative.

The void reactivity feedback at the middle-of-equilibrium-cycle exposure is close to the cycle maximum value. The analysis at Point A_{fr} assumed equilibrium Xenon, which is consistent with operation at high rod lines and the analysis at Point B assumed no Xenon, which is a conservative assumption relative to normal startup operations in this region. The above assumptions yield a conservative analysis with respect to stability.

For the LOFH event, the analysis was performed for Points A_{fr} and B with the initial steady-state conditions described above. A change of 100°F equivalent rated feedwater temperature was assumed for this feasibility study and the stability analysis was performed at the new, higher, equilibrium power level. A loss of feedwater heating of 100°F equivalent rated is large relative to plant experience and yields a conservative analysis relative to stability.

For the flow reduction event, the analysis was performed for two cases that start at rated power. One case starts at rated flow and the other at minimum flow, and both end at the Exclusion Region analytical boundary. It was determined that end-of-cycle conditions yield higher core decay ratios. This can be seen in results of the Perry confirmation cases presented in Section 5.4 of Reference 1. End-of-cycle conditions are conservative because the core average axial power shape at the end of cycle is top-peaked at rated conditions and, when the flow is reduced, the axial power shape becomes double-peaked, with peaks in the top and bottom of the core. This is the worst axial power shape as discussed in Section 9; the peak in the bottom of the core creates a low boiling boundary and the peak in the top of the core generates increased void reactivity feedback - both adverse to stability.

Perry Cycle 2 was used in the analysis because it was determined that the axial power shape at the End-of-Cycle 2 condition was more limiting with respect to stability. Both cases assumed rated Xenon and rated feedwater temperature. Rated feedwater temperature is assumed, since the condition immediately after the flow reduction was analyzed. Equilibrium conditions are reached after several minutes, allowing the operator time to take action. Justification for this analysis condition is presented in Section 4.

E.1.3.2 Restricted Region Boundary Validation

Validation analysis was performed for the Restricted Region boundary of the demonstration plant for steady-state conditions and the LOFH transient.

For steady-state conditions, the analysis was performed at Points A' and B' on the analytical boundary. These are the highest power and lowest flow conditions attainable outside of the Restricted Region taking into account setpoint uncertainty. No stability control is required at Points A' and B', therefore control rod patterns were developed to yield low boiling boundaries for worst stability performance. This was accomplished by minimizing shallow control rod insertion.

The analysis at Point A' was performed with the End-of-Cycle 2 exposure while the analysis at Point B' was performed with the middle-of-equilibrium-cycle exposure. The combination of these exposures with the selected state points and rod patterns minimized the core average boiling boundary and maximized the void reactivity feedback above it.

The analysis at Point A' assumed equilibrium Xenon which is consistent with operation at high rod lines and the analysis at Point B' assumed no Xenon, which is a conservative assumption relative to normal startup operations in this region. The analysis at Point B' also assumes high radial peaking (1.7 radial peaking factor) and Feedwater Heater Out of Service (FWHOOS) equivalent to loss of 100°F rated feedwater temperature. The assumptions for the steady-state conditions are conservative with respect to stability.

For the LOFH event, the analysis was performed for Points A' and B' with the initial steady-state conditions described above. A change of 100°F equivalent rated feedwater temperature was assumed and the stability analysis was performed at the new, higher, equilibrium power level. Again, loss of feedwater heating of 100°F equivalent rated is large relative to plant experience and yields a conservative analysis relative to stability.

E.1.3.3 Demonstration Validation Matrix Summary

The results of the Demonstration Validation Matrix analysis show that the cases with stability control have low core decay ratios and hot channel decay ratios near zero. This can be seen in Figure E-7 for Points A_{fr} and B. The low decay ratios illustrate the effectiveness of the core average boiling boundary as a stability control. The very low hot channel decay ratio also indicates significant margin to Regional Mode oscillations.

The favorable results of the intermediate flow reduction events support the generic application of the Exclusion Region flow clamp at 40% of rated core flow.

The LOFH events from A' and B' provide the largest decay ratio results for the DVM. The results show that the steady-state and LOFH cases with controls are non-limiting. This supports a generic basis for the selection of cases to be analyzed in the IVM and the RVM.

E.1.4 Additional and Bounding Validation Analysis

Additional validation analysis was performed for the demonstration plant. The state points analyzed are shown in Figure E-8. They consist of steady-state conditions at Points A, B and A'; intermediate flow events from H0, H1 and H2;

and LOFH events from A and A'. A detailed summary of the results is provided in Table E-4 and the decay ratio results are shown relative to the ODYSY stability criterion in Figure E-9. Table E-4 and Figures E-8 and E-9 contain bounding events that are described in Section E.1.4.2. Additional validation analysis is provided in Section E.2.

E.1.4.1 Additional Validation Analysis

The steady-state case at Point B provides a sensitivity to boiling boundary height. Steady-state Point B in the DVM had a 3.8 foot boiling boundary, whereas this additional case has a 4.5 foot boiling boundary. The higher boiling boundary was achieved by changing the no-Xenon conditions in the DVM to equilibrium Xenon conditions. The core decay ratio is significantly lower (0.13 versus 0.38) for the case with the higher boiling boundary as can be seen comparing Figures E-7 and E-9.

The steady-state and LOFH validation analysis at Point A are performed with the same assumptions as were used for Point A_{fr}. Point A is at a higher core flow along the highest rod line and corresponds to the endpoint of the application of the generic shape function to the Exclusion Region boundary. As expected, the decay ratios are lower for the Point A analysis relative to the analysis results at Point A_{fr}.

The steady-state validation analysis at Point A' demonstrates the effect of lower feedwater temperature on stability margin. A FWHOOS reduction of 100°F equivalent rated feedwater temperature is assumed for steady-state conditions. All other assumptions are the same as for the DVM analysis. The decay ratio results with the FWHOOS are higher than those for the DVM as shown in Figure E-9. This analysis illustrates the importance of choosing anticipated or permitted reactor operating modes when performing a Reference Cycle calculation for the initial application of Enhanced Option I-A.

The additional flow reduction events analyzed consist of a flow reduction from H2 at the 105% rod line to the exclusion region analytical boundary with the initial rated feedwater temperature and the equilibrium temperature and the flow reduction events of the DVM with equilibrium feedwater temperature. The flow reduction conditions with equilibrium feedwater temperature are shown as shaded circles in Figures E-8 and E-9. The decay ratio results are higher with equilibrium feedwater temperature as expected.

E.1.4.2 Bounding Validation Analysis

Bounding analysis has been performed for selected transients that are identified precursors to reactor instability. These events were analyzed for two reasons.

First, it is important to demonstrate that the analytical methodology and tools produce high decay ratios for operating conditions expected to be unstable. This confirms the ability of the methods used in the Enhanced Option I-A analysis to differentiate among the spectrum of all events and conditions that lead to reduced stability margin.

Second, the severity of bounding events must be compared with those events that are reasonably limiting. The licensing methodology for Enhanced Option I-A is designed to prevent reactor instability for reasonably limiting transients. Bounding transients lie outside the licensing basis for the stability solution and are addressed by defense-in-depth. A general understanding of the severity of bounding events is necessary to judge the adequacy and robustness of the solution defense-in-depth features.

A bounding LOFH analysis was performed at Point A'. This point represents the lowest flow/highest power state condition where stability controls to limit core power distribution are not required. To make this event bounding, a FWHOOS reduction of 100°F equivalent rated feedwater temperature is imposed on the initial reactor state. The transient involves an additional 100°F equivalent rated feedwater temperature loss. These temperatures bound the expected values for a LOFH transient from FWHOOS conditions and the combination is limiting. As a result, the calculated terminal state condition decay ratios exceed the ODYSY stability criterion. However, the decay ratio values still remain close to the criterion, and the loss of stability margin during the transient is expected to occur slowly. Therefore, defense-in-depth measures will be effective in responding to bounding LOFH events that are not within the solution licensing basis and that exceed reasonably limiting conditions.

In addition, a bounding IFRE has been analyzed. This event involves a severe power shape at the end-of-cycle resulting from spectral shift operation. The initial reactor state at rated power exhibits a very high axial flux peak at the core top which enhances neutronic feedback. Following the flow reduction to Point A_{fr}, a significant bottom peak develops, resulting in a double-peaked axial power shape which enhances both the thermal-hydraulic and neutronic feedback. In addition, the initial reactor state point on the highest licensed flow-control line and at rated power results in limiting terminal conditions following the flow reduction. The EOC conditions are associated with highly negative void reactivity feedback. This combination of conditions is considered bounding and is not expected during normal reactor operations. The calculated equilibrium state condition decay ratios exceed the ODYSY stability criterion only slightly and the loss in stability margin will occur relatively slowly (i.e., 5 to 7 minutes). Defense-in-depth measures are effective in responding to this bounding event, which is not considered within the solution licensing basis.

E.2 Initial Application Supporting Analysis

Feasibility analysis has been performed for the demonstration plant to support the basis for portions of the initial application process. The IVM is analyzed for the Perry equilibrium cycle (used as the Current Cycle) both with and without the assumption of FWHOOS. In addition, analysis to validate the state points selected to compute the Reference Cycle to Standard Cycle decay ratio bias correction is performed using the Perry equilibrium cycle as the Reference Cycle.

E.2.1 Initial Validation Matrix

The IVM was applied to the demonstration plant to confirm the validation analysis process. The IVM excludes three non-limiting points from the DVM: the steady-state and LOFH cases at the high rod line with stability controls (Point A_{fr}) and the LOFH case at natural circulation with stability controls (Point B). The seven state point conditions that are analyzed include: steady-state conditions at Points B, A' and B'; LOFH at Points A' and B'; and flow reduction from H0 and H1. The IVM state points are shown in Figure E-10.

The IVM analysis was performed at end-of-cycle Haling conditions with nominal feedwater heating. A detailed summary of the analysis results is provided in Table E-5 and the decay ratio results are shown relative to the ODYSY boundary validation stability criterion in Figure E-11. An additional set of IVM analysis was performed with a FWHOOS reduction of 50°F equivalent rated feedwater temperature. A detailed summary of the FWHOOS results is provided in Table E-6 and the decay ratio results are shown relative to the ODYSY boundary validation stability criterion in Figure E-12.

The IVM procedure, as described in Section 7, was implemented. A four-foot core average boiling boundary was achieved at Point B with radially uniform shallow control rod insertion and the deep control rods for all of the steady-state cases were inserted to either notch position 00 or 08 in a radially uniform manner. The analysis at Points B and B' was performed based on a Haling depletion to rated power at minimum flow (H1). This is inconsistent with the procedure which prescribes the use of a Haling depletion to rated power and flow (H0) to minimize control rod insertion at the off-rated conditions. Sensitivity studies show no difference in decay ratio values for analysis performed with either Haling depletion at common off-rated state point conditions. The procedure uses H0, which has less excess reactivity, to allow more flexibility in changing rod patterns.

No Xenon is assumed for the analysis at Points B and B' for the IVM with nominal heating as prescribed by the procedure. For the analysis at these points with FWHOOS, equilibrium Xenon was used and the results are included for information. For a plant-specific initial application analysis, these cases should be analyzed with no Xenon. No-Xenon conditions are expected to result in higher decay ratios.

A LOFH event with a 60°F equivalent rated feedwater temperature reduction was assumed at Points A' and B'. Intermediate flow reduction events were analyzed at the analytical Exclusion Region boundary. One flow reduction case was based on a Haling depletion to rated flow and power and the other case was based on a Haling depletion to minimum flow at rated power. The flow reduction cases for the IVM were analyzed with rated feedwater temperature. The same cases were analyzed with equilibrium feedwater temperature and are shown in Table E-7 for information. The ODYSY analysis results show that all decay ratios are higher at the equilibrium feedwater temperature conditions as expected.

The steady-state case at Point B with stability controls in place is non-limiting, as expected, for both the nominal heating and the FWHOOS conditions, as shown in Figures E-11 and E-12. This provides further evidence that core average boiling boundary is an effective stability control. The intermediate flow reduction events show considerable margin with rated feedwater temperature assumed. The flow reduction post-event powers are higher for the FWHOOS analysis and result in higher decay ratio results. The analysis results for the FWHOOS cases at Point B' are less limiting than the rated feedwater temperature cases as a result of the equilibrium Xenon assumption. The limiting point in each analysis is the LOFH at Point B'. The close proximity of the results to the ODYSY stability criterion demonstrates that the validation process is a valid analytical test of the region boundaries.

E.2.2 Reference Cycle Decay Ratio Bias

The Enhanced Option I-A initial application process for generating the region boundaries requires decay ratios to be calculated for both the Standard Cycle and the Reference Cycle at three different flow rates along the rated flow-control line with a best-estimate stability code (Section 7.2.2.2). The difference in decay ratios between the two fuel cycles at each state point is used to compute a decay ratio bias correction. Calculations are performed at natural circulation (N/C), N/C plus 10% of rated core flow and N/C plus 25% of rated core flow for an end-of-cycle Haling condition with equilibrium feedwater temperature and rated Xenon concentration. The cases may be run with no Xenon, all rods out, if higher decay ratio values are needed.

ODYSY calculations were performed for the demonstration plant to confirm the analysis process. The Perry equilibrium cycle is assumed to be the Reference Cycle. The state points analyzed are shown in Figure E-13 and the decay ratio results are presented in Table E-8. It can be seen that performing a calculation with no Xenon, all rods out (Point 4), yields significantly larger decay ratio values.

1

IMAGE EVALUATION TEST TARGET (MT-3)

150mm

6"

1

IMAGE EVALUATION TEST TARGET (MT-3)

150mm

6″

1

IMAGE EVALUATION TEST TARGET (MT-3)

150mm

6"

1

IMAGE EVALUATION TEST TARGET (MT-3)

150mm

6"

E.3 Stability Controls Supporting Analysis

Stability analysis has been performed with ODYSY for the demonstration plant to provide an analytical basis for the selection of the stability control and its associated operational limit. The results of this analysis are presented and discussed in Section 9 of this report. The values of the key parameters for this analysis are tabulated in Tables E-9 through E-14. As evidenced by the data, a core average boiling boundary of four feet provides an effective means of stability control.

Additional analysis was also performed for a BWR/3 to provide assurance that the trends observed for the demonstration plant, a BWR/6, are generic in nature and common to all BWR types. The analysis included sensitivities to boiling boundary height, two-phase axial power shape, radial peaking, and feedwater temperature. The core average boiling boundary of four feet was found to provide the same level of control for the BWR/3 as the BWR/6. This is as expected due to the generic nature of the physics and general reactor systems design characteristics of GE BWRs.

E.4 Reference

1. NEDO-31960, "BWR Owners' Group Long-Term Stability Solutions Licensing Methodology", June 1991.

Table E-1: Region Boundary Generation Calculations

Point	Power (% of Rated)	Flow (% of Rated)	Core Decay Ratio	Hot Channel Decay Ratio
High Rod Line				
1	71.0	45.0	0.89	0.59
2	72.9	47.0	0.82	0.55
3	75.4	50.0	0.74	0.49
4	80.3	56.0	0.55	0.41
5	86.5	64.0	0.43	0.34
6	92.4	72.0	0.36	0.30
Natural Circulation				
7	36.7	30.0	0.88	0.47
8	35.0	30.0	0.81	0.44
9	31.0	30.0	0.66	0.39
10	25.0	30.0	0.52	0.14
11	15.0	30.0	0.24	0.00

Table E-2: Region Boundary Intercepts

State Point	Nominal Boundary		Analytical Boundary	
	Power (% of Rated)	Flow (% of Rated)	Power (% of Rated)	Flow (% of Rated)
A_{fr}	66.8	40.0	62.3	35.0
A	73.8	48.0	69.5	43.0
A'	80.3	56.0	76.3	51.0
B	34.7	30.0	37.7	30.0
B'	27.9	30.0	30.9	30.0

Table E-3: Demonstration Validation Matrix Results

Category	Analytical State Point	Power[1] (%)	Flow (%)	Exposure /Cycle[2]	Xenon[3]	Feedwater Temp.[4]	Stability Control[5]	Core DR	Channel DR
Steady State	A_{fr}	62.3	35.0	MOEC	Equil.	Equil.	Yes	0.16	0.00
	B	38.4	30.0	MOEC	None	Equil.	Yes[6]	0.38	0.00
	A'	76.1	51.0	EOC2	Equil.	Equil.	No	0.43	0.33
	B'	30.4	30.0	MOEC	None	FWHOOS	No	0.55	0.39
Flow Events	H0	45.5	35.0	EOC2	Rated	Rated	No	0.34	0.15
	H1	56.9	35.0	EOC2	Rated	Rated	No	0.61	0.46
LOFH Events	A_{fr}	73.6	35.0	MOEC	Equil.	LOFH	Yes	0.23	0.03
	B	42.3	30.0	MOEC	None	LOFH	Yes	0.50	0.03
	A'	84.8	51.0	EOC2	Equil.	LOFH	No	0.81	0.42
	B'	33.2	30.0	MOEC	None	FWHOOS + LOFH	No	0.71	0.47

1. Power for steady-state cases B, A' and B' is approximately equal to values in Table E-2.
2. MOEC = Middle of equilibrium cycle, EOC2 = End of Cycle 2.
3. Equil. = Equilibrium Xenon at state power, None = No Xenon, Rated = Equilibrium Xenon at rated power.
4. Equil. = Equilibrium feedwater temperature at state power, FWHOOS = Feedwater Heater Out of Service of -100°F of rated temperature equilibrated at state power, Rated = Rated feedwater temperature, LOFH = Loss of Feedwater Heating of -100°F of rated temperature equilibrated at state power.
5. Yes = Core average boiling boundary of 4.0 feet, No = No axial power distribution stability control.
6. Core average boiling boundary for this case is 3.8 feet, the maximum attainable for these state conditions.

Table E-4: Additional and Bounding Demonstration Validation Results

Category	Analytical State Point	Power[1] (%)	Flow (%)	Exposure /Cycle[2]	Xenon[3]	Feedwater Temp.[4]	Stability Control[5]	Core DR	Channel DR
Steady State	A	69.5	43.0	MOEC	Equil.	Equil.	Yes	0.10	0.00
	B	38.4	30.0	MOEC	Equil.	Equil.	Yes[6]	0.13	0.00
	A'	75.8	51.0	EOC2	Equil.	FWHOOS	No	0.65	0.35
Flow Events	H0	53.4	35.0	EOC2	Rated	Equil.	No	0.68	0.42
	H1[7]	64.0	35.0	EOC2	Rated	Equil.	No	1.02	0.66
	H2	48.9	35.0	EOC2	Rated	Rated	No	0.40	0.26
	H2	56.4	35.0	EOC2	Rated	Equil.	No	0.75	0.45
LOFH Events	A	81.4	43.0	MOEC	Equil.	LOFH	Yes	0.14	0.02
	A'[7]	83.0	51.0	EOC2	Equil.	FWHOOS + LOFH	No	0.97	0.44

1. Power for steady-state cases B and A' is approximately equal to values in Table E-2.
2. MOEC = Middle of equilibrium cycle, EOC2 = End of Cycle 2.
3. Equil. = Equilibrium Xenon at state power, Rated = Equilibrium Xenon at rated power.
4. Equil. = Equilibrium feedwater temperature at state power, FWHOOS = Feedwater Heater Out of Service of -100°F of rated temperature equilibrated at state power, Rated = Rated feedwater temperature, LOFH = Loss of Feedwater Heating of -100°F of rated temperature equilibrated at state power.
5. Yes = Core average boiling boundary of 4.0 feet, No = No axial power distribution stability control.
6. Core average boiling boundary for this case is 4.5 feet.
7. Bounding IFRE and LOFH event.

Table E-5: Initial Validation Matrix Demonstration Results

Category	Analytical State Point	Power (%)	Flow (%)	Exposure /Cycle[1]	Xenon[2]	Feedwater Temp.[3]	Stability Control[4]	Core DR	Channel DR
Steady State	B	37.7	30.0	EOEC	None	Equil.	Yes	0.20	0.00
	A'	76.3	51.0	EOEC	Equil.	Equil.	No	0.40	0.19
	B'	30.9	30.0	EOEC	None	Equil.	No	0.73	0.27
Flow Events	H0	41.8	35.0	EOEC	Rated	Rated	No	0.28	0.01
	H1	50.4	35.0	EOEC	Rated	Rated	No	0.36	0.21
LOFH Events	A'	82.2	51.0	EOEC	Equil.	LOFH	No	0.60	0.32
	B'	32.3	30.0	EOEC	None	LOFH	No	0.83	0.36

1. EOEC = End of equilibrium cycle.
2. None = No Xenon, Equil. = Equilibrium Xenon at state power, Rated = Equilibrium Xenon at rated power.
3. Equil. = Equilibrium feedwater temperature at state power, Rated = Rated feedwater temperature, LOFH = Loss of Feedwater Heating of -60°F of rated temperature equilibrated at state power.
4. Yes = Core average boiling boundary of 4.0 feet, No = No axial power distribution stability control.

Table E-6: Initial Validation Matrix FWHOOS Demonstration Results

Category	Analytical State Point[1]	Power (%)	Flow (%)	Exposure /Cycle[2]	Xenon[3]	Feedwater Temp.[4]	Stability Control[5]	Core DR	Channel DR
Steady State	B	37.7	30.0	EOEC	Equil.	Equil.	Yes	0.31	0.01
	A'	76.3	51.0	EOEC	Equil.	Equil.	No	0.53	0.32
	B'	30.9	30.0	EOEC	Equil.	Equil.	No	0.66	0.02
Flow Events	H0	44.2	35.0	EOEC	Rated	Rated	No	0.42	0.00
	H1	53.4	35.0	EOEC	Rated	Rated	No	0.58	0.37
LOFH Events	A'	80.9	51.0	EOEC	Equil.	LOFH	No	0.69	0.30
	B'	32.2	30.0	EOEC	Equil.	LOFH	No	0.74	0.05

1. All cases assume Feedwater Heater Out of Service of -50°F of rated temperature equilibrated at state power.
2. EOEC = End of equilibrium cycle.
3. Equil. = Equilibrium Xenon at state power, Rated = Equilibrium Xenon at rated power.
4. Equil. = Equilibrium feedwater temperature at state power (FWHOOS), Rated = Rated feedwater temperature (FWHOOS), LOFH = Loss of Feedwater Heating of -60°F of rated temperature equilibrated at state power (FWHOOS).
5. Yes = Core average boiling boundary of 4.0 feet, No = No axial power distribution stability control.

Table E-7: IVM Demonstration Flow Events with Equilibrium Feedwater Temperature Results

Category	Analytical State Point	Power (%)	Flow (%)	Exposure /Cycle[1]	Xenon[2]	Feedwater Temp.[3]	Stability Control[4]	Core DR	Channel DR
Flow Events	H0	49.2	35.0	EOEC	Rated	Nominal	No	0.56	0.35
	H1	58.0	35.0	EOEC	Rated	Nominal	No	0.68	0.41
	H0	49.2	35.0	EOEC	Rated	FWHOOS	No	0.58	0.02
	H1	59.4	35.0	EOEC	Rated	FWHOOS	No	0.84	0.42

1. EOEC = End of equilibrium cycle.
2. Rated = Equilibrium Xenon at rated power.
3. Nominal = Nominal equilibrium feedwater temperature at state power,
   FWHOOS = Feedwater Heater Out of Service of -50°F of rated temperature equilibrated at state power.
4. No = No axial power distribution stability control.

Table E-8: Reference Cycle Decay Ratio Bias Demonstration

Point	Power (%)	Flow (%)	Core DR	Channel DR
Rated Xenon, All Rods Out				
1	70.2	55.0	0.31	0.06
2	55.8	40.0	0.47	0.33
3	41.5	30.0	0.77	0.37
No Xenon, All Rods Out				
4	62.4	30.0	1.33	1.28

Table E-9: Boiling Boundary Height Sensitivity

Power (%)	Flow (%)	Z_{bb} (feet)	Core DR	Channel DR
36.0	30.0	4.0	0.14	0.00
36.0	30.0	3.0	0.36	0.02
36.0	30.0	2.0	0.55	0.39

Table E-10: Two-Phase Axial Flux Shape Sensitivity

Power (%)	Flow (%)	Z_{bb} (feet)	Axial Peaking Factor	Peak Node from bottom	Core DR	Channel DR
30.4	30.0	4.0	1.43	10/25	0.30	0.00
30.4	30.0	4.0	1.71	22/25	0.44	0.00

Table E-11: Radial Peaking Sensitivity

Power (%)	Flow (%)	Z_{bb} (feet)	Radial Peaking Factor	Core DR	Channel DR
38.4	30.0	4.5	1.35	0.13	0.00
38.4	30.0	4.5	1.73	0.08	0.00
38.4	30.0	4.5	1.96	0.05	0.00

Table E-12: Feedwater Temperature Sensitivity

Power (%)	Flow (%)	Z_{bb} (feet)	Rated FW Temp. (°F)	Core DR	Channel DR
30.4	30.0	4.0	420	0.30	0.00
30.4	30.0	4.0	320	0.38	0.03

Table E-13: Hot Region Sensitivity

Power (%)	Flow (%)	Z_{bb} (feet)	MCPR	Radial Peaking Factor	Core DR	Channel DR
36.0	30.0	4.0	2.14	1.45	0.14	0.00
36.0	30.0	4.0	1.92	1.81	0.22	0.44
36.0	30.0	4.0	1.64	2.05	0.30	0.49
36.0	30.0	3.6	1.41	2.30	0.89	0.82

Table E-14: Cycle Exposure Sensitivity

Power (%)	Flow (%)	Z_{bb} (feet)	Cycle Exposure	Core DR	Channel DR
38.4	30.0	3.8	MOC	0.38	0.00
38.4	30.0	4.3	BOC	0.18	0.00
38.4	30.0	5.6	EOC	0.09	0.00

Figure E-1: Region Boundary Definition Analysis Points

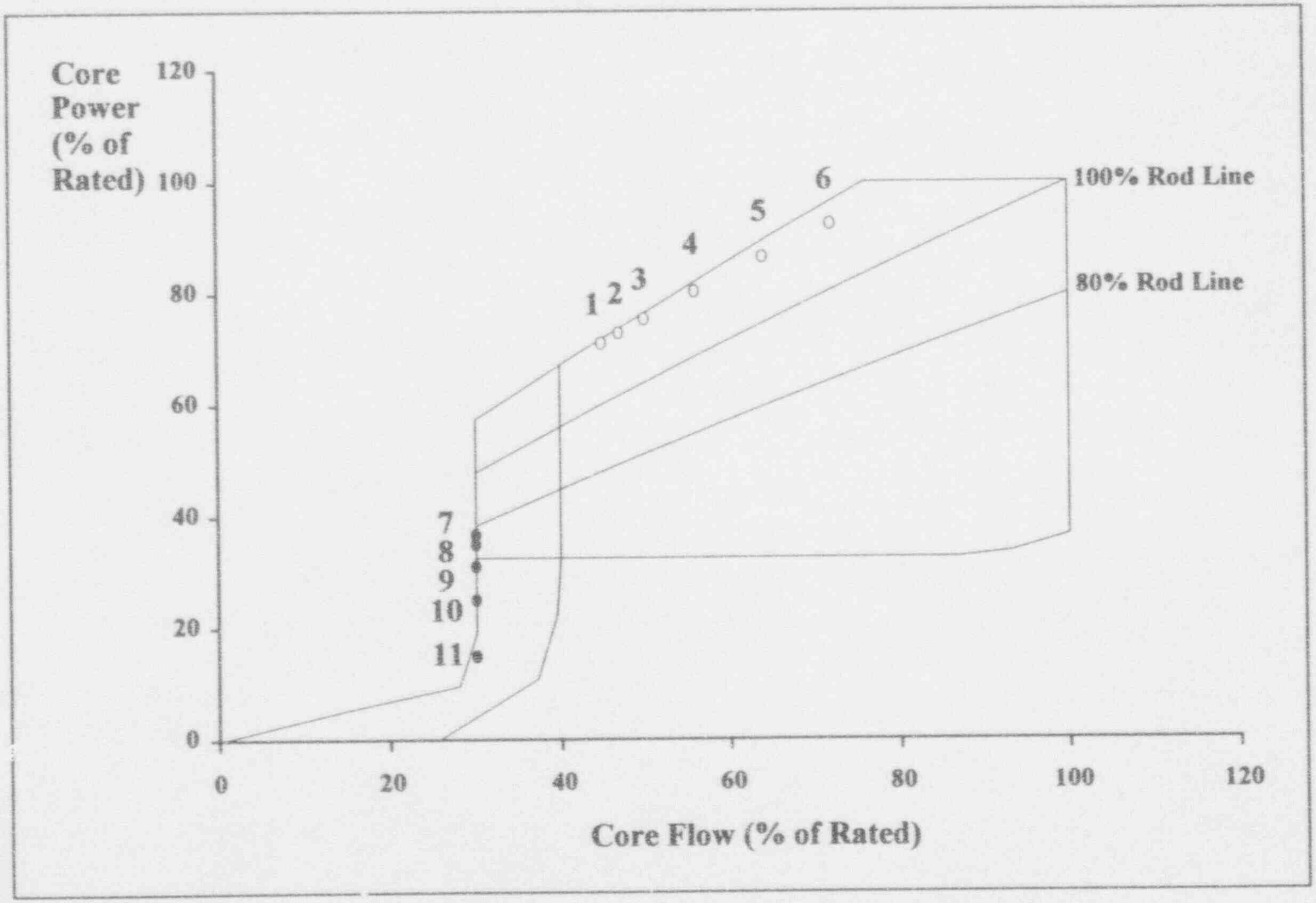


Figure E-2: Stability Criteria Map Application

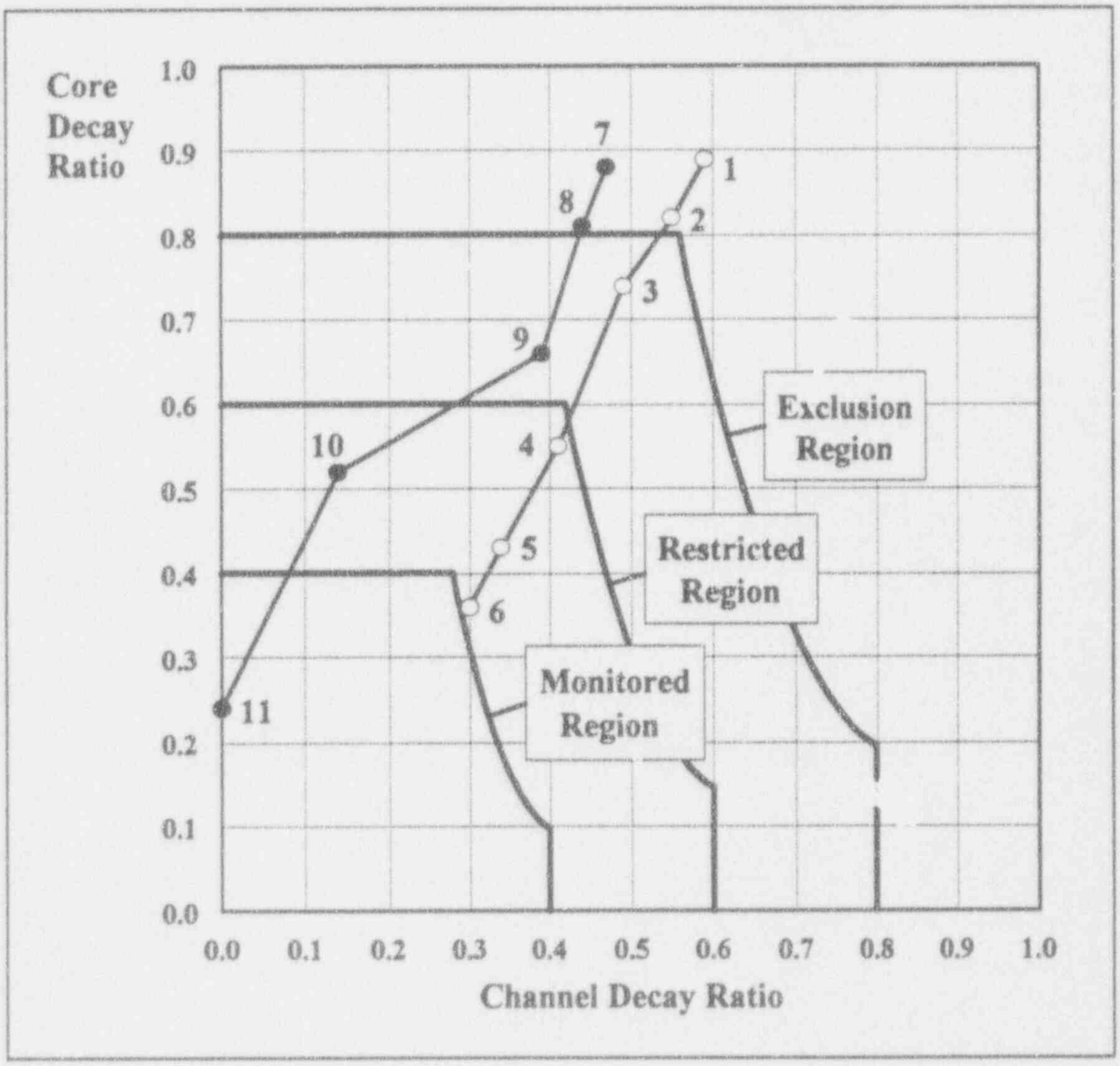


Figure E-3: Initial Demonstration Region Boundary Definition

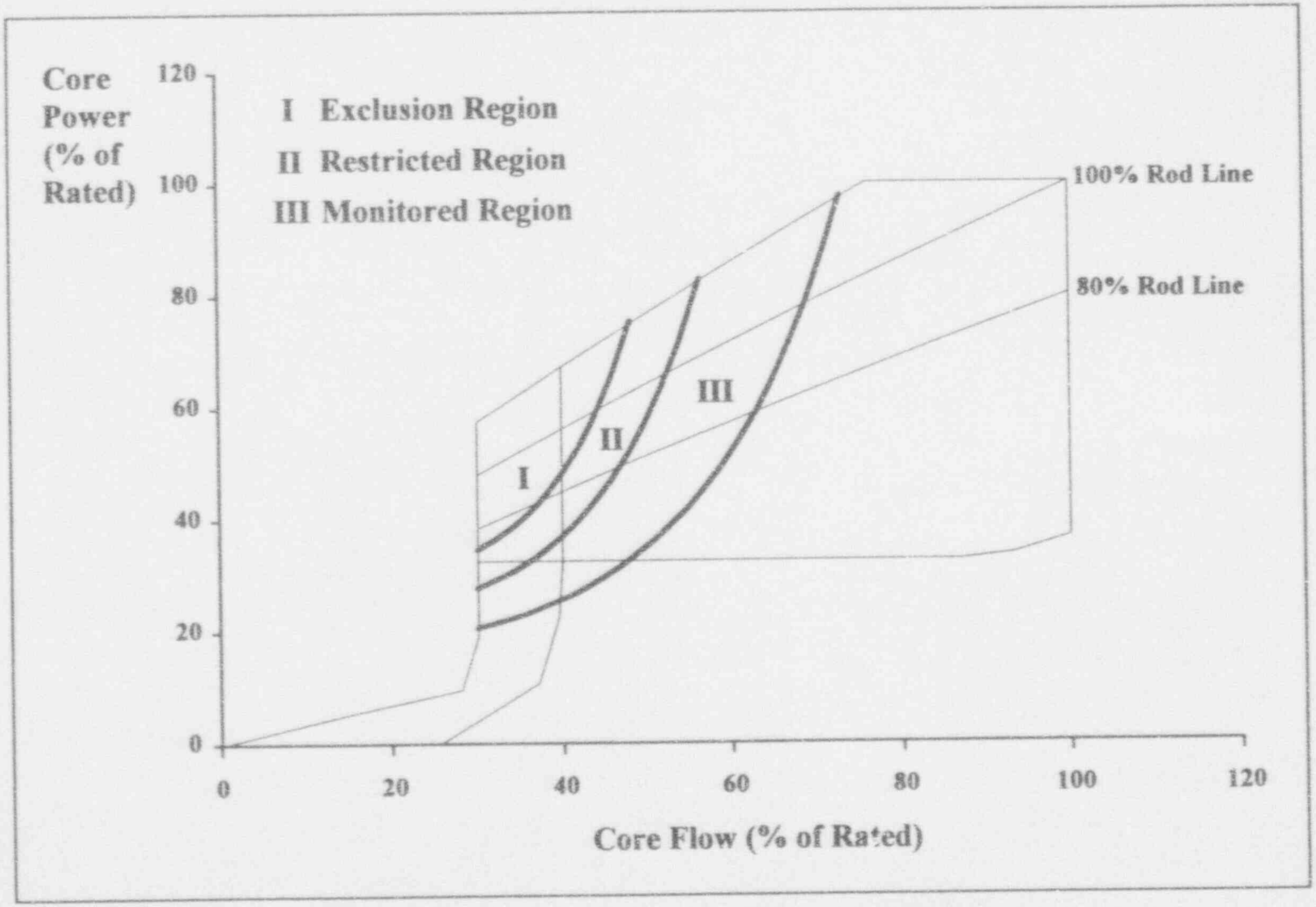


Figure E-4: Final Demonstration Region Boundaries

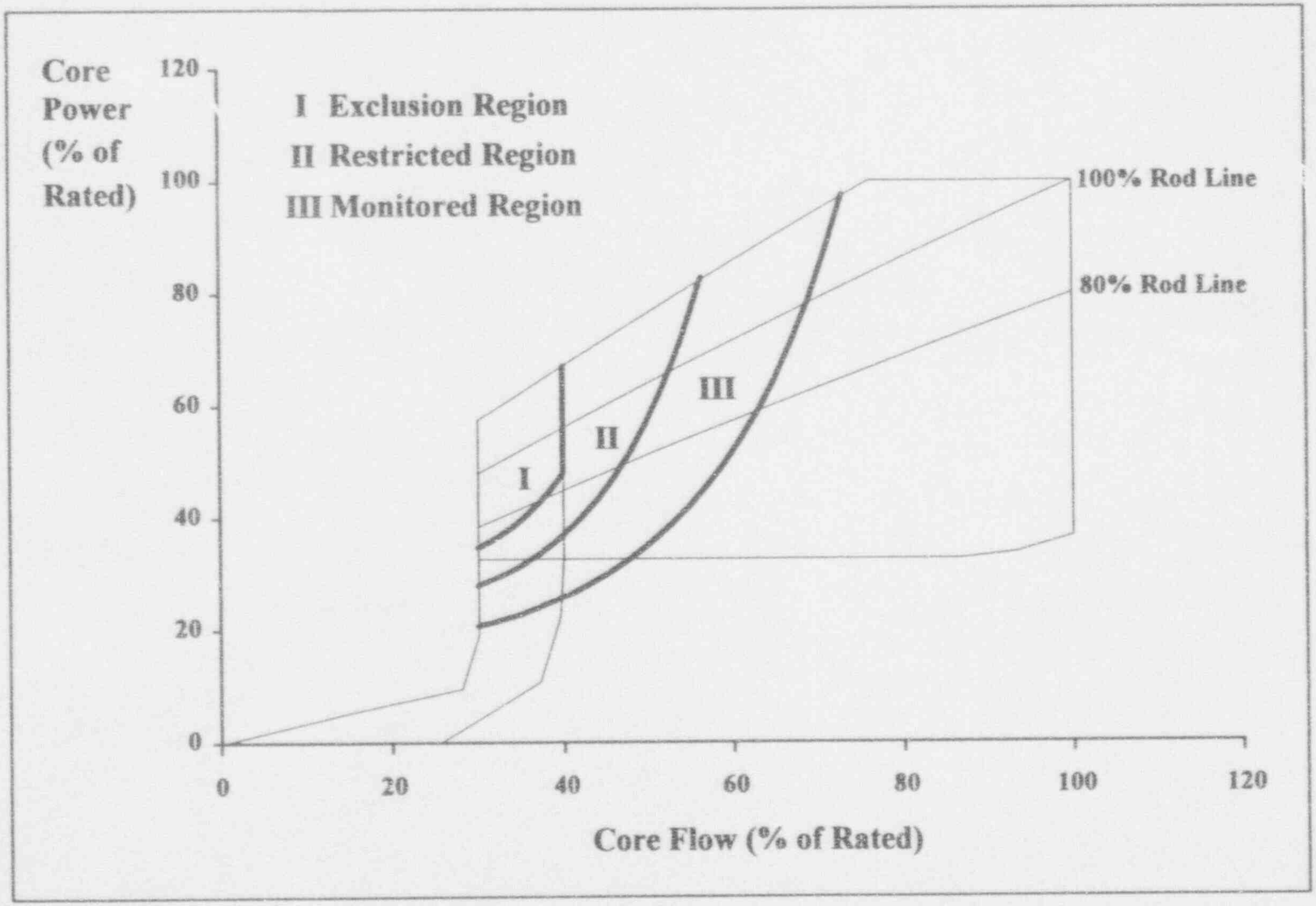


Figure E-5: Nominal and Analytical Boundaries

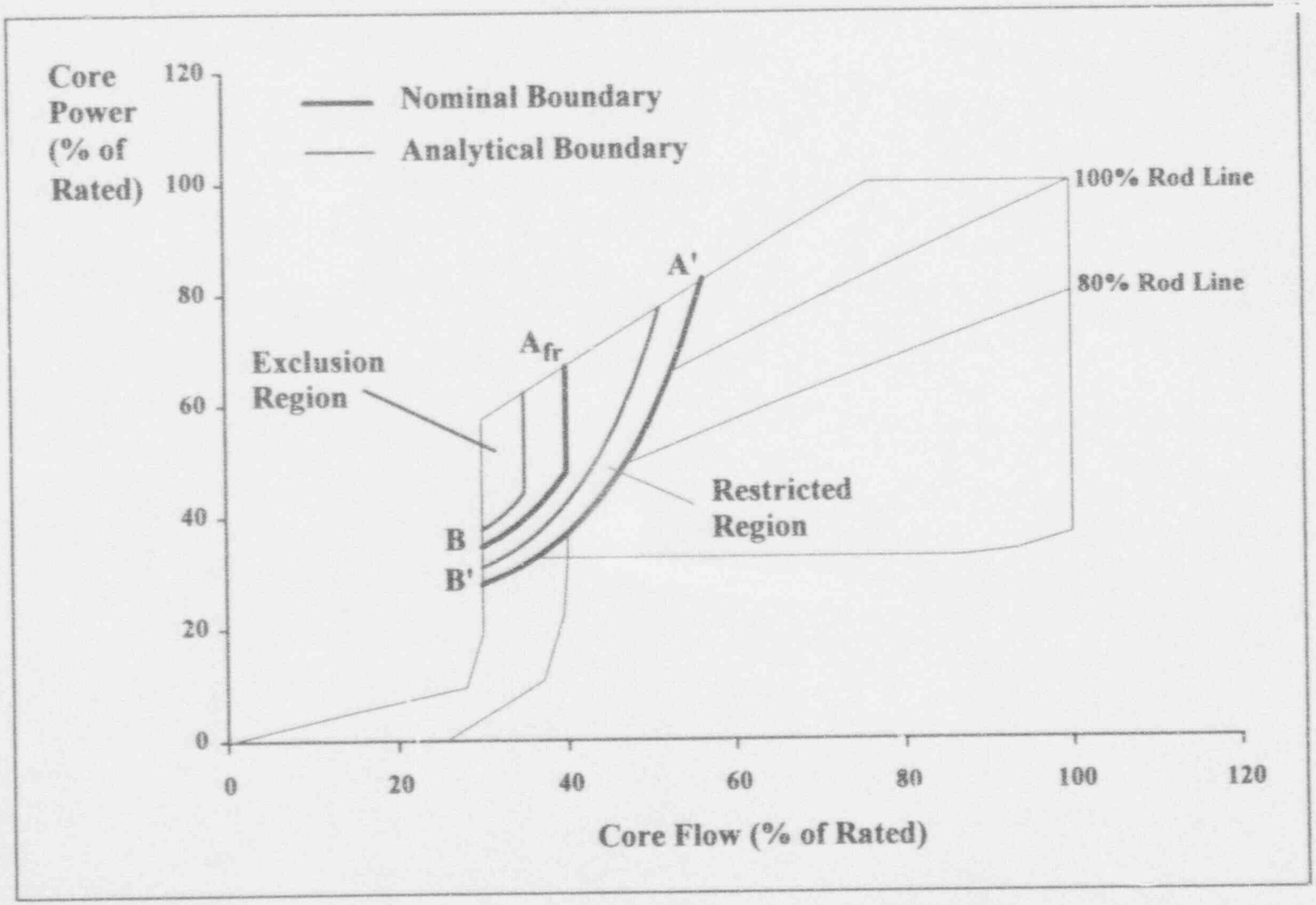


Figure E-6: Demonstration Validation Matrix State Points

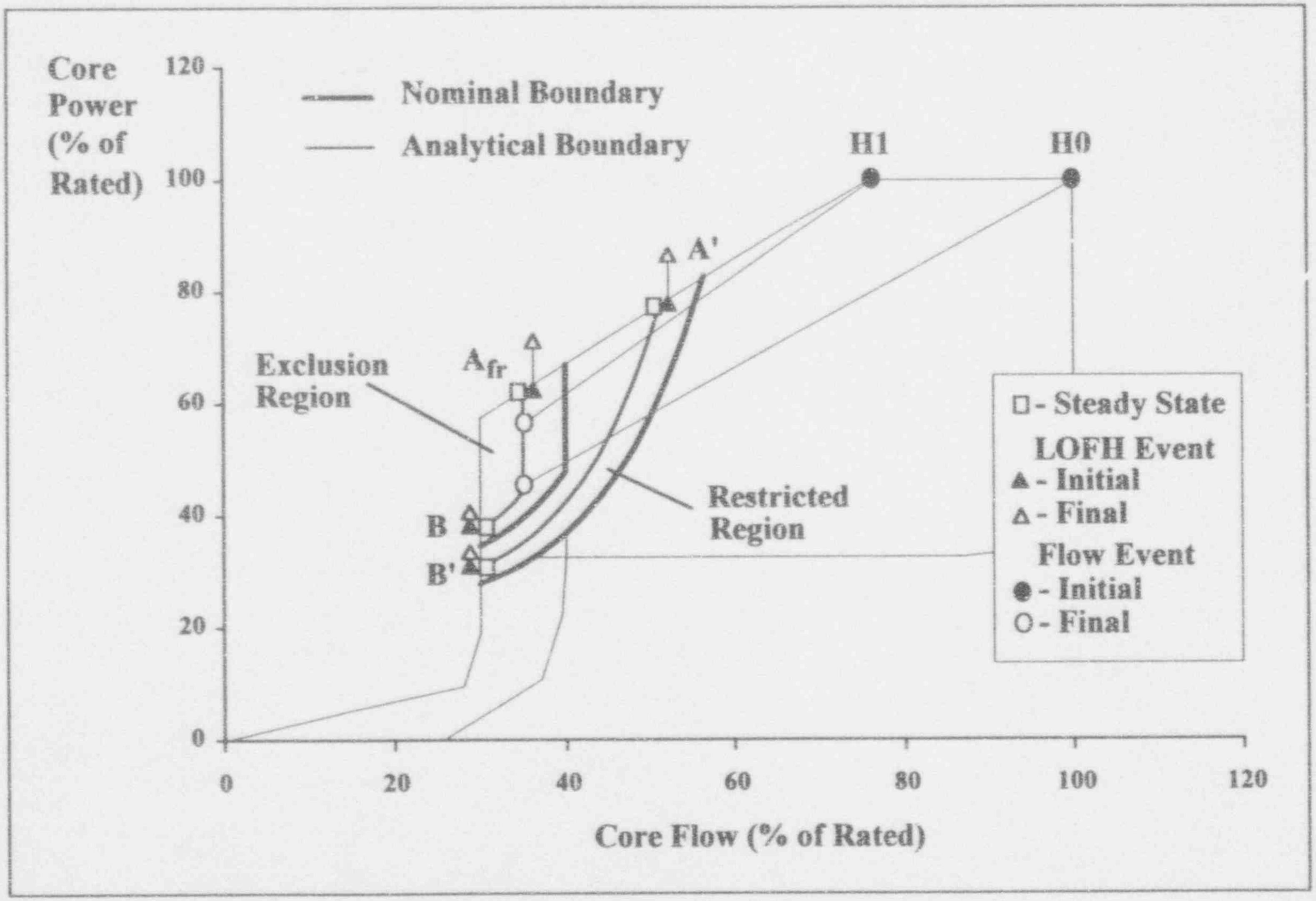


Figure E-7: Demonstration Validation Matrix Results

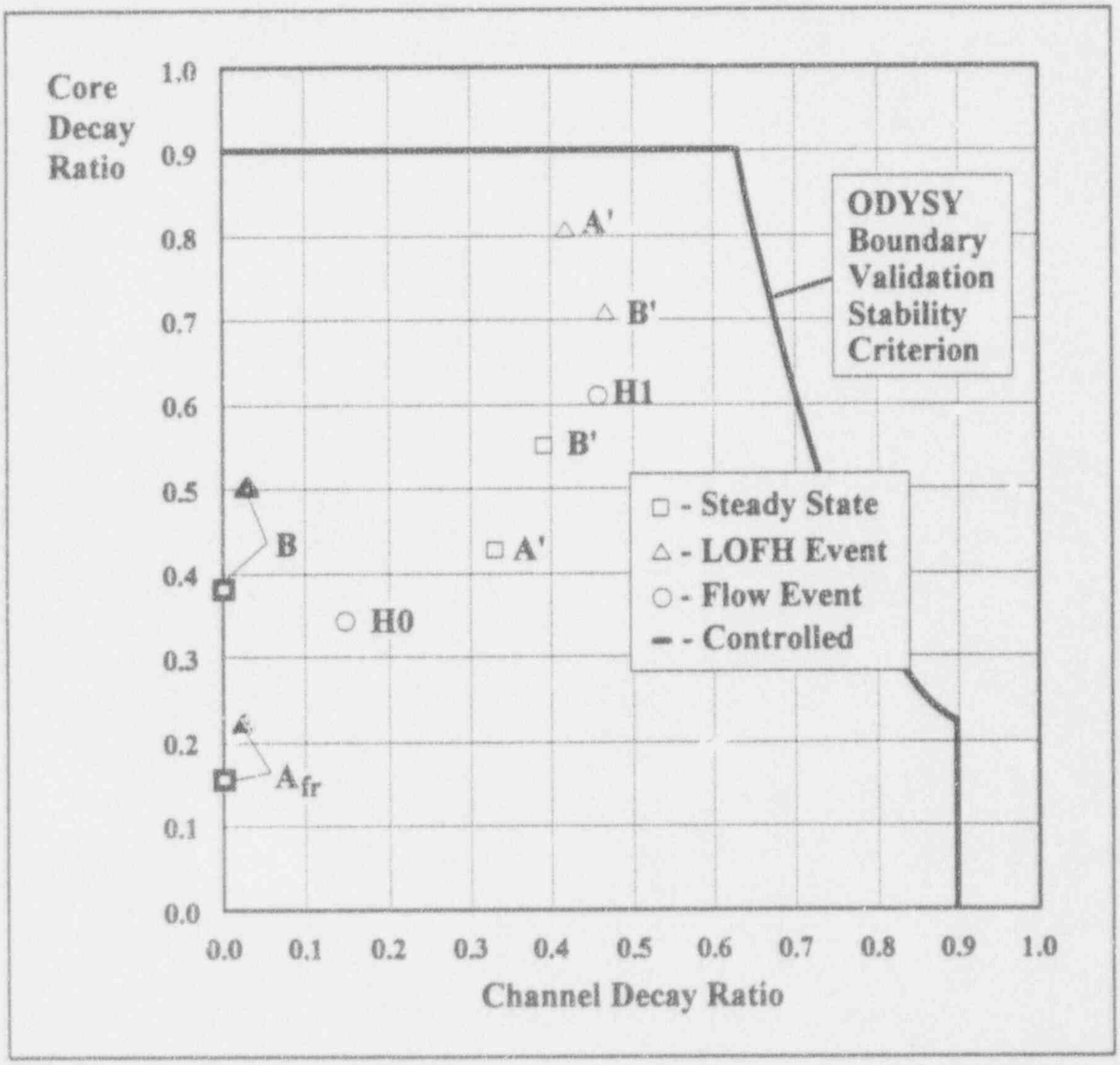


Figure E-8: Additional and Bounding Demonstration Validation State Points

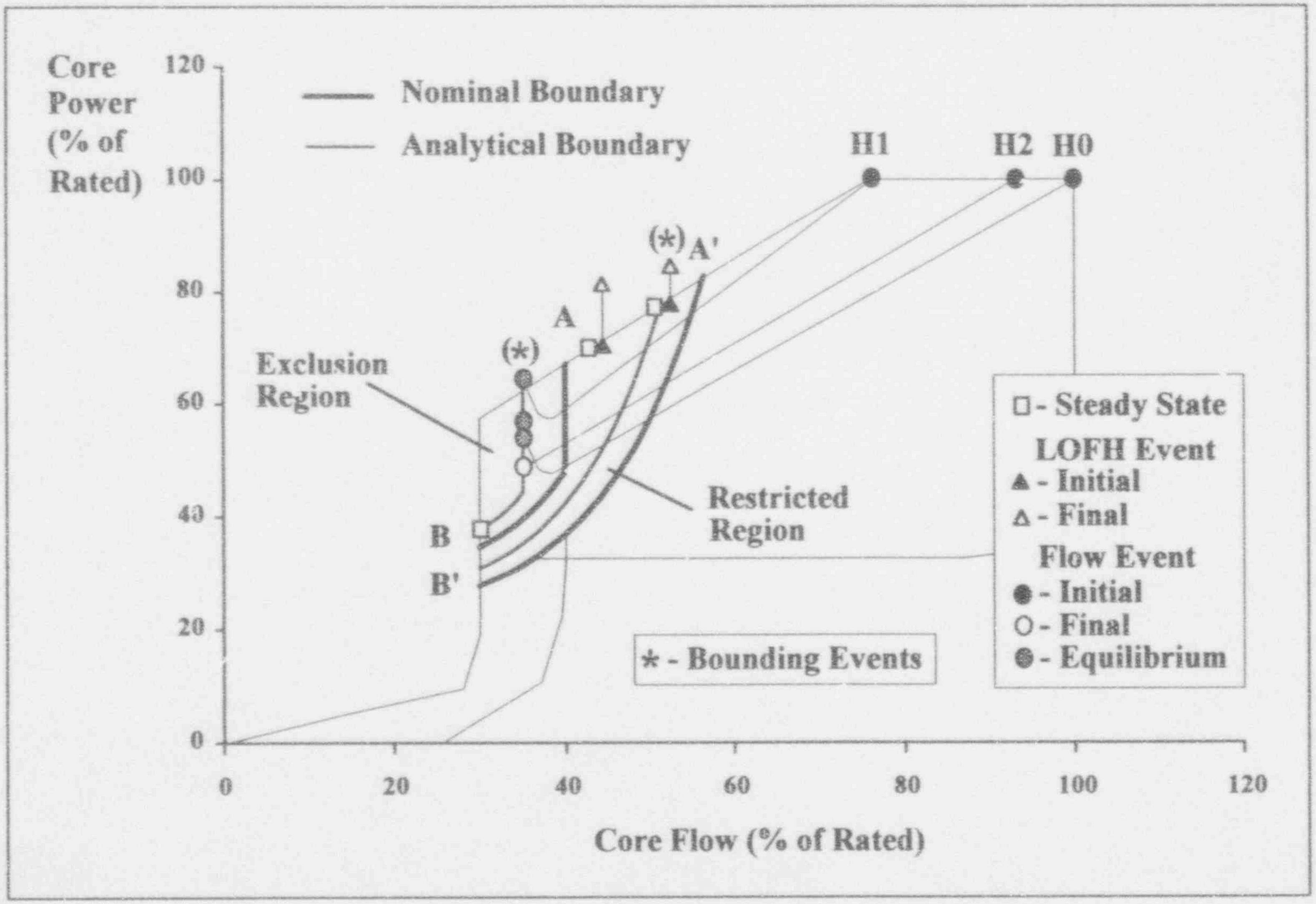


Figure E-9: Additional and Bounding Demonstration Validation Results

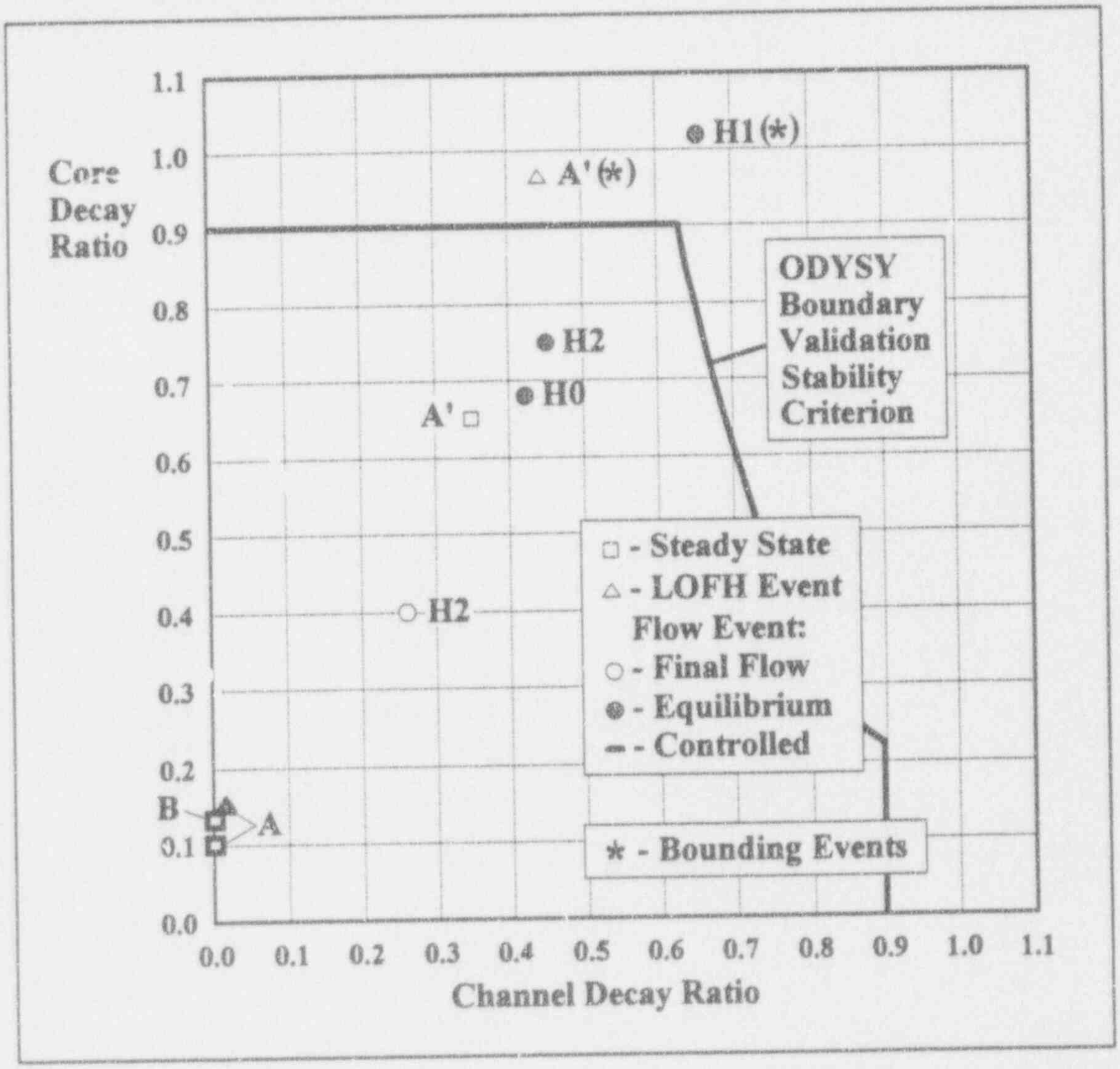


Figure E-10: Initial Validation Matrix State Points

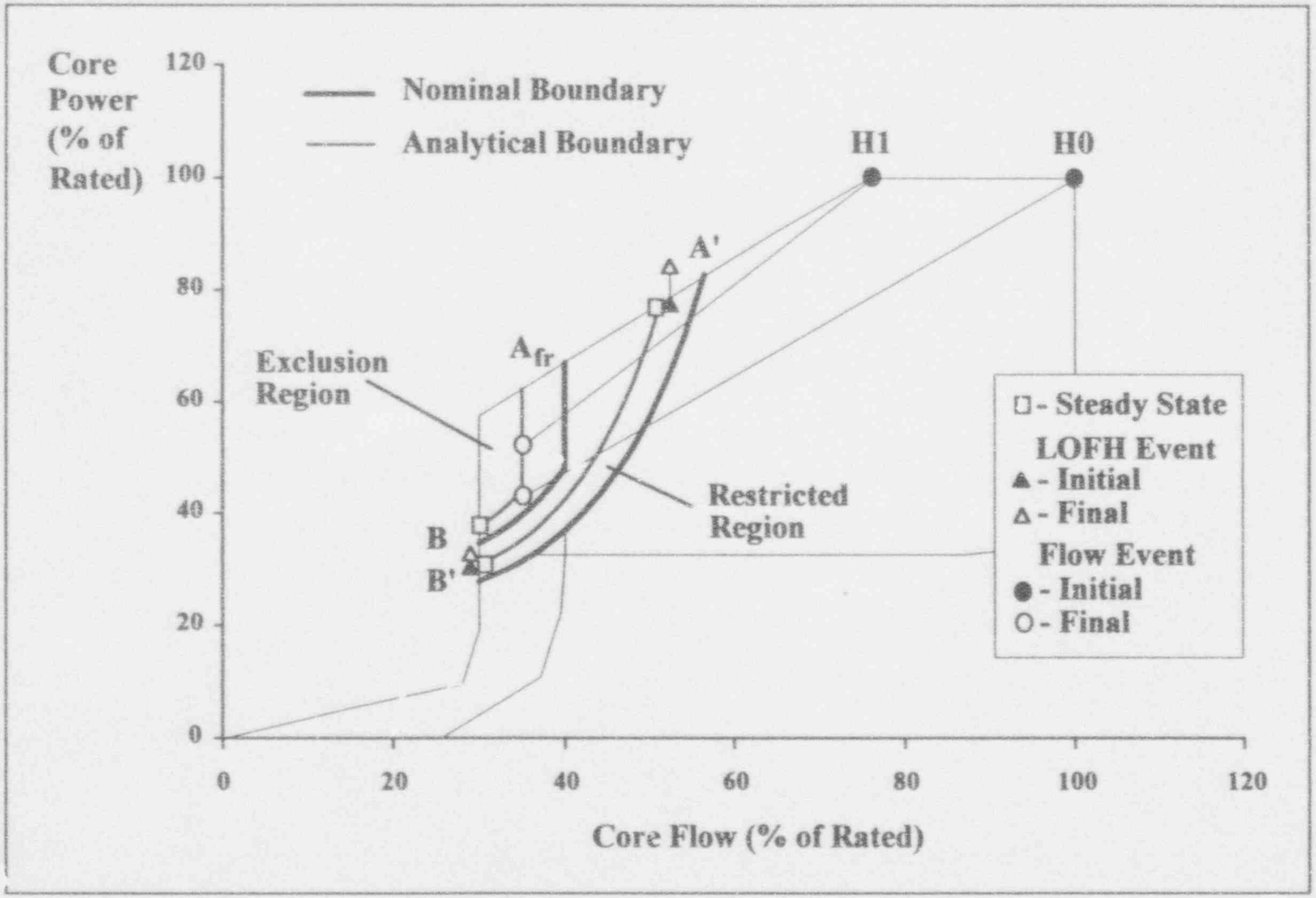


Figure E-11: Initial Validation Matrix Demonstration Results

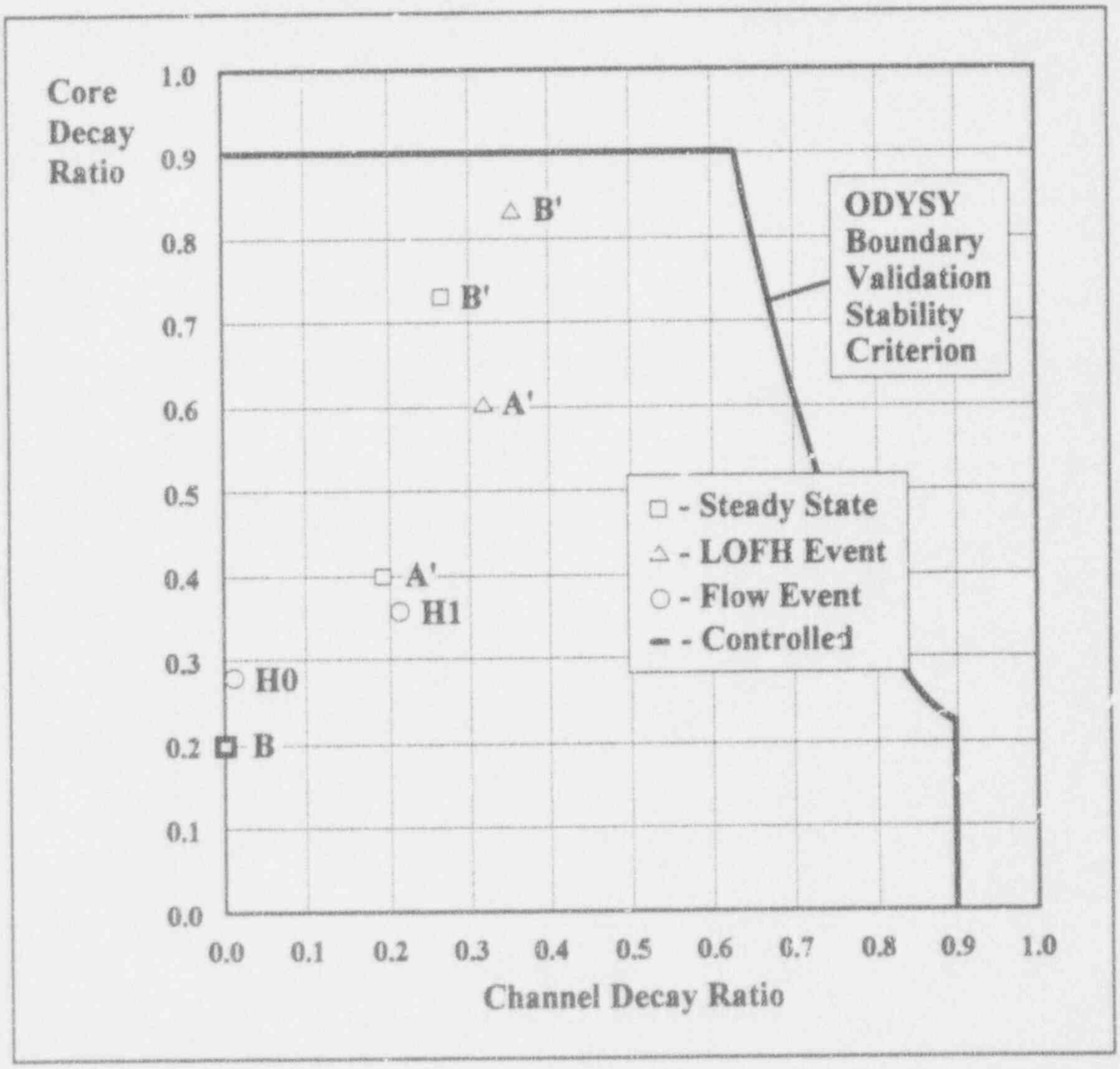


Figure E-12: Initial Validation Matrix FWHOOS Demonstration Results

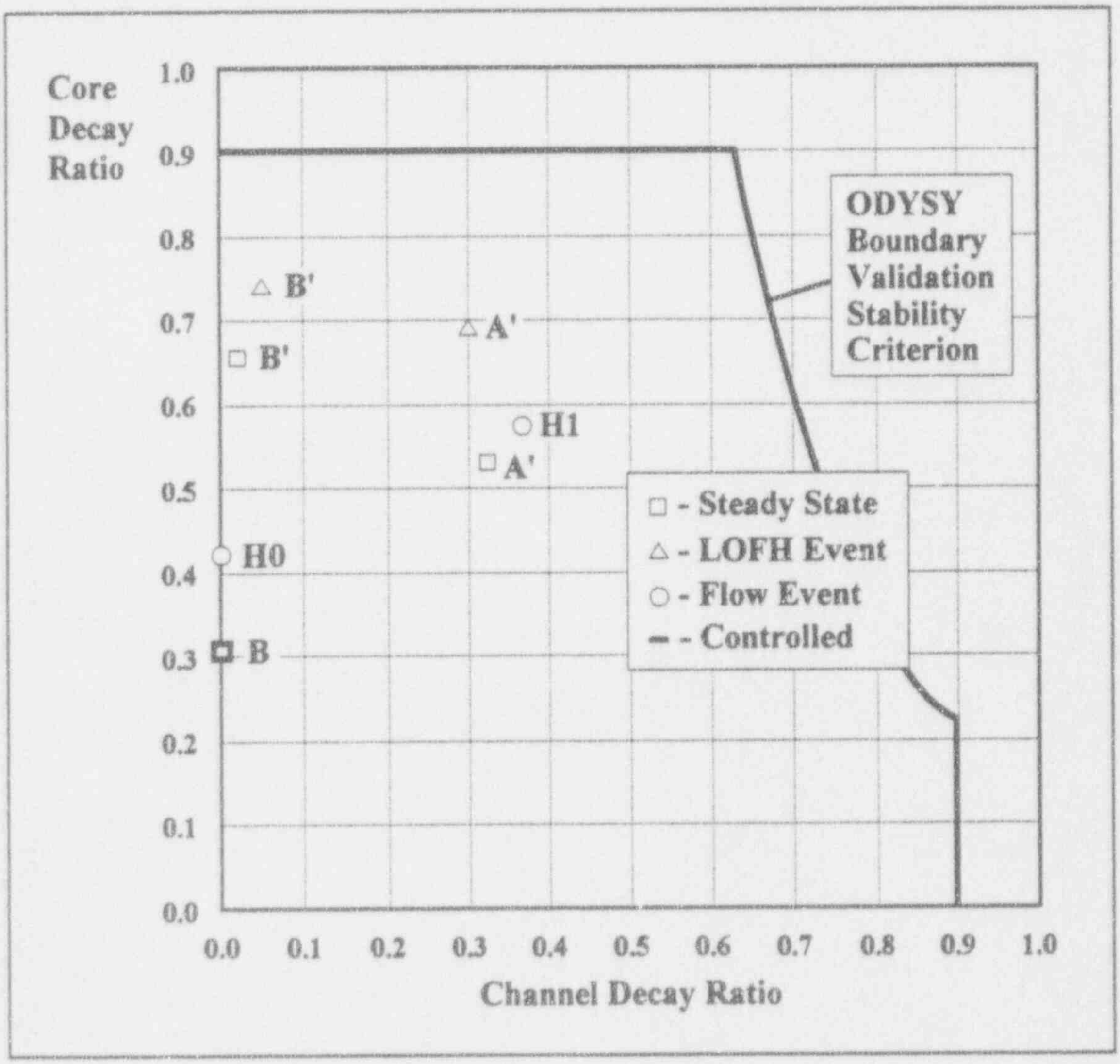
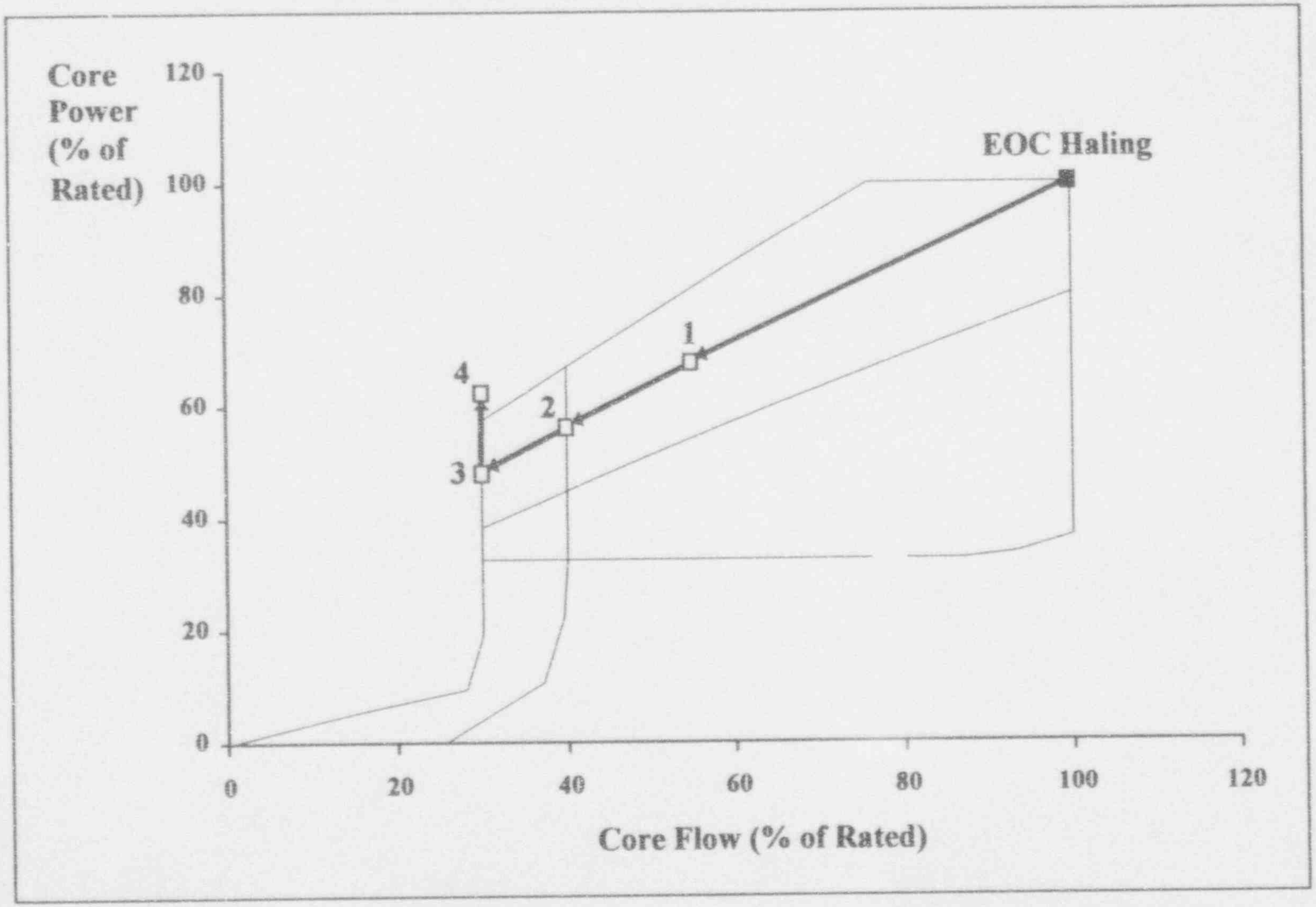


Figure E-13: Reference Cycle Decay Ratio Bias Demonstration State Points



APPENDIX F: GENERIC REGION BOUNDARY SHAPE FUNCTION

F.1 Objective

A generic shape function for generating the boundaries of Enhanced Option I-A stability regions is defined by relating the percent of rated core power (P) and percent of rated core flow (W) based on available state point data from Vermont Yankee (Cycle 15), Fitzpatrick (Cycle 10), Monticello (Cycle 15), Duane Arnold (Cycle 10), and Perry (Cycle 2 and Equilibrium Cycle). This function, P(W), is designed to satisfy:

i. $P=P_B$ at $W=W_B$, and $P=P_A$ at $W=W_A$, where (W_B,P_B) and (W_A,P_A) are the two state points which meet the established Enhanced Option I-A boundary generation stability criteria along the natural circulation and the highest rod-line, respectively.

ii. At each available state point datum flow rate, W_d, and power, P_d,

$$P(W_d) - P_d \leq 2\%.$$

F.2 Method

A proposed form of P(W) is

$$\ln P = \alpha + \beta W + \gamma W^2. \tag{F-1}$$

Evaluations of Equation F-1 at (W_B,P_B) and (W_A,P_A) eliminate α and β, and yield

$$\ln P = \frac{W_A \ln P_B - W_B \ln P_A}{W_A - W_B} + \frac{\ln P_A - \ln P_B}{W_A - W_B} W + \gamma\left[W_B W_A - (W_B + W_A)W + W^2\right] \tag{F-2}$$

Equation F-2 is further simplified by introducing a new parameter:

$$c = \gamma \frac{(W_A - W_B)^2}{\ln P_A - \ln P_B}, \tag{F-3}$$

where the parameter c represents the curvature of P(W). Equation F-2 becomes:

$$\ln \frac{P}{P_B} = \left\{ \frac{W - W_B}{W_A - W_B} + c\left[\frac{(W - W_A)(W - W_B)}{(W_A - W_B)^2} \right] \right\} \ln \frac{P_A}{P_B}, \tag{F-4}$$

or alternatively,

$$P = P_B \left(\frac{P_A}{P_B} \right)^{\frac{W - W_B}{W_A - W_B} + c\left[\frac{(W - W_A)(W - W_B)}{(W_A - W_B)^2} \right]}. \tag{F-5}$$

Table F-1 shows the minimum value of c required in Equation F-5 to bound the state points of the available six data sets.

The minimum value of c to bound all data points is thus 0.75, limited by the Vermont Yankee data. However, with c=0.50, the Vermont Yankee data meet criterion (ii) by being no more than 1.25% outside of the boundary. Thus a value of 0.50 is selected for c which is sufficient to represent or bound all data points. Plots of the generic shape function with c=0.50 for all plants are presented in Figures F-1 through F-6.

F.3 Summary

For each set of two state points (W_B,P_B) and (W_A,P_A) along the natural circulation and highest flow-control line, respectively, determined by FABLE, a region boundary for Enhanced Option 1-A can be determined by a generic shape function as follows,

$$\ln\frac{P}{P_B} = \left\{\frac{W - W_B}{W_A - W_B} + c\left[\frac{(W - W_A)(W - W_B)}{(W_A - W_B)^2}\right]\right\}\ln\frac{P_A}{P_B}. \qquad \text{(F-6)}$$

The value of c is selected to be 0.50 based on available plant data. Equation F-6 is then reduced to

$$\ln\frac{P}{P_B} = \frac{1}{2}\left[\frac{W - W_B}{W_A - W_B} + \left(\frac{W - W_B}{W_A - W_B}\right)^2\right]\ln\frac{P_A}{P_B}, \qquad \text{(F-7)}$$

or,

$$P = P_B\left(\frac{P_A}{P_B}\right)^{\frac{1}{2}\left[\frac{W - W_B}{W_A - W_B} + \left(\frac{W - W_B}{W_A - W_B}\right)^2\right]}. \qquad \text{(F-8)}$$

Table F-1: Minimum Value of c to Bound Plant State Point Data

PLANT	c_{min} for Equation F-5
Vermont Yankee	0.7461
Fitzpatrick	-0.0394
Monticello	0.2380
Duane Arnold	0.3633
Perry (Cycle 2)	-0.1597
Perry (Equil. Cycle)	0.5185

Figure F-1: Generic Shape Function for Vermont Yankee (Cycle 15)

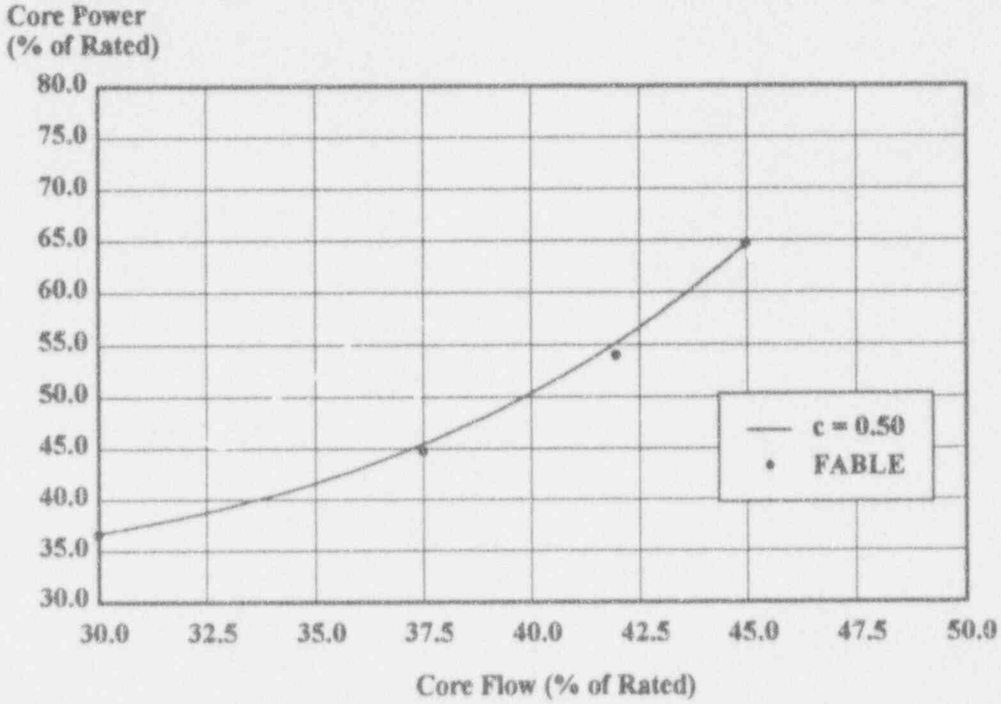


Figure F-2: Generic Shape Function for Fitzpatrick (Cycle 10)

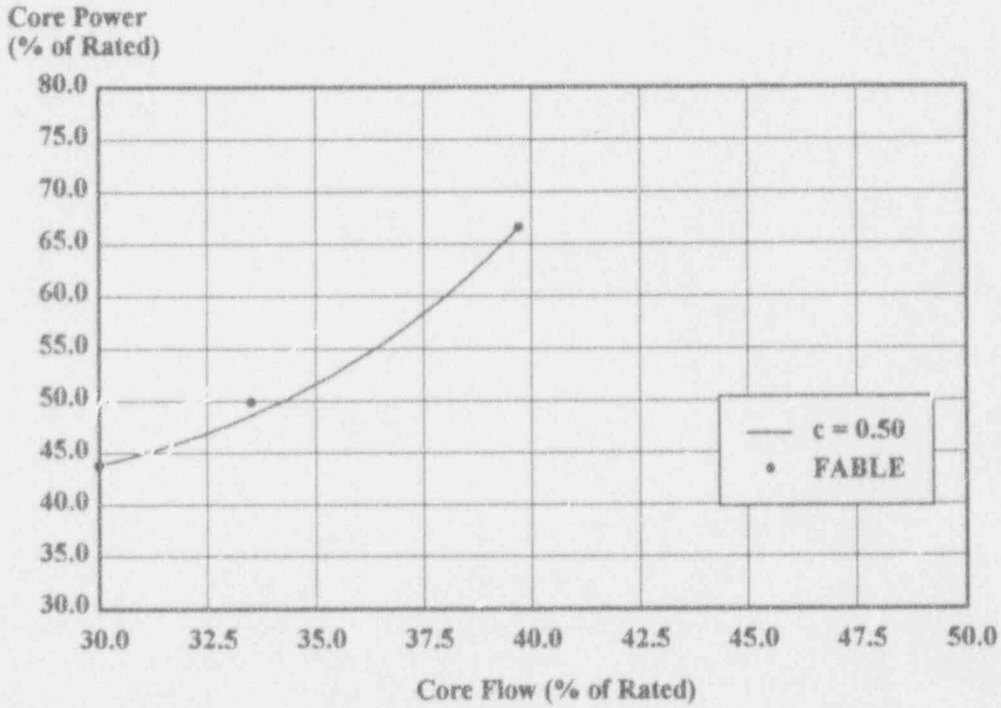


Figure F-3: Generic Shape Function for Monticello (Cycle 15)

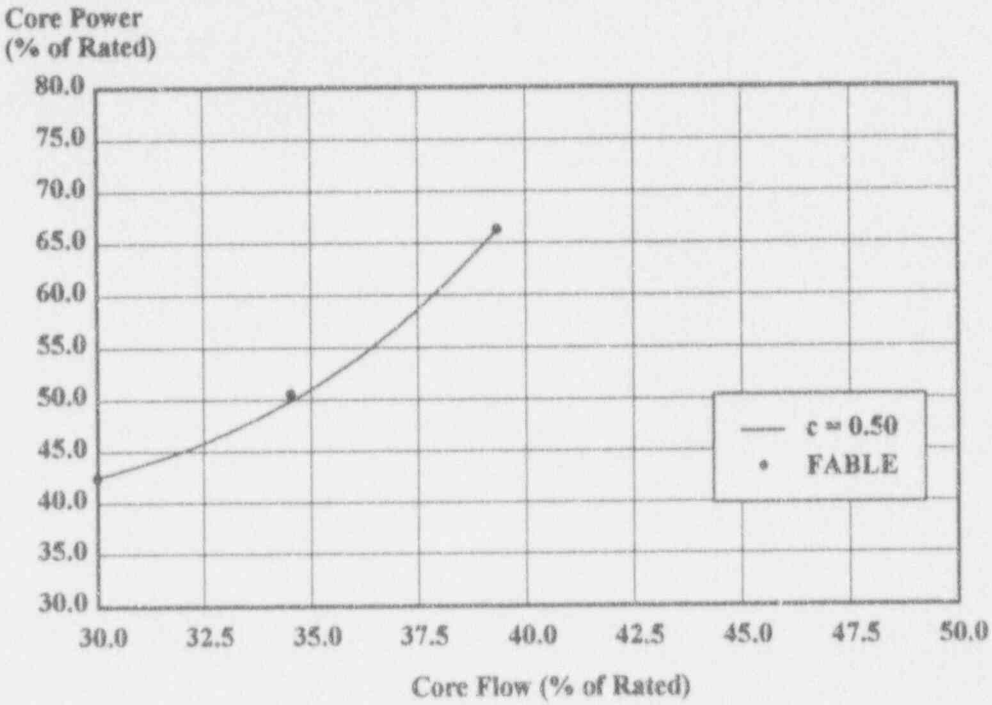


Figure F-4: Generic Shape Function for Duane Arnold (Cycle 10)

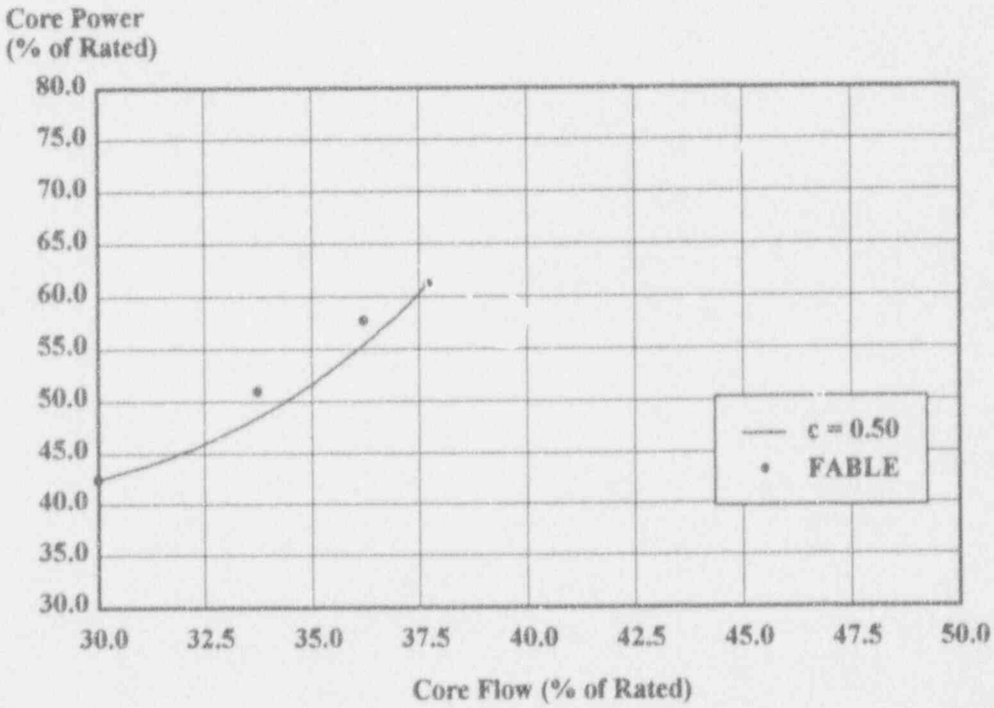


Figure F-5: Generic Shape Function for Perry (Cycle 2)

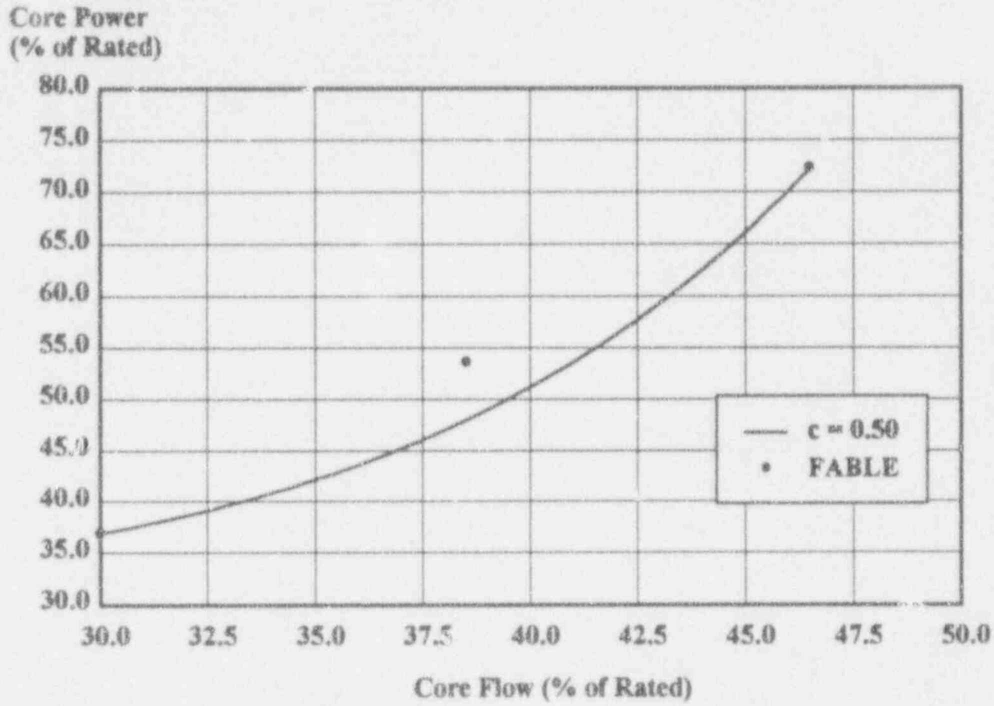
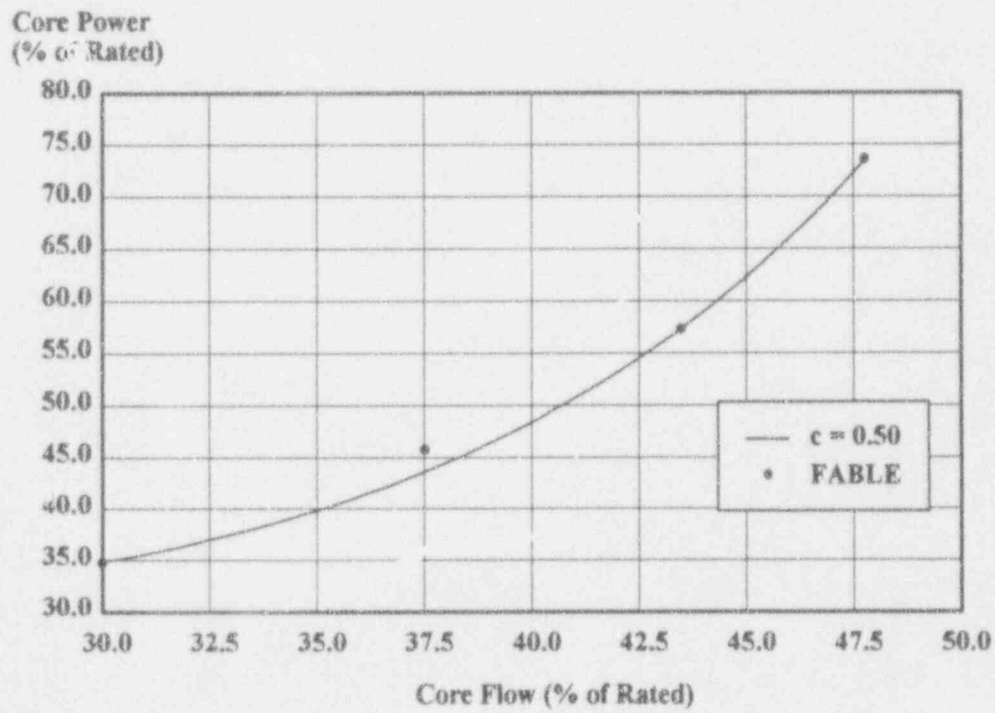


Figure F-6: Generic Shape Function for Perry (Equilibrium Cycle)



APPENDIX G: STABILITY REGION SETPOINT REQUIREMENTS FOR SINGLE-LOOP OPERATION

G.1 Introduction

The Flow Control Trip Reference Card (FCTRC) provides a power reference, for both the APRM flux trip and the control rod block, that is a function of inferred core flow rate. The flow indication used by the FCTRC is reactor recirculation drive flow. Therefore the FCTRC trip reference algorithm must accommodate the correlation between recirculation drive flow and core flow for all operating modes including natural circulation, two-loop operation and single-loop operation. In general there is no significant difference between the correlation under different recirculation operating modes and there is no need to adjust the Exclusion Region and Restricted Region boundary setpoints for each operating mode.

G.2 Plant Application

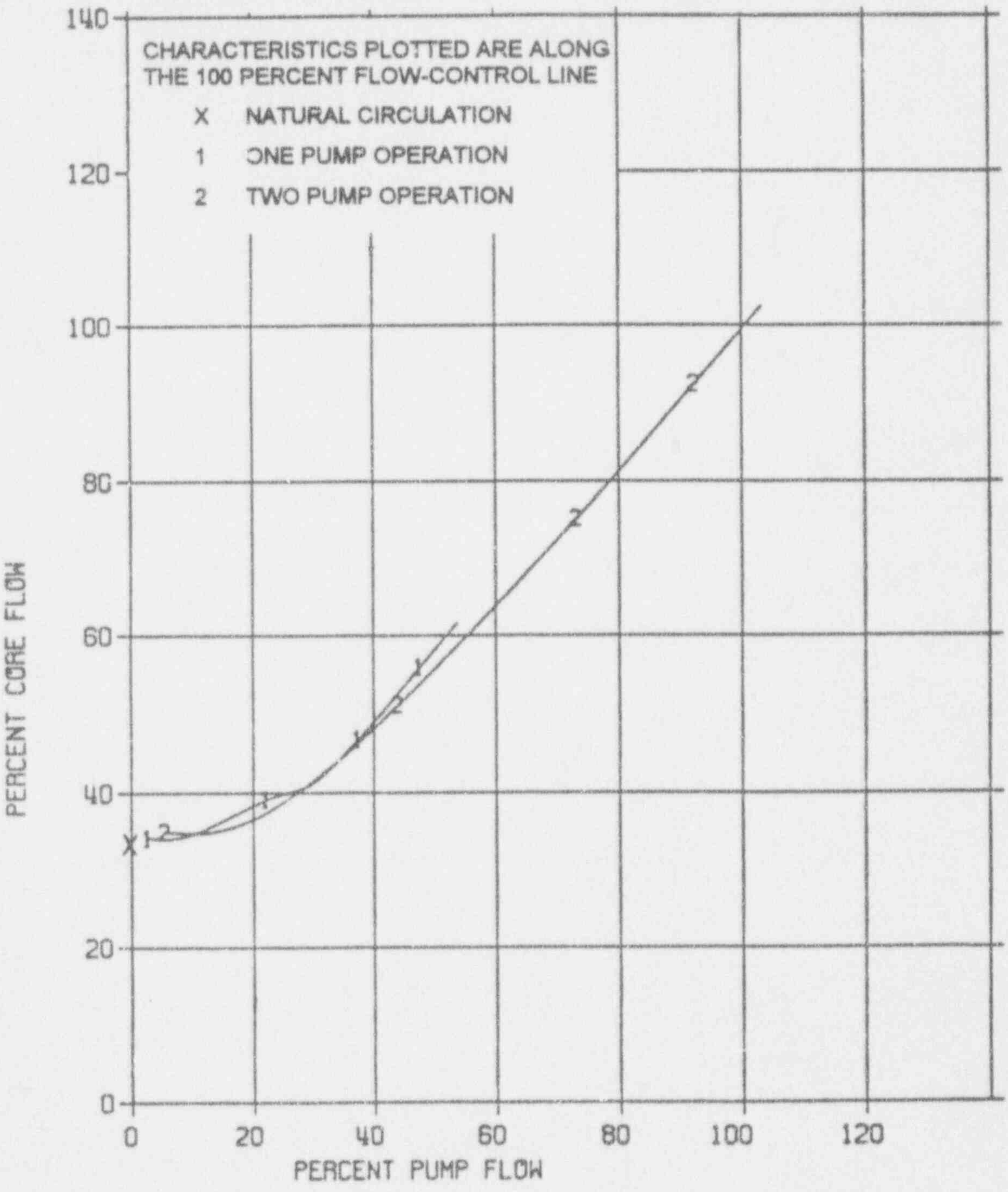
An important plant operating feature which must be considered when specifying the APRM flow-biased flux trip and control rod block setpoints for the stability regions is the difference in the recirculation drive flow and core flow correlation between the single-loop and two-loop operating modes. A typical BWR/6 correlation is shown in Figure G-1. As Figure G-1 indicates, there is very little difference between the single-loop and two-loop percent pump (drive) flow vs. percent core flow curves. In fact, above approximately 10% pump flow, single-loop operation produces a core flow which is greater than or equal to the two-loop core flow for a given recirculation drive flow. This results from the canceling effects of reverse flow through idle jet pumps and removal of parallel recirculation pump operation that occur when transitioning from two-loop to single-loop operation. Considering the shape of the Enhanced Option I-A region boundaries, a correlation which uses the two-loop curve when a single recirculation loop is operating would result in an underprediction of the core flow and a corresponding underprediction of the associated flux trip and rod block setpoints. Therefore, the Exclusion Region and Restricted Region boundary setpoints for two-loop operation are applicable for all reactor recirculation system operating modes.

This conclusion provides assurance that the Enhanced Option 1-A stability solution region boundaries are appropriate to protect against flow reduction events. In particular, intermediate flow reduction events that result from inadvertent entry into single-loop operation do not challenge the Exclusion Region boundary location. IFREs are generally expected to terminated above the 40% core flow clamp of the Exclusion Region. Since determining this core flow value from recirculation drive flow does not significantly change based on the recirculation operating mode, the single flow clamp value provides protection during events involving inadvertent changes in recirculation operating modes. Following such an event, manual operator action to switch the FCTRC trip references to the correct operating mode setpoints is required. However, these changes are unrelated to reactor stability and are required manual actions during plant operation.

G.3 Conclusion

Implementation of Enhanced Option I-A stability region boundary setpoints requires a correlation between recirculation pump (drive) flow and core flow. Since this correlation is, in general, conservative for single-loop operations and does not vary significantly between natural recirculation, single-loop, and two-loop operating modes, the same APRM flow-biased flux trip and control rod block setpoints that define the Exclusion Region and Restricted Regions provide stability protection in different operating modes. The existing plant-specific setpoints for core flows above the Restricted Region boundaries remain unchanged by implementation of the Enhanced Option I-A stability solution.

Figure G-1: Percent Core Flow vs. Percent Pump Flow for Single-Loop and Two-Loop Operation for a Typical BWR/6



4

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